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SP-100 PROGRAM

Power Plant System Assessment-Final Report

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SP-100 PROGRAM

SPACE REACTOR SYSTEM

AND

SUBSYSTEM INVESTIGATIONS

POWER PLANT SYSTEM ASSESSMENT

FINAL REPORT

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I. INTRODUCTION AND SUMMARY

A. PURPOSE AND SCOPE

At present, planners in the Department of Defense, Department of Energy, and National Aeronautics and Space Administration are evaluating the potential of reactor electric systems to provide power in space. This evaluation is being performed at the system level, in parallel with evaluations of mission requirements being performed by users and user agencies. As part of this evaluation, the Tri-Agency SP-100 Program contracted Rockwell International to conduct a 6-month power plant system assessment for a 100-kWe-class space reactor electric system.

The purpose of this assessment was to provide system-level insights into 100-kWe-class space reactor electric systems. Using these insights, Rockwell was to select and perform conceptual design studies on a "most attractive" system that met the preliminary design goals and requirements of the SP-100 Program. About 4 of the 6 months were used in the selection process. The remaining 2 months were used for the system conceptual design studies. Rockwell completed these studies at the end of FY 1983. This report summarizes the results of the power plant system assessment and describes our choice for the most attractive system — the Rockwell SR-100G System (Space Reactor, 100 kWe, Growth) — a lithium-cooled UN-fueled fast reactor/Brayton turbo-electric converter system.

The scope of the power plant system assessment encompassed nine key activities:

- Update Rockwell's previous subsystem and system studies in light of technical requirements of the SP-100 Program.
- Preliminarily characterize candidate systems and select the promising ones.
- Select the technically plausible candidate systems that fully meet requirements.

- Briefly characterize these technically plausible candidate systems.
- Identify the technology feasibility issues for these technically plausible candidate systems.
- Select the most attractive system that fully meets the requirements.
- Perform conceptual design studies for the most attractive system.
- Perform preliminary technology demonstration planning for the most attractive system.
- Select a reduced-power backup system that provides the best combination of performance and technical maturity, and preliminarily characterize this system.

These power plant system assessment activities were performed to a set of preliminary design goals and requirements for the SP-100 Program. The key design goals and requirements are shown in Table I-1.

TABLE I-1
KEY SP-100 DESIGN GOALS AND REQUIREMENTS

Requirement	Value
Power output	100 kWe
Full-power operation	7 yr
Reliability (first 2 yr)	95%
Neutron fluence (7 yr) ^a	10^{13} neutrons/cm ²
Gamma dose (7 yr) ^a	5×10^5 rads
Total power system mass	3000 kg
Power system length	6.1 m
Power system diameter	4.5 m maximum

^aAt payload, 25 m from the reactor face

B. SUMMARY OF SELECTIONS

This section summarizes the process used for selecting the most attractive system and briefly describes the SR-100G System. The information is presented by tracing the nine key activities identified in Section I.A.

The first key activity of the power plant system assessment was to update Rockwell's previous subsystem and system studies in light of the technical requirements of the SP-100 Program. The starting point for this activity was the recently completed Rockwell Subsystem Technology Assessment Report.¹ The key element from this prior assessment was a matrix of the various combinations of reactors and power converters that could be coupled to form a reactor electric system. The original matrix had 529 combinations of reactors and power converters (23 x 23). We reviewed this matrix with regard to the SP-100 design goals and requirements and with regard to launch time frame. Forty logical reactor-power converter matchups were selected that were available for early and mid-1990s launch time frames. These 40 candidate systems are shown in Figure I-1 in their original positions on the 23 x 23 matrix.

The second activity of the power plant system assessment was to preliminarily characterize the candidate systems and select the promising ones. Using a basic set of system design ground rules, the key performance parameters and configurational features of each of the 40 candidate systems were determined. The systems, identified by type of reactor and power converter, are summarized in Table I-2. The table shows the power capability of the system and the system launch availability. This second activity of the assessment continued with selection of promising ones from these 40 candidate systems. First, the 40 systems were characterized. Then, a decision model was used to evaluate each system in terms of:

- Ability to meet requirements
- Development risk

Figure I-1. Reduction of the Compatibility Matrix to 40 Candidate Systems

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TABLE I-2
SUMMARY OF THE 40 CANDIDATE SYSTEMS

SYSTEM NUMBER	REACTOR TYPE	CONVERSION TYPE	POWER CAPABILITY (kW _e)	AVAILABLE FOR LAUNCH	
				EARLY 1990s	MID 1990s
1	LMFR	TE	15		X
2	HPR	TE	15		X
3	HPR	TE	38		X
4	LMTR	TE	23	X	
5	LMFR	TE	23	X	
6	LMTR	TE	19	X	
7	LMFR	TE	19	X	
8	LMFR	TE	34		X
9	HPR	TE	43		X
10	TIR	TI	35	X	
11	TIR	TI	58		X
12	LMTR	CCB	39	X	
13	LMTR	CCB	47	X	
14	GCTR	CCB	25	X	
15	GCFR	CCB	57	X	
16	FBR	CCB	25		X
17	LMFR	CCB	37	X	
18	LMFR	CCB	57		X
19	GCTR	CCB	25		X
20	GCFR	CCB	59		X
21	LMFR	CCB	58		X
22	HPR	CCB	58		X
23	HPR	CCB	57		X
24	LMTR	OR	39	X	
25	LMTR	OR	39		X
26	LMFR	OR	37	X	
27	LMFR	OR	37		X
28	LMTR	MR	19	X	
29	LMTR	MR	20		X
30	LMFR	MR	18	X	
31	LMFR	MR	19	X	
32	LMFR	PR	32		X
33	LMFR	PR	25		X
34	BPFR	PR	44		X
35	HPR	PR	37		X
36	HPR	PR	37		X
37	LMTR	FPS	63	X	
38	LMFR	FPS	60	X	
39	LMTR	FPS	66	X	
40	LMFR	FPS	69		X

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LEGEND	
LMFR	LIQUID METAL (COOLED) FAST REACTOR
LMTR	LIQUID METAL (COOLED) THERMAL REACTOR
HPR	HEAT PIPE REACTOR
TIR	THERMIONIC REACTOR (IN CORE)
GCTR	GAS COOLED THERMAL REACTOR
GCFR	GAS COOLED FAST REACTOR
FBR	FIXED (PARTICULATE) BED REACTOR
BPFR	BOILING POTASSIUM (COOLED) FAST REACTOR
TE	THERMOELECTRIC
TI	THERMIONIC (IN CORE)
CCB	CLOSED CYCLE (GAS) BRAYTON
OR	ORGANIC RANKINE
MR	MERCURY RANKINE
PR	POTASSIUM RANKINE
FPS	FREE PISTON STIRLING

- Design complexity
- Life potential
- Scalability.

These five separable decision factors were used in a weighted, linear decision model to select the promising candidate systems. Using the model, the 12 systems listed in Table I-3 were selected as promising candidates. Five of the systems could be available for an early 1990s launch, and the remaining seven systems could be available for a mid-1990s launch. For those systems available for both launch dates, the system launched later is a more advanced version of the system launched earlier. Section II provides further details on this and the previous activities.

None of the systems would provide 100 kWe while meeting all other SP-100 design goals and requirements. These systems used enhanced technologies and were meant to meet the design goals and requirements of the SP-100 Program, which included the following: "If a requirement cannot be met with the design power of 100 kW, then the design power level shall be treated as a design variable." It soon became apparent that the real needs of the SP-100 Program were inconsistent with power levels less than 100 kWe (i.e., that power level should not be treated as a variable). The requirements were then changed, and our activities concentrated on identifying the advanced technologies needed to meet the 100-kWe goal.

The third activity of the power plant system assessment was to select the technically plausible candidate systems that fully meet requirements. In other words, to select the systems that could plausibly provide 100 kWe. This activity was accomplished by starting with each of the promising candidate systems shown in Table I-3. Operating temperatures were increased until the system provided 100 kWe and fully met the SP-100 preliminary design goals and

TABLE I-3
THE 12 PROMISING CANDIDATE SYSTEMS

SYSTEM NUMBER ^a	REACTOR TYPE	CONVERTER TYPE	POWER (kWe) AND LAUNCH AVAILABILITY	
			EARLY 1990s	MID-1990s
5, 8	LIQUID-METAL- COOLED FAST	THERMOELECTRIC	23	34
—, 9	HEAT-PIPE - COOLED FAST	THERMOELECTRIC	—	43
10, 11	THERMIONIC	THERMIONIC	35	58
15, 20	GAS-COOLED FAST	GAS BRAYTON	57	59
17, 18	LIQUID-METAL- COOLED FAST	GAS BRAYTON	37	57
—, 34	BOILING POTASSIUM FAST	POTASSIUM RANKINE	—	44
38, 40	LIQUID-METAL- COOLED FAST	FREE PISTON STIRLING	60	69

^a FIRST NUMBER REFERS TO A MID-TERM SYSTEM,
SECOND TO A MORE ADVANCED MID-TERM SYSTEM

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requirements. Once the necessary operating temperatures had been determined, the systems were reviewed for technical plausibility, and certain of the systems were eliminated. The five systems considered technically plausible are shown, with their reactor operating temperatures, in Table I-4. The systems use deployed or nondeployed radiators, as necessary; in some cases, both types were evaluated.

TABLE I-4
OPERATING TEMPERATURES FOR THE FIVE TECHNICALLY
PLAUSIBLE CANDIDATE SYSTEMS

Reactor	Power Converter	Reactor Outlet Temperature (K)	
		Deployed Radiator	Nondeployed Radiator
Li heat pipe fast	Thermoelectric	-	1620
NaK-cooled stainless steel fast	In-core thermionic	-	880
Li-cooled refractory fast	Closed cycle Brayton turboelectric	1465	1690
Li-cooled refractory fast	Potassium Rankine turboelectric	-	1470
Li-cooled refractory fast	Free piston Stirling reciprocating	1085	1285

The fourth activity of the power plant system assessment was to briefly characterize these technically plausible candidate systems. A set of ground rules and assumptions was used to evaluate each system. Operating statepoint and performance diagrams were developed, masses of the individual subsystems were determined, and system mass summaries were calculated. For each system,

Rockwell attempted to keep the mass somewhat below the 3000-kg requirement, but this was not always achieved. The mass and area characteristics of the systems are shown in Table I-5.

TABLE I-5
MASS AND AREA CHARACTERISTICS OF THE FIVE PLAUSIBLE CANDIDATE SYSTEMS

Reactor	Power Converter	System Mass (kg)	System Radiator Area (m ²)
Li heat pipe fast	Thermoelectric	2650	58 (nondeployed)
NaK-cooled stainless steel fast	In-core thermionic	2970	46 (nondeployed)
Li-cooled refractory fast	Dual 50-kWe closed-cycle Brayton turbo-electric	2750	85 (deployed)
		2730	52 (nondeployed)
Li-cooled refractory fast	Dual 50-kWe potassium Rankine turbo-electric	3390	30 (nondeployed)
Li-cooled refractory fast	Free piston Stirling reciprocating (50% extra cylinders)	3270	108 (deployed)
		3190	46 (nondeployed)

The fifth activity of the power plant system assessment was to identify the technology feasibility issues for these technically plausible candidate systems. The purpose was to answer two questions:

- Are there potential technology feasibility issues that could be "showstoppers" farther down the line?
- Are there tests that could be performed in the next 2 to 3 yr to determine which, if any, of these potential issues really are "showstoppers"?

The technology feasibility evaluation was performed by identifying the types of tests that could be performed to address the technology feasibility issues. The tests identified are shown in Table I-6. The potential technology feasibility issues identified for each type of reactor and power

converter employed in the plausible candidate systems were reviewed with respect to these tests. This review concluded that no reactor or power converter could be identified that did not have potential technology feasibility issues. Certain reactors and power converters had fewer issues than others, and this information was factored into the selection of the most attractive system.

TABLE I-6
TESTS WHICH ADDRESS TECHNOLOGY FEASIBILITY ISSUES

Designation	Time Required to Obtain Necessary Data
Performance test	Hours to days
Endurance (steady state)	Year (plus)
Endurance (cyclic loading)	Up to a year (plus)
Damage fraction summation (also known as accelerated life testing)	As required (generally a year plus)
Test to failure	As required
Compatibility, corrosion, erosion, mass transfer	Months to year (plus)

The sixth activity of the power plant system assessment was to select the most attractive system that fully meets the requirements. In making this selection, the Rockwell team believed that there were four key considerations:

- Most plausibly meeting all design goals and requirements
- Having good potential for meeting safety goals
- Being amenable to a fallback power converter that could also produce 100 kWe
- Having power growth potential with better-than-linear growth capability.

Within these considerations, no system was a clear-cut winner. Each technically plausible candidate system had several potential technology feasibility issues. We selected a most attractive system within these considerations: this was a lithium-cooled UN-fueled fast reactor coupled to a closed-cycle Brayton turboelectric converter. A preliminary statepoint diagram for the system is shown in Figure I-2, and preliminary system characteristics are presented in Table I-7.

The reactor and converter in this system operate at temperatures exceeding the applicable technology bases, but the needed advances in these bases are considered more plausible than those for the other systems. The system, more specifically the reactor, has good potential for meeting safety goals. The safety technology base for space reactors is specific to the liquid-metal-cooled pin-type reactor, and detailed safety features will be engineered into the system when the mission is more well defined. A Stirling power converter could be used as a fallback option with this system, depending on its development status. The reactor can grow over a factor of ten in thermal power and still remain compact, and the power converter can grow or can be used modularly. Thus, the system exhibits better-than-linear growth potential.

Section III further discusses the third, fourth, fifth, and sixth activities of the assessment.

The seventh activity of the power plant system assessment was to perform conceptual design studies for the most attractive system. This activity encompassed several key studies as shown in Table I-8. The studies were supported by subcontracted efforts of the Garrett Turbine Engine Company (Brayton converter) and Mechanical Technology Incorporated (Stirling engine). These studies led to several important changes in the system features. One of the most important was the switch to a radiator cooled directly by the converter working fluid (He-Xe gas). This reduced system mass and size by eliminating a

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Figure I-2. Preliminary Statepoint Diagram for the Most Attractive System — Lithium-Cooled Reactor With Dual Brayton Turboelectric Converters

TABLE I-7
PRELIMINARY CHARACTERISTICS OF THE MOST ATTRACTIVE SYSTEM

● SYSTEM	
■ NET POWER	100 kWe
■ MASS	2750 kg
● REACTOR	
■ POWER	420 kWt
■ FUEL	UO ₂ OR UN
■ MATERIALS	REFRACTORY ALLOY
■ COOLANT	LITHIUM
■ COOLANT OUTLET TEMPERATURE	1465 K
● SHIELD	
■ GAMMA	TUNGSTEN ALLOY
■ NEUTRON	LiH
● POWER CONVERTER	
■ TYPE	CLOSED-CYCLE BRAYTON TURBOELECTRIC
■ FLUID	He-Xe
■ NUMBER OF UNITS	DUAL HALF-POWER COUNTERROTATING
■ TURBINE INLET TEMPERATURE	1400 K
■ COMPRESSOR INLET TEMPERATURE	475 K
● RADIATOR	
■ TYPE	DEPLOYED, PUMPED NaK
■ AREA	85 m ²
■ AVERAGE TEMPERATURE	555 K

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TABLE I-8

CONCEPTUAL DESIGN STUDIES PERFORMED ON THE MOST ATTRACTIVE SYSTEM

- **OVERALL SYSTEM**
 - DESIGN POINT SELECTION
 - SYSTEM CONTROL
 - STARTUP AND RESTART
 - COMPONENT LOCATION
 - DEPLOYMENT OR NON-DEPLOYMENT
 - RELIABILITY
 - SURVIVABILITY
- **REACTOR SUBSYSTEM**
 - FUEL ELEMENT SIZE
 - REFLECTOR CONTROL CONFIGURATION
 - FUEL AND CLADDING MATERIALS SELECTION
- **SHIELDING SUBSYSTEM**
 - SHIELD THERMAL CONTROL
 - GAMMA SHIELD MATERIAL SELECTION
- **PRIMARY HEAT TRANSPORT SUBSYSTEM**
 - PUMP SELECTION
 - AFTERGLOW HEAT REMOVAL
 - DORMANCY THERMAL CONTROL
 - STARTUP
 - STRUCTURAL MATERIALS SELECTION
- **POWER CONVERSION AND PROCESSING SUBSYSTEM**
 - CLOSED CYCLE BRAYTON PERFORMANCE
 - USE OF REDUNDANT CONVERTERS
 - MATERIALS SELECTION
 - POWER PROCESSOR DESIGN
 - CONTROL
 - ALTERNATE USE OF STIRLING CONVERTER
- **HEAT REJECTION SUBSYSTEM**
 - CONFIGURATION AND PERFORMANCE
 - FLUID SELECTION
 - DEPLOYMENT MECHANISM
 - MATERIALS SELECTION

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heat exchanger and the temperature drop that occurred across it. Consequently, the Brayton conversion cycle could be reoptimized to reduce the required radiator area. The result was a system with two fully redundant 100-kWe converters using a fixed (nondeployed) gas-cooled radiator. We called this system the SR-100G. The system normally operates with both converters running at 50 kWe, but is designed to deliver a full 100 kWe with only a single converter operating. The statepoint diagram for this "design condition" is shown in Figure I-3. An artist's concept of the system is shown in Figure I-4, and its key features are presented in Table I-9. As can be seen from this table, the system mass slightly exceeded 3000 kg when the final accounting was done. Further detail on the system's key features and on the analyses performed to define those features is presented in Section IV.

The eighth activity of the power plant system assessment was to perform preliminary technology demonstration planning for the most attractive system. This activity started from the technology feasibility issues identified for the most attractive system. A technology demonstration plan was prepared based on these identified issues. This is an integrated overall plan for technology demonstration, leading to system demonstration at the earliest possible time. Four conditions are key to the plan:

- A detailed, in-depth, cost- and schedule-effective test program plan for the technology demonstration activities of the program. This test program plan should be in place before testing is begun.
- A well-defined baseline system design meeting the program-approved system specification. This must be established first to define the required test conditions/ranges to be investigated in the test program.
- A well-defined set of system materials specifications. The materials used in the tests should conform to program approved system materials specifications.
- Adequate test facilities. These must be capable of imposing test conditions that duplicate (as closely as practicable) the anticipated component, subsystem, or system operating conditions.

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Figure I-3. SR-100G System Statepoint Diagram (Design condition — one of two Brayton converters operating)

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Figure I-4. SR-100G Reactor Turboelectric Power System

TABLE I-9
KEY FEATURES OF THE SR-100G SYSTEM

● System	
Net power	100 kWe
Full-power life	7 yr
Maximum diameter	4.3 m
Length	6.1 m
Mass	3053 kg
● Reactor	
Fuel	UN
Cladding	W-lined, ASTAR-811C
Coolant	Li
Structure	ASTAR-811C
● Shield	
Gamma	CMW-3950 (W alloy)
Neutron	SS-canned LiH
● Power Converter	
Type	Closed-cycle Brayton turboelectric
Fluid	He-Xe
Number	Dual, full-power
Materials	ASTARs
● Radiator	
Type	Fixed, gas cooled
Fluid	He-Xe
Channel	Dual D-tube with shaped fin
Material	SS-clad Cu

This plan is summarized in Figure I-5, and further details are given in Section V.

The ninth activity of the power plant system assessment was to select a reduced-power backup system that provides the best combination of performance and technical maturity, and to preliminarily characterize this system. This activity complemented the others in the assessment because it was aimed at identifying a nearer-term system — one that would necessarily provide less than 100 kWe (within the mass and envelope constraints) — but one that would have little or no development risk. In the overall performance of the power plant system assessment, this activity was subordinate to studies for the most attractive system. However, during those studies, various reactor-power converter combinations were evaluated as backup systems and their characteristics determined. As it evolved, the first consideration for the backup system became the technical maturity of the combination. In considering this "best combination," there is a dilemma. The reactor with the most technical maturity is a liquid-metal-cooled, UO_2 -fueled, pin type that operates in the 975 K (1300°F) coolant outlet range (or lower) (i.e., an LMFBF-type reactor at SNAP reactor temperatures). The power converter with the most technical maturity for space power applications is the existing thermoelectric converter (or a slight improvement to it). But to achieve, within the imposed SP-100 mass and envelope requirements, reasonable system performance (i.e., power output) with thermoelectrics appears to require reactor outlet temperatures well in excess of the 975 K of the technically mature reactor. In other words, the technically mature reactor and the technically mature power converter are not compatible.

For a backup system, we chose a high-temperature reactor coupled to thermoelectric converters. The reactor, a lithium-cooled, UN-fueled fast reactor, operates at conditions essentially identical to those for the most attractive system reactor (e.g., 1470 K design point outlet temperature). Although a 1470 K reactor cannot be classified as technically mature, it does (1) obtain reasonable performance from the thermoelectrics, (2) maximize technology carrythrough from the most attractive system, and (3) provide for potential future growth in system power output.

Figure I-5. Technology Demonstration Plan Summary Schedule for Key Elements

However, as the backup system characterization progressed, it appeared more prudent to also provide technical maturity for the reactor (outlet temperature of 975 K or less). The potential power converters are then the Rankine and Brayton turboelectrics, the Stirling engine and the in-core thermionics. In-core thermionics are attractive but not mature since they may suffer from limited lifetimes due to insulator degradation and fuel swelling. Rankine turboelectrics are attractive but not mature since they face the questions of condensing and vapor liquid interface control in zero-g environments. The Stirling engine is attractive but cannot be classified as technically mature at this time because of its development status.

The Brayton turboelectric converter technology may be regarded as relatively mature at temperatures (turbine inlet) of 1150 K (1600°F) and below. The Brayton can be coupled to a mature reactor (the reactor at 975 K would limit the operating temperature of the Brayton to more mature values). Such a "low-temperature" Brayton system appears both attractive and mature. Figure I-6 presents some very preliminary power versus reactor temperature data for such a system, from the most mature temperature to much higher temperatures. The range of masses is due to the very preliminary calculations that generated this curve. Possibly more power could be available at the lower temperatures.

C. CONCLUSIONS

During the power plant system assessment in FY 1983, Rockwell concluded that an attractive 100-kWe system can be configured — a system with potential to meet the SP-100 design goals and requirements. However, this system would employ equipment operating at extremely high temperatures (approaching 1500 K). Rockwell believes that this high-temperature operation can be achieved — but to do so could require significant development time and funding. We believe that the system represents a starting point for interaction between power system designers and power system users, and also offers potential when reconfigured to operate at lower "mature technology" temperatures.

Figure I-6. Electrical Power versus Temperature Characteristics for a Reduced Temperature (Backup) Reactor - Brayton Turboelectric System

II. SELECTING THE PROMISING CANDIDATE SYSTEMS

A. DEFINING THE 40 CANDIDATES

The selection of the most attractive system meeting SP-100 Program requirements employed Rockwell's reactor and power conversion subsystems compatibility matrix. This matrix had been generated for the Rockwell Subsystem Technology Assessment Report.¹ The matrix had been assembled by reviewing technology advancements and enhancements on the reactor and power conversion subsystems. The matrix (Figure II-1) presents 529 potential combinations of reactors and power converters that could be used to generate electric power in space. The combinations shown in this figure are categorized by reactor type, power converter type, and level of technology.

These 529 combinations were purposefully assembled without a specific mission or set of functional requirements in mind. By applying such requirements to the matrix, it could be narrowed down into a group of promising candidate systems. This narrowing-down process was performed by reviewing the SP-100 Program requirements. These requirements are provided in Appendix A, and the key requirements are summarized in Table II-1. A further requirement, one that cannot yet be fully translated into specifics, is safety. Safety is a key requirement for the system under SP-100 Program guidelines.

With these requirements in mind, the first step in reducing the compatibility matrix was to investigate the time frame for system launch. The matrix had been assembled with three launch time frames in mind – early 1990s, mid-1990s, and late 1990s. Although no specific SP-100 requirements were provided on the launch time frame, it was decided (after discussion with program personnel) that the program needs would best be served by the early or mid-1990s launch time frames. In the terminology of the reactor-power converter compatibility matrix, this meant retaining only near-term or mid-term technologies. The far-term technologies (i.e., those with higher temperatures and/or

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TABLE II-1
KEY SP-100 DESIGN GOALS AND REQUIREMENTS

Requirement	Value
Power output	100 kWe
Full-power operation	7 yr
Total system lifetime	10 yr
Regulated power	100 to 400 Vdc
Load following	Limited
Reliability (first 2 yr) ^a	95%
Neutron fluence (7 yr) ^a	10^{13} neutrons/cm ²
Gamma dose (7 yr)	5×10^5 rads
Total power system mass	3000 kg
Power system length	6.1 m
Power system diameter	4.5 m maximum

^aOn a surface, at least 4.5 m in diameter, 25 m from the reactor face

higher efficiencies) would be available only in the late 1990s and thus were eliminated. The resulting reduced matrix, shown in Figure II-2, has 240 reactor-power converter combinations.

The next step was to reduce the matrix to a set of candidate systems. The matrix itself was only a tool for showing the interrelationships between types of reactors and power converters versus technology level. Investigation of the 240 remaining combinations showed that only 40 were logical matchups within the requirements of the SP-100 Program. In other words, combinations such as a gas-cooled reactor coupled to a thermionic converter could logically be eliminated. In the resulting matrix, shown in Figure II-3, the 40 candidate systems were numbered for convenience. These system numbers are used in this section of the report to refer to those systems.

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Figure II-2. Compatibility Matrix of Reactor-Power Converter Combinations for Early or Mid-1990s Launch Timeframes

4 CANDIDATE SYSTEM NUMBER

(REFERENCE TABLE II 4 ESG DOE 13398)

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Figure II-3. Reduction of the Matrix to 40 Candidate Systems

At this point in the selection process, the 40 candidate systems were only logical combinations of a type and technology level of reactor with a type and technology level of power converter. Next, these two major subsystems were assembled into overall system configurations with specific performance parameters and with shielding, heat transport, and heat rejection subsystems. The systems thus assembled specifically met the mass and envelope requirements for the SP-100 Program. Safety was considered key, and each system was assembled with safety in mind. Reliability was accommodated by redundant, active components in the subsystems. Deployable radiators were used where they were required and were reasonable. The systems used conical-cylindrical configurations with the reactor at the smaller (top) end of the conical section. A 15° conical half-angle was used, but this was varied where required. From all power converters, 10% overpower was required to account for degradation effects, power conditioning losses, and transmission line losses. Launch time frame was carried as an independent variable.

The above discussion does not include power as a key system requirement. For the purpose of these studies, power was considered a variable. This was due to the statement in the design goals document that if any other requirement could not be met at the design power level of 100 kW, the design power level would be treated as a design variable. In other words, power would be reduced to meet mass, envelope, reliability, and safety requirements.

The systems that resulted from this analysis are summarized in Tables II-2 and II-3. Table II-2 gives the system number, type of reactor employed, type of power converter employed, system power, and the launch availability date (either the early 1990s or mid-1990s). Table II-3 briefly describes the systems. We evaluated these 40 systems and downselected to determine the promising candidate systems for further evaluation.

In designing the systems, safety was considered to the extent possible at this point in the design process. Many safety issues are at the detail level of the design process. The purpose here was to identify whether a system

TABLE II-2
THE 40 CANDIDATE SYSTEMS

SYSTEM NUMBER	REACTOR TYPE	CONVERSION TYPE	POWER CAPABILITY (kWe)	AVAILABLE FOR LAUNCH	
				EARLY 1990s	MID 1990s
1	LMFR	TE	15		X
2	HPR	TE	15		X
3	HPR	TE	38		X
4	LMTR	TE	23	X	
5	LMFR	TE	23	X	
6	LMTR	TE	19	X	
7	LMFR	TE	19	X	
8	LMFR	TE	34		X
9	HPR	TE	43		X
10	TIR	TI	35	X	
11	TIR	TI	58		X
12	LMTR	CCB	39	X	
13	LMTR	CCB	47	X	
14	GCTR	CCB	25	X	
15	GCFR	CCB	57	X	
16	FBR	CCB	25		X
17	LMFR	CCB	37	X	
18	LMFR	CCB	57		X
19	GCTR	CCB	25		X
20	GCFR	CCB	59		X
21	LMFR	CCB	58		X
22	HPR	CCB	58		X
23	HPR	CCB	57		X
24	LMTR	OR	39	X	
25	LMTR	OR	39		X
26	LMFR	OR	37	X	
27	LMFR	OR	37		X
28	LMTR	MR	19	X	
29	LMTR	MR	20		X
30	LMFR	MR	18	X	
31	LMFR	MR	19	X	
32	LMFR	PR	32		X
33	LMFR	PR	25		X
34	BPFR	PR	44		X
35	HPR	PR	37		X
36	HPR	PR	37		X
37	LMTR	FPS	63	X	
38	LMFR	FPS	60	X	
39	LMTR	FPS	66	X	
40	LMFR	FPS	69		X

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LEGEND	
LMFR	LIQUID METAL (COOLED) FAST REACTOR
LMTR	LIQUID METAL (COOLED) THERMAL REACTOR
HPR	HEAT PIPE REACTOR
TIR	THERMIONIC REACTOR (IN CORE)
GCTR	GAS COOLED THERMAL REACTOR
GCFR	GAS COOLED FAST REACTOR
FBR	FIXED (PARTICULATE) BED REACTOR
BPFR	BOILING POTASSIUM (COOLED) FAST REACTOR
TE	THERMOELECTRIC
TI	THERMIONIC (IN CORE)
CCB	CLOSED CYCLE (GAS) BRAYTON
OR	ORGANIC RANKINE
MR	MERCURY RANKINE
PR	POTASSIUM RANKINE
FPS	FREE PISTON STIRLING

TABLE II-3
 DESCRIPTIONS OF THE 40 CANDIDATE SYSTEMS
 (Sheet 1 of 16)

NO	DESCRIPTOR	SUBSYSTEM DATA					ADDITIONAL COMMENTS
		REACTOR	SHIELDING	PRIMARY HEAT TRANSPORT	POWER CONVERSION	HEAT REJECTION	
1	TE SiGe/UO ₂ FAST REACTOR	UO ₂ FUEL Na COOLED 1100K OUTLET TEMP SUPERALLOY CLAD REFLECTOR CONTROLLED	LiH AND W ~ 50° CONE HALF ANGLE ACTIVE SHIELD COOLING REQUIRED	PUMPED LOOP REDUNDANT EM PUMPS AND VAUS WITH ISOLATION D SHAPED TUBES DISTRIBUTION NETWORK	BINARY DOPED SiGe BONDED TO D TUBES WITH HOT SHOE AT 1025K T _c ~ 630K	DIRECT RADIATING COLD SHOES WITH BONDED FINS ε = 0.9 T _{rej} ~ 590K	RADIATOR SIZE INCREASED BY VARYING SHIELD CONE HALF ANGLE TO OPTIMIZE POWER OUTPUT
2	TE SiGe/Na HEAT PIPE REACTOR	UO ₂ FUEL Na/SUPER ALLOY HEAT PIPE COOLED 1100K EVAPORATOR TEMP BE REFLECTOR CONTROLLED	LiH AND W ~ 50° CONE HALF ANGLE ACTIVE SHIELD COOLING REQUIRED	Na HEAT PIPES SUPERALLOY WALL 0.9mm THICKNESS 17.3mm I.D., 9.2 kWt/PIPE, 1.5 POWER DESIGN MARGIN	BINARY DOPED SiGe CONDUCTIVE COUPLING TO HP T _H ~ 1025K T _c ~ 635K	DIRECT RADIATING COLD SHOES WITH BONDED FINS, ε = 0.9 T _{rej} ~ 600K	RADIATOR SIZE INCREASED BY VARYING SHIELD CONE HALF ANGLE TO OPTIMIZE POWER OUTPUT
3	TE SiGe/Li HEAT PIPE REACTOR	UO ₂ FUEL Li/Mo Re HEAT PIPE COOLED, ~1440K EVAP TEMPERATURE BE REFLECTOR CONTROLLED	LiH AND W ~ 37° CONE HALF ANGLE ACTIVE SHIELD COOLING RE QUIRED	Li HEAT PIPES Mo Re WALL 0.8mm WALL THICKNESS, 16 mm I.D., 10.7 kWt/PIPE 1.5 POWER DESIGN MARGIN	BINARY DOPED SiGe, RADIATIVE COUPLING HOT SHOE TEMP ~ 1250K T _c ~ 780K	DIRECT RADIATING COLD SHOES WITH BONDED FINS, ε = 0.9 T _{rej} ~ 740K	RADIATOR SIZE INCREASED BY VARYING SHIELD CONE HALF ANGLE TO OPTIMIZE POWER OUTPUT

TABLE II-3
DESCRIPTIONS OF THE 40 CANDIDATE SYSTEMS
(Sheet 2 of 16)

NO	DESCRIPTOR	SUBSYSTEM DATA					ADDITIONAL COMMENTS
		REACTOR	SHIELDING	PRIMARY HEAT TRANSPORT	POWER CONVERSION	HEAT REJECTION	
4	TE PbTe COMPACT/ UZrH REACTOR	UZrH FUEL NaK COOLANT HAST N CLAD 920K OUTLET TEMP, BE REFLECTOR CONTROLLED	LiH AND W ~15° CONE HALF ANGLE	PUMPED NaK LOOP REDUNDANT EM PUMPS AND VAUs ^a WITH ISOLATION	PbTe COMPACT CON VERTER (TUBULAR MODULES) $\bar{T}_H \sim 850K$ $\bar{T}_c \sim 530K$ NaK HEATING AND COOLING LOOPS	SECONDARY NaK HR LOOP WITH HX TO DTA ^b HEAT REJECTION LOOP $\bar{T}_{rej} \sim 520K$ REDUNDANT PUMPS AND VAUs FOR BOTH HRL ^c SUBSYSTEMS	DEPLOYABLE RADIATOR, WITH ORGANIC COOLANT AND ARMOR (NPP 0.99 FOR INITIAL LIFE), DEPLOYMENT ADDS ~ 10% TO RADIATOR MASS
5	TE PbTe COMPACT UO ₂ FAST REACTOR	UO ₂ FUEL NaK COOLED SST CLAD 920K OUTLET TEMP, BE REFLECTOR CONTROLLED	LiH AND W ~15° CONE HALF ANGLE	PUMPED NaK LOOP REDUNDANT EM PUMPS AND VAUs WITH ISOLATION	PbTe COMPACT CON VERTER (TUBULAR MODULES) $\bar{T}_H \sim 850K$ $\bar{T}_c \sim 530K$ NaK HEATING AND COOLING LOOPS	SECONDARY NaK HR LOOP WITH HX TO DTA HEAT REJECTION LOOP $\bar{T}_{rej} \sim 520K$ REDUNDANT PUMPS AND VAUs FOR BOTH HRL SUBSYSTEMS	DEPLOYABLE RADIATOR, WITH ORGANIC COOLANT ^d AND ARMOR (NPP 0.99 FOR INITIAL LIFE), DEPLOYMENT ADDS ~ 10% TO RADIATOR MASS
6	TE TAGS CONVERTER UZrH REACTOR	UZrH FUEL NaK COOLANT HAST N CLAD 920K OUT LET TEMP, BE REFLECTOR CONTROLLED	LiH AND W ~15° CONE HALF ANGLE	PUMPED NaK LOOP REDUNDANT EM PUMPS AND VAUs WITH ISOLATION	TAGS CONVERTER MOD ULES SIMILAR IN DESIGN TO 4 NaK HEATING AND COOLING LOOPS $\bar{T}_H \sim 820K$ $\bar{T}_c \sim 510K$	SECONDARY NaK HR LOOP WITH HX TO DTA HEAT REJECTION LOOP $\bar{T}_{rej} \sim 500K$ REDUNDANT PUMPS AND VAUs FOR BOTH HRL SUBSYSTEMS	DEPLOYABLE RADIATOR, WITH ORGANIC COOLANT AND ARMOR (NPP 0.99 FOR INITIAL LIFE), DEPLOYMENT ADDS ~ 10% TO RADIATOR MASS
7	TE TAGS CONVERTER UO ₂ FAST REACTOR	UO ₂ FUEL NaK COOLED SST CLAD 920K OUTLET TEMP, BE REFLECTOR CONTROLLED	LiH AND W ~15° CONE HALF ANGLE	PUMPED NaK LOOP REDUNDANT EM PUMPS AND VAUs WITH ISOLATION	TAGS CONVERTER MOD ULES SIMILAR IN DESIGN TO 4 NaK HEATING AND COOLING LOOPS $\bar{T}_H \sim 820K$ $\bar{T}_c \sim 510K$	SECONDARY NaK HR LOOP WITH HX TO DTA HEAT REJECTION LOOP $\bar{T}_{rej} \sim 500K$ REDUNDANT PUMPS AND VAUs FOR BOTH HRL SUBSYSTEMS	DEPLOYABLE RADIATOR, WITH ORGANIC COOLANT AND ARMOR (NPP 0.99 FOR INITIAL LIFE), DEPLOYMENT ADDS ~ 10% TO RADIATOR MASS

^aVAU VOLUME ACCUMULATOR UNIT
^bDTA DOWTHERM "A" COOLANT
^cHRL HEAT REFLECTION LOOP
^dNPP NONPUNCTURE PROBABILITY

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TABLE II-3
DESCRIPTIONS OF THE 40 CANDIDATE SYSTEMS
(Sheet 3 of 16)

NO	DESCRIPTOR	SUBSYSTEM DATA					ADDITIONAL COMMENTS
		REACTOR	SHIELDING	PRIMARY HEAT TRANSPORT	POWER CONVERSION	HEAT REJECTION	
8	TE SiGe GAP/ FAST Li COOLED REACTOR	UN FUEL, Li COOLANT, T 111 CLAD, 1350K OUTLET TEMP, Be REFLECTOR CONTROLLED	LiH AND W ~23° CONE HALF ANGLE, ACTIVE SHIELD COOLING REQUIRED	PUMPED LOOP, REDUNDANT EM PUMPS AND VAUs WITH ISO LATION, D-SHAPED TUBES DISTRIBUTION NETWORK	TERNARY DOPED SiGe (GaP), BONDED TO D TUBES WITH HOT SHOE @ 1250K T _c ~ 770K	DIRECT RADIATING COLD SHOES WITH BONDED FINs, ε = 0.9 T _{rej} ~ 740K	RADIATOR SIZE INCREASED BY VARY- ING SHIELD CONE HALF-ANGLE TO OPTIMIZE POWER OUTPUT
9	TE SiGe GAP/ Li HEAT-PIPE REACTOR	UO ₂ FUEL Li/Mo Re HEAT PIPE COOLED, ~1420K EVAPORATOR TEMP, Be REFLECTOR CONTROLLED	LiH AND W 33° CONE HALF ANGLE, ACTIVE SHIELD COOLING REQUIRED	Li HEAT PIPES Mo Re WALL 0.8mm WALL THICKNESS, 16mm I.D., 10.7 kWt/PIPE, 1.5 POWER DESIGN MARGIN	TERNARY DOPED SiGe (GaP), RADIATIVE COUPLING, HOT SHOE TEMP. ~1250K, T _c ~ 780K	DIRECT RADIATING COLD SHOES WITH BONDED FINs, ε = 0.9 T _{rej} ~ 740K	RADIATOR SIZE INCREASED BY VARY- ING SHIELD CONE HALF ANGLE TO OPTIMIZE POWER OUTPUT

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TABLE II-3
DESCRIPTIONS OF THE 40 CANDIDATE SYSTEMS
(Sheet 4 of 16)

NO	DESCRIPTOR	SUBSYSTEM DATA					ADDITIONAL COMMENTS
		REACTOR	SHIELDING	PRIMARY HEAT TRANSPORT	POWER CONVERSION	HEAT REJECTION	
10	IN CORE THERMIONICS	UO ₂ FUELED TFEs UZrH DRIVER FUEL ELEMENTS ADDED FOR CRITICALITY WITH 3.3cm DIAMETER TFEs NaK COOLED WITH 850K REACTOR OUTLET TEMP	LiH AND W 17° CONE HALF ANGLE	CESIUM WORKING FLUID, T _{Res} ~ 600K, 0.25mm INTER ELECTRODE GAP	BASED ON THE GA MARK 7A (2E2 TEST SERIES) CONVERTER TUNGSTEN DUPLEX EMITTER, TRILAYER COLLECTOR WITH Al ₂ O ₃ INSULATOR T _E ~ 1850K, T _C ~ 880K VENTING OF FISSION GAS TO SEPARATE PLENUM 10% REDUNDANCY IN NUMBER OF CONVERTERS	PUMPED NaK LOOP REDUNDANT EM PUMPS AND VAUs WITH ISOLATION, FIXED RADIATOR WITH ARMOR (NPP 0.99 FOR INITIAL LIFE) T _{rej} ~ 800K	RADIATOR SIZE INCREASED BY VARYING SHIELD CONE HALF ANGLE TO OPTIMIZE POWER OUTPUT
11	IN CORE THERMIONICS	UO ₂ FUELED TFEs UZrH DRIVER FUEL ELEMENTS ADDED FOR CRITICALITY WITH 3.3cm DIAMETER TFEs NaK COOLED WITH 880K REACTOR OUTLET TEMP	LiH AND W 15° CONE HALF ANGLE	CESIUM WORKING FLUID, T _{Res} ~ 580K, 0.50mm INTER ELECTRODE GAP	SIMILAR TO DESIGN IN 10 BUT CONVERTER OXYGENATED TO LOWER BARRIER INDEX T _E ~ 1820K, T _C ~ 910K	PUMPED NaK LOOP REDUNDANT EM PUMPS AND VAUs WITH ISOLATION, FIXED RADIATOR WITH ARMOR (NPP 0.99 FOR INITIAL LIFE) T _{rej} ~ 840K	RADIATOR SIZE INCREASED BY VARYING SHIELD CONE HALF ANGLE TO OPTIMIZE POWER OUTPUT

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TABLE II-3
 DESCRIPTIONS OF THE 40 CANDIDATE SYSTEMS
 (Sheet 5 of 16)

NO.	DESCRIPTOR	SUBSYSTEM DATA					ADDITIONAL COMMENTS
		REACTOR	SHIELDING	PRIMARY HEAT TRANSPORT	POWER CONVERSION	HEAT REJECTION	
12.	CLOSED-CYCLE BRAYTON/UZrH REACTOR	UZrH FUEL HAST.-N CLAD NaK COOLED 920K OUTLET TEMP. BE REFLECTOR CONTROL	LiH AND W 15° CONE HALF-ANGLE	PUMPED NaK LOOPS WITH HXs ^a AND LOOP ISOLATION CAPABILITY REDUNDANT PHTS LOOPS WITH EM PUMPS AND VAUs	FOUR SEPARATE 50% CAPACITY CCGT UNITS WORKING FLUID IS He-Xe MIXTURE (MW=40) T _{INLET} ~900K C _{INLET} ~410K R _c ~1.80	PUMPED ORGANIC COOLANT LOOPS REDUNDANT HR LOOPS DEPLOYABLE RADIATOR T _{rej.} ~450K	DEPLOYABLE RADIATOR WITH ORGANIC COOLANT AND ARMOR (NPP-0.99 FOR INITIAL LIFE), DEPLOYMENT ADDS ~10% TO RADIATOR MASS. ALTERNATOR COOLING PROVIDED FROM PRIMARY RADIATOR COOLANT.
13.	CLOSED-CYCLE BRAYTON/UZrH REACTOR	UZrH FUEL HAST.-N CLAD NaK COOLED 980K OUTLET TEMP. BE REFLECTOR CONTROL	LiH AND W 15° CONE HALF-ANGLE	PUMPED NaK LOOPS WITH HXs AND LOOP ISOLATION CAPABILITY REDUNDANT PHTS LOOPS WITH EM PUMPS AND VAUs	FOUR SEPARATE 50% CAPACITY CCGT UNITS WORKING FLUID IS He-Xe MIXTURE (MW=40) T _{INLET} ~960K C _{INLET} ~415K R _c ~1.85	PUMPED ORGANIC COOLANT LOOPS REDUNDANT HR LOOPS DEPLOYABLE RADIATOR T _{rej.} ~460K	DEPLOYABLE RADIATOR WITH ORGANIC COOLANT AND ARMOR (NPP-0.99 FOR INITIAL LIFE), DEPLOYMENT ADDS ~10% TO RADIATOR MASS. ALTERNATOR COOLING PROVIDED FROM PRIMARY RADIATOR COOLANT.
14.	CLOSED-CYCLE BRAYTON/HTGR DERIVATIVE	HTGR DERIVATIVE Epi THERMAL SPECTRUM GRAPHITE CORE UC ₂ FUEL BEADS PyC CLAD He-Xe COOLANT (MW=40) 1100K OUTLET TEMP SUPERALLOY STRUCTURE REFLECTOR CONTROLLED	LiH AND W AUGMENTED SHIELD COOLING REQUIRED 15° CONE HALF-ANGLE	DUAL REACTOR GAS OUTLET DUCTS 10-15cm DIAMETER.	FOUR 50% CAPACITY DIRECT CYCLE BRAYTON UNITS WITH HT LOOP ISOLATION CAPABILITY T _{INLET} ~1100K C _{INLET} = 425K R _c ~2.0	PUMPED ORGANIC COOLANT LOOPS REDUNDANT HR LOOPS WITH FIXED RADIATOR. T _{rej.} ~480K	ALTERNATOR COOLING PROVIDED FROM PRIMARY RADIATOR COOLANT.

^aHX: HEAT EXCHANGER

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TABLE II-3
DESCRIPTIONS OF THE 40 CANDIDATE SYSTEMS
(Sheet 6 of 16)

NO.	DESCRIPTOR	SUBSYSTEM DATA					ADDITIONAL COMMENTS
		REACTOR	SHIELDING	PRIMARY HEAT TRANSPORT	POWER CONVERSION	HEAT REJECTION	
15.	CLOSED-CYCLE BRAYTON/GAS-COOLED METAL CLAD REACTOR	UO ₂ FUEL HA ST . X CLAD FAST SPEC TRUM He-Xe(MW=40) COOLANT 1100K OUTLET TEMP. BeO REFLECTED AND CONTROLLED	LiH AND W AUGMENTED SHIELD COOLING REQUIRED 15° CONE HALF-ANGLE	DUAL REACTOR GAS OUTLET DUCTS 10-15cm DIAMETER	FOUR 50% CAPACITY DIRECT CYCLE BRAYTON UNITS WITH HT ^a LOOP ISOLATION CAPABILITY T _{INLET} ~ 1100K T _{INLET} ^b = 425K R _c ~ 2.0 ^c	PUMPED ORGANIC COOLANT LOOPS REDUNDANT HR LOOPS DEPLOYABLE RADIATOR T _{rej.} ~ 480K	DEPLOYABLE RADIATOR WITH ORGANIC COOLANT AND ARMOR (NPP-0.99 FOR INITIAL LIFE), DEPLOYMENT ADDS ~10% TO RADIATOR MASS. ALTERNATOR COOLING PROVIDED FROM PRIMARY RADIATOR COOLANT.
16.	CLOSED CYCLE BRAYTON/FIXED BED REACTOR	PACKED BED TRISO FUEL PARTICLES THERMAL SPECTRUM CONCENTRIC POROUS FRITS (INCONEL) IN ANNULAR GEOMETRY He Xe(MW=40) COOLANT 1100K OUTLET TEMP. BE MODERATED AND EXTERNAL BE REFLECTOR CONTROLLED	LiH AND W AUGMENTED SHIELD COOLING REQUIRED 15° CONE HALF ANGLE	DUAL REACTOR GAS OUTLET DUCTS 10-15cm DIAMETER.	FOUR 50% CAPACITY DIRECT CYCLE BRAYTON UNITS WITH HT LOOP ISOLATION CAPABILITY T _{INLET} ~ 1100K C _{INLET} = 425K R _c ~ 2.0	PUMPED ORGANIC COOLANT LOOPS REDUNDANT HR LOOPS WITH FIXED RADIATOR. T _{rej.} ~ 480K	ALTERNATOR COOLING PROVIDED FROM PRIMARY RADIATOR COOLANT.
17	CLOSED CYCLE BRAYTON/UO ₂ FAST REACTOR	UO ₂ FUEL SST CLAD NaK COOLED 920K OUTLET TEMP BE REFLECTOR CONTROL	LiH AND W 15° CONE HALF ANGLE	PUMPED NaK LOOPS WITH HXs AND LOOP ISOLATION CAPABILITY REDUNDANT PHTS LOOPS WITH EM PUMPS AND VALVES	FOUR SEPARATE 50% CAPACITY UNITS WORKING FLUID IS He Xe MIXTURE (MW=40) T _{INLET} ~ 900K C _{INLET} ~ 410K R _c ~ 1.80	PUMPED ORGANIC COOLANT LOOPS REDUNDANT HR LOOPS DEPLOYABLE RADIATOR T _{rej.} ~ 450K	DEPLOYABLE RADIATOR WITH ORGANIC COOLANT AND ARMOR (NPP-0.99 FOR INITIAL LIFE), DEPLOYMENT ADDS ~10% TO RADIATOR MASS. ALTERNATOR COOLING PROVIDED FROM PRIMARY RADIATOR COOLANT

^aHT: HEAT TRANSFER

^bC_{INLET}: COMPRESSOR INLET TEMPERATURE

^cR_c: COMPRESSOR PRESSURE RATIO

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TABLE II-3
 DESCRIPTIONS OF THE 40 CANDIDATE SYSTEMS
 (Sheet 7 of 16)

NO	DESCRIPTOR	SUBSYSTEM DATA					ADDITIONAL COMMENTS
		REACTOR	SHIELDING	PRIMARY HEAT TRANSPORT	POWER CONVERSION	HEAT REJECTION	
18	CLOSED-CYCLE BRAYTON/LMCFR	UO ₂ FUEL FAST SPEC TRUM SUPERALLOY CLAD Na COOLED 1127K OUTLET TEMP BE REFLECTOR CONTROLLED	LiH AND W AUGMENTED SHIELD COOL ING REQUIRED 15° CONE HALF ANGLE	Na PUMPED LOOPS WITH HXs REDUNDANT PHTS LOOPS WITH EM PUMPS AND VAUs AND LOOP ISOLATION CAPABILI TY	FOUR SEPARATE 50% CAPACITY CCGT UNITS WORKING FLUID IS He Xe MIXTURE (MW=40) T _{INLET} ~ 1100K C _{INLET} ~ 425K R _c ~ 2.0	PUMPED ORGANIC COOLANT LOOPS REDUNDANT HR LOOPS WITH ISOLA TION CAPABILITY DEPLOYABLE RAD IATOR T _{rej} ~ 480K	DEPLOYABLE RAD IATOR WITH ORGA NIC COOLANT AND ARMOR (NPP-0 99 FOR INITIAL LIFE). DEPLOYMENT ADDS ~10% TO RADIATOR MASS ALTERNATOR COOL ING PROVIDED FROM PRIMARY RADIATOR COOLANT ↓
19	CLOSED CYCLE BRAYTON/HTGR DERIVATIVE	HIGH TEMP VERSION OF REACTOR IN 14 INCLUDES ADVANCED FUEL BEAD DESIGN 1200K OUT LET TEMP AND REFRAC TORY METAL ALLOY STRUCTURE	LiH AND W AUGMENTED SHIELD COOL ING REQUIRED 15° CONE HALF ANGLE	DUAL REACTOR GAS OUTLET DUCTS 9 14cm DIAMETER	FOUR 50% CAPACITY DIRECT CYCLE BRAYTON UNITS WITH HT LOOP ISOLATION CAPABILITY T _{INLET} ~ 1200K C _{INLET} ~ 440K R _c ~ 2.1	PUMPED ORGANIC COOLANT LOOPS WITH ISOLATION CAPABILITY REDUNDANT HR LOOPS WITH DE PLOYABLE RADIA TOR T _{rej} ~ 500K	
20	CLOSED CYCLE BRAYTON/GCFR	UO ₂ OR W UO ₂ FUEL MATRIX REFRACTORY METAL CLAD AND STRUC TURE FAST SPEC TRUM He Xe (MW=40) COOLANT MIXTURE 1200K OUT LET TEMP BeO REFLECT ED AND CON TROLLED	LiH AND W AUGMENTED SHIELD COOL ING REQUIRED 15° CONE HALF ANGLE	DUAL REACTOR GAS OUTLET DUCTS 9 14cm DIAMETER	FOUR 50% CAPACITY DIRECT CYCLE BRAYTON UNITS WITH HT LOOP ISOLATION CAPABILITY T _{INLET} ~ 1200K C _{INLET} ~ 440K R _c ~ 2.1	PUMPED ORGANIC COOLANT LOOPS WITH ISOLATION CAPABILITY REDUNDANT HR LOOPS WITH DE PLOYABLE RADIA TOR T _{rej} 500K	

TABLE II-3
DESCRIPTIONS OF THE 40 CANDIDATE SYSTEMS
(Sheet 8 of 16)

NO	DESCRIPTOR	SUBSYSTEM DATA					ADDITIONAL COMMENTS
		REACTOR	SHIELDING	PRIMARY HEAT TRANSPORT	POWER CONVERSION	HEAT REJECTION	
21	CLOSED CYCLE BRAYTON/LMCFR	UN FUEL T III CLAD Li COOLED 1230K OUTLET TEMP. FAST SPECTRUM BE REFLECTOR CONTROL	LiH AND W AUGMENTED SHIELD COOLING REQUIRED 15° CONE HALF-ANGLE	PUMPED Li LOOPS WITH HXs AND LOOP ISOLATION CAPABILITY REDUNDANT PHTS LOOPS WITH EM PUMPS AND VAUs	FOUR SEPARATE 50% CAPACITY CCGT UNITS WORKING FLUID IS He-Xe MIXTURE (MW=40) T _{INLET} ~1200K C _{INLET} ~440K R _c ~2.1	PUMPED ORGANIC COOLANT LOOPS WITH ISOLATION CAPABILITY REDUNDANT HR LOOPS WITH DEPLOYABLE RADIATOR T _{rej.} ~500K	DEPLOYABLE RADIATOR WITH ORGANIC COOLANT AND ARMOR (NPP-0.99 FOR INITIAL LIFE), DEPLOYMENT ADDS ~10% TO RADIATOR MASS. ALTERNATOR COOLING PROVIDED FROM PRIMARY RADIATOR COOLANT.
22	CLOSED CYCLE BRAYTON/Li HP REACTOR	UO ₂ FUEL Li/Mo Re HEAT PIPE COOLED 1250K EVAPORATOR TEMP BE REFLECTOR CONTROLLED	LiH AND W AUGMENTED SHIELD COOLING REQUIRED 15° CONE HALF-ANGLE	Li-HEAT PIPES WITH 2 HXs Mo-Re WALL .75mm WALL THICKNESS 14mm I.D. 5.1 KWt/PIPE	FOUR SEPARATE 50% CAPACITY CCGT UNITS WORKING FLUID IS He-Xe MIXTURE (Mwt=40) T _{INLET} ~1200K C _{INLET} ~440K R _c ~2.1	PUMPED ORGANIC COOLANT LOOPS WITH ISOLATION CAPABILITY REDUNDANT HR LOOPS WITH DEPLOYABLE RADIATOR T _{rej.} ~500K	
23	CLOSED CYCLE BRAYTON/Na HP REACTOR	UO ₂ FUEL Na/SUPER ALLOY HEAT PIPE COOLED, 1120K EVAP. TEMPERATURE BE REFLECTOR CONTROLLED	LiH AND W AUGMENTED SHIELD COOLING REQUIRED 15° CONE HALF-ANGLE	Na-HEAT PIPES WITH 2HXs SUPERALLOY 0.9mm WALL THICKNESS 14mm I.D. 6.5 kWt/PIPE 1.5 DESIGN POWER MARGIN	FOUR SEPARATE 50% CAPACITY CCGT UNITS WORKING FLUID IS He-Xe MIXTURE (MW=40) T _{INLET} ~1100K C _{INLET} ~425K R _c ~2.0	PUMPED ORGANIC COOLANT LOOPS REDUNDANT HR LOOPS WITH ISOLATION CAPABILITY DEPLOYABLE RADIATOR T _{rej.} ~480K	

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TABLE II-3
DESCRIPTIONS OF THE 40 CANDIDATE SYSTEMS
(Sheet 9 of 16)

NO	DESCRIPTOR	SUBSYSTEM DATA					ADDITIONAL COMMENTS
		REACTOR	SHIELDING	PRIMARY HEAT TRANSPORT	POWER CONVERSION	HEAT REJECTION	
24	ORGANIC RANKINE/ UZrH REACTOR	UZrH FUEL NaK COOLANT, HAST N CLAD, 690K OUTLET TEMP. BE REFLECTOR CONTROLLED	LiH AND W ~15° CONE HALF ANGLE	TWO REDUNDANT NaK PUMPED LOOPS WILL LOOP ISOLATION CAPABILITY EACH LOOP EQUIPPED WITH AN EM PUMP AND VAU	FOUR SEPARATE 50% CAPACITY ORGANIC RANKINE PCUs ^a WITH NaK DTA BOILER WITH DIPS TYPE SUBSYSTEM TINLET ~ 630K EACH PCU HAS HT LOOP ISOLATION CAPABILITY	TWO REDUNDANT HR LOOPS FOR ORGANIC WORKING FLUID DEPLOYABLE RADIATOR T _{rej} ~ 440K	ALTERNATOR COOL ING PROVIDED BY PRIMARY RADIATOR COOLANT OUTLET LINE
25	ORGANIC RANKINE/ UZrH REACTOR	UZrH FUEL NaK COOLANT, HAST N CLAD, 700K OUTLET TEMP, BE REFLECTOR CONTROLLED	LiH AND W ~15° CONE HALF ANGLE	TWO REDUNDANT NaK PUMPED LOOPS WILL LOOP ISOLATION CAPABILITY EACH LOOP EQUIPPED WITH AN EM PUMP AND VAU	FOUR SEPARATE 50% CAPACITY ORGANIC RANKINE PCUs WITH NaK DTA BOILER WITH DIPS ^b TYPE SUBSYSTEM TINLET ~ 645K EACH PCU HAS HT LOOP ISOLATION CAPABILITY	TWO REDUNDANT HR LOOPS FOR ORGANIC WORKING FLUID DEPLOYABLE RADIATOR T _{rej} ~ 450K	ALTERNATOR COOL ING PROVIDED BY PRIMARY RADIATOR COOLANT OUTLET LINE
26	ORGANIC RANKINE/ UO ₂ FAST REACTOR	UO ₂ FUEL NaK COOLED, SST CLAD, 690K OUTLET TEMP BE REFLECTOR CONTROLLED	LiH AND W ~15° CONE HALF ANGLE	TWO REDUNDANT NaK PUMPED LOOPS WILL LOOP ISOLATION CAPABILITY EACH LOOP EQUIPPED WITH AN EM PUMP AND VAU	FOUR SEPARATE 50% CAPACITY ORGANIC RANKINE PCUs WITH NaK DTA BOILER WITH DIPS TYPE SUBSYSTEM TINLET ~ 630K EACH PCU HAS HT LOOP ISOLATION CAPABILITY	TWO REDUNDANT HR LOOPS FOR ORGANIC WORKING FLUID DEPLOYABLE RADIATOR T _{rej} ~ 440K	ALTERNATOR COOL ING PROVIDED BY PRIMARY RADIATOR COOLANT OUTLET LINE
27	ORGANIC RANKINE/ UO ₂ FAST REACTOR	UO ₂ FUEL NaK COOLED, SST CLAD, 700K OUTLET TEMP BE REFLECTOR CONTROLLED	LiH AND W ~15° CONE HALF ANGLE	TWO REDUNDANT NaK PUMPED LOOPS WILL LOOP ISOLATION CAPABILITY EACH LOOP EQUIPPED WITH AN EM PUMP AND VAU	FOUR SEPARATE 50% CAPACITY ORGANIC RANKINE PCUs WITH NaK DTA BOILER WITH DIPS TYPE SUBSYSTEM TINLET ~ 645K EACH PCU HAS HT LOOP ISOLATION CAPABILITY	TWO REDUNDANT HR LOOPS FOR ORGANIC WORKING FLUID DEPLOYABLE RADIATOR T _{rej} ~ 450K	ALTERNATOR COOL ING PROVIDED BY PRIMARY RADIATOR COOLANT OUTLET LINE

^aPCU POWER CONVERTER UNIT
^bDIPS DYNAMIC ISOTOPE POWER SYSTEM

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TABLE II-3
 DESCRIPTIONS OF THE 40 CANDIDATE SYSTEMS
 (Sheet 10 of 16)

NO	DESCRIPTOR	SUBSYSTEM DATA					ADDITIONAL COMMENTS
		REACTOR	SHIELDING	PRIMARY HEAT TRANSPORT	POWER CONVERSION	HEAT REJECTION	
28	MERCURY RANKINE/ UZrH REACTOR	UZrH FUEL NaK COOLANT HAST N CLAD 920K OUTLET TEMP BE REFLECTOR CONTROLLED	LiH AND W ~15° CONE HALF ANGLE	TWO REDUNDANT NaK PUMPED LOOPS WITH ISOLATION CAPABI- LITY EACH LOOP HAS A THERMOELECTRIC PUMP AND EXPANSION COMPENSATOR COLD SHOE OF TE PUMP USES MERCURY WORKING FLUID AS COOLANT AND PRE- HEATS MERCURY IN BOILER FEED LINE	FOUR 50% CAPACITY "SNAP 2 TYPE" MER- CURY RANKINE PCUs TINLET ~ 895K TWO PARALLEL TRAINS OF TWO CRU ^a /BOILER UNITS EACH ARE CON- NECTED TO THE REDUN- DANT PHTS ^b LOOPS	MERCURY TUR- BINE EXHAUST LINES FROM EACH SET OF PCUs ARE CONNECTED WITH A COMMON EXHAUST MANI- FOLD WHICH HAS ISOLATION CAPA- BILITY. DIRECT CONDENSING RADIATOR USING SHARED FIN DE- SIGN T _{rej} ~ 589K	FIXED RADIATOR ALTERNATOR COOL- ED BY MERCURY WORKING FLUID RETURN FROM RADIATOR/CON- DENSER. MERCURY LUBRICATED BEARINGS FOR CRUs. ARMOR NPP-0.99 FOR INITIAL LIFE.
29.	MERCURY RANKINE/ UZrH REACTOR	UZrH FUEL NaK COOLANT HAST N CLAD 980K OUTLET TEMP., BE REFLECTOR CONTROLLED	LiH AND W ~15° CONE HALF ANGLE	TWO REDUNDANT NaK PUMPED LOOPS WITH ISOLATION CAPABILITY EACH LOOP HAS A NaK PUMP/INDUCTION MOTOR DRIVEN, VOLUME EXPANSION COMPENSATOR, AND DRIVES TWO "SNAP 8 TYPE" MERCURY BOILERS.	FOUR 50% CAPACITY "SNAP 8 TYPE" MER- CURY RANKINE PCUs WITH 20% TURBINE BLADE EFFICIENCY IMPROVEMENT TINLET ~ 950K TWO PARALLEL TRAINS OF TWO PCUs EACH ARE CONNECTED TO THE REDUNDANT PHTS LOOPS	TWO REDUNDANT PUMPED NaK LOOPS PROVIDE HEAT REJECTION FOR THE FOUR MER- CURY CONDEN- SERS. T _{rej} ~ 560K EACH LOOP HAS ISOLATION CAPA- BILITY. A SEPA- RATE ORGANIC COOLANT LOOP WITH REDUNDANT PUMPS PROVIDES HEAT REMOVAL FROM THE NaK AND MERCURY PMA ^d AND ALTER- NATOR. T _{rej} ~ 380K	PRIMARY RADIATOR IS A FIXED TUBE AND FIN USING PUMPED NaK FLUID. ARMOR NPP-0.99 FOR INITIAL LIFE) ^c TAA HAS ORGANIC LUBRICATED BEARINGS ORGANIC LUB/COOLANT HR RADIATOR IS A FIXED TUBE AND FIN DESIGN. SAME ARMOR DESIGN AS THE PRIMARY RADIATOR.

^aCRU: COMBINED ROTATING UNIT

^bPHTS: PRIMARY HEAT TRANSPORT SUBSYSTEM

^cTAA: TURBINE ALTERNATOR ASSEMBLY

^dPMA: PUMP MOTOR ASSEMBLY

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TABLE II-3
DESCRIPTIONS OF THE 40 CANDIDATE SYSTEMS
(Sheet 11 of 16)

NO	DESCRIPTOR	SUBSYSTEM DATA					ADDITIONAL COMMENTS
		REACTOR	SHIELDING	PRIMARY HEAT TRANSPORT	POWER CONVERSION	HEAT REJECTION	
30	MERCURY RANKINE/ UO ₂ FAST REACTOR	UO ₂ FUEL NaK COOLED SST CLAD 920K OUTLET TEMP BE REFLECTOR CONTROLLED	LiH AND W ~15° CONE HALF ANGLE	TWO REDUNDANT NaK PUMPED LOOPS WITH ISOLATION CAPABIL ITY EACH LOOP HAS A THERMOELECTRIC PUMP AND EXPANSION COMPENSATOR COLD SHOE OF TE PUMP USES MERCURY WORK ING FLUID AS COOL ANT AND PREHEATS MERCURY IN BOILER FEED LINE	FOUR 50% CAPACITY "SNAP 2 TYPE" MER CURY RANKINE PCUs T _{INLET} ~ 895K TWO PARALLEL TRAINS OF TWO CRU/BOILER UNITS EACH ARE CON NECTED TO THE REDUNDANT PHTS LOOPS	MERCURY TUR BINE EXHAUST LINES FROM EACH SET OF PCUs ARE CONNECTED WITH A COMMON EXHAUST MANI FOLD WHICH HAS ISOLATION CAPA BILITY DIRECT CONDENSING RADIATOR USING SHARED FIN DESIGN T _{rej} ~ 589K	FIXED RADIATOR ALTERNATOR COOLED BY MERCURY WORK ING FLUID RETURN FROM RADIATOR/ CONDENSER MER CURY LUBRICATED BEARINGS FOR CRUs ARMOR NPP 0 99 FOR INITIAL LIFE
31	MERCURY RANKINE/ UO ₂ FAST REACTOR	UO ₂ FUEL SUPERALLOY CLAD NaK COOLED 1000K OUT LET TEMP REFLECTOR CONTROLLED	LiH AND W ~15° CONE HALF ANGLE	TWO REDUNDANT NaK PUMPED LOOPS WITH ISOLATION CAPABILITY EACH LOOP HAS A NaK PUMP/INDUCTION MOTOR DRIVEN, VOLUME EXPANSION COMPENSATOR, AND DRIVES TWO "SNAP 8 TYPE" MERCURY BOILERS	FOUR 50% CAPACITY "SNAP 8 TYPE" MER CURY RANKINE PCUs WITH 20% TURBINE BLADE EFFICIENCY IMPROVEMENT, T _{INLET} ~ 950K TWO PARALLEL TRAINS OF TWO PCUs EACH ARE CONNECTED TO THE REDUNDANT PHTS LOOPS	TWO REDUNDANT PUMPED NaK LOOPS PROVIDE HEAT REJECTION FOR THE FOUR MER CURY CONDEN SERS T _{rej} ~ 560K EACH LOOP HAS ISOLATION CAPA BILITY A SEPA RATE ORGANIC COOLANT LOOP WITH REDUNDANT PUMPS PROVIDES HEAT REMOVAL FROM THE NaK AND MERCURY PMA _s AND ALTER NATOR T _{rej} ~ 380K	PRIMARY RADIATOR IS A FIXED TUBE AND FIN USING PUMPED NaK FLUID (ARMOR NPP 0 99 FOR INITIAL LIFE) TAA HAS ORGANIC LUBRICATED BEARINGS ORGANIC LUB/COOLANT HR RADIATOR IS A FIXED TUBE AND FIN DESIGN SAME ARMOR DESIGN AS THE PRIMARY RADIATOR

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TABLE II-3
DESCRIPTIONS OF THE 40 CANDIDATE SYSTEMS
(Sheet 12 of 16)

NO	DESCRIPTOR	SUBSYSTEM DATA					ADDITIONAL COMMENTS
		REACTOR	SHIELDING	PRIMARY HEAT TRANSPORT	POWER CONVERSION	HEAT REJECTION	
32	POTASSIUM UO ₂ FAST REACTOR	UO ₂ FUEL SUPERALLOY CLAD Na COOLED 1145K OUTLET TEMP, BE REFLECTOR CONTROLLED	LiH AND W AUGMENTED SHIELD COOLING REQUIRED 15° CONE HALF ANGLE	TWO REDUNDANT Na PUMPED LOOPS WITH ISOLATION CAPABILITY EACH LOOP HAS A Na PUMP/INDICATION MOTOR DRIVEN, VOLUME EXPANSION COMPENSATOR AND DRIVES TWO Na/K BOILERS.	FOUR 50% CAPACITY PCUs COUPLED INTO TWO PARALLEL POWER TRAINS. EACH PCU CONSISTS OF A LOW PRESSURE KTA, ^a A SEPARATE TURBINE DRIVEN POTASSIUM PUMP, AND BOILER FEED HEATER. T _{INLET} ~1110K	POTASSIUM TURBINE EXHAUST LINES FROM EACH SET OF PCUs ARE CONNECTED WITH A COMMON EXHAUST MANIFOLD WHICH HAS ISOLATION CAPABILITY. DIRECT CONDENSING PRIMARY RADIATOR. T _{rej.} ~833K	KTA AND PHTS Na PUMPS HAVE POTASSIUM LUBRICATED BEARINGS AND COOLING BY AN AUXILIARY LIQUID POTASSIUM PUMPED LOOP HR SYSTEM. T _{rej.} ~580K
33	POTASSIUM RANKINE/FAST Li COOLED REACTOR	UN FUEL, Li COOLANT, T 111 CLAD, 1325K OUTLET TEMP, BE REFLECTOR CONTROLLED	LiH AND W AUGMENTED SHIELD COOLING REQUIRED 15° CONE HALF ANGLE	TWO REDUNDANT Li PUMPED LOOPS WITH ISOLATION CAPABILITY EACH LOOP HAS A Li PUMP/INDUCTION MOTOR DRIVEN, VOLUME EXPANSION COMPENSATOR AND DRIVES TWO Li/K BOILERS	FOUR 50% CAPACITY PCUs COUPLED INTO TWO PARALLEL POWER TRAINS T _{INLET} ~1280K EACH PCU CONSISTS OF A MODERATE PRESSURE KTA EQUIPPED WITH MOISTURE SEPARATORS, BOILER FEED HEATER AND INDUCTION MOTOR DRIVEN POTASSIUM PUMP	TWO REDUNDANT PUMPED NaK LOOPS PROVIDE HEAT REJECTION FOR THE FOUR POTASSIUM CONDENSERS. EACH HR LOOP HAS ISOLATION CAPABILITY. PUMPED NaK LOOP PRIMARY RADIATOR. T _{rej.} ~950K	KTA, PHTS Li PUMPS, AND POTASSIUM PUMPS HAVE A SEPARATE LUB./COOLANT HR LOOP USING A LIQUID POTASSIUM COOLANT. T _{rej.} ~580K

^aKTA: POTASSIUM TURBINE ALTERNATOR

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TABLE II-3
DESCRIPTIONS OF THE 40 CANDIDATE SYSTEMS
(Sheet 13 of 16)

NO	DESCRIPTOR	SUBSYSTEM DATA					ADDITIONAL COMMENTS
		REACTOR	SHIELDING	PRIMARY HEAT TRANSPORT	POWER CONVERSION	HEAT REJECTION	
34	POTASSIUM RANKINE/DIRECT CYCLE BKFR	BOILING POTASSIUM COOLANT UO ₂ FUEL FAST SPECTRUM SST CLAD 1110K OUTLET TEMP REFLECTOR CONTROLLED	LiH AND W AUGMENTED SHIELD COOLING REQUIRED 15° CONE HALF ANGLE	REDUNDANT POTASSIUM VAPOR DUCTS WITH ISOLATION CAPABILITY	FOUR 50% CAPACITY PCUs COUPLED INTO TWO PARALLEL POWER TRAINS EACH PCU CONSISTS OF A LOW PRESSURE KTA, A SEPARATE TURBINE DRIVEN POTASSIUM PUMP, AND BOILER FEED HEATER T _{INLET} ~1110K	POTASSIUM TURBINE EXHAUST LINES FROM EACH SET OF PCUs ARE CONNECTED WITH A COMMON EXHAUST MANIFOLD WHICH HAS ISOLATION CAPABILITY DIRECT CONDENSING PRIMARY RADIATOR T _{rej} ~833K	KTA HAS POTASSIUM LUBRICATED BEARINGS AND COOLING BY AN AUXILIARY LIQUID POTASSIUM PUMPED LOOP HR SYSTEM T _{rej} ~580K
35	POTASSIUM RANKINE/Na HP REACTOR	UO ₂ FUEL, Na/SUPERALLOY HEAT PIPE COOLED, 1125K EVAPORATOR TEMP BE REFLECTOR CONTROLLED	LiH AND W AUGMENTED SHIELD COOLING REQUIRED 15° CONE HALF ANGLE	Na HEAT PIPES SUPERALLOY WALL 0.9 mm THICKNESS 17.3 mm I.D., 9.2 kWt/PIPE 1.5 POWER DESIGN MARGIN REDUNDANT, Na HEAT PIPE DRIVEN POTASSIUM BOILERS PROVIDE EACH SET OF PCUs WITH POTASSIUM VAPOR THE BOILER VAPOR DUCTS HAVE ISOLATION CAPABILITY	FOUR 50% CAPACITY PCUs COUPLED INTO TWO PARALLEL POWER TRAINS EACH PCU CONSISTS OF A LOW PRESSURE KTA, A SEPARATE TURBINE DRIVEN POTASSIUM PUMP, AND BOILER FEED HEATER T _{INLET} ~1110K	POTASSIUM TURBINE EXHAUST LINES FROM EACH SET OF PCUs ARE CONNECTED WITH A COMMON EXHAUST MANIFOLD WHICH HAS ISOLATION CAPABILITY DIRECT CONDENSING PRIMARY RADIATOR T _{rej} ~833K	KTA HAS POTASSIUM LUBRICATED BEARINGS AND COOLING BY AN AUXILIARY LIQUID POTASSIUM PUMPED LOOP HR SYSTEM T _{rej} ~580K

BKFR - BOILING POTASSIUM FAST REACTOR

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TABLE II-3
DESCRIPTIONS OF THE 40 CANDIDATE SYSTEMS
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NO	DESCRIPTOR	SUBSYSTEM DATA					ADDITIONAL COMMENTS
		REACTOR	SHIELDING	PRIMARY HEAT TRANSPORT	POWER CONVERSION	HEAT REJECTION	
36	POTASSIUM RANKINE/LI-HP ^a REACTOR	UO ₂ FUEL, Li/Mo-Re HEAT PIPE COOLED, 1300K EVAPORATOR TEMP BE REFLECTOR CONTROLLED	LiH AND W AUGMENTED SHIELD COOLING REQUIRED 15° CONE HALF ANGLE	Li/Mo Re HEAT PIPES 0.75 mm WALL THICKNESS, 14 mm I.D., 7.1 kWt/PIPE REDUNDANT Li-HEAT PIPE DRIVEN POTASSIUM BOILERS PROVIDE EACH SET OF PCUs WITH POTASSIUM VAPOR THE BOILER VAPOR DUCTS HAVE ISOLATION CAPABILITY	FOUR 50% CAPACITY PCUs COUPLED INTO TWO PARALLEL POWER TRAINS TINLET ~1280K EACH PCU CONSISTS OF A MODERATE PRESSURE KTA EQUIPPED WITH MOISTURE SEPARATORS, BOILER FEED HEATER, AND INDUCTION MOTOR DRIVEN POTASSIUM PUMP	TWO REDUNDANT PUMPED NaK LOOPS PROVIDE HEAT REJECTION FOR THE FOUR POTASSIUM CONDENSORS EACH HR LOOP HAS ISOLATION CAPABILITY PUMPED NaK LOOP PRIMARY RADIATOR $\bar{T}_{rej} \sim 950K$	KTA, AND POTASSIUM PUMPS HAVE A SEPARATE LUB / COOLANT HR LOOP USING A LIQUID POTASSIUM COOLANT $\bar{T}_{rej} \sim 580K$
37	STIRLING ENGINE/UZrH REACTOR	UZrH FUEL, NaK COOLANT, HASTIN CLAD, 920K OUTLET TEMP BE REFLECTOR CONTROLLED	LiH AND W 15° CONE HALF ANGLE	REDUNDANT PUMPED NaK LOOPS WITH ISOLATION CAPABILITY PHTS LOOP HAS EM PUMP, VAU, AND HX TUBE BUNDLE FOR EACH PCU	FREE PISTON STIRLING ENGINE REDUNDANT PCUs CONSISTING OF A FOUR CYLINDER OPPOSED CONFIGURATION, WITH HOT END COUPLED PAIRS OF PISTONS RUNNING IN OPPOSITION TO MINIMIZE VIBRATION $T_H \sim 895K$ $T_C \sim 478K$ FOUR LINEAR ALTERNATORS COUPLED TO THE PISTONS IN EACH CYLINDER PROVIDE THE ELECTRICAL POWER OUTPUT	REDUNDANT ORGANIC COOLANT LOOPS ARE PROVIDED WITH ISOLATION CAPABILITY FOR EACH PCU PUMPED LOOP TUBE AND FIN RADIATOR DESIGN WITH ARMOR AND DEPLOYMENT MECHANISM $\bar{T}_{rej} \sim 460K$	LINEAR ALTERNATOR COOLING IS PROVIDED BY HEAT REJECTION THROUGH FPSE ^b COOLING JACKET TO PRIMARY RADIATOR COOLANT (DTA) PHTS THERMOELECTRIC PUMP MODULES REJECT HEAT TO PRIMARY RADIATOR COOLANT

^aLI-HP. LITHIUM HEAT PIPE
^bFPSE FREE PISTON STIRLING ENGINE

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TABLE II-3
DESCRIPTIONS OF THE 40 CANDIDATE SYSTEMS
(Sheet 15 of 16)

NO	DESCRIPTOR	SUBSYSTEM DATA					ADDITIONAL COMMENTS
		REACTOR	SHIELDING	PRIMARY HEAT TRANSPORT	POWER CONVERSION	HEAT REJECTION	
38	STIRLING ENGINE/ UO_2 FAST REACTOR	UO_2 FUEL, NaK COOLED, SST CLAD, 920K OUTLET TEMP BE REFLECTOR CONTROLLED	LiH AND W 15° CONE HALF-ANGLE	REDUNDANT PUMPED NaK LOOPS WITH ISOLATION CAPABILITY PHTS LOOP HAS EM PUMP, VAU, AND HX TUBE BUNDLE FOR EACH PCU	FREE PISTON STIRLING ENGINE REDUNDANT PCUs CONSISTING OF A FOUR CYLINDER OPPOSED CONFIGURATION, WITH HOT END COUPLED PAIRS OF PISTONS RUNNING IN OPPOSITION TO MINIMIZE VIBRATION $T_H \sim 895\text{K}$ $T_C \sim 478\text{K}$ FOUR LINEAR ALTERNATORS COUPLED TO THE PISTONS IN EACH CYLINDER PROVIDE THE ELECTRICAL POWER OUTPUT	REDUNDANT ORGANIC COOLANT LOOPS ARE PROVIDED WITH ISOLATION CAPABILITY FOR EACH PCU PUMPED LOOP TUBE AND FIN RADIATOR DESIGN WITH ARMOR AND DEPLOYMENT MECHANISM $T_{rej} \sim 460\text{K}$	LINEAR ALTERNATOR COOLING IS PROVIDED BY HEAT REJECTION THROUGH FPSE COOLING JACKET TO PRIMARY RADIATOR COOLANT (DTA) PHTS THERMOELECTRIC PUMP MODULES REJECT HEAT TO PRIMARY RADIATOR COOLANT
39	STIRLING ENGINE/ UZrH REACTOR	UZrH FUEL HAST N CLAD NaK COOLED 980K OUTLET TEMP BE REFLECTOR CONTROLLED	LiH AND W 15° CONE HALF-ANGLE	REDUNDANT PUMPED NaK LOOPS WITH ISOLATION CAPABILITY PHTS LOOP HAS EM PUMP VAU, AND HX TUBE BUNDLE FOR EACH PCU	FREE PISTON STIRLING ENGINE REDUNDANT PCUs CONSISTING OF A FOUR CYLINDER OPPOSED CONFIGURATION, WITH HOT END COUPLED PAIRS OF PISTONS RUNNING IN OPPOSITION TO MINIMIZE VIBRATION $T_H \sim 950\text{K}$ $T_C \sim 510\text{K}$ FOUR LINEAR ALTERNATORS COUPLED TO THE PISTONS IN EACH CYLINDER PROVIDE THE ELECTRICAL POWER OUTPUT	REDUNDANT ORGANIC COOLANT LOOPS ARE PROVIDED WITH ISOLATION CAPABILITY FOR EACH PCU PUMPED LOOP TUBE AND FIN RADIATOR DESIGN WITH ARMOR AND DEPLOYMENT MECHANISM $T_{rej} \sim 490\text{K}$	LINEAR ALTERNATOR COOLING IS PROVIDED BY HEAT REJECTION THROUGH FPSE COOLING JACKET TO PRIMARY RADIATOR COOLANT (DTA) PHTS THERMOELECTRIC PUMP MODULES REJECT HEAT TO PRIMARY RADIATOR COOLANT

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TABLE II-3
DESCRIPTIONS OF THE 40 CANDIDATE SYSTEMS
(Sheet 16 of 16)

NO	DESCRIPTOR	SUBSYSTEM DATA					ADDITIONAL COMMENTS
		REACTOR	SHIELDING	PRIMARY HEAT TRANSPORT	POWER CONVERSION	HEAT REJECTION	
40	STIRLING ENGINE/ UO ₂ FAST REACTOR	UO ₂ FUEL SUPERALLOY CLAD 1050K OUTLET TEMP REFLECTOR CONTROLLED	LiH AND W AUGMENT SHIELD COOLING MAY BE REQUIRED. 15° CONE HALF- ANGLE.	REDUNDANT PUMPED Na LOOPS WITH ISO- LATION CAPABILITY. PHTS LOOP HAS EM PUMP, VAU, AND HX TUBE BUNDLE FOR EACH PCU	IMPROVED FREE PISTON STIRLING ENGINE. REDUNDANT PCUs CONSISTING OF A FOUR-CYLINDER OPPOSED CONFIGURA- TION, WITH HOT END COUPLED PAIRS OF PISTONS RUNNING IN OPPOSITION TO MINI- MIZE VIBRATION T _H ~1010K T _C ~540K FOUR LINEAR ALTER- NATORS COUPLED TO THE PISTONS IN EACH CYLINDER PROVIDE THE ELECTRICAL POWER OUTPUT.	REDUNDANT ORGANIC COOLANT LOOPS ARE PROVIDED WITH ISOLATION CAPABILITY FOR EACH PCU PUMPED LOOP TUBE AND FIN RADIATOR DESIGN WITH ARMOR AND DEPLOYMENT MECHANISM T _{rej} ~520K	LINEAR ALTERNATOR COOLING IS PROVIDED BY HEAT REJECTION THROUGH FPSE COOL- ING JACKET TO PRI- MARY RADIATOR COOLANT (DTA). PHTS THERMOELEC- TRIC PUMP MODULES REJECT HEAT TO PRI- MARY RADIATOR COOLANT

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could be made safe and whether additional design complexities would be required to provide this safety. The key safety requirements affecting the systems are hydrogenous fluid subcriticality (submerging the core in and flooding the core with water or another fluid and having the core remain inactive) and reentry dispersal (designing the core such that the dispersal of radioactive materials would be sufficient to meet nuclear safety requirements upon reentry). The requirement that most affects the complexity of the reactor is hydrogenous fluid subcriticality. All types of reactors would require additional engineered safety features to assure this subcriticality. These engineered safety features most probably would complicate the reactor design, since they must be able to survive a launch abort accident. In all the systems assembled, consideration was given the overall complexity resulting from this subcriticality equipment and also from additional reentry dispersal features.

B. SELECTING THE PROMISING CANDIDATE SYSTEMS USING A DECISION MODEL

At this point in the downselection process, Rockwell decided to use a formalized decision model to assist in selecting the most promising from the 40 system candidates. The purpose of the decision model was to assist in the decision-making process. The model supplemented — but did not replace — engineering analysis and judgement. Engineering judgment was still a valid decision tool in the selection process, but the decision model did treat tangible and intangible factors consistently, and this helped the process. The model properly utilized the expertise of many individuals and helped document the basis for decisions.

The modeling approach was to break the problem into separate pieces. This allowed a complex problem to be treated as individual, simpler problems. These separate problems were then independently evaluated, and the results were combined using mathematical tools. Because experts disagree as to how the pieces should be combined and also because the overall problem was not totally specified, our approach was to use the knowledge gained in this

process to help make proper decisions. Our approach was not to perform a mathematical evaluation of the pieces and then define a distinct winner on a purely numerical basis.

The decision model employed five independent, complete, and clearly defined decision factors:

- Ability to meet requirements
- Development risk
- Design complexity
- Life potential
- Scalability.

These factors were not simply on detailed functional requirements, such as system reliability or system mass. Each system was then ranked, relying primarily on engineering analysis, for each factor. The five decision factors are discussed below.

The ability to meet requirements was quantified as the electrical power that the system could produce, for an initial 2-yr operating life, relative to 100 kWe (constrained to a 3000-kg mass and 6.1-m envelope). Thus, if the system met all requirements and produced 100 kW, it received 1.0 for this decision factor. If it was necessary to reduce system power to meet all requirements, then the factor was reduced linearly in relation to the power produced by the system. Requirements for safety, reliability, survivability, etc., were addressed for all systems, and concepts were employed for a safe, reliable, survivable, etc., system.

Development risk was considered important in the selection process because it quantified the amount of technology development required. Systems with no risk and a flight demonstration infrastructure in place received 1.0 for this factor. Low-risk systems, those requiring only straightforward engineering, were assigned a value between 0.7 and 0.9. Medium-risk systems,

those in which the technical issues needing resolution were reasonably achievable, received a value between 0.5 and 0.7. High-risk systems, where resolving the required technical factors could significantly affect the program, received a value between 0.3 and 0.5. Very high-risk systems, those involving large and uncertain program impacts, received a value between 0.1 and 0.3.

Design complexity was considered important because it viewed one system versus another in terms of the total amount of complexity. Systems with single transport loops (with passive startup) received a value of 1.0. Single-loop systems (with active startup) were evaluated lower, and multiloop systems (with active startup) were assigned values as low as 0.1.

Life potential is the potential of the system to provide power for 7 yr. This decision factor complements decision factor 1, in which the system's ability to produce power for an initial 2-yr operating life was the measurement value. Systems with demonstrated 7-yr lifetimes in space received a value of 1.0. Systems with no apparent current limitations relative to lifetime potential received values of around 0.8. Where significant testing would be required to support the lifetime potential, the systems received values between 0.5 and 0.7. Where known technical factors made the projection extremely difficult, the systems received values as low as 0.1.

Scalability is the capability of a system to be scaled to higher power. At present, the program requires 100 kWe within the mass and envelope limits. However, the program director has noted several times that the goals of the program are to design concepts for 100-kWe-class systems, meaning systems with powers from 50 kWe to 500 kWe (depending on the actual mission requirements that evolve). Rockwell therefore considered scalability to be an important decision factor. Higher power systems would be heavier and larger. But certain systems at higher powers may encounter inherent limits for reasonable mass and envelope limits. Where a system was demonstrated scalable over the 50- to 500-kWe range, it received a value of 1.0. Systems with no inherent

technical problems in scaling received values between 0.7 and 0.9. Systems where known factors indicated that scalability was not reasonable received values as low as 0.1.

Once the decision model factors had been defined, each member of Rockwell's design team was briefed on the factors and briefed on the basic design features of the 40 candidate systems. Each member of the design team was asked to independently assess each of the 40 systems against each of the five factors using a 0 to 1.0 scale for each factor. After each individual had independently assessed the systems, the team was assembled for a review of the results. In this review, the team members were again briefed on the features of the systems. For each factor, the team reviewed each system with thorough discussion about why a given value was assigned for a given factor. As it turned out, there was much interaction between team members and significant negotiation in assigning factor values. In all cases, the team, which consisted of 17 persons, agreed on a single value (measured in tenths, from 0 to 1.0). Records were kept of the discussion results and of the initial range of factor values provided by the team members.

The decision factor values (agreed to by the team) for the 40 candidate systems are shown in Table II-4. The "meet requirements" factor uses two significant figures, because this is a measure of the system kilowatt electric output versus the 100-kW requirement. The remaining four factors were assessed only on the basis of a single significant figure.

Once the decision factor values had been agreed on, they were weighted and combined linearly. The weightings, shown in Table II-5, were agreed to during a separate review session attended by the entire team. The table shows baseline, low-range, and high-range weightings. The baseline weightings were the negotiated values agreed to by the review team. The low- and high-range values represented the bounds on the weightings at the beginning of this negotiation process. The baseline weighting was used for most assessments, but the low- and high-range values were used in sensitivity analysis.

TABLE II-4
 DECISION FACTOR VALUES FOR THE 40 CANDIDATE SYSTEMS
 (Sheet 1 of 2)

SYSTEM NUMBER	REACTOR TYPE	CONVERSION TYPE	DECISION FACTOR VALUES				
			MEET REQUIREMENTS	DEVELOPMENT RISK	DESIGN COMPLEXITY	LIFE POTENTIAL	SCALABILITY
1	LMFR	TE	.15	.7	.8	.6	.1
2	HPR	TE	.15	.5	.9	.5	.1
3	HPR	TE	.38	.3	.9	.3	.2
4	LMTR	TE	.23	.8	.4	.6	.2
5	LMFR	TE	.23	.8	.4	.8	.2
6	LMTR	TE	.19	.8	.4	.6	.2
7	LMFR	TE	.19	.8	.4	.8	.2
8	LMFR	TE	.34	.4	.8	.3	.3
9	HPR	TE	.43	.3	.9	.2	.3
10	TIR	TI	.35	.5	.8	.2	.6
11	TIR	TI	.58	.4	.8	.2	.6
12	LMTR	CCB	.39	.7	.3	.6	.4
13	LMTR	CCB	.47	.6	.3	.5	.4
14	GCTR	CCB	.25	.6	.5	.7	.6
15	GCFR	CCB	.57	.6	.5	.6	.6
16	FBR	CCB	.25	.3	.5	.6	.6
17	LMFR	CCB	.37	.7	.3	.8	.6
18	LMFR	CCB	.57	.6	.2	.6	.7
19	GCTR	CCB	.25	.5	.4	.7	.8
20	GCFR	CCB	.59	.5	.4	.6	.8

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TABLE II-4
DECISION FACTOR VALUES FOR THE 40 CANDIDATE SYSTEMS
(Sheet 2 of 2)

SYSTEM NUMBER	REACTOR TYPE	CONVERSION TYPE	DECISION FACTOR VALUES				
			MEET REQUIREMENTS	DEVELOPMENT RISK	DESIGN COMPLEXITY	LIFE POTENTIAL	SCALABILITY
21	LMFR	CCB	.58	.4	.2	.5	.8
22	HPR	CCB	.58	.3	.3	.3	.6
23	HPR	CCB	.57	.4	.3	.4	.6
24	LMTR	OR	.39	.6	.4	.7	.4
25	LMTR	OR	.39	.6	.4	.7	.4
26	LMFR	OR	.37	.6	.4	.7	.5
27	LMFR	OR	.37	.6	.4	.7	.5
28	LMTR	MR	.19	.6	.5	.5	.4
29	LMTR	MR	.20	.5	.5	.3	.4
30	LMFR	MR	.18	.6	.5	.6	.5
31	LMFR	MR	.19	.6	.5	.6	.5
32	LMFR	PR	.32	.4	.2	.5	.8
33	LMFR	PR	.25	.3	.2	.3	.8
34	BPFR	PR	.44	.3	.8	.4	.8
35	HPR	PR	.37	.3	.5	.3	.5
36	HPR	PR	.37	.2	.3	.3	.5
37	LMTR	FPS	.63	.6	.5	.4	.3
38	LMFR	FPS	.60	.6	.5	.5	.4
39	LMTR	FPS	.66	.5	.5	.3	.3
40	LMFR	FPS	.69	.5	.4	.4	.4

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TABLE II-5
DECISION FACTOR WEIGHTINGS AND
WEIGHTING SENSITIVITY RANGES

Decision Factor	Decision Factor Weighting (%)						
	Low Range		Baseline			High Range	
Meets requirements	20	25	30	35	40	45	50
Development risk		15	20	25	30	35	
Design complexity		10	15	20	25	30	
Life potential			5	10	15		
Scalability			5	<u>10</u>	15		
				100			

By combining the weightings with the decision factors in a linear model, the baseline ranking was established as shown in Figure II-4. This figure shows the 40 candidate systems ordered by this ranking. This baseline ranking is represented by the solid black dot. The shaded regions on this figure are the results of sensitivity analyses on the baseline ranking. This figure does not differentiate among reactor type, power converter type, or launch date. The ranking information is grouped by reactor type Figure II-5 and by power converter type in Figure II-6. These figures show which types of reactor and which types of power converter have the higher rankings.

A more interesting review of the decision model results is shown in Figure II-7. In this figure, the system launch timeframes are separated – the early 1990s timeframe is shown on the left, the mid-1990s timeframe on the right. Comparing this with Figure II-4 shows system 15 to be first overall and first in the ranking for early 1990s launch. Similarly, system 20 is first in the mid-1990s launch and second overall.

Figure II-8 gives further insight into the sensitivity of the overall ranking to variations in the weighting factors. These appropriately named "barbed-wire" charts show how ranking varies when different weightings are

Figure II-4. The 40 Candidate Systems Ordered by Baseline Ranking

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Figure II-5. Candidate System Ranking, Grouped by Reactor Type

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Figure II-6. Candidate System Ranking, Grouped by Power Converter Type

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Figure II-7. Candidate System Ranking, Grouped by Launch Readiness

Figure II-8. Effects of Decision Factor Weighting Variations

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employed. There are five plots in this figure, one for variations in each weighting. When one weighting was changed, the other four weightings were varied linearly so that the total weighting added to 100%. Remembering that only the top ten systems are shown in this figure, it is apparent that the ranking is insensitive to weighting variation. This is very important, because a decision model is only as good as the decision factors selected and the weightings applied to those factors.

Once this decision model information had been established, it was combined with the engineering judgment gained through application of the model. Together, results of the model plus this judgment formed the basis for selecting the promising candidate systems. In this selection process, high overall ranking from the decision model was important. In other words, within the sensitivity weighting of the decision factors, the systems with high overall rankings were thought to provide the best general attributes. Also considered, however, were those systems that had high specialized rankings. Specialized ranking refers to limit analysis results. In limit analysis, one or two of the factors are given the total weighting, and the other decision factors are ignored. Such analyses identify systems with good special attributes. Technology carrythrough was also considered in the selection process. This refers to a technology that exists for the early 1990s' systems and can be "carried through" to a mid-1990s' system. Such systems retain a common technology base. Finally, Rockwell believed that mass sensitivity should not be ignored in the selection of the promising candidate systems.

In the limit analysis, Rockwell believed it important to identify the systems with lowest risk and those with the highest potential. The system with the lowest risk was identified by heavily weighting the development risk decision factor. Evaluated this way, the top ranking system used a liquid-metal-cooled reactor with thermoelectric power converters. This is almost obvious from engineering judgment, because the liquid-metal-cooled reactor and the thermoelectric converters have the most mature technology bases. Note that this specific combination of reactor and converter did not have a high

baseline ranking because it would not produce high electric power. However, it appears to have low risk. The highest potential system was identified by heavily weighting the scalability and life potential decision factors. The top ranking systems used liquid-metal-cooled or gas-cooled reactors with Brayton or potassium Rankine power converters. These systems have the potential to produce very high powers.

Technology carrythrough was reviewed by investigating the systems in one time frame and extrapolating them to the next time frame. Systems that retained basic technology features were given special consideration.

As a final engineering judgment, each system was evaluated in terms of the tradeoff between power and mass. The left side of Figure II-9 is a histogram showing the number of 3000-kg systems providing power in a specific range. The range has been divided in decades from 0 to 100 kWe. The right side of the figure shows the same type of information for systems with a mass of 4000 kg. Note that there are six 4000-kg systems that provide 100 kWe, but no 3000-kg systems that provide more than 70 kWe. The systems providing the higher power tend to use dynamic power converters.

The results of the decision model and engineering judgment are shown in Table II-6. In its selection, Rockwell chose to divide the systems into those available for early 1990s launch and those available for mid-1990s launch. The table shows the power converter and reactor types for each system. The system numbers correspond to the descriptions in Section II.A. The results show that a thermoelectric system coupled with a liquid-metal-cooled fast reactor is applicable to both the early 1990s and mid-1990s launch dates. Neither system produces the required 100 kWe (system 5 produces 23 kWe, and system 8 produces 34 kWe), but both satisfy the low-risk assessment and were selected. Thermoelectrics can also be combined with a heat-pipe-cooled reactor, and this is shown as system 9, which is available for mid-1990s launch. There was technology carrythrough between systems 5 and 8, but it is our belief that the heat-pipe-cooled reactor would not be available until the

Figure II-9. Effects of System Mass on System Power

TABLE II-6
THE 12 PROMISING CANDIDATE SYSTEMS

<u>POWER CONVERSION TYPE</u>	<u>SELECTED SYSTEMS</u>		<u>REACTOR TYPE</u>
	<u>FOR EARLY 1990S</u>	<u>FOR MID 1990S</u>	
THERMOELECTRIC	5	TECHNOLOGY ADVANCE → 8	LIQUID-METAL-COOLED FAST
	—	9	HEAT PIPE FAST
THERMIONIC	10	TECHNOLOGY ADVANCE → 11	THERMIONIC
BRAYTON	15	TECHNOLOGY ADVANCE → 20	GAS-COOLED FAST
	17	TECHNOLOGY ADVANCE → 18	LIQUID-METAL-COOLED FAST
POTASSIUM RANKINE	—	34	BOILING POTASSIUM FAST
STIRLING	38	TECHNOLOGY ADVANCE → 40	LIQUID-METAL-COOLED FAST

later launch timeframe. Thermionic reactor systems, systems 10 and 11, can produce powers between 35 and 58 kWe (the lower number for the earlier system, the higher number for the later system). These are in-core thermionic reactors, cooled by liquid metal. The Brayton rotating power converter has been combined with both gas-cooled fast and liquid-metal-cooled fast reactors. For each of these combinations, there are early 1990s and mid-1990s systems, with technology carrythrough between the two. The gas-cooled fast reactor with Brayton power converter systems are numbers 15 and 20 and produce 57 and 59 kWe, respectively. The liquid-metal-cooled fast reactor with Brayton power converter systems are numbers 17 and 18 and produce 37 and 57 kWe, respectively. The potassium Rankine power converter was combined with the boiling potassium fast reactor. This is considered to be a system applicable only to a mid-1990s launch. This system, number 34, produces 44 kWe. Lastly, the Stirling reciprocating power converter was combined with a liquid-metal-cooled fast reactor. This system, though high on development risk because of the infant state of Stirling engine technology, produced high electric powers. System 38 produced 60 kWe, and system 40 produced 69 kWe. There is technology carrythrough between the two systems.

In general, the decision modeling and downselection process resulted in five systems that would be ready for launch in the early 1990s and seven systems that would be ready for launch in the mid-1990s. These systems used many types of reactors and power converters – specifically, five different types of reactors and five different types of power converters. These systems formed the basis for the selection of the most attractive system (discussed in Section III).

III. SELECTING THE MOST ATTRACTIVE SYSTEM

A. THE PROMISING CANDIDATE SYSTEMS

The starting point for selecting the most attractive system was the group of 12 promising candidate systems identified in Table II-6. Before discussing the selection process, it is important to note that the ground rules for selection changed slightly at this point in the studies. Under prior ground rules, electric power could be reduced to meet the 3000-kg mass limit and 6.1 m length limit for the systems. Under the new ground rules (provided by the Jet Propulsion Laboratory SP-100 Project Office), technologies should be advanced until 100 kWe could be provided within these mass and envelope limits. Of the 12 promising systems, 5 could be available in the early 1990s and seven in the mid-1990s. None of these systems provided 100 kWe. To identify the most attractive system, the selection process started with the 7 mid-1990 systems, as shown in Table II-6. These were the most advanced of the 12 promising candidates and required the least technology improvement to achieve 100 kWe.

To obtain 100 kWe from each system within the mass and envelope constraints of the specification document, reactor temperatures were increased. All systems still employed a 15° conical half-angle, and this was varied slightly for systems in which radiators were fixed (nondeployed). For systems with deployed radiators, a 15° cone half-angle was used.

The seven systems used both static and dynamic power converters. For the static thermoelectric converters, total redundancy was not provided since the systems were highly modular. Instead, the system was slightly oversized, as noted later. The same applies to the static thermionic converters. For the turboelectric converters (rotating equipment such as Brayton and Rankine), three power conversion arrangement options were considered. The first was a single rotating unit providing 100 kWe. This would provide no redundancy and

no cancellation of torques. The second option was dual 50-kWe counter-rotating units. Since both units would operate continuously, there would be no redundancy. Torque cancellation would be provided. The third option was two pairs of 50-kWe counter-rotating units. This would provide torque cancellation and a totally redundant set of power converters. For the dynamic reciprocating converters (Stirling), two options were considered. The first option was four pairs of opposed cylinders in a single unit with 100-kWe power capability. The second option was six pairs of opposed cylinders in a single unit, providing 50% installed redundancy.

For all power converter options, either static or dynamic, the power conversion system was oversized by 10 to 15% (depending on the specific system) to account for power processing losses, transmission losses, power converter degradation (as applicable), and liquid pumping power (where required).

First, the technically plausible candidates were selected from the seven mid-1990s systems. These candidates are identified in Table III-1.

System 8 – the liquid-metal-cooled fast reactor with thermoelectric converters – was eliminated (not a plausible system) because a reactor temperature over 1700 K was needed to meet mass and envelope limits. This system was thus not considered further for possible selection as the most attractive system.

System 9 – the heat pipe fast reactor using thermoelectric converters – was selected as plausible because the mass and envelope limits could be met at a reactor temperature near 1600 K. As shown in Table III-1, this system employs 220,000 converters and uses a nondeployed radiator.

System 11 – the in-core thermionic reactor – was selected as plausible because the mass and envelope limits could be met with SNAP-proven liquid-metal temperatures and technologies. As shown in Table III-1, this system uses a non-deployed radiator and employs 750 converters.

TABLE III-1

THE FIVE TECHNICALLY PLAUSIBLE CANDIDATE SYSTEMS

Reactor	Power Converter	Deployed	Nondeployed	Number of Converters
Lithium heat-pipe fast	Thermoelectric		X	220,000
NaK-cooled stainless steel fast	In-core thermionic		X	750
Lithium-cooled refractory fast	Closed-cycle Brayton turboelectric	X	X	One or two
Lithium-cooled refractory fast	Potassium Rankine turboelectric		X	One or two
Lithium-cooled refractory fast	Free piston Stirling reciprocating	X	X	Four to six pairs

System 20 – the gas-cooled fast reactor using a Brayton turboelectric power converter – was eliminated as not being plausible because of gas duct size concerns and the incompatibility of the gas-cooled reactor with any other power converter.

System 18 – the liquid-metal-cooled fast reactor using a Brayton turboelectric power converter – was selected as plausible because the mass and envelope limits could be met with reactor temperatures in the 1450 to 1650 K range. As shown in Table III-1, this system can use deployed or nondeployed radiators. The higher reactor temperatures occur with nondeployed radiators, because less radiator area is allowed and the radiator must operate at a higher temperature.

System 34 – the boiling potassium fast reactor using a potassium Rankine turboelectric power converter – was not directly selected as plausible in its

original configuration. The system was modified to use a lithium-cooled fast reactor, with the potassium Rankine converter system being coupled through a lithium-to-potassium boiler. The system was then selected as plausible because the mass and envelope limits could be met with a nonboiling reactor operating at about 1450 K. As shown in Table III-1, this system uses a nondeployed radiator.

Lastly, system 40 — the liquid-metal-cooled fast reactor using a Stirling reciprocating power converter — was selected as plausible because the mass and envelope limits could be met with reactor temperatures in the 1100 to 1300 K range. As shown in Table III-1, this system can use a deployed or nondeployed configuration. The nondeployed configuration for the radiator results in the higher reactor temperature.

The 12 systems listed in Table III-2 result from these combinations. The table also identifies the figures (Figures III-1 through III-12) that give the statepoint diagram and mass breakdown for each system.

As each system was defined, certain key considerations were taken into account in the design process:

- Safety
- Reliability
- Power processing
- Heat transport startup.

The effects of each consideration on the candidate systems are discussed below.

The two key safety considerations are designing for hydrogenous fluid subcriticality and designing for reactor reentry, disassembly, and fuel dispersal. In designing for hydrogenous fluid subcriticality (includes water and other fluids such as hydrazine), Rockwell believes that intrinsic subcriticality is not practical for the reactors considered. In other words, there

TABLE III-2
PLAUSIBLE CANDIDATE SYSTEMS

System Type	Appears in Figure
Lithium heat pipe fast reactor, thermoelectric converter, nondeployed radiator	III-1
NaK-cooled stainless steel fast reactor, in-core thermionic converter, nondeployed radiator	III-2
Lithium-cooled refractory fast reactor, closed-cycle Brayton turboelectric converter, deployed radiator, single 100-kWe combined rotating unit	III-3
Lithium-cooled refractory fast reactor, closed-cycle Brayton turboelectric converter, deployed radiator, dual 50-kWe combined rotating units	III-4
Lithium-cooled refractory fast reactor, closed-cycle Brayton turboelectric converter, nondeployed radiator, single 100-kWe combined rotating unit	III-5
Lithium-cooled refractory fast reactor, closed-cycle Brayton turboelectric converter, nondeployed radiator, dual 50-kWe combined rotating units	III-6
Lithium-cooled refractory fast reactor, potassium Rankine turboelectric converter, nondeployed radiator, single 100-kWe combined rotating unit	III-7
Lithium-cooled refractory fast reactor, potassium Rankine turboelectric converter, nondeployed radiator, dual 50-kWe combined rotating units	III-8
Lithium-cooled refractory fast reactor, free piston Stirling reciprocating converter, deployed radiator, no extra cylinders	III-9
Lithium-cooled refractory fast reactor, free piston Stirling reciprocating converter, deployed radiator, 50% extra cylinders	III-10
Lithium-cooled refractory fast reactor, free piston Stirling reciprocating converter, nondeployed radiator, no extra cylinders	III-11
Lithium-cooled refractory fast reactor, free piston Stirling reciprocating converter, nondeployed radiator, 50% extra cylinders	III-12

83-AU1-74-8

Figure III-1. Lithium Heat Pipe Fast Reactor, Thermoelectric Converter, Nondeployed Radiator

Figure III-2. NaK-Cooled Stainless Steel Fast Reactor, In-Core Thermionic Converter, Nondeployed Radiator

83-AU1-74-18

Figure III-3. Lithium-Cooled Refractory Fast Reactor, Closed-Cycle Brayton Turboelectric Converter, Deployed Radiator, Single 100-kWe Combined Rotating Unit

Figure III-4. Lithium-Cooled Refractory Fast Reactor, Closed-Cycle Brayton Turboelectric Converter, Deployed Radiator, Dual 50-kWe Combined Rotating Units

83-AU1-74-17

Figure III-5. Lithium-Cooled Refractory Fast Reactor, Closed-Cycle Brayton Turboelectric Converter, Nondeployed Radiator, Single 100-kWe Combined Rotating Unit

Figure III-6. Lithium-Cooled Refractory Fast Reactor, Closed-Cycle Brayton Turboelectric Converter, Nondeployed Radiator, Dual 50-kWe Combined Rotating Units

83-AU1-74-14

Figure III-7. Lithium-Cooled Refractory Fast Reactor, Potassium Rankine Turboelectric Converter, Nondeployed Radiator, Single 100-kWe Combined Rotating Unit

NOTE: SECOND POWER CONVERSION
UNIT NOT SHOWN

83-AUI-74-13

Figure III-8. Lithium-Cooled Refractory Fast Reactor, Potassium Rankine Turboelectric Converter, Nondeployed Radiator, Dual 50-kWe Combined Rotating Units

83-AU1-74-12

Figure III-9. Lithium-Cooled Refractory Fast Reactor, Free Piston Stirling Reciprocating Converter, Deployed Radiator, No Extra Cylinders

NOTE: ONLY ONE
CYLINDER SHOWN

83-AU1-74-11

Figure III-10. Lithium-Cooled Refractory Fast Reactor, Free Piston Stirling Reciprocating Converter, Deployed Radiator, 50% Extra Cylinders

NOTE: ONLY ONE
CYLINDER SHOWN

83-AU1-74-10

Figure III-11. Lithium-Cooled Refractory Fast Reactor, Free Piston Stirling Reciprocating Converter, Nondeployed Radiator, No Extra Cylinders

NOTE: ONLY ONE
CYLINDER SHOWN

83-AU1-74-9

Figure III-12. Lithium-Cooled Refractory Fast Reactor, Free Piston Stirling Reciprocating Converter, Nondeployed Radiator, 50% Extra Cylinders

is not enough thermalization of the neutron spectrum in the core to allow using thermal resonance absorbers or other intrinsic means of maintaining subcriticality. The lack of thermalization can be seen from the standard Safonov curve for U^{235} -water mixtures (Figure III-13). This curve shows the critical mass versus bare sphere radius for various molecular ratios of water to uranium. As the molecular ratio increases, the critical mass initially decreases because thermalization is occurring. A well thermalized sphere occurs at the bottom of the curve. For the fast reactors considered in this study, the range of interest of molecular ratios is 0.5 to 1 for core immersion and complete flooding (with water or hydrazine). Little thermalization occurs in the core, and thermal resonance absorbers are not applicable for intrinsic subcriticality.

Therefore, we must consider engineered subcriticality: the placement of engineered features on the reactor that render it subcritical in the hydrogenous fluid immersion and flooding situation. Fast spectrum poisons were

83-AUI-74-3

Figure III-13. Standard Safonov Curve Shows Almost No Thermalization in Range of Interest (0.5 to 1 H₂O/U²³⁵ Ratios)

selected as the probable method. These poisons must be removable in space, but must also withstand launch abort conditions. As shown below, a preliminary version of such an engineered device is included in the reactor design.

The other key safety consideration is designing for reactor reentry, disassembly, and fuel dispersal. Typically, as the reactor begins reentry, burn through of the reactor vessel or support occurs. The core is ejected from the vessel as the vessel burns through. The fuel elements are disassembled from the core bundle, and the radioactivity in the fuel is dispersed sufficiently to meet the safety goals. The actual design of a reactor for fuel dispersal cannot be confirmed until the final spacecraft configuration and its orbit have been established. However, inherent design features can be incorporated to enhance fuel dispersal, and Rockwell has incorporated these features in our most attractive reactor system design.

The next design consideration is reliability. For the systems under review, the key difference between systems with static or dynamic converters is the degree of redundancy that can be provided. Static converters provide inherent redundancy because they are modular. Dynamic converters, on the other hand, provide significant powers per unit and have real mass increases when redundancy is provided. The question is whether acceptable reliability goals can be achieved for dynamic converters without unacceptable system mass penalties. For these converters, reliability growth in the equipment and the proper and wise use of redundancy will be required to achieve system reliability goals.

Using the Brayton turboelectric power converter as an example, Figure III-14 shows typical ranges of failure rates per hour for rotating equipment and the required failure rates for a system using various combinations of converters. The failure rates for a system using a single converter and having a 2-yr life and 7-yr life are shown. The failure rates for the 2-yr and 7-yr lives of a dual- and triple-redundant converter are also shown. In any case,

*FOR RELIABILITY ALLOCATION OF 0.98 (SYSTEM RELIABILITY OF 0.95)

83-AU1-74-33

Figure III-14. Reliability Growth Will Be Needed for Dynamic Power Converters

reliability growth nearing one order of magnitude (a reduction in failure rate of one order of magnitude) will be the minimum required to achieve 95% reliability for a 7-yr life. This appears to be an achievable goal.

The next system design consideration is power processing. There are different types of power processors for static and turboelectric power converters. Figure III-15 shows a typical schematic of a power processor for thermoelectric power converters. Figure III-16 shows a power processing concept for a turboelectric power converter. Table III-3 summarizes the considerations for power processing. Turboelectric processors need only dissipate 0.5 kWe of power and have continuous control. There are two power stages. Thermoelectric power converters require approximately 2.5 kWe of power processor dissipation and use discontinuous step control with multiple power stages. The differences in power dissipation and the differences in masses for the power processing systems were both considered in determining the system masses.

The final design consideration is heat transport loop startup. The systems considered here would either use pumped liquid-metal coolants or employ heat pipes. Heat pipes can probably be self-started, but engineering considerations must be provided for starting liquid-metal coolant loops. Figure III-17 shows a typical pumped coolant loop for primary heat transport. There are two startup options, including maintaining the coolant in a liquid state or defrosting the coolant at startup. We currently believe that it is better to maintain the coolant in a liquid state because of the difficulties in defrosting and the danger of rupturing a system during defrosting. The primary heat transport loop can be maintained in a liquid state by operating the reactor at low power or by providing separate heating through an isotope heat source, batteries, photovoltaics, fuel cells, etc.

B. THE TECHNOLOGY FEASIBILITY ISSUES

The next activity in selecting the most attractive system was evaluating the technology base available for the reactors and power converters to be used in the

83-AUI-74-28

Figure III-15. Power Processor Concept for Thermoelectric Power Converters

83-AUI-74-29A

Figure III-16. Power Processor Concept for Turboelectric Power Converters

TABLE III-3

POWER PROCESSORS – COMPARISON FOR
THERMOELECTRIC AND TURBOELECTRIC CONVERTERS

Type	Turbo-electric	Thermoelectric
Design rating (kV·A)	2	100
Steady state processor dissipation (kWe)	0.5	2.5
Type of control	Continuous	Discontinuous step
Number of power stages	2	50

PRIMARY HEAT TRANSPORT LOOP

83-AUI-74-23

Figure III-17. Startup for Pumped Coolant Loops

technically plausible candidate systems. This evaluation had to answer two basic questions:

- Are there potential technology feasibility issues that could be show stoppers further down the line?
- Are there tests that could be performed in the next 2 to 3 yr to determine which, if any, of these potential issues really are show stoppers? The premise here is that systems with fewer technology feasibility issues are probably more attractive systems from a selection standpoint.

Technology feasibility was reviewed by investigating the types of tests that could shed light on the issue. These tests, listed in Table III-4, include performance tests, steady-state endurance tests, cyclic endurance tests, accelerated damage fraction tests, tests to failure, and material tests. The designator codes provided in the table for each test are used in subsequent tables. Each type of test can be performed on a component, sub-assembly, subsystem, or system. Some require only hours or days to perform, while others require one or more years to perform. The time frame associated with the test is the key to determining if technology feasibility show stoppers can be identified in the next 2 to 3 years.

The technology feasibility evaluation was performed by reviewing each type of reactor and each type of power converter with respect to the tests shown in Table III-4. The potential technology issues associated with a reactor or power converter were noted. The time to the first manifestation of the issue was then estimated. Then the question was asked, "Can a test be run in the next 2 to 3 yr to verify whether this is or is not an issue?" The answer was either yes or no, and the type of test, using the designator code from Table III-4, was shown. The answer for all of the reactors and converters is that none of them had all "yeses," i.e., each had several potential technology feasibility issues. But certain of the reactors and converters had fewer issues than others, and this adds to the basis for selecting the most attractive system.

TABLE III-4
TESTS WHICH ADDRESS TECHNOLOGY FEASIBILITY ISSUES

Test Code	Designation	Time to Obtain Necessary Data
P	Performance test	Hours to days
ESS	Endurance (steady state)	Year (plus)
EC	Endurance (cyclic loading)	Up to a year (plus) ^a
DF	Damage fraction summation (also known as accelerated life testing)	As required (generally year, plus)
F	Test to failure	As required
M	Compatibility, corrosion, erosion, mass transfer	Months to year (plus)

^aDepending on relaxation time between cycles.

The first reactor to be evaluated for technology feasibility issues was what we called the lithium refractory fast reactor. This is actually either the lithium-cooled (pumped liquid lithium) fast reactor or the heat-pipe-cooled (lithium working fluid) fast reactor. We noted that an advantage of using the technology feasibility evaluation approach is in identifying reactor groups that have similar technology feasibility issues. The potential issues are shown in Table III-5. The table identifies fuel growth and fission gas release, corrosion, mass transfer, refractory materials degradation, and reflector dimensional stability as issues that potentially could not be resolved in the next 2 to 3 yr.

The second reactor to be evaluated was the in-core thermionic reactor. The potential technology feasibility issues for this reactor are shown in Table III-6. The table identifies fuel growth, fission gas venting, insulation degradation, and refractory materials degradation issues as potentially not being resolvable in the next 2 to 3 yr.

TABLE III-5
POTENTIAL TECHNOLOGY FEASIBILITY ISSUES FOR THE
LITHIUM-REFRACTORY FAST REACTOR

Potential Issue	Estimated Time to First Manifestation (yr)	Can a Test Be Run in Next 2-3 Yr to Verify? (Test Type)
Fuel - materials interaction	1/2 to 1	Yes (M)
Accommodation of fuel growth and fission gas release	1 (Minimum)	No (ESS)
Corrosion/mass transfer	2 - 3	No (M)
Self-welding of control mechanisms	<1	Yes (M)
Refractory materials degradation	>1	No (ESS)
Actuator degradation	<1	Yes (EC)
Producibility, welding, joining of refractories	N.A.	Yes (F)
Accommodation of in-situ formed noncondensable gas	1 (minimum)	Yes (P)
Dimensional stability of BeO reflector (if used)	2 - 3	No (ESS)

TABLE III-6
POTENTIAL TECHNOLOGY FEASIBILITY ISSUES FOR THE
IN-CORE THERMIONIC REACTOR

Potential Issue	Estimated Time to First Manifestation (yr)	Can a Test Be Run in Next 2-3 Yr to Verify? (Test Type)
Output failure due to growth	1 - 2	No (ESS)
Fuel-cladding interaction	1/2 - 1	Yes (M)
Fission gas venting	1 - 2	No (ESS)
Insulation degradation	1 - 2	No (ESS)
Accommodation of in-situ formed noncondensable gas	1 (minimum)	Yes (P)
Refractory materials degradation (emitter-collector)	>1	No (ESS)
Producibility, welding, joining of refractories	N.A.	Yes (F)

Potential technology feasibility issues for the thermoelectric converter are shown in Table III-7. A key issue identified for this converter was converter material performance. The figure of merit (Z) for the converter is a measure of its performance. In our studies, we used a figure of merit for thermoelectric converters of $1 \times 10^{-3} \text{ K}^{-1}$. This Z has not yet been achieved, and a technology breakthrough will be required to achieve it. The other issues appear to be resolvable once the material has been defined and isolated.

The potential technology feasibility issues for the closed-cycle Brayton turboelectric converter are shown in Table III-8. Resolution of torque cancellation, performance demonstration, and heat exchanger integrity are identified marginal in the 2- to 3-yr timeframe.

The potential technology feasibility issues for the potassium Rankine turboelectric converter are identified in Table III-9. The issues concerning the zero-g effects on vapor-liquid interface control, condenser performance, and performance demonstration do not appear to be resolvable in the next 2 to 3 yr.

The potential technology feasibility issues for the free-piston Stirling reciprocating converter are shown in Table III-10. This converter is early in its development, and there are several basic questions to be answered about it. However, it appears that many of these questions may be resolvable in the next 2 to 3 yrs, although questions on vibration, creep distortion of the engine head, and long-term performance may not be resolvable.

Potential technology feasibility issues exist also for the overall system; these are shown in Table III-11. No test can be run in the next 2 to 3 yr to isolate and verify these issues because they must be done in a system environment.

TABLE III-7
POTENTIAL TECHNOLOGY FEASIBILITY ISSUES FOR THE
THERMOELECTRIC CONVERTER

Potential Issue	Estimated Time to First Manifestation (yr)	Can a Test Be Run in Next 2-3 yr to Verify? (Test Type)
TE material with $Z \sim 1 \times 10^{-3}$ for operation at $T \sim 1500$ K	Technology breakthrough required - Timeframe indeterminate -	
Degradation of output	<1	Yes, contingent (E_{SS}/E_C) on material definition
Bonding of materials	1-2	Yes, contingent (E_{SS}/E_C) on material definition
Material sublimation	<1	Yes, contingent (E_{SS}/E_C) on material definition
Insulation degradation	>1	Yes (M)
Performance demonstration	Hours to days	? (P)

TABLE III-8
POTENTIAL TECHNOLOGY FEASIBILITY ISSUES FOR THE
CLOSED-CYCLE BRAYTON TURBOELECTRIC CONVERTER

Potential Issue	Estimated Time to First Manifestation (yr)	Can a Test Be Run in Next 2-3 yr to Verify? (Test Type)
Turbine wheel/blading	Anytime	Yes (E_{SS}/E_C)
Bearing	Anytime	Yes (E_C)
Transition joint	Anytime	Yes (F)
Torque cancellation	Immediate	? (P)
Insulation degradation	>1	Yes (M)
Producibility, welding, joining of refractories	NA	Yes (F)
Performance demonstration	Hours to days	? (P)
Heat exchanger integrity	1/2 (Plus)	? (E_C)

TABLE III-9

POTENTIAL TECHNOLOGY FEASIBILITY ISSUES FOR THE
POTASSIUM RANKINE TURBOELECTRIC CONVERTER

Potential Issue	Estimated Time to First Manifestation (yr)	Can a Test Be Run in Next 2-3 yr to Verify? (Test Type)
Turbine wheel/blading	Anytime	Yes (E_{SS}/E_C)
Excessive erosion/corrosion of blades	<1	Yes (M)
Bearing	Anytime	Yes (E_C)
Torque cancellation	Immediate	? (P)
Bore seal	Anytime	Yes (E_C)
Effects of zero-g on vapor-liquid interface control	Within hours	No (P)
Boiler and condenser performance	Within hours	Boiler - Yes (P) Condenser - No (P)
Insulation degradation	>1	Yes (M)
Producibility, welding, joining of refractories	NA	Yes (F)
Transition joint	Anytime	Yes (M)
Performance demonstration	Hours to days	? (P)

TABLE III-10

POTENTIAL TECHNOLOGY FEASIBILITY ISSUES FOR THE
FREE-PISTON STIRLING RECIPROCATING CONVERTER

Potential Issue	Estimated Time to First Manifestation (yr)	Can a Test be Run in Next 2-3 yr to Verify? (Test Type)
Vibration	Immediate	? (P)
Gas bearing	Anytime	Yes (E_{SS}/E_C)
Transition joint	Anytime	Yes (F)
Insulation degradation	>1	Yes (M)
Creep distortion of engine head	<1	? (E_{SS}/E_C)
Performance demonstration	Hours to days	? (P)

TABLE III-11

POTENTIAL TECHNOLOGY FEASIBILITY ISSUES
FOR THE OVERALL SYSTEM

Potential Issue	Estimated Time to First Manifestation (yr)	Can a Test be Run in Next 2-3 yr to Verify? (Test Type)
Performance demonstration	Days	No (E_{SS}/E_C)
Lifetime	7	No (E_{SS}/E_C)
Reliability	Anytime	No (E_{SS}/E_C)
Dormancy/restart	Anytime	Yes (?) (P)
Safety	Event and design dependent	No (P)
Survivability	Event dependent	No (P)

The technology feasibility evaluation served to identify the reactors and power converters that have fewer potential showstoppers in future years. The systems studies (Section III.A) discussed the system-level implications of using these reactors and power converters. Section III.C discusses the selection process for determining the most attractive system. This process was based on a combined evaluation using the system-level studies and the feasibility issues.

C. THE SELECTED MOST ATTRACTIVE SYSTEM

The purpose of the studies was to select the most attractive system that met the requirements. This section describes that selection. The selection of a fallback power converter that could optionally be used with the most attractive system is also described.

The key considerations in selecting the most attractive system were that it most plausibly meet all technical requirements, have a good potential for meeting safety goals, be amenable to a fallback power converter that could produce 100 kWe, and have higher power potential with better-than-linear growth. The latter consideration is quite important. As time goes on, missions will require higher powers. As power systems grow to provide this power, they must grow at a better than linear rate.

The reactor and power converter combinations that have been considered for the most attractive system are shown in Figure III-18. The studies performed to combine these reactors and power converters into systems were reviewed, as were the technology feasibility issues for the reactors and power converters. A lithium-refractory fast reactor using a closed-cycle Brayton turboelectric converter was selected as the most attractive system. This system uses a baseline reactor that is lithium cooled. A heat pipe reactor could be substituted for the lithium-cooled reactor, but this is not now the preferred choice. The lithium-cooled reactor can achieve the necessary temperatures and is compatible with most power converters. The Brayton power

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Figure III-18. Reactor-Power Converter Candidates for the Most Attractive System

converter has a good technology base, and the system can produce 100 kWe within the SP-100 mass limits. The system using this reactor and power converter has good potential to meet the reliability goals through component-level redundancies. There is also the potential to put in dual-redundant power conversion systems, although this option needs further evaluation. The system has good growth capability, into the MWe range. The system can also employ a Stirling reciprocating converter as a fallback option and still provide 100 KWe. The use of this converter reduces reactor temperatures by almost 400 K.

The key question here is why were the other systems not selected. The system using a heat-pipe reactor and advanced thermoelectrics is a high-temperature system. Its growth capabilities are limited, and a breakthrough in technology would be required to achieve 100 kWe within the mass and envelope constraints of the SP-100 Program. Rockwell believed it was not in the best interests of the program to select this system. The in-core thermionic reactor system has potential but also has several technology feasibility issues, and the reactor-converter combination is a closed-end option. This places all the technology issues in one basket, and the in-core thermionic reactor system is not amenable to other reactor or power converter fallback options. The lithium-cooled fast reactor coupled to a potassium Rankine system has zero-g, erosion, moisture control, and corrosion concerns. This system appears to have the most growth capability, but has somewhat more technology feasibility issues than the other rotating system, the Brayton system.

The preliminary characteristics of the most attractive system are shown in Table III-12. The system produces a net power of 100 kWe at a mass of 2750 kg. This is under the 3000-kg system limit and provides the potential for adding redundancy to the power converters in the system. The reactor, shield, power converter, and radiator features are shown in this table. The preliminary reactor and shield concept for the system are shown in Figure III-19.

TABLE III-12
CHARACTERISTICS OF THE MOST ATTRACTIVE SYSTEM

● System	
- Net power	100 kWe
- Mass	2750 kg
● Reactor	
- Power	420 kWt
- Fuel	UO ₂ or UN
- Materials	Refractory alloy
- Coolant	Lithium
- Coolant outlet temperature	1465K
● Shield	
- Gamma	Tungsten alloy
- Neutron	LiH
● Power Converter	
- Type	Closed-cycle Brayton turboelectric
- Fluid	He-Xe
- Number of units	Dual half-power counter rotating
- Turbine inlet temperature	1400 K
- Compressor inlet temperature	475 K
● Radiator	
- Type	Deployed, pumped NaK
- Area	85m ²
- Average temperature	555 K

83-AU1-74-60

Figure III-19. Preliminary Reactor and Shield Concept for the Most Attractive System

This is a pin-type fuel reactor using molten lithium as a coolant. The reactor preliminarily employs a central poison plug for hydrogenous fluid subcriticality. Tapered beryllium reflectors are used for control, and segments of the reflector slide down using mechanical actuation to provide neutron leakage windows. A gamma shield and thermal shield are provided between the reactor and the canned lithium hydride neutron shield. The reflector segments are actuated below the shield by mechanical actuators with gear trains. Further information on the reactor is provided in Figure III-20.

This reactor is sized to combine with a Brayton turboelectric power converter and provide liquid lithium at a reactor outlet temperature of 1465 K. This reactor could also be combined with the Stirling reciprocating converter as a fallback option. Although this converter is not as well demonstrated as the Brayton converter, it has the advantage of allowing the reactor to operate in the 1100- to 1300-K range, and at the same or a lower thermal power.

Once the most attractive system described above had been selected, it was used as a starting point for the next 2 months of Rockwell studies. Many design and performance improvements were implemented, and the system was designated the SR-100G (Space Reactor, 100 kWe, Growth). This system is described in Section IV.

83-AU-74-61

Figure III-20. Preliminary Reactor Concept for the Most Attractive System

IV. ROCKWELL SR-100G SYSTEM

A. SYSTEM DESCRIPTIONS

The system selected as the most attractive is a lithium-cooled refractory alloy reactor coupled to a Brayton turboelectric converter. This is called the Rockwell SR-100G System, meaning space reactor, 100 kWe, growth system. Figures IV-1, 2, and 3 give, respectively, an artist's conception of the system, a cutaway view, and an interface view. The key features of the system are presented in Table IV-1.

In overview, system heat is supplied from the reactor to heat exchangers using lithium coolant circulated by electromagnetic (EM) pumps. The reactor heat thus transported is employed in redundant Brayton turboelectric converters to provide 100 kWe. Waste heat is rejected from the system through a radiator that is gas cooled by the working fluid of the Brayton turboelectric converters. Redundant tubes are provided on the radiator, and each set of tubes is capable of rejecting the total amount of waste heat at 100 kWe.

The system was designed for the worst-case condition where only one of the dual Brayton turboelectric converters is operating. This, the design point, is used to size all components and equipment. The heat balance and statepoints at the design point condition are shown in Figure IV-4. Only one of the two Brayton loops is shown on this figure because only one is operating.

Each of the subsystems that compose the SR-100G System are explained in subsequent sections. But much can be learned about the system by tracing the statepoint diagram. Thermal power of 508 kWt is generated in the reactor and is removed by the lithium coolant. The reactor is fueled with UN and is controlled by opening neutron leakage windows in the beryllium radial reflector. The coolant enters the reactor at 1400 K and exits at 1473 K. The majority of the 7 kW of heat lost from the reactor outlet piping is lost from a 0.3 m²

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Figure IV-1. Artist's Conception of SG-100G Reactor System

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Figure IV-2. Cutaway View of SR-100G System

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Figure IV-3. Key Envelope and Interface Dimensions of the SR-100G System

TABLE IV-1
KEY FEATURES OF SR-100G SYSTEM

● SYSTEM	
■ NET POWER	100 kWe
■ DESIGN LIFE	10 yr
■ FULL-POWER LIFE	7 yr
■ MAXIMUM DIAMETER	4.3 m
■ LENGTH	6.1 m
■ MASS	3053 kg
● REACTOR	
■ FUEL	UN
■ CLADDING	W-LINED, ASTAR-811C
■ COOLANT	Li
■ STRUCTURE	ASTAR-811C
● SHIELD	
■ GAMMA	CMW-3950 (W ALLOY)
■ NEUTRON	SS-CANNED LiH
● POWER CONVERTER	
■ TYPE	CLOSED-CYCLE BRAYTON TURBOELECTRIC
■ FLUID	He-Xe
■ NUMBER	DUAL, FULL-POWER
■ MATERIALS	ASTARs
● RADIATOR	
■ TYPE	FIXED, GAS-COOLED FIN AND TUBE
■ FLUID	He-Xe
■ CHANNEL	DUAL D-TUBE
■ FIN MATERIAL	SS-CLAD Cu

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Figure IV-4. Design Point Heat Balance and Statepoints (Only One Converter Operating)

uninsulated area on the dual outlet pipes. This area is provided to reject afterglow heat when the system is shut down. This heat loss causes a slight drop in temperature before the lithium enters the first of the two lithium-to-gas heat exchangers.

These heat exchangers are connected in an integrated series fashion so that the lithium passes back and forth between the heat exchangers. This is done so that both heat exchangers can simultaneously operate at half power with similar temperature profiles. Lithium leaving the heat exchangers enters the first and then the second helical-induction EM pump. These pumps, powered by ac power from an inverter, circulate the lithium in the primary heat transport loop.

Two radiation shields attenuate the gamma and neutron radiation from the reactor to required levels at the dose plane, located 25 m from the top of the reactor. The tungsten-alloy gamma shield is located directly below the reactor. The neutron shield, shaped like the frustum of a cone, is a stainless steel can filled with lithium hydride. Steel and mesh structures are located inside the can to compartmentalize the shield and to provide structural ties between the reactor and the equipment below the shield.

The reactor heat transferred to the heat exchanger is used to heat the Brayton cycle working fluid. This fluid is a combination of helium and xenon with a molecular weight of 40. When a single Brayton turboelectric converter is being used (i.e., the design point condition), the turbine inlet temperature is 1400 K at a pressure of 2214 kPa. The working fluid at this temperature is expanded through the turbine, where work is extracted. The working fluid exits the turbine at 1127 K and at about 54% of the entering pressure. This working fluid then enters the recuperator, where 442 kWt is transferred to the working fluid reentering the lithium-to-gas heat exchanger. The recuperator for this system was optimized at a relatively low effectiveness of 76.5%, which kept its mass low. The working fluid exiting the recuperator then enters the gas-cooled radiator. Before entering the radiator, a small amount of working fluid is mixed into this flow; this is working fluid that

has been used to cool the combined rotating unit gas bearings and the lithium pump coolant channels. The combined fluid enters the radiator at 801 K. The radiator comprises a parallel arrangement of D-shaped gas flow ducts. Each Brayton converter has its own set of ducts; these ducts alternate on the inside and then the outside of the radiator. The total radiator area is 61 m^2 , which includes 33 m^2 on the conical section, 18 m^2 on the cylindrical section, and 10 m^2 on the annular disk at the back of the system (see Figure IV-2). The D-shaped gas ducts are part of a stainless-steel-clad copper radiator fin. The average fin effectiveness is 0.85, and the fins are coated with calcium titanate with an emissivity of 0.89. The radiator is sized to reject 400 kWt of thermal power; 396 kWt of this is waste heat from the Brayton cycle, and 4 kWt is input to the radiator from the solar flux (based on a worst-case sun-related orbit).

The working fluid exiting the radiator ducts is collected in a manifold at a temperature of 514 K. It is then ducted to the compressor, where work is input. The pressure of the fluid is increased about 92%, up to 2241 kPa. The fluid exits the compressor at a temperature of 695 K. Most of it then travels into the generator cooling area, but a small portion is bled to the bearings and then to the lithium pump coolant passages. Essentially the full gas flow is used to cool the generator. As shown on the statepoint diagram, the generator provides 109 kWe of gross electric power, 9 kWe of which is consumed parasitically or used for system housekeeping. The generator mechanical efficiency is 95%. The generator also requires a peak power of 4 kWe for field excitation, thus the net generator efficiency is 91.5%. The total heat that must be removed from the generator is 10 kWt, and this is removed by the working fluid exiting the compressor and entering the generator cooling area. Heat from the generator raises the temperature of the working fluid about 7 K before it enters the recuperator. In the recuperator, 442 kWt of power is input to the working fluid, raising its temperature to 1028 K before it enters the lithium-to-gas heat exchangers, completing the loop.

The electric power generated in the combined rotating unit is delivered to the power processor. This processor is located on the deployment boom,

near the payload. The power processor uses a saturable reactor, magnetic amplifiers, and a set of ballast resistors to control parasitically the electric power delivered to the user. The power processor is designed for continuous power demand control from 0 to 100 kWe. If the user demands zero power, the entire 100 kWe is dissipated in the ballast resistors. If the user demands 100 kWe, no power is radiated to space by the ballast resistors. There is a separate redundant power processor, with control equipment and ballast resistors, for each Brayton converter.

The reactor and shield are structurally "tied together" based on reentry considerations (the reactor and shield should reenter together, after the radiator has disassembled, to provide aerodynamic stabilization for reactor-forward reentry for higher potential of burnup). The reactor-shield combination is structurally tied to the radiator. The Brayton converter and deployment boom are supported by the radiator support rings via an internal structure.

The system was designed to meet the SP-100 preliminary design goals and requirements, and it has a high potential for meeting those requirements. The system was designed for 7 yr at full power of 100 kWe within the specified envelope and mass constraints. The overall envelope is 4.3 m OD by 6.1 m long, as shown in Figure IV-3. The total mass is 3053 kg, as detailed in Table IV-2. This mass somewhat exceeds the 3000 kg SP-100 limit, but is considered reasonably close at this stage of the design process.

Most importantly, the system was designed with safety in mind. When installed in the shuttle orbiter bay, all lithium in the system (which will be molten and in circulation) will be doubly contained and inert gas blanketed. The portion of the double containment covering the reactor and shields will be removed when the system is deployed from the orbiter.

The reactor is configured to be subcritical when immersed and flooded with water. This is accomplished by filling the open neutron leakage windows (used for control) with B_4C poison blocks. Should the reactor remain intact

TABLE IV-2
MASS SUMMARY OF THE SR-100G SYSTEM

<u>REACTOR SUBSYSTEM</u>		<u>HEAT REJECTION SUBSYSTEM</u>	
FUEL ELEMENTS	180	FINS	248
INTERNAL STRUCTURE	18	D-TUBES	114
VE SSEL	44	D-TUBE ARMOR	301
REFLECTOR ASSEMBLY	105	EMISSION COATING	10
ACTUATOR ASSEMBLIES	22	RADIATOR STRUCTURE	86
INSULATION	2	MANIFOLDS	38
LITHIUM	7	PAYLOAD THERMAL SHIELD	15
DOUBLE CONTAINMENT COVER*	42	POWER PROCESSING RADIATOR	22
LAUNCH SUBCRITICALITY			834
ASSEMBLY*	42		
	462		
<u>SHIELD SUBSYSTEM</u>		<u>POWER PROCESSING AND CONTROL SUBSYSTEM</u>	
GAMMA SHIELD	62	FIELD CONTROL AMPLIFIERS	3
NEUTRON SHIELD ASSEMBLY	354	CONTROL RECTIFIER ASSEMBLIES	9
PATCH SHIELDS	18	BALLAST CONTROL ELEMENTS	23
	434	PARALLELING AND SWITCHING	2
<u>PRIMARY HEAT TRANSPORT SUBSYSTEM</u>		POWER TRANSMISSION LINES	35
Li TO GAS HEAT EXCHANGERS (2)	191	REACTOR CONTROL ELEMENTS	22
Li PIPING AND INSULATION	29	BATTERIES	140
VOLUME ACCUMULATOR	36	EM PUMP POWER SUPPLIES	24
EM PUMPS	24	STARTUP INVERTERS	12
LITHIUM	9		270
	289		
<u>POWER CONVERSION SUBSYSTEM</u>		<u>MISCELLANEOUS</u>	
COMBINED ROTATING UNITS (2)	337	DOUBLE CONTAINMENT BULK	65
RECUPERATORS (2)	146	BULKHEAD	
GAS DUCTS AND INSULATION	150	COMPONENT MOUNTING	40
GAS MANAGEMENT EQUIPMENT	25	STRUCTURE	105
HELIUM-XENON	1		
	659		
		TOTAL	3053 kg
		*REMOVED AFTER LAUNCH	

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following a launch abort accident, the poison blocks would maintain the reactor subcritical, if immersed and flooded with water.

The reactor and its shields are also configured to provide maximum reentry heating to the reactor top face. This and the design of the reactor vessel (burnable lips) provide good potential for reentry burnup and the necessary dispersal of irradiated fuel.

Having provided this overview of the system, the following six subsections of the report describe the reactor, shield, primary heat transport loop, power converter, controller and power processor, and radiator in further detail.

1. Reactor Description

The SR-100G System uses a lithium-cooled, uranium-nitride-fueled fast spectrum reactor. The reactor configuration is shown in Figure IV-5. Key data on the reactor configuration are provided in Table IV-3.

The reactor configuration selected is based on the knowledge gained from studies of compact reactors combined with current analyses using data on the selected reactor fuels and materials. The core has 121 fuel elements with uranium nitride fuel pellets. The fuel element cladding is ASTAR-811C refractory alloy. This alloy is also used for the reactor vessel and structural materials. Lithium enters the inlet plenum at the bottom of the reactor through dual opposed inlet pipes. The domed inlet plenum and inlet plenum surge vanes distribute the lithium through holes in the gridplates, and the lithium flows up through the core. The fuel elements are spirally wrapped with wire for spacing and to enhance mixing and flow distribution. The heated lithium exits the core through the upper grid plate into the outlet plenum. Lithium exits the outlet plenum through dual opposed outlet pipes which turn and travel down cutouts in the periphery of the reactor fixed reflector.

Figure IV-5. Reactor Configuration for the SR-100G System

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TABLE IV-3
REACTOR DATA FOR THE SR-100G SYSTEM

FUEL		CORE	
MATERIAL	UN	NUMBER OF FUEL ELEMENTS	121
DENSITY (% TD)	94	EQUIVALENT DIAMETER (m)	0.241
ENRICHMENT (% U-235)	93.5	FUEL HEIGHT (m)	0.356
PELLET DIAM. x HEIGHT (mm)	17.27 x 25.4	FUEL ELEMENT PITCH (mm)	20.83
CLADDING		GRID PLATES	
MATERIAL	ASTAR-811C	MATERIAL	ASTAR-811C
THICKNESS (mm)	0.508	THICKNESS (mm)	12.7
BARRIER MATERIAL	W	COOLANT HOLE DIAM. (mm)	6.35
BARRIER THICKNESS (mm)	0.0254	REACTOR VESSEL	
FUEL ELEMENT		MATERIAL	ASTAR-811C
OD (mm)	19.05	OD x WALL THICKNESS (mm)	267 x 3.2
FUEL-CLAD DIAMETRAL GAP (mm)	0.764	NOMINAL HEIGHT (m)	0.7
LENGTH (m)	0.54	REFLECTOR	
SPACER WIRE DIAMETER (mm)	1.27	MATERIAL	Be
SPACER WIRE PITCH (m)	0.152	MIDHEIGHT THICKNESS (m)	0.101
GAS PLENUM LENGTH (m)	0.152	TAPER ANGLE (DEGREES)	7
		LENGTH (m)	0.356
		CONTROL SEGMENT STROKE (m)	0.178

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The reactor employs radial reflector control. The reflector material is clad beryllium metal, and there are fixed and movable portions, as shown in Figure IV-5. The movable reflector section is segmented into two sliding sectors: these move up and down over a 0.18 m stroke to open and close neutron leakage windows on the core periphery. The windows are opened near the core mid height for maximum control worth. The segment stroke (total travel) is one-half of the core height, and the two sliding segments subtend a total angle of 240° about the core circumference. Each segment is controlled through an actuator drive rod, as shown in Figure IV-6. The actuator rod, a hollow tube filled with lithium hydride, penetrates the neutron shield. The actuators, of the SNAP-8 type, are hardened high-temperature stepper motors with brakes. The rotational to translational motion is accommodated by dry lubricated gears and recirculating ball screws. The latter were tested at temperature in the early 1970s as part of the SNAP program. Two actuators are provided, one for each segment. Were it necessary to further split the segments, the 240° of control could be provided by four separate segments with four actuator mechanisms at a small increase in system mass (approximately 25 kg).

The fuel material selected was uranium nitride, 94% theoretical density enriched to 93.5% U-235. The pellets are loaded into 1.9-cm-OD ASTAR-811C cladding. A large diameter fuel element was selected based on the excellent thermal conductivity of UN. A large diameter element permits a larger fuel fraction in the core for increased neutron economy. The large diameter element is also stiffer, thus reducing the potential for thermally and hydraulically induced distortion and vibration, which could damage the core or cause reactivity fluctuations.

The cladding material selected, ASTAR-811C, shows good strength at the high temperature being considered for the reactor. Further data on the selection of ASTAR are presented in Section IV.C.2. Because the tantalum-based alloy ASTAR-81C is not compatible with UN fuel, a thin (0.025 mm thick) coating of tungsten is CVD deposited on the inside surfaces of the cladding. The cladding thickness of 0.51 mm was selected based on manufacturability and

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Figure IV-6. Reactor, Shield, and Control Actuators for the SR-100G System

on the anticipated pressure buildup inside the element (from the release of fission product gases) over the 7-yr full-power life of the reactor. The cladding thickness was not sized on any consideration of restraining fuel growth: as discussed later, the gap between fuel and cladding was selected to preclude any contact between the fuel and cladding at end of life.

The reactor uses a sealed fuel element, and the fission gases are retained in the element. A 0.152-m-long gas plenum is provided at the lithium inlet end of the fuel pin. A spacer holds the fuel above the plenum. The spacer, similar to a long "collar button," is basically two discs with a bent wire in between. As the fuel grows axially, the wire distorts to accommodate the growth.

The amount of fission gas released was calculated in two ways. First, the number of fission gas atoms generated in the SNAP-50 reactor was used.² SNAP-50 was designed to operate at 2200 kWt for 10,000 h. Second, the number of fission gas atoms generated in 2.2×10^4 MW h was calculated using the basic gas yield data from Ref. 3. Comparing the two (Table IV-4) shows the SNAP-50 data to be about 10% higher than the data calculated from Ref. 3.

TABLE IV-4
GASEOUS FISSION PRODUCT PRODUCTION

Fission Product	Boiling Point (K)	Fission Yield	Atoms Produced in 2.2×10^4 MW h	
			Calculated Using Yield	SNAP-50
Xe	166	0.21247	5.22×10^{23}	5.25×10^{23}
Cs	690	0.18160	4.46×10^{23}	4.59×10^{23}
Kr	121	0.0518	1.27×10^{23}	1.27×10^{23}
Rb	961	0.027	6.6×10^{22}	1.08×10^{23}
I	457	0.01534	3.8×10^{22}	8.6×10^{22}
			1.20×10^{24}	1.31×10^{24}

For conservatism, the SNAP-50 data were used to calculate the production of gaseous fission product in the SR-100G reactor. In the SR-100G, the energy produced in 7 yr of operation at 525 kWt (the design power) would be 3.2×10^4 MW h. By ratioing this to the SNAP-50 data from Table IV-4, the number of gas atoms generated in the maximum powered element was calculated to be 2.6×10^{22} . Based on data from Refs. 4 and 5, it was conservatively assumed that 20% of the atoms produced would be released to the plenum. This plenum, located at the lithium inlet end of the reactor, has a temperature of about 1400 K. The gas pressure due to the gaseous fission products at the end of the 7-yr life is about 2600 kPa. The partial pressure due to the helium-nitrogen gap bonding gas is 100 kPa at temperature; thus, the total end-of-life plenum pressure is approximately 2700 kPa.

Using this calculated end-of-life plenum pressure, the minimum cladding thickness due to pressure considerations was calculated using the following expression (which conforms to MIL-STD-1522):

$$t = \frac{PR}{\sigma E - 0.6P} ,$$

where

- t = thickness (cm)
- P = internal pressure (kPa)
- R = inside cladding radius (cm)
- σ = allowable stress limits (kPa)
- E = efficiency.

For an allowable stress of 69,000 kPa for the ASTAR-811C cladding and an efficiency of 1.0, the required cladding thickness is 0.36 mm. For fabricability reasons, the cladding thickness selected was 0.51 mm, which is more than adequate.

The gap between fuel and cladding was sized at 0.76 mm (diametral) at the beginning of life (cold). The fuel gap is loaded with a mixture of helium and nitrogen to provide a 100-kPa pressure at operating temperature (hot, clean). At this condition, the partial pressure of nitrogen is about 14 kPa. The nitrogen overpressure is provided to reduce the transport of fuel to the cladding during operation. The diametral gap between the fuel and cladding was sized to preclude contact between fuel and cladding in the peak-peak fuel element at the end of the 7-yr full-power life.

The fuel used is 94% of theoretical density (T.D.). There is a reasonable data base for UN fuel swelling, but this includes data on porous UN (densities in the 70 to 90% T.D. range), pellets with central holes, and solid pellets. To determine the fuel swelling characteristics in the SR-100G reactor, only the data available on solid (>90% T.D.) UN pellets were used. All or most of the fuel-swelling data on such pellets were collected and are shown in Figure IV-7. The test times and burnups for these data are shown in Table IV-5, and further information and references are provided in Section IV.C.2.

Using these data, a "maximum" fuel growth correlation was constructed by drawing an upper bound curve that encompassed all of the data (see Figure IV-7). This maximum fuel growth correlation was then used in a Rockwell fuel growth code called TIM. The TIM code calculates fuel growth for radial rings of fuel (20 rings were used in the current calculations) using a stress relaxation algorithm. The code, which accounts for fuel ratchetting phenomena, was checked using UO_2 fuel swelling data for GGA-type in-core thermionic convertors (because fuel growth versus time was determined for these by neutron radiography). The data reviewed encompassed about 95,000 h of in-core radiation test results from 12 convertors with metal burnup values to 1.6 at. %. The fueled emitter cans of the convertors were in contact with the UO_2 fuel surfaces for most of the radiation time and were periodically removed from the test reactor and accurately measured for emitter can growth effects.

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Figure IV-7. Maximum Growth Correlation Used To Size Fuel-Clad Gap

TABLE IV-5
UN SOLID PELLET, FULL DENSITY FUEL TESTS

Test Organization	Number of Tests	Burnup Range ^a (10 ²⁰ fissions/cm ³)	Irradiation Time (h)	
			Longest Single Test	Total Test Time
JAERI	3	3.9 to 6.8	b	b
ORNL UN-3	3	3.5 to 4.5	5,800	16,400
LRL-1	8	2.0 to 2.5	3,558	28,464
CANEL-57	15	1.9 to 15	11,985	113,530
BRR-3	1	2.5	3,888	3,888
BRR-10	5	1.9 to 2.3	3,888	19,440
BRR-13	3	1.6 to 2.0	2,592	7,776
ORR	2	6.0 to 6.5	b	b
				189,498

^aPeak burnup in the SR-100G is approximately 5.7×10^{20}

^bNot reported

The TIM fuel growth calculation was performed for the temperatures of the peak-peak fuel element (peak power element radially at peak axial fuel temperature location). The temperatures are shown in Figure IV-8. For this hottest location of the hottest element, the calculations indicated a residual end-of-life diametral gap of 0.08 mm. These calculations used the beginning-of-life temperatures over the 7-yr growth period. The initial heat transfer across the fuel to cladding gap is about 40% conductive and 60% radiative, and this heat transfer was used over the 7-yr life. The use of BOL temperatures was considered conservative. The reduction in the fuel cladding diametral gap over time at full power is shown in Figure IV-9. A hand calculation performed as a check indicated imminent gap closure at end of life.

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Figure IV-8. Fuel Element Temperature Profile (BOL)

Figure IV-9. Fuel-To-Cladding Gap Closure
With Operating Time

An additional concern in selecting a fuel to cladding gap was the potential for vaporization of the UN fuel and its subsequent transport and deposition on the cladding. Figure IV-10 (from Battelle studies) shows the effect of operating nitrogen overpressure (and fuel operating temperature) on the potential transport of UN. The SR-100G fuel element design uses a sealed pin. A helium-nitrogen fill gas provides 100 kPa of total pressure at operating temperatures with about 14 kPa of nitrogen partial pressure. The nitrogen partial pressure is provided because it reduces the transport of fuel to the cladding during operation. The transport and deposition of UN on the cladding was calculated as less than 5×10^{-5} mm over the entire 7-yr operating life of the reactor. The key to low transport is the nitrogen (and to a certain extent, helium) overpressure. If the design used a "vented-to-space" fuel design, UN transport could be a major technology issue.

The core loading of 121 fuel elements was chosen based on criticality considerations. This results in a relatively benign peak linear power density of about 19.5 kWt/m for the fuel. A core height of about 36 cm was selected to be consistent with the 121 fuel elements and to provide sufficient reactivity for a 7-yr life. A 1.27 mm spacer wire is wound around each fuel element to provide spacing between the elements and to promote coolant mixing between flow subchannels.

The fuel elements are sealed at each end with welded end plugs; the plugs contain tapered end pins that mate with holes in the 1.27-cm-thick upper and lower grid plates. This positively positions the fuel elements in a triangular array with a center-to-center spacing of 2.1 cm. The grid plates also contain coolant passages drilled between the fuel element positioning holes.

The grid plates are mounted in the reactor vessel; the vessel has a 0.27-m OD and a 0.32-cm wall thickness. The nominal vessel height is 0.7 m, but the end closure heads are concave.

Figure IV-10. Effect of Nitrogen Pressure On Transport in UN
(Source: Ref. 16)

A beryllium reflector radially surrounds the reactor. The reflector has a nominal thickness at mid height of 0.1 m and is tapered with a conical half angle of 7° . The height of the reflector is equal to the core height (including the fuel element gas plenum height). The flattened reactor outlet coolant pipes pass through cutouts on the outer surface of the stationary segments of the reflector. Ten layers of multifoil insulation surround the reactor and insulate it from the reflector. This serves two purposes: to eliminate heat loss from the reactor and to reduce temperatures in the beryllium reflector. Nuclear heating calculations performed on the beryllium reflector showed that it could be kept to a maximum temperature of 760 K next to the reactor. This material was thus selected over beryllium oxide.

The reactor has no axial reflectors, either outside the coolant plenums or inside the fuel elements. Because of the presence of the lithium outlet plenum on top of the reactor and of the fuel element gas plenum and the lithium inlet plenum on the bottom of the reactor, external reflectors were not used. The neutronic analysis showed that the reactor could perform as required without external reflectors. Internal axial reflectors (inside the fuel elements) are also not used: analysis showed them to be less "mass-effective" at increasing the reactivity of a compact reactor than increasing the fuel length. In addition, selecting a proper reflector material is always a problem because of concerns over fuel-cladding compatibility as well as over nuclear worth.

The reactor is controlled by moving the sliding segments of the radial reflector up or down to close or open the neutron leakage windows. This type of control has several advantages. First, the reflector itself can be relatively thin because only the portion of the reflector material that really has neutronic worth need be provided. Second, the reflector can be tapered from top to bottom to reduce the diameter at the top of the reactor. This reduces shield mass with only a minimal penalty on control worth. Third, the segments provide a window with close access to the reactor vessel. As further discussed below, poison blocks can be inserted in these windows to provide sub-criticality during a water flooding and immersion accident. This eliminates

the need for central poisoned holes in the reactor. These three advantages all tend to reduce reactor diameter significantly and by so doing reduce the mass of the gamma and neutron shields.

Two sliding segments are used; they encompass 240° of the reactor periphery. The segments have been configured so that either could operate the reactor at full power for 7 yr if the other segment fails. Operating the reactor with unequal segment positions will affect the power distribution, and detailed three dimensional calculations need to be performed to determine this effect. Along these lines, both the reliability and the power distribution effects could be improved (if necessary) by splitting the two segments into four and controlling each individually. This would add two more sets of actuators, drivelines, and segment guides, and the mass penalty would be under 25 kg.

Reactor performance is summarized in Table IV-6. The reactor core is designed to produce 525 kWt for 7 yr at a reactor outlet temperature of 1473 K. At these conditions, the fuel element peak linear power density is approximately 19.5 kW/m, which is very much below the capability of UN fuel. A smaller number of larger diameter fuel elements would yield a higher linear power, but the selection made was felt to be the appropriate compromise between fuel element diameter, clad thickness, and fission gas plenum length.

The calculated cold BOL excess reactivity available for temperature defect and burnup is 5.9% $\Delta k/k$ (57.9). The reactivity swing from cold BOL conditions to hot EOL conditions has not yet been evaluated in detail. However, we estimated that 5% $\Delta k/k$ would be required: 3% from the temperature defect and 2% from burnup. The 3% value is based on an overall reactor temperature coefficient estimate of $-2.5 \times 10^{-5} \Delta k/k$ per K. This is considered the maximum negative value for the coefficient: the actual value, when calculated in detail, is expected to be somewhat less than this (in absolute value) but still negative. However, because of the high fuel enrichment, the Doppler

TABLE IV-6
REACTOR PERFORMANCE SUMMARY

THERMAL

POWER (kWt)	525*
OUTLET TEMPERATURE (K)	1473
CORE TEMPERATURE RISE (K)	75
COOLANT FLOW RATE (kg/sec)	1.66
LINEAR POWER (kW/m)	
PEAK	19.5
AVERAGE	12.2

NUCLEAR

REACTIVITY (% $\Delta K / K$)	
COLD BOL	5.9
CONTROL WORTH	9.7
SHUTDOWN MARGIN	3.8
FUEL BURNUP (10^{20} FISSIONS/cm ³)	
PEAK	5.7
AVERAGE	3.6
NEUTRON FLUX (10^{13} n/cm ² -s)	
PEAK	7.2
AVERAGE	4.4
NEUTRON FLUENCE [10^{21} n/cm ² (E > 0.1 MeV)]	
PEAK	14
AVERAGE	8.5

*FOR DESIGN PURPOSES — EXCEEDS 508 kWt SYSTEM DESIGN POINT REQUIREMENT

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coefficient for the reactor may be slightly positive (as was the predicted Doppler coefficient for the SNAP-50 reactor). This is more than compensated by the prompt fuel temperature coefficient, the gridplate coefficients, etc. With the expected time constant of the UN fuel pellet (approximately the same as the time constant for a standard LMFBR uranium oxide fuel pellet), the small positive Doppler coefficient should not cause any control problems. However, this needs to be analyzed in detail.

With the 5% $\Delta k/k$ cold BOL to hot EOL swing, and the cold BOL excess reactivity of 5.9% $\Delta k/k$, there is approximately a 20% reactivity margin to cover uncertainties in the calculational method. The calculated reactivity worth of both sliding control segments is 9.7% $\Delta k/k$ (\$12.9). The control segment stroke of 0.18 m is divided evenly on either side of the core mid height. Dividing the segment stroke in this way results in a differential control worth that is nearly linear, based on the detailed calculations performed on earlier SNAP systems using this control method.

Subtracting the control worth from the available excess reactivity gives a cold BOL shutdown margin of 3.8% $\Delta k/k$ (\$5).

The calculated peak burnup in the fuel based on the cold volume is 5.9×10^{20} fissions/cm³. This is equivalent to a 1.8 at. % burnup. The peak total neutron flux of 7.2×10^{13} n/cm² s occurs at the core center. The peak neutron fluence at neutron energies greater than 0.1 meV is 1.4×10^{22} n/cm² for 7-yr operation at full power. This peak fluence also occurs at the core center.

Power distributions were calculated for the SR-100G reactor based on a two-dimensional synthesis of one-dimensional transport calculation in R-Z geometry. This calculation uses a leakage iteration between two one-dimensional calculations. The ANISN discrete ordinate code was used to perform the one-dimensional

transport theory solution. Comparing the results from this methodology for compact reactors with those from two-dimensional DOT calculations shows sufficient accuracy for conceptual design studies. The cross sections used in the analysis were a 42-group library that had been processed through the 1DX code to perform the necessary resonance self-shielding of the infinitely dilute cross sections using the core composition of the SR-100G. No spatial self-shielding was done due the lack of internal moderation.

The radial and axial power distributions for the conditions with the reflector segments fully closed is shown in Figure IV-11. At this EOL condition, the overall peak-to-average power (radial times axial) is about 1.65. Although calculations were not performed in two dimensions of the R-Z power spectrum or in three dimensions for the R-Z- θ power, such calculations had been performed in the early 1970s on Rockwell's 110-kWt reactor for the 5-kWe system. At the time of that study, Rockwell tried to assure that no problems would arise due to radial, axial, or azimuthal power distributions caused by the reflector segment windows (that reactor was also controlled in the same manner as the SR-100G reactor). The BOL and EOL R-Z power contours for that reactor are shown in Figures IV-12 and -13.

At end-of-life, this epithermal UZrH reactor had an overall peak-to-average power factor of 1.8, which is slightly higher than that calculated for the SR-100G reactor. This should be expected because the harder spectrum of the SR-100G reactor tends to flatten the profile. The azimuthal power variation that occurred at the interface between the fixed and movable reflector segments was also calculated in the 1970s for that 110-kWt reactor. The results, shown in Figure IV-14, show the azimuthal variation to be quite small. However, for either of these reactors, the most severe power profile would occur if one segment were fully inserted and the reactor were controlled with the other segment. Such an analysis needs to be performed for the SR-100G reactor. However, by reviewing Figures IV-12, -13, and -14 for the 5-kWe system reactor, a fairly good perspective on the "one segment in, one

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Figure IV-11. Reactor Core Power Profiles

6531 4616

Figure IV-12. 5-kWe System BOL Power Contours, Segments in Normal Position

6531 4517

Figure IV-13. 5-kWe System EOL Power Contours Segments, Fully Closed

6531-4603

Figure IV-14. 5-kWe System Peak Power Per Element Relative to Core Average

segment out" power profile can be obtained. The peak power would occur somewhat off core centerline. In the hard spectrum SR-100G, the overall peak-to-average factor at this location would probably be well below 2.0, and the powers at the core boundaries would not tend to be as low as with the epithermal spectrum reactor used in the 5-kWe system.

A major driver to the reactor design was nuclear safety. Two key nuclear safety requirements affect the design of the reactor: (1) the reactor must remain subcritical when immersed and flooded in a hydrogenous fluid (such as water or hydrazine) and (2) the reactor should burn up and disperse its fuel on reentry.

Several methods can provide hydrogenous fluid subcriticality. In the past SNAP reactor programs, it was attempted to make the core intrinsically subcritical when immersed and flooded in water. "Intrinsic" here means that resonance neutron absorbers or other "built in" features are provided that cause the reactor to be subcritical by depleting a portion of the neutron spectrum. This was not possible in the SNAP reactors, because they were epithermal reactors, and very little shift in spectrum occurred when the reactor was immersed and flooded in water. It was hoped that this intrinsic technique might be applicable to a fast spectrum reactor, where a considerable spectrum shift might occur during the water immersion and flooding accident. This was found not to be the case — very little spectrum change occurs.

The engineered feature method was thus used to ensure hydrogenous fluid subcriticality. In other words, some sort of removable device is put in or near the reactor that precludes it from becoming critical when the core remains intact but is fully immersed and flooded with the hydrogenous fluid. For the SR-100G reactor, the first method tried was successful. This method was to place removable poison sleeves in the windows created by the sliding beryllium control reflector segments. These sleeves, shown schematically in Figure IV-15, are located about the centerline of the core and are approximately one-half the core height. Neutronic calculations were performed for

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Figure IV-15. Hydrogenous Fluid Subcriticality Control

the normal, submerged, and flooded-submerged configurations. The results, shown in Figure IV-16, show the reactor to be slightly subcritical when both flooded and immersed.

The flooded and immersed configuration requires that the beryllium reflector and the poison blocks remain in place. The mechanical design concept to provide this is shown in Figure IV-17. In this design, the two poison blocks that fit in the reflector segment windows are permanently mounted to a retaining band. This "launch subcriticality assembly" is mounted onto the reactor. Before startup of the system (either after the system has been removed from the shuttle orbiter or is in its destined orbit), the launch subcriticality assembly would be removed. It is envisioned that this could be performed by the remote manipulator arm on the orbiter or that alternate mechanical means could be provided if it is to be removed in a higher orbit.

This approach differs substantially from that on the SNAP reactors designed in the 1960s. In those reactors, the reflector was designed to separate from the core during the accident, and the cores were critical in water. In the SR-100G reactor, the reflectors are held to the reactor structure by a stout support mechanism, and the launch subcriticality assembly retaining band holds the reflectors and the poison blocks to the reactor during the accident event.

The second key nuclear safety requirement is reentry burnup of the reactor with subsequent fuel dispersal. It is not now possible to completely design the required safety features of the reactor or fully predict reactor burnup. This is because burnup is strongly dependent on material properties, the exact reentry configuration, the aerodynamics of reentry, and the actual orbit. However, one important event that must occur to enhance the probability of fuel burnup and dispersal is the early burnup of the reactor vessel and the release of the core. This event must occur at a sufficiently high reentry altitude for there to be sufficient time to disassemble the core into individual fuel elements and ablate and disperse the fuel element materials.

	<u>CONFIGURATION</u>	<u>REACTIVITY (% $\Delta K/K$)</u>
<u>NORMAL</u>		5.9
<u>SUBMERGED</u>		-4.4
<u>FLOODED AND SUBMERGED</u>		-0.6

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Figure IV-16. Reactivity With Core Flooded and Submerged

Figure IV-17. Launch Subcriticality Assembly

A necessary condition for the early release of the core from the vessel is to maintain a stable "reactor-first" reentry mode. This reactor-first attitude provides maximum aerodynamic heating to the reactor vessel. Extensive wind tunnel testing in the SNAP-10A Aerospace Safety Program showed that the complete reactor power system (reactor, shield, and conical radiator) had a very stable reactor-first reentry mode. Tests also showed that the reactor-shield configuration was stable, although there were some oscillations. For a reactor only configuration, tests indicated that the reactor would tumble. For a tumbling reentry mode, the reactor core bundle is released at a much lower altitude.

The Rockwell SR-100G System is aerodynamically similar to the configurations tested in the SNAP-10A Aerospace Safety Program. The system is expected to have similar reentry heating characteristics. To assure reactor-first reentry, the design of the system provided a high-integrity structural tie between the reactor vessel and the structure of the neutron shield. This high-integrity structure maximizes the time the reactor vessel is exposed to peak aerodynamic heating. The vessel is equipped with a burnable vessel lip at the top that enhances burnup of the tie between the end closure on the vessel and the sides of the vessel. However, there are the possibilities of self-welding of metal surfaces in the reactor, including self-welding between the gridplates and fuel elements.

The system is configured to have maximum potential for reentry burnup and dispersal of the fuel based on the test data available. These test data are, however, limited to other types of alloys and materials, and the data must be made available for the alloys and materials to be used in such high temperature systems.

2. Shield Description

The SR-100G System uses a heavy-metal gamma shield and a lithium hydride neutron shield configured to meet the radiation requirements presented in Table IV-7. The shield configuration and pertinent dimensions are presented

TABLE IV-7
REQUIREMENTS ON REACTOR-INDUCED RADIATION TO THE USER

- 1) Life fluence and dose (7-yr full-power operation) the maximum life radiation levels to the user shall be

Neutron fluence: 10^{13} neutrons/cm² (1 MeV equivalent)
Gamma dose: 5×10^5 rads (silicon)
- 2) Peak flux: TBD
- 3) Location of requirement: On a surface of at least 4.5 m in diameter which is 25 m distant from the forward face of the reactor and perpendicular to a line from the center of the reactor through the center of the radiation shield.

in Figure IV-18. There are actually three different types of shields. Starting at the reactor and moving toward the dose plane, the first shield is the gamma shield. This is a heavy-metal shield made of a tungsten alloy. This shield is placed near the reactor to minimize its mass. The shield is a disc with an outer annular section that is one-half the thickness of the disc. The inner disc section is the same diameter as the reactor core and provides the primary shielding of the core from the dose plane. The thinner annular ring provides additional shielding to cover the dose plane and sufficient attenuation for gamma rays from the reactor which are scattered toward the dose plane by the beryllium reflector.

There is also a secondary reason for placing the gamma ray shield next to the core. As configured, the gamma ray shield is next to the lithium inlet plenum, and nuclear heating in the gamma ray shield is transferred from the shield to the plenum, providing additional heat input to the lithium coolant.

The next shield is the primary neutron shield. It is stainless-steel-canned lithium hydride. The neutron shield is insulated from the gamma shield by a layer of superinsulation because it is necessary to maintain the neutron shield well below the melting point of lithium hydride. The lithium hydride

Figure IV-18. Shielding Configuration

shield covers the entire system cone. This is necessary because significant attenuation occurs in the shield, and all neutrons scattered by any portion of the reactor must be attenuated by the shield. The top and bottom of the shield are segments of spheres, where the top shield segment height is equal to one-eighth of the LiH shield thickness, and the bottom shield segment height is one-fourth of the thickness. The neutron shield contains an internal stainless steel structure which provides compartmentization. Should the outer shell be punctured, these compartments would slow the leakage of hydrogen from the shield. The stainless steel can between the tungsten alloy gamma shield and the neutron shield is borated. This boration reduces neutron capture gamma rays in the tungsten.

The third type of shield is the neutron patch shield. There are two such shields, and they are also formed from stainless-steel-canned lithium hydride. These shields are located where the dual sets of inlet and outlet lithium pipes traverse the periphery of the primary neutron shield. Neutrons scattered from these pipes are attenuated by the patch shields.

The neutron and gamma shield designs were based on one-dimensional transport calculations and analogy to a system designed by Rockwell in the 1970s — the 5-kWe reactor thermoelectric system. The conceptual and preliminary design for this latter system was completed in 1973 under a contract to DOE and NASA. The analogy method was used because the SR-100G System and the 5-kWe system are very similar in shield design, and detailed shielding analyses had been performed by Oak Ridge National Laboratory (ORNL) for the 5-kWe system.

The analysis at ORNL had been performed to determine important effects, such as the thickness of the annular ring gamma shield, the lack of need for gamma shielding the total subtended cone of the system, the gamma and neutron attenuation provided by the power conversion subsystem (which were modular thermoelectrics), and the neutrons streaming through piping penetrations in the neutron shield (there were two pipe penetrations through the 5-kWe system neutron shield).

The analysis performed at that time to design the gamma ray and neutron shields used the DOT/SPACETRAN code. The DOT code, which is a two-dimensional transport code, was used in R-Z geometry to determine the leakage from the LiH neutron shield and also to shape the shield. A P_3 21-neutron and 18-gamma ray energy group coupled cross-section library was used. The DOT code incorporated a 298-angle quadrature with 84 angles between 0 and 8°. This correctly determined the neutron and gamma flux leakage as a function of angle from the bottom surface of the LiH shield. The SPACETRAN code was then coupled to the DOT code, and it summed the neutron and gamma fluxes at points on the dose plane.

The results of this analysis on the 5-kWe system are very pertinent to the SR-100G System. The resulting gamma shield design used a central disc shield that was the diameter of the core. The gamma shield also included an outer annular ring of the same material but only half the thickness. The outside diameter of this annular shield ring was set by the angle between the top of the reactor fuel elements and the payload dose plane. This shield did not need to extend to the full system cone angle subtended by the outside diameter of the reactor control reflectors and the dose plane. This has the effect of making the gamma shield much lighter.

The neutron shield design for the SR-100G System was designed using a neutron source strength obtained from axial neutron leakage analyses. A synthesized two-dimensional analysis was performed by using a buckling iteration of the ANISN one-dimensional transport code, called APC. The resulting neutron leakage was 33% higher than that for a UZrH reactor. This was expected because the SR-100G has a fast spectrum, and the UZrH reactor has an epithermal spectrum.

The gamma shield was designed using a gamma ray source strength based on a comparison of the present UN-fueled reactor with the 5-kWe UZrH-fueled reactor. The gamma ray leakage was 20% lower for the SR-100G reactor because the UN-fueled core has a higher material density than the UZrH-fueled core. The neutron and gamma ray shield attenuation coefficients for the SR-100G System

were then obtained by analogy with the 5-kWe system to reduce the source strengths to the values required at the dose plane. No credit was taken for attenuation provided by the power conversion equipment or other equipment located below the neutron shield in the SR-100G System. In addition, the shielded solid angle for the SR-100G System (the solid angle between the reactor and the dose plane) is only one-third that for the 5-kWe system, and thus it is probably possible to reduce the size of the SR-100G gamma shield even further.

The radiation maps for neutrons and gamma rays for the SR-100G System are shown in Figure IV-19. This shielding concept works for this specific system because the attenuation provided by the gamma shield itself is relatively low. For the SR-100G System (as it was for the 5-kWe system), the gamma ray dose is attenuated by only a factor of eight in the shield. There are gamma rays that emanate from the core and are scattered by the beryllium radial control reflectors. At least a single scatter in the beryllium is required for a gamma ray to reach the dose plane. When gamma rays scatter, they lose a large portion of their energy. For example, a 2-MeV gamma ray loses half its energy when scattered by only 40° . With the reduced energy of the scattered gamma rays, the gamma ray shielding effectiveness of the lithium hydride neutron shield is improved. If the gamma ray dose requirement at the payload were much lower, or if the reactor power were much higher, it would probably be necessary to extend the gamma shield to the full cone angle of the system with a subsequent increase in gamma shield mass.

The material for the gamma shield was selected after studying various high-density shield materials, including uranium (in the form of U-8 Mo), tantalum (in the form of Ta-10 W), tungsten (in the form of a tungsten alloy), lead, and borated steel. Lead was rejected because of its low melting point (about 600 K). Borated steel was rejected because of its excessive mass. U-8 Mo was rejected for its high internal heating (due to fast neutron fissions) and instability at high temperatures. The final two candidates (i.e., tantalum and tungsten alloys) were analyzed. The tungsten alloy was selected

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Figure IV-19. Radiation Maps

due to its slightly lower neutron capture cross section (lower secondary gamma ray generation rates). Table IV-8 presents the neutron absorption cross sections for both tantalum and tungsten for 28 neutron energy groups. The tungsten absorption cross sections are lower than those of tantalum in most of the energy groups.

TABLE IV-8
COMPARISON OF Ta AND W NEUTRON ABSORPTION CROSS SECTIONS

Group	E (eV)	Neutron Absorption Cross Section (barns)	
		Ta	W
1	1.0×10^7	-0.178 ^a	-0.272 ^a
2	6.5×10^6	0.0103	0.0156
3	4.0×10^6	0.0237	0.0269
4	2.5×10^6	0.0669	0.0536
5	1.4×10^6	0.120	0.0954
6	9.0×10^5	0.163	0.103
7	4.0×10^5	0.292	0.144
8	1.0×10^5	0.782	0.440
9	1.7×10^4	2.55	0.965
10	3.0×10^3	8.61	2.96
11	5.5×10^2	27.3	10.2
12	1.0×10^2	88.4	12.5
13	3.0×10	98.4	192.0
14	1.0×10	298.0	75.5
15	3.0	5.43	4.06
16	1.0	4.16	3.40
17	6.75×10^{-1}	4.41	3.83
18	5.5×10^{-1}	4.78	4.29
19	4.05×10^{-1}	5.51	5.08
20	2.7×10^{-1}	6.41	5.98
21	2.07×10^{-1}	7.79	7.26
22	1.35×10^{-1}	9.16	8.49
23	1.05×10^{-1}	10.51	9.68
24	7.75×10^{-2}	12.91	11.76
25	4.75×10^{-2}	17.36	15.61
26	2.4×10^{-2}	24.79	22.03
27	1.2×10^{-2}	37.14	32.75
28	4.0×10^{-3}	65.91	57.85

^aNegative due to (n,2n) reaction.

The selected tungsten alloy is made by Contacts Metal Welding, Inc., and the alloy is called CMW-3950 (previously known as Mallory 3950). It is 95% W, 3.5% Ni, and 1.5% Fe. It has a high density (18.0 g/cm^3), is machinable, can be manufactured in the size required, and can withstand high temperatures (1470 K).

Also important to shield design are the temperatures of the shields; temperature was analyzed for both the gamma and neutron shields. The heat generation in the tungsten-alloy gamma shield was based on a similar study for the SNAP-8 reactor. In this SNAP program study, Ta-10 W was used for the gamma ray shield. The study data were modified to account for the density difference of Ta-10 W and the CMW-3950 that is used in the SR-100G System. Figure IV-20 presents the heat generation rate in the gamma shield as a function of thickness with the reactor operating at 500 kWt.

The heat generation in the lithium hydride neutron shield was based on two studies, a SNAP-8 study and a SNAP-50 study. Figure IV-21 presents heat generation in the SR-100G neutron shield for the reactor operating at 500 kWt.

In addition, the heat generation in the molybdenum foils of the super-insulation (generated by gamma heating) was calculated to be $1.0 \times 10^{-2} \text{ W/cm}^3$.

A two-dimensional steady-state temperature analysis was performed for the temperatures of most concern, those in the neutron shield. The model included heating in the gamma shield that was transferred down through the superinsulation into the reactor support structure and then into the top of the neutron shield. A diagram of the thermal model is shown in Figure IV-22. The model is constructed so that conduction heat transfer is simulated in both the longitudinal and radial directions. In the model, the central core is made up of circular discs. The succeeding annular sections are all of equal volumes except for the nodal volumes that form the outer surfaces. The surface

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Figure IV-20. Internal Heat Generation in the Gamma Shield

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Figure IV-21. Internal Heat Generation in the Neutron Shield

- AXISYMMETRIC 2-D STEADY-STATE MODEL
- INTERNAL NUCLEAR HEATING (VARIES WITH DISTANCE FROM REACTOR)
- CONDUCTIVE COUPLING TO REACTOR SUPPORT
- RADIATIVE COUPLING TO SPACE (0.85 EMISSIVITY)

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Figure IV-22. Neutron Shield Thermal Model

emissivity of the outer conical frustum section of the shield is 0.85, and this is the only surface that radiates heat to space. The steady-state isotherm lines within the neutron shield at full-power operation are shown in Figure IV-23. Inspection of these profiles shows that the hottest temperature is located at the upper part of the shield, where the structure attaches to the reactor support. The significant portion of the nuclear heating is in this upper region of the shield.

The important consideration here is to maintain the temperature of the lithium hydride well below its melting point of 960 K. This was accomplished with the shield and insulation design for the SR-100G System. Also, the shield must be kept above about 600 K to permit reannealing. The bottom portions of the shield are cooler than this, and it may be necessary to insulate some of the outer surface to increase the temperature of this part of the shield.

The thermal analysis showed that the shield design for the system provides temperatures that do not exceed the melting temperature of lithium hydride with good margins. Based on the heat generation rates in the model, the reactor power level at which the temperatures would exceed the lithium hydride melting point were approximated. This reactor power level is about 1000 kWt. In this case, the heat generation in the shield can be reduced by having a two-region shield. The top region uses lithium hydride enriched in Li^7 (the Li^6 is reduced to 0.01% instead of the normal 7.5%). By reducing the Li^6 concentration, the heat generation can be reduced. The lower region will be natural LiH.

3. Primary Heat Transport Loop Description

The primary heat transport loop comprises many components. The lithium outlet piping carries the high-temperature lithium from the reactor outlet plenum down around the neutron shield to the lithium-to-gas heat exchangers. These tube-and-shell heat exchangers contain lithium inside the tubes. The

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Figure IV-23. Neutron Shield Temperatures at System Design Point

lithium tubes traverse back and forth between the dual heat exchangers to provide approximately similar temperature profiles to each of the redundant Brayton converters. The lower temperature lithium exiting the heat exchangers enters the first and then the second of the dual EM pumps. These pumps provide the driving force for the lithium in the loop. The reactor inlet piping carries the lithium back up around the neutron shield to the reactor inlet plenum. Connected to the loop, but located in a lower temperature portion of the system, is the volume accumulator. This provides compensation for the volumetric expansion of the lithium coolant and the refractory alloy of the loop. The accumulator also provides a positive overpressure for the lithium to prevent boiling and pump cavitation.

A final set of components associated with the primary heat transport loop is the double-containment bulkhead and the double-containment cover (see Figure IV-2). The bulkhead is a thin stainless steel shell that doubly contains all lithium in the primary loop. The bulkhead terminates at a flange below the LiH neutron shield. The double-containment cover mounts to this flange and completes the double containment for the primary loop by enclosing the reactor and lithium piping on the periphery of the shield. The bulkhead and cover are provided for safe launch of the system. The system is launched with the lithium in a molten condition. Since lithium at this temperature might react with the nitrogen atmosphere of the shuttle orbiter bay, double containment was provided for the lithium in the primary loop. The volume between the cover and the primary loop component is inerted. The cover would be removed once the system is out of the orbiter bay.

The inlet and outlet lithium piping outside the shield and on the periphery of the reactor reflector are similar in construction. The pipe cross section is flat with semicircular ends, and the total thickness across the outside of the flats is about 1 cm. There are four such pipes — the two from the lower reactor inlet plenum travel around the shield and the two from the

upper reactor outlet plenum travel around the reflectors and shield. The piping is flattened to offer a smaller scattering cross section for neutrons, thereby reducing the size of the neutron patch shields. The flow area of the piping is equivalent to that of 3-cm-ID piping. The piping transitions to 3.8-cm piping as it passes under the LiH shield. At design conditions, the maximum lithium velocity in the piping is about 2.4 m/s. The piping is constructed of ASTAR-811C refractory alloy.

The lithium-to-gas heat exchangers are mounted below the neutron shield inside the double-containment bulkhead. Key dimensions and data on the design of the heat exchangers are shown in Figure IV-24. These are shell-and-tube designs. The lithium flows on the tube side of the heat exchanger with the helium-xenon Brayton cycle working fluid on the shell side. This was done to minimize pressure drop on the gas side. There are six passes through each heat exchanger on the lithium side. After making a pass through one exchanger, the lithium passes through the crossover tube to the other heat exchanger. This is done to equalize temperature distributions in the heat exchanger so either heat exchanger can operate its associated Brayton cycle at full power and still be capable of working in parallel with the other heat exchanger at half power. Each heat exchanger is sized for the full duty of the Brayton turboelectric system at 100 kWe net output (see Figure IV-3). ASTAR-1611C was selected for the heat exchangers for its high strength-to-weight ratio.

The lithium pump (Figure IV-25) is a helical induction EM pump. It uses alternating current to generate forces in the fluid that move the fluid along the double-volute, single-pass flow channel. The pump design is based on a composite of similar pumps described in Refs. 6 and 7. Further data on the pump are provided in Table IV-9. The single-pass design provides simpler hydraulics than other types of pumps and minimizes inlet cavitation problems while improving pump efficiency. The double volute was selected to achieve an optimal length to width ratio for the pumping duct. Additionally, the duct pitch is larger at the pump ends, which lowers the fluid velocity and thus

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Figure IV-24. Li-To-Gas Heat Exchangers

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Figure IV-25. Lithium Pump Configuration

TABLE IV-9
LITHIUM PUMP CHARACTERISTICS AND PERFORMANCE

PUMP TYPE	EM HELICAL INDUCTION
FLUID PUMPED	Li
FLUID TEMPERATURE (K)	1400
FLOW RATE (Kg/sec)	1.66
DEVELOPED PRESSURE (kPa)	22.3
NET POSITIVE SUCTION HEAD (kPa)	48.2
PUMP EFFICIENCY (%)	18
INPUT TO PUMP (kV A)	0.91
PHASES	3
HERTZ	60
POWER FACTOR (%)	48
INPUT POWER (kWe)	0.437
VOLTAGE (volts)	500
INPUT POWER TO INVERTER (kWe)	0.545
PUMP COOLING LOAD (kWt)	0.63
LOSS THRU SUPPORTS (kWt)	0.13
PUMP COOLANT GAS	He-Xe
INLET PRESSURE (kPa)	2241
INLET TEMPERATURE (K)	695
ΔT (K)	55.5
FLOW (kg/sec)	0.025
PUMP WEIGHT, EACH (kg)	12
PUMP MATERIALS	
STRUCTURE	ASTAR-811C
MAGNETIC	Si-Fe
WINDINGS	Ag-Ni
INSULATION	ALUMINA

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lessens velocity head losses. The stator housing of the pump provides lithium containment in case of duct failure. At the temperature of the pumped fluid (which is the cold leg temperature of the lithium loop), the magnetic susceptibility of the stator and flux return stacks approaches 1.0. Therefore, to get a reasonable flux density in the pump requires cooling the stacks. The stacks are cooled with working fluid (gas) from the Brayton turboelectric converters. Each pump has dual redundant gas cooling channels, each of which is connected independently to each of the Brayton converters. By providing such cross connection, either lithium pump can operate with either Brayton converter. Cooling the pumps provides a 60% recovery of the magnetic susceptibility, which is adequate to provide good pumping efficiency.

The pump arrangement is shown in Figure IV-26. The pumps are installed in series to provide 100% full flow redundant pumping capability without the use of valves. The developed head of each pump is sized to include the pressure losses through a nonoperating pump. The overall pump efficiency of about 18% is very good for an EM device. Three-phase ac input to the pump comes from an inverter, and the total input power to the inverter is about 0.5 kWe.

The volume accumulator for the SR-100G System (Figure IV-27) is based on work performed on the 5-kWe reactor thermoelectric system. That system incorporated three such accumulators because it had two separate sodium loops (and the secondary loop had a large sodium inventory). Only one accumulator is needed for the single lithium loop of the SR-100G System.

The accumulator consists of a containment shell and a movable head. The movable head is supported by two bellows: one for primary containment, the other for secondary containment. There are thus three separate cavities inside the volume accumulator. The first, the primary containment cavity, is located between the bottom dome, the primary containment bellows, and the movable head. This is the volume occupied by lithium as it expands. The second, the secondary containment cavity, is located between the cylindrical

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Figure IV-26. Arrangement of Redundant Lithium Pumps

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Figure IV-27. Lithium Loop Volume Accumulator

shell and the two containment bellows. This cavity normally is evacuated. The third, the gas cavity, is located between the top gas dome, the movable head, and the secondary bellows. This cavity normally is gas charged. The charging gas provides a force on the movable head, which provides an over-pressure to the lithium loop.

The characteristics of the volume accumulator are given in Table IV-10. The accumulator is in an insulated section inside the double-containment bulkhead and is connected to the lithium loop by a small piping run. This allows operating the accumulator at 800 K, at which temperature materials such as Inconel can be used for the bellows. The accumulator has been sized to accommodate the maximum differential volume between the lithium and the loop material.

TABLE IV-10
LITHIUM LOOP VOLUME ACCUMULATOR CHARACTERISTICS

Maximum temperature (K)	800
Minimum temperature (K)	533
Pressure at maximum temperature (kPa)	234
Pressure at minimum temperature (kPa)	48
Δ volume (m ³)	0.0062
Effective diameter (m)	0.25
Length (m)	0.3
Dry weight (kg)	36
Bellows deflection (m)	0.127

Three important conditions for the primary heat transport loop were analyzed during the conceptual design studies: (1) heat losses to space before reactor startup, (2) reactor afterglow heat removal, and (3) heat losses during dormancy periods. The first, or prestartup, condition was analyzed to determine input power requirements. To accommodate startup of the lithium

loop, the lithium is maintained in a molten condition following lithium charging of the reactor and primary heat transport loop in the ground facility. The minimum temperature of 533 K was selected to be well above the freezing point of lithium. The primary loop lithium must be maintained in this condition in the shuttle orbiter bay (where the primary loop is totally surrounded by the double-containment bulkhead and cover) and in space before startup (where the double-containment cover has been removed). Analyses were performed to determine the heat losses prior to reactor startup with the double-containment cover removed. Because the temperature required to maintain the lithium molten is relatively low, the thermal radiation (for a reactor that has never been operated) to the space environment at an equivalent space temperature of 200 K is about 70 W. Table IV-11 breaks this total into its major components.

TABLE IV-11
HEAT LOSS PRIOR TO REACTOR STARTUP IN SPACE AT 533 K

Component	Heat Loss (W)
Reactor vessel portion exposed directly to space (insulated)	5
Reactor support structure connecting to shield (uninsulated)	20
Piping (insulated ^a)	6
Reactor reflectors	4
Heat transport equipment inside double-containment bulkhead	35
Total	70

^aWith decay heat removal sections insulated

This total heat loss must be supplied while the system is in the shuttle orbiter bay and while the system is in space before startup. In addition, the

primary heat transport lithium must be circulated to achieve temperature uniformity around the loop. In this instance, the inefficiency of the EM pump is beneficial. Almost all electrical power entering the pump is transformed to heat in the lithium, either through the inefficiency of the pump (because the gas cooling circuits are not operating at this temperature) or through frictional heating of the lithium in the loop. The EM pumps require a total of 50 We to provide sufficient lithium circulation in the primary heat transport loop. By supplying a total of slightly over 70 We to the pumps, the lithium is both heated and circulated to maintain it in a molten and uniform condition.

The second key condition analyzed was reactor afterglow heat removal. Figure IV-28 shows the reactor afterglow heat in watts versus full-power operating time and time after shutdown. With several years of full-power operation, the initial afterglow heat can peak out at values approaching 6% of the design reactor power, or 31 kWt. However, this power decays rapidly: about 1 h after reactor shutdown the power is only about 7.4 kWt. Because the Brayton turboelectric converters will spin down for about 1 h and can be used for removing the peak afterglow heat, the "1-h" afterglow heat was used to size a decay heat removal area. At 1500 K, the decay heat removal area required is only 0.03 m^2 . With such a small area, it was decided to use two uninsulated sections on the reactor outlet pipes where they pass through cut-outs in the reactor reflectors. These two areas will reject the 7.4 kWt at a temperature of 1500 K, as shown in Figure IV-29. As the afterglow heat decreases, the temperature of the reactor decreases. An attractive feature of this concept is that no active control is required and the system will operate at an equilibrium temperature dependent on the afterglow heat produced by the reactor. During equilibrium operation, 50 W of electric power is supplied to the EM pumps to circulate the lithium. The only mechanical action required for this concept is the removal of small strips of "multifoil" insulation from the 0.03-m^2 decay heat removal area on the reactor outlet piping. This is accomplished by mechanically tying the insulation to the movable segments of the reflector. When the segments are moved during the second stage of initial

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Figure IV-28. Reactor Afterglow Heat After Full-Power Operation

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Figure IV-29. Afterglow Heat Removal Capability

reactor startup, the insulation is irreversibly removed from the decay heat removal area on the piping.

The third key condition analyzed was thermal loss during dormancy periods. During dormancy periods, the reactor and liquid lithium are maintained at 533 K and circulated by the EM pumps. The Brayton converter is not operating, and no gas is circulating through the radiator. The heat losses from the primary heat transport system are the sum of those that occur before reactor startup in space and those that occur from the decay heat removal area at 533 K. These losses (see Figure IV-29) are 70 W and 115 W, respectively, and hence the total amount of thermal power that must be provided during dormancy periods to maintain the loop in a liquid condition is approximately 185 W. Again, this would be supplied by operating the EM pumps, and this power would be drawn from a hardened on-board supply.

4. Power Converter Description

The SR-100G System employs two redundant Brayton turboelectric converters. This arrangement is shown in Figure IV-30. Each converter is capable of providing 100 kWe net system power output. The redundant converters are arranged to permit each to be operated at 50 kWe net through the system. This "both operating" mode is more benign than the "one operating, one shut down" mode. This latter mode, used to design the Brayton turboelectric equipment, was called the "design point" condition. The statepoint diagram for this design point condition is shown in Figure IV-31. This diagram provides the operating temperatures, pressures, and flow rates at the key points of the Brayton turboelectric converter loop.

The design statepoints for the Brayton turboelectric converter were determined through a parametric study coupled with the system design point selection. This is discussed in Section IV.C.1. The Brayton turboelectric system is described below.

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Figure IV-30. Redundant Brayton Turboelectric Converter Arrangement

Figure IV-31. Design Point Heat Balance and Statepoints (Only One Converter Operating)

The Brayton turboelectric converters were specifically designed to provide redundancy with a true "on line" spare. To provide this capability, there are two identical and completely autonomous Brayton converters. Each converter can produce fully rated electric power if fully charged with He-Xe working fluid. With both Brayton converters operating (i.e., each producing 50 kWe of the 100 kWe total power), each of the converters is charged with approximately 50% of its rated working fluid inventory. If one converter fails, it would stop operation and the gas management equipment on the operating converter would double the working fluid inventory of that converter. This operating converter would then provide the full electrical power of the system.

The special features incorporated to accommodate these two modes were in the Li-to-gas heat exchanger and the gas radiator. The heat exchangers have piping access to every second pass of their respective tube bundles so that the single primary lithium heat transport loop simultaneously provides heat to the two heat exchangers operating in parallel (on their gas sides). This allows both converters to operate at nearly the same turbine inlet temperatures. The radiator accommodates the operation of both converters by having dual gas passages that share common fins.

The Brayton turboelectric converter includes the gas side of the Li-to-gas heat exchanger, the hot leg gas ducting, the combined rotating unit (CRU) (turbine, generator, and compressor), the recuperator, the cold leg gas ducting, and the radiator gas ducting (this is discussed in Section IV.A.7). The converter equipment concepts used in the SR-100G System are similar in many respects to those from previous systems designed and/or fabricated in government programs. These include the mini-BRU (Brayton rotating unit) for the Brayton Isotope Power System (BIPS), which was built and tested in the 1970s, and the Turboalternator Compressor (TAC) Brayton Turbogenerator, which was designed (and some components built) in the early 1970s. The Brayton turboelectric converter is described below. The Brayton turboelectric converter concept was prepared by the Garrett Turbine Engine Company of Phoenix under subcontract to Rockwell International for the SR-100G System design.

The Li-to-gas heat exchanger is described above as part of the discussion of the primary lithium heat transport loop. The hot leg gas ducting was sized at about 10 cm diameter, and C-103 material was selected for the hot leg.

The CRU consists of a radial inflow turbine, a four-pole (1500-Hz) Rice generator, and a centrifugal compressor linked by a single shaft. The shaft is supported by gas-lubricated journal and thrust bearings. A concept drawing and design characteristics for the CRU are shown in Figure IV-32. Specific characteristics of the Rice generator are shown in Table IV-12. The aerodynamic performance of the turbine and the compressor are consistent with current Garrett Turbine Engine Company production units adjusted for the specific speed, size, pressure ratio, and working fluid at the converter design point. The efficiency at the design point is projected to be 90% for the turbine and about 85% for the compressor.

The foil-type journal bearings that support the shaft radially have inherent hydrodynamic start, steady-state, and stop capabilities. The thrust bearing, also a foil bearing, is a double-acting configuration with the two thrust foils mounted face-to-face and separated by a gauged spacer. Both sides of the thrust bearing incorporate hydrodynamic start, steady-state, and stop capabilities.

A labyrinth seal is located at the compressor and turbine ends of the CRU to minimize leakage of the bearing cavity pressurization gas. Compressor discharge bleed gas is ducted to the thrust bearing and exits from the journal bearings to provide bearing cooling. As can be seen from the statepoint diagram, this bearing cooling gas is then ducted to cool the lithium EM pumps before it reenters the Brayton loop.

The generator stator assembly, end bells, and generator heat exchanger are sandwiched between two frame assemblies to which all structural ties are made. The wound stator assembly contains no organic material and will be sufficiently out gassed and canned to preclude contamination of the working

ROTOR SPEED (rpm)	45,000
COMPRESSOR ROTOR DIAMETER (cm)	15.6
COMPRESSOR BLADE TIP WIDTH (cm)	0.622
TURBINE ROTOR DIAMETER (cm)	17.1
GENERATOR ROTOR DIAMETER (cm)	13.7
GAS JOURNAL BEARING DIAMETER (cm)	5.84
GAS JOURNAL BEARING LENGTH (cm)	5.84
GAS THRUST BEARING DIAMETER (cm)	14.0
NUMBER OF POLES	4
FREQUENCY (Hz)	1500
MATERIALS	
TURBINE	TZM
TURBINE HOUSING	ASTAR-1611C
COMPRESSOR	17-4 pH
COMPRESSOR HOUSING	INCONEL 625

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Figure IV-32. Design Characteristics for Brayton Combined Rotating Unit

TABLE IV-12
GENERATOR FOR BRAYTON TURBOELECTRIC CONVERTER

Type	Rice
Design power (kWe)	110
Speed (rpm)	45,000
Number of poles	4
Frequency (Hz)	1500
Rotor diameter (cm)	13.7
Efficiency (%)	95
Voltage control	Active field
Maximum field power (kWe)	4
Cooling	Full compressor discharge
Cooling temperature (K)	695
Startup	Starter winding

fluid. In addition, a metallic or ceramic bore seal is provided to further preclude contamination of the working fluid. A liquid-cooled heat exchanger is incorporated in each frame assembly. This coolant transfers windage and electromagnetic losses in the generator to the working fluid from the compressor discharge.

The recuperator (Figure IV-33) is a compact, pure counterflow, plate-fin heat exchanger of all brazed and welded construction. It is fabricated of Hastelloy X. It is similar in construction and function to the Mini-Brayton recuperator designed, built, and operated for the BIPS program. The internal construction of the recuperator is also illustrated in the figure. Heat transfer takes place predominantly in the rectangular counterflow section where high-performance rectangular offset fins are used. Triangular end sections are used to distribute and collect gas flow to and from the counterflow section. End sections incorporate plain rectangular fins to uniformly distribute gas flow, with minimum pressure drop.

Figure IV-33. Recuperator Concept for the Brayton Turboelectric Converter

An important feature of the recuperator, one specifically designed for the SR-100G System, is that it has a relatively low effectiveness (0.78) compared with recuperators designed, built, and tested for other applications (where the effectiveness approached 0.98). The reduced effectiveness was part of the overall system optimization and was the most mass-effective value to provide performance for the converter within the mass and envelope constraints of the SP-100 Program.

The cold leg piping that connects the recuperator and CRU to the radiator is constructed from Inconel 617.

Two key concerns were addressed in designing the Brayton turboelectric converter: wearout modes and the compatibility of the converter materials with contaminants in the working fluid. In the context of mechanical wear of rubbing surfaces, wearout modes have been virtually eliminated by using gas foil bearings. These bearings have hydrodynamic start, steady-state, and stop capabilities. However, the bearings and shaft do make contact at the beginning of startup and at the end of shutdown. Materials must be provided that will withstand the number of cycles projected over the 10-yr life of the system. This does not appear to be a problem since many material combinations are available. The other context of a wearout mode is extension to such phenomena as stress-rupture, excessive creep, fatigue failures, etc. The rotating and stationary equipment of the Brayton converter is admittedly subject to such "wearout" modes, and proper design of the unit is required to reduce its potential susceptibility to such modes to acceptable levels. For example, in the past, this has been accommodated in the turbine by designing for a specific amount of creep over the steady-state and cyclic life of the converter.

The second concern — the compatibility with contaminated working fluid — has been effectively eliminated by the design of the system and the turboelectric converter. The concern relates to the transport of contaminant impurities in the working fluid around the Brayton loop. Since the loop contains high-pressure He-Xe, it can be hypothesized that the moving gas could

transport the contaminants (or tramp gases) and embrittle the refractory metal alloys employed in the loop. To keep this from happening, all sources of such tramp gases have been minimized or fully excluded from the converter. The possible sources are (1) contaminants entering through the heat exchanger (the heat source), (2) contaminants outgassing from electrical materials in the generator, (3) contaminants outgassing from lubricating materials in the bearings, and (4) contaminants outgassing from the converter structural materials themselves.

The first source has been eliminated. The lithium-to-gas heat exchanger is not the source of tramp gas. In fact, any gases entering the lithium loop from the reactor would be tied up in the lithium coolant. The second source, the electrical materials, has also been eliminated. The approach here is to have a hermetically sealed set of generator stator windings. In addition to being sealed from the working fluid with ceramic or metallic bore seals, the insulations used are nonorganic and would not in themselves be a source. The third source, lubricants of the bearings, has been eliminated. The bearings are lubricated by the working fluid itself (gas bearings).

The fourth source, the actual materials of construction of the converter, has been minimized by careful design and material processing. The Brayton converter must be heated and evacuated before use to outgas its various surfaces. It can then be charged with the working fluid. If necessary, the converter could be outgassed and sealed for an extended period before this charging. The working fluid itself will contain less than 1 ppm oxygen, which is considered to be the "worst tramp" impurity that could cause embrittlement. Oxygen can be gettered. All of the refractory metal alloys that would be used in the converter contain appreciable amounts of hafnium or zirconium. These two elements are effective getters of oxygen, and when the alloys are annealed at 1400 K to 1600 K, it is assured that all oxygen in the alloys will be in the stable form of HfO_2 and ZrO_2 . Any oxygen transport within the closed Brayton converter will therefore be dependent on the vapor pressures of these oxides. These vapor pressures are very low, as shown in Figure IV-34.

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Figure IV-34. Vapor Pressure-Temperature Relationships for Refractory Metals and Oxides

No significant transport of material would occur for vapor pressures below 10^{-8} mm Hg.

5. Controller and Power Processor Description

The control equipment and power processor equipment are an integrated package designed to perform the operational control functions for the SR-100G System. A schematic of the power processing and control equipment is shown in Figure IV-35. A more complete block diagram of the equipment is shown in Figure IV-36.

The SR-100G control and power processor equipment operates and protects the system during all phases of the mission, including ground handling, pre-launch, launch, initial startup, and operation (including power management, shutdown, restart, and switching from two converters to one converter). The control equipment (the left side of Figure IV-36) controls the reactor power by moving the sliding reflector segments. This control equipment brings the reactor up to power and then maintains the lithium coolant temperature essentially constant. The equipment is also used for reactor shutdown. Lithium temperature is to some degree also inherently self-regulated by the negative overall temperature coefficient of reactivity. Long-term adjustment of the reactor coolant outlet temperature is provided by the speed trim controller and the load trim controller to keep the shaft speeds of the CRUs within their control range and to keep the generator and power processing equipment from exceeding their ratings (and also in situations where the payload instructs the power system to provide only low power).

The main elements of the reactor control equipment are the reflector segment motor actuators, the power switches, the alternator and delay circuits, the controller package, the speed trim controller, and the load trim controller (see Figure IV-36). Moving the reflector segments changes reactivity by increasing or decreasing the neutron leakage through the window in the reflector at the axial mid-plane of the core. The position of each segment is

Figure IV-35. SR-100G System Power Processing and Control Schematic

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Figure IV-36. SR-100G Power Processing and Control Equipment Block Diagram

adjusted by means of an electrical actuator driving a recirculating ball screw. The resulting translational motion slides the segments up and down. The actuator is a reluctance stepping motor with four stator phases. Rotation is produced by sequentially flowing current through each phase coil. The direction of rotation is determined by the order in which the coils are supplied power.

The segments are driven in discreet steps corresponding to a single cyclic activation of the four phase coils. When not being driven in this manner, the actuator is locked against rotation by means of an internal spring-loaded toothed brake. This brake must be released before rotation begins and must be set at the end of rotary motion.

To limit the complexity of the flight hardware, the minimum of instrumentation to guide the operation of the controller is used. Thus, no segment position sensors are provided, no monitoring of the reactor power by neutron or gamma detectors is provided, and controller operation is based on a pre-established program with a minimum of information that is obtained from existing instrumentation or simple additions of instrumentation.

A typical functional block diagram for the reactor controller package is shown in Figure IV-37. Two such controller packages are actually provided. A pair of coupled oscillators (one from each controller package) supplies the basic pulse sequence used to initiate control operations. During startup, pulses from the oscillators are passed by the startup blocks (subject to conditions established by the pulse counter and temperature indicators) to the delay block and then to the alternator, where the pulses are alternately passed to one or the other of the phase sequencers. The active phase sequencer produces a series of pulses that turn the power switches, in order, passing current to the actuator brake release coil and the four actuator phase coils. At the completion of the cycle, the brake is reset. The period between the series of pulses determines the stepping rate. To limit power and temperature transients during startup and the beginning of the power operating

Figure IV-37. Reactor Controller Package Block Diagram

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phase, and to meet the startup time requirements for the system, three stepping rates are used.

The startup blocks determine the segment insertion rate by dividing the pulse rate from the oscillators. The insertion rate is selected according to segment position and the lithium temperature. The segment position relative to the starting position is determined by counting current pulses to two of the phases of each actuator. The order of these pulses identifies the direction of the actuator step.

Segment drive commands originating the startup or operating blocks pass to the delay block, where they are inhibited unless more than 1 s has elapsed since the previous command was passed. This ensures that adequate time has elapsed for completion of an actuator step before a subsequent drive command is received.

The alternator functions to share insertion and withdrawal commands equally between the two segments. So long as segment motion is maintained in one direction, each segment is stepped alternately. If the motion is reversed, the segment stepped first is the one that had just been stepped.

The phase sequencer that is selected by the alternator generates a series of phase signals whose order is determined by the drive direction required. These signals initiate release of the actuator brake and control the sequential application of current to the four stator phase coils in the actuator.

A second set of equipment performs the power processing functions for the SR-100G. These functions include load control, voltage control, CRU speed control, load paralleling, and CRU startup control. The equipment for one of the two power processors is shown on the right side of Figure IV-36. Further detail on the processor is shown in Figure IV-38. This figure actually shows the details of both power processors. The power processor equipment must radiate energy to space and is located in an area of low neutron and gamma

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Figure IV-38. Electric Power, Voltage, and Load Control for the Rockwell SR-100G System

radiation. The processor also includes high-temperature ballast resistors capable of rejecting 100 kWe to space. The potential location of the power processor, its radiator, and the ballast resistor is shown in Figure IV-39.

The power processing equipment parasitically draws power from the system. This housekeeping power is approximately 7.6 kWe with a 100 kWe net system power output. The housekeeping power budget is shown in Table IV-13. Note that a margin of 1.4 kWe has been provided. Each Brayton converter is sized to provide a gross power of 109 kWe, with the system net power being 100 kWe when the margin is accounted for.

The overall power processor shown in Figure IV-38 provides control for the CRUs and the starter winding in the CRUs. During normal operation, each CRU delivers approximately 50% of the net system output (i.e., 50 kWe). However, the converters are sized for 100 kWe net, and the associated power processor equipment for each converter is also rated at the full 100-kWe capacity. In other words, each CRU has its own dedicated power processor and full ballast loading capability.

As shown in Figure IV-36, the turbine acting as the prime mover drives the compressor and generator. The generator field is the active electromagnetic type, with the field mounted in the generator stator (Rice machine). There are no windings in the four-pole rotor of the generator. Nominal output frequency is 1500 Hz with three or six phases.

The main elements of the power processor shown in Figure IV-38 are the voltage error preamplifiers, the saturable reactors for ballast control, the controlled rectifier assemblies, the load sharing and control, the magnetic power amplifiers, the magnetic preamplifiers, the ballast loads and their radiators, and the switching-parallel-protection circuitry. Magnetic components were selected wherever possible because of their inherent higher hardness to most radiation effects. Due to the high-frequency (1500 Hz) operation, this selection does not result in significant mass and size penalties.

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Figure IV-39. Potential Location of Power Processor and Control Electronics

TABLE IV-13
HOUSEKEEPING POWER BUDGET AT DESIGN POINT
OF THE SR-100G SYSTEM

<u>SYSTEM HOUSEKEEPING</u>	<u>POWER (kWe)</u>
<u>Li PUMP</u>	
TRANSMISSION	0.05
CONDITIONING	0.10
PUMP ELECTRICAL	<u>0.45</u>
SUBTOTAL	0.60
<u>SYSTEM CONTROL</u>	0.20
<u>POWER PROCESSING</u>	
SHAPING NETWORK	0.10
PREAMP	0.10
POWER STAGE	0.50
FIELD POWER	<u>4.00</u>
SUBTOTAL	4.70
<u>SPEED CONTROL</u>	
SHAPING NETWORK	0.10
SATURABLE REACTOR	<u>0.50</u>
SUBTOTAL	0.60
<u>POWER TRANSMISSION</u>	0.50
<u>POWER CONTROL RECTIFIERS</u>	1.00
<u>MARGIN</u>	<u>1.40</u>
<u>TOTAL</u>	9.00
<u>SYSTEM NET POWER</u>	<u>100.00</u>
<u>SYSTEM GROSS POWER</u>	109.00

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Discreet power semiconductors are used discriminately to raise survivability levels above those available with CMOS technology.

The power processor operates in general by error detection and control. The error detector senses any deviation of the generator voltage from the preset reference level and responds by commanding the solid-state preamplifier to drive the magnetic preamplifier and the magnetic amplifier until the deviation has been reduced to less than a predefined acceptable amount.

The voltage regulation is achieved by reducing the power to the active field of the generator when the output exceeds the preset limits (or vice versa). Thus the voltage regulator acts as a fast servomechanism to protect the user loads from damaging, stressful conditions. It also compensates for speed variations, environmental effects, and transients caused by external load variations and input excursions. The regulated voltage provided by the system can be designed for a specific nominal value from 100 to 400 Vdc. The voltage is adjustable over a 20% range around this nominal value.

The main smoothing effect on load variations is accomplished by the ballast load control. The ballast load control switching is performed by static saturable reactor (SR) magnetic components. The saturable reactor is essentially a linear controller that operates like a magnetic amplifier and offers infinite resolution. The ballast control assists in voltage regulation, but its main function is load control and thus frequency and speed regulation.

The power input from the reactor to the CRU shaft is fairly constant, and adjustment to preset levels of reactor power require long periods as compared with the ballast load control time constant. Excess power input or diminished user load is detected by the load and speed sensor, which then acts to drive the SR in a direction that modifies the ballast load demand as required. In other words, when the user load increases up to the 100-kWe capability of the system, the load dissipated in the ballast resistor decreases. The ballast

resistors are configured in a redundant series-parallel arrangement. The ballast resistors for each independent power processor are capable of rejecting the complete 100-kWe net output of the system. Thus, as configured, the system can continuously follow user load changes from 0% to 100% of full power output. It has this infinite resolution because the processor uses magnetic amplifiers and saturable reactors. The slewing rate response of the processor is about 1 kWe per cycle. At a processor frequency of 1500 Hz, the processor can thus follow user power changes at rate up to 1500 kWe/s.

The loads delivered by the CRUs are continuously monitored. When an overload is detected, the load control system initiates action to reduce the preset level of reactor output and thus prevent damage or system runaway. When the load detector senses insufficient power input, it raises the power command level. The SR's efficiency does not have to be good. Load dissipation in this system is beneficial. Thus the SRs are small and compact.

Switching, protection, and load sharing are accomplished by the two fully rated (100 kWe) silicon controlled rectifier (SCR) banks, the current sharing transducers, and the switching network (shown in Figure IV-40). The switching network does not normally interrupt power; rather it offers three-way optional decoupling via dry switching. The SCRs remain off until nominal speed has been achieved and both voltage and speed regulation action have been stabilized (see Figure IV-35). When the paralleling comparators indicate acceptable voltage and speed conditions (dc output), the SCRs are commanded to "turn on" and load current may begin to flow. The voltage of both machines is now equal. The transducers compare generator currents and adjust the field excitation to ensure equalization. Current transformers in the SCRs ballast loops act in a daisy loop fashion to ensure equal overall torque and speed output.

Although no specific standards exist for the design of this power processor, it is expected that standard electrical interfaces will be used, where applicable, including MIL-STD-S04, MIL-E-6051 (EMC requirements, systems),

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Figure IV-40. Typical Generator Disconnect Method

MIL-STD-1541 (EMI characteristics, requirements, equipment), and MIL-STD-462 (EMI characteristics, measurements). The control equipment and power processor equipment are designed to accommodate user failure without resultant system failures.

6. Radiator Description

The Rockwell SR-100G System uses a conical-cylindrical-annular disk, gas-cooled radiator. The overall radiator configuration is shown in Figures IV-41 and -42. The conical radiator section provides 33 m^2 of area, the cylindrical section provides 18 m^2 , and the disk section provides 10 m^2 , for a total of 61 m^2 of radiator area.

The radiator is of all-metallic construction and uses D-shaped gas flow channels in the arrangement shown in Figure IV-43. This figure shows the cross section of a fin and its D-tube channels in the cylindrical portion of the radiator. The radiator, actually two radiators in one, can be called a dual D-tube or shared-fin design. One of the gas-flow channels is connected to one of the Brayton turboelectric power converters. The other gas flow channel (on the side of the fin opposite the first channel) is connected to the other converter. These connections are alternated as one progresses around the radiator, i.e., the next set of channels is connected oppositely. The method of connection to the gas manifolds (either the manifolds at the top of the radiator or at the bottom of the radiator) are shown in Figure IV-44.

Hot Brayton working fluid from the recuperator enters the top manifold of the radiator and is distributed to the radiator D-tubes. The working fluid flows down the D-tubes and transfers heat to the metallic fin. The working fluid is collected in manifolds on the bottom of the radiator (at the ID of the annular disk) and is returned to the inlet of the CRU compressor. The design point characteristics of the radiator are shown in Table IV-14. These conditions are for only one of the two Brayton turboelectric converters (i.e., design point conditions). The radiator has been designed to reject the total

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Figure IV-41. Artist's Conception of SR-100G
Reactor System

Figure IV-42. SR-100G Turboelectric Power System

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Figure IV-43. Radiator Dual Tube-Fin Configuration

Figure IV-44. Radiator Manifold Arrangement

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TABLE IV-14
He-Xe-COOLED RADIATOR CHARACTERISTICS
AT DESIGN POINT

Heat rejected	400 ^a
Radiator coolant	He-Xe
Radiator inlet temperature (K)	801
Radiator outlet temperature (K)	514
Radiator area (m ²)	61
Surface emissivity	0.89
Radiator mass flow rate (kg/s)	2.59
Average gas pressure (kPa)	1176
Radiator tube diameter (cm)	1
Number of D tubes	106
Average fluid to wall ΔT (K)	15
Fin effectiveness (average)	0.85
Radiator assembly mass (kg)	
Radiator tubes	114
Radiator fins	248
Radiator armor	301
Emissivity coating	10
Radiator structure	86
Manifolds	38

^aIncludes 4 kWt of solar input
0129C/nth

amount of waste heat from the Brayton turboelectric cycle with only one converter operating. With both converters operating, gas flows through all D-tubes.

There are 106 D-tubes for each converter. The radiator was analyzed using an axial step-wise integration to determine heat rejection capability. Fin effectiveness was calculated at each axial section. The average fin effectiveness for the radiator is 0.85. The average temperature drop between the working fluid and the wall is 15 K. The overall heat rejected by the radiator includes 4 kWt of input from absorbed solar energy in a worst-case orbit. In earth orbit, the system will intercept up to about 26 kW of radiant power from the sun. Materials similar to the calcium titanate coating on the radiator have a solar absorptivity of about 0.15. Assuming this value for calcium titanate reduces the power received by the radiator from the sun to less than 4 kWt. If the system is in low earth orbit it could receive up to 3 kWt more from the earth. This was accounted for by using an effective temperature of space of 200 K.

The basic materials of construction for the radiator are 300-series stainless steel and copper. The radiator fins are a sandwich of stainless-steel-clad copper (Figure IV-43). This material was chosen for its thermal effectiveness. It is also used as part of the overall system structure. The outer and inner D-tubes are fabricated from 300-series stainless steel. This was selected for fabricability and armor capability. The D-tubes are brazed to the stainless steel cladding of the fin. Calcium titanate is used as the emissivity coating and is applied to the stainless steel outer surface of the fin and to the outer surface of the armored D-tube. This coating was selected for its stability, proven bonding capability, high emissivity (0.89), low absorptivity, and low Z atomic number (for survivability effects).

The amount of armor required to protect the radiator tubing from penetration by micrometeorites was calculated using the following equation from Ref. 8:

$$\delta = \gamma_a \left(\frac{6}{\pi \rho_p} \right)^{1/3} \left(\frac{\rho_p}{\rho_a} \right)^{1/2} \left(\frac{V_p}{\sqrt{\frac{E_a g}{\rho_a}}} \right)^{2/3} \left(\frac{\alpha A_v}{-\ln P(0)} \right)^{1/3B} \left(\frac{1}{B+1} \right)^{1/3B}$$

This equation was used in conjunction with the meteorite flux coefficients from Ref. 9 for the near-earth environment:

$$\text{LOG}_{10} N_t = -14.37 - 1.213 \text{ LOG}_{10} m.$$

Using these equations, the required armor thickness for the outer D-tubes is 0.31 cm for a 0.99 probability of nonpuncture over the 10-yr life of the system. This armor thickness is based on the vulnerable area of one-half of the outer tubes, because each Brayton converter uses half of the outer tubes and the alternate half of the interior tubes. This provides the same exterior nonpuncture probability for each of the two converter trains.

There are two potential types of secondary damage from micrometeorites. In the first, a micrometeorite punctures the fin and penetrates one of the interior tubes (on the back side of the fin). In the second, a meteorite enters the opening at the back of the system and penetrates one of the interior tubes. A review of the available literature for secondary damage effects from micrometeorites led to the conclusion that the nonpuncture probability for the interior tubes was probably somewhat above 0.99 for meteorites penetrating the fins. This could not be fully quantified because the data are sparse. Similarly, the nonpuncture probability for meteorites entering through the 1.4-m-diameter opening in the back of the system is also in excess of 0.99. These are both worst-case estimates for the nonpuncture probability. The overall nonpuncture probability for each of the independent radiator gas flow paths will be somewhat below 0.99 because of these effects.

The radiator mass also includes that for the stabilizing rings structure, the manifolds, the payload thermal shield, and the power processing radiator. The payload thermal shield is tentatively located at the power processor and the user interface (Figure IV-45). As noted for the power processor, this insulation could be located anywhere along the deployment mast and need not be located at the user interface.

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Figure IV-45. Power System Thermal Radiation Shield

B. SYSTEM OPERATIONS

1. Startup and Restart

This section discusses the initial startup and restart of the SR-100G. Certain information on startup is classified SECRET and is provided in the classified addendum¹⁰ to this report.

A simplified diagram showing the basic steps for the initial system startup is shown in Figure IV-46. As installed in the shuttle orbiter bay, the system is configured as follows. The reactor, never having been operated, has essentially zero fission product inventory. The reflector control segments are fully retracted, and the open windows created by the segments in this position are filled with the launch subcriticality poison block assembly. The reactor is subcritical by about \$6. The double-containment cover is installed to the flange on the double-containment bulkhead. Power is being supplied to the dual EM pumps from the shuttle power supply. This power is maintaining a lithium coolant flow at about 5% of rated value and is also being used to heat the lithium and maintain it liquid at 533 K. Both Brayton turboelectric converters are shut down. There are no maintenance requirements for the converters or the gas-cooled radiator. These conditions are maintained through STS launch, ascent, and orbit.

Following SR-100G system deployment, but before its ascent to orbit, the double-containment cover and the launch-subcriticality assembly are removed. Power is transferred from the shuttle power supply to internal battery-supplied power. At this time, the multifoil insulation covering the 0.03-m^2 afterglow heat removal area on the lithium outlet piping is still in place. Only 70 W of thermal power is required to maintain the lithium molten and uniformly distributed in this configuration. Before ascent to final orbit, the two controller drives are checked by stepping them once in the insertion direction and once in the retraction direction. The segments are then rapidly

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Figure IV-46. SR-100G System Startup Sequence

inserted to a predetermined position that leaves the reactor in a 50% subcritical condition (the position is predetermined through zero-power critical checks during reactor ground checkout). At this point, the small sections of multifoil insulation covering the afterglow heat removal surfaces on the reactor lithium outlet piping are still in place. Thus, a total of 70 W is required to maintain the lithium liquid and in a uniform condition. This thermal input is supplied through the EM pumps. At this point, the system is ready for transfer to operating orbit.

Following transfer to a stable orbit, a startup command signal initiates a preprogrammed startup sequence. The system controller is preprogrammed with three insertion rates for the two reactor control segment drives. The segments are stepped alternately at either the slow rate (250-s stepping period), the moderate rate (25-s stepping period), or the fast rate (4-s stepping period). The startup command first causes the lithium pumps to increase lithium coolant flow from about 5% to 50%. This reduces any effect that reactor power perturbations could have on system temperatures. With 50% flow achieved, the first reactivity insertion sequence is begun. The segments are inserted at the fastest rate, bringing the reactor from a 50% subcritical condition to a zero-power critical position (preprogrammed position). The next sequence brings the reactor from the zero-power critical condition to the sensible heat condition (about 1% power). This sequence is accomplished at the slow reactivity insertion rate. This rate is required to prevent excessive reactor power peaking. With sensible heat provided by the reactor, the reactor outlet temperature has increased about 50 K.

The next part of the startup sequence is performed at the moderate reactivity insertion rate. This rate is still low enough to allow the reactor's negative temperature coefficient of reactivity to modify the rate of reactor power increase. As reactivity is added, the reactor power increases slightly at a slow rate. The increased power increases the reactor and primary heat transport loop temperatures essentially isothermally. The temperature of the reactor and loop are controlled by the thermal losses from the system.

As this portion of the reactivity insertion sequence continues, a startup command is triggered for power converter 1 when the reactor coolant outlet temperature reaches 1100 K. This signal activates the startup drive on the first combined rotating unit (CRU 1), as shown in Figure IV-47. The startup drive draws power from the startup batteries of the system, which provide three-phase ac power to the separate startup windings on the CRU. Startup power is provided for about 20 s, until the CRU is self-sustaining and will accelerate rapidly under its own power. The self-sustaining speed is estimated to be about 50% of the full operating speed of the CRU. Startup winding slip is sensed, and when it approaches zero, startup power is discontinued. CRU 1 then accelerates to its operating speed within about 20 to 60 s.

As CRU 1 accelerates to operating speed, the alternator begins producing three-phase ac voltage as the field flashing circuitry begins to operate. A self-sustaining circuit is developed. As the CRU reaches operating speed, the power processor automatically cuts in the ballast load resistors to regulate CRU speed and power. As the CRU "spinup" power begins to be rejected by the radiator, the cooled gas is transported to the lithium-to-gas heat exchanger. This causes a temporary decrease in the reactor coolant inlet temperature. As reactivity continues to be inserted, the reactor inlet and outlet lithium temperatures resume their steady-state values. The temperature differential across the reactor increases as a result of the increasing ballast load and heat rejected by the gas-cooled radiator.

The power processor is increasing the amount of power dumped to the ballast resistor for power converter 1. As this power increases, the reactor coolant outlet temperature increases.

Just before converter 1 provides 50 kWe to its ballast resistor, the startup controller commands CRU 2 to spin up. The second converter is started in the same manner as the first, except that the time for this converter to develop 50 kWe in its ballast resistor is much shorter since system temperatures are very close to their normal operating values. Reactivity continues

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Figure IV-47. Power Processor and Control Block Diagram

to be inserted as the second converter is developing power in its ballast load. About 25 K before the reactor lithium outlet temperature set point is reached (normal operating condition), the control mode is switched from start-up to operating. The reactivity rate is then changed back to the slow rate to prevent temperature overshoot at the start of temperature regulation. At this point, each converter is operating at 50-kWe net output from the system into the ballast load resistors. Reactor control segment insertion or withdrawal is directed in response to temperature regulation as adjusted by the speed and load trim controllers. In this mode, anywhere from 0 to 100 kWe can be transferred to the user, with the remaining power being dissipated in the ballast load resistors.

At this point, mechanical links between the multifoil insulation covers (covering the afterglow heat removal area on the lithium outlet pipes) and the control segments cause these covers to be removed and discarded. This increases the heat losses from the primary loop piping. These losses are compensated by the power being developed in the reactor.

2. Operation at Power

In the normal mode of operation of the SR-100G System, the reactor provides equal thermal power to both turboelectric converters. This is shown schematically in Figure IV-48. As discussed in Section IV.A, the system was designed to operate with only one converter providing the full 100-kWe system output. This is the "design point" condition. In the "operating point" condition, each converter provides about 50 kWe of the total system output.

The operating point condition is, for several reasons, a more benign condition than the design point condition. One reason is that the recuperators and lithium-to-gas heat exchangers are oversized for the operating point condition. In addition, the margin in housekeeping power (an additional 1.4 kWe) was included in the design point condition and must be accounted for. The more benign operating environment with both converters performing is shown in

9467-264

Figure IV-48. Redundant Brayton Turboelectric Converter Arrangement

Figure IV-49. The key design point and operation point conditions are compared in Table IV-15. The table shows that the reactor power is actually somewhat higher at the operating point than at the design point. This is because the increased effectiveness of the heat exchangers and recuperators lower system temperatures. The reduced temperatures reduce the overall system Carnot efficiency and thus the reactor must produce slightly more power to produce 100-kWe net output. However, the reactor was designed for 525 kWt at 1473 K coolant outlet temperature.

With both converters operating, the control and power processor equipment is controlling system load, paralleling between the converters, voltage, CRU speed, and reactor outlet temperature. The method of this control is discussed in Section IV.A.6.

Should one of the Brayton turboelectric converters begin to fail, the failure will be sensed by the load-sharing transducer in the power processor for that converter (it may also be sensed by monitoring sensors on the CRU itself). At this point, the control system will decide whether to fully enable the operating converter. If so, the gas management equipment on the operating converter will begin to inject working fluid into that converter to increase its pressure. This one-shot injection essentially doubles the working fluid inventory and pressures in the operating converter. The disabled converter is shut down, and the operating converter now provides the full electrical power for the system. The system temperatures increase up to their design-point values as shown in Table IV-15.

3. Shutdown and Restart

The SR-100G System is designed for multiple shutdowns and restarts over the system life. There are two shutdown modes: normal (temperatures are gradually decreased) and rapid shutdown (used under special circumstances).

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Figure IV-49. Preliminary Operating Point Heat Balance and Statepoints
 (Both Converters Operating)

TABLE IV-15
DESIGN POINT VERSUS OPERATING POINT CHARACTERISTICS

	<u>DESIGN POINT</u> (ONE CONVERTER OPERATING)	<u>OPERATING POINT</u> (BOTH CONVERTERS OPERATING)
● SYSTEM		
■ NET POWER (kWe)	100	100
■ HOUSEKEEPING POWER (kWe)	9.0*	7.6
■ OVERALL EFFICIENCY (%)	19.7	19.3
● REACTOR		
■ POWER (kWt)	508	519
■ COOLANT OUTLET TEMPERATURE (K)	1473	1420
● POWER CONVERTER		
■ GROSS POWER (kWe)	109	107.6
■ TURBINE INLET TEMPERATURE (K)	1400	1357
■ COMPRESSOR INLET TEMPERATURE (K)	514	527
● RADIATOR		
■ HEAT REJECTED (kWt)	400	412
■ INLET TEMPERATURE (K)	801	798
■ OUTLET TEMPERATURE (K)	514	527

*INCLUDED 1.4 kWe MARGIN

In the normal shutdown mode, the reactor control segments are withdrawn slowly causing reactor power to decrease slowly. The turbine inlet temperature to the turboelectric converters decreases gradually. The user has been disconnected from the power system, and electrical load is being dissipated in the ballast load resistors. When the turbine inlet temperature drops below the self-sustaining temperature for the CRU, the CRU shaft speed begins to decrease. Based on the inertia of the system, the CRU will take about 1 h to completely spin down. Heat load during this period is being removed through the radiator.

The heat that the CRU is removing is the afterglow heat in the reactor. If the reactor has been operated for a long period, this afterglow heat can approach 6% of the full reactor power, or about 31 kWt. The afterglow heating from the reactor is decreasing with time in parallel with the CRU's shaft speed decreasing with time. Because a significant amount of heat can be rejected by the large system radiator through the Brayton working fluid, the reactor outlet lithium temperature decreases after the shutdown to an equilibrium value with the radiator.

As the shaft speed gets lower and lower, the heat that can be rejected decreases, but so does the decay heat load that must be rejected. The temperatures begin to increase again as the shaft speed approaches zero. When the CRU stops, heat rejection from the system radiator stops. This occurs approximately 1 h after the beginning of shutdown. In the worst-case situation, the maximum afterglow heating from the reactor at this point is 7.4 kWt. The afterglow heat removal areas on the lithium outlet pipes have been designed to reject this heat at 1500 K. If prior to shutdown the reactor had not been in operation for a long period, the afterglow heat will be less than this maximum value and the temperature will be correspondingly lower at the shutdown condition. As time from shutdown progresses, and the afterglow heating from the reactor decreases, the lithium temperature will decrease. The minimum allowed lithium temperature is 533 K (to maintain the lithium molten and uniform). When this temperature is reached, additional power is provided to the lithium pumps to maintain the lithium temperature.

The rapid shutdown sequence is similar to the normal shutdown sequence, except that the reactor control segments are rapidly withdrawn by a separate mechanism, thus effectively shutting down the reactor instantaneously. This causes harsher temperature transients in the system, but the system is designed for these transients. A preliminary estimate of the parameter changes that would occur during the rapid shutdown is shown in Figure IV-50. The discussion of temperature changes for the normal shutdown can be applied as well to the rapid shutdown shown in this figure. At the end of the sequence, the same terminal condition is achieved by either the normal or rapid shutdown sequence.

When it is necessary to start up the system, power is supplied from the on-board system startup batteries. These batteries are maintained in a charged condition during a period of dormancy. The restart sequence is similar to the initial startup sequence described in Section IV.B.1. The only exception is that the reactor can be at temperatures above the 533K minimum lithium molten temperature, depending on the amount of afterglow heating and the time after shutdown.

4. Safety

The SR-100G System has been configured to be a safe system. Safety was a key design concern for all phases of the system life, including ground handling, launch, operation, and safe end-of-life disposition.

With regard to ground handling, the system, including the reactor, will be nonradioactive, and the reactor will be configured to be subcritical in all potential ground handling accidents. It is envisioned that this subcriticality can be assured by installation of the launch subcriticality assembly, which places poison blocks around the periphery of the core. When the reflectors are not installed on the reactor, a complete poison sleeve will be installed in their place to assure subcriticality in the event of ground handling accidents. The chief concern with ground handling will be the temperature of the reactor and primary lithium piping which is maintained at 533 K.

*BASED ON PRIOR LONG-TERM REACTOR OPERATION

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Figure IV-50. System Rapid Shutdown Sequence

This can be accommodated by appropriate handling procedures. On installation in the shuttle orbiter bay, the reactor and high-temperature piping will be surrounded by the double-containment cover. This precludes accident events due to single-point leaks in the lithium-containing boundary.

Concerning the launch phase, three conditions must be considered. First is the containment of the liquid lithium. As discussed, double containment is provided by the double-containment cover. Second is structural integrity. The system structure, including lithium boundaries and working fluid pressure boundaries, was designed to MIL-STD-1522. This is the standard invoked by NASA Handbook NHB 1700.7A. Third is a launch abort accident; that condition is discussed below.

For a launch abort accident, the principal concern is a scenario in which the reactor remains intact and ends up in a hydrogenous fluid (water, for example) environment where criticality could occur, with subsequent destruction and dispersal of the reactor fuel. Other scenarios may result in damage or destruction of the core due to fire or explosion. However, at launch, the core is nonradioactive because it has not yet been operated, and these scenarios are of lesser concern than the hydrogenous fluid subcriticality requirement. As discussed in Section IV.A.2, the reactor has been designed to accept a launch subcriticality assembly that places neutronic poison blocks in the windows created by withdrawing the reflector control segments. These blocks are structurally tied to the reactor and the reflector. In any type of event that does not destroy the reactor, the launch subcriticality assembly maintains the reactor in a subcritical condition when the lithium has been displaced by water and the reactor is fully immersed in water.

Concerning the operational phase of the system, the SR-100G has been configured with redundant components for control and shutdown of the system. There are events, such as catastrophic failures of the reactor, that could irreversibly shut down the system. However, these events do not decrease the safety features of the system. On the other hand, certain of the events could actually enhance the final burnup of the system during reentry.

Concerning final disposition, the reactor and shield have been configured to enhance reactor burnup and fuel dispersal. Since the system would be operated in a safe orbit with at least a 300-yr orbital life, the major portion of system daughter products will have decayed before reentry. On reentry, the system must be passively designed to disassemble and burn up to disperse reactor fuels. Features such as burnable lips on the reactor and a strong structural tie between the reactor and shield (for aerodynamic stability) have been provided on the system. These are described in Section IV.A.2. Further design analysis and test verification must be performed to assure reactor burnup and fuel dispersal.

5. Reliability

The reliability effort was guided by a program reliability goal of 95% probability of success for 2 yr for the first flight unit, with eventual growth to 7 yr of full-power operation and 10 yr of system life. A brief qualitative reliability assessment and preliminary top-level reliability allocation for the most attractive design concept was made. This design concept uses a liquid-metal-cooled fast reactor and heat transport loop with two independent and redundant Brayton-cycle power conversion, processing, and heat rejection trains. Reliability was one factor that influenced the choice of this design concept.

The principal failure modes and effects for the SR-100G System include: (1) single-point failure modes, (2) double-point failure modes, and (3) slow system degradation failure modes. The key single-point failure modes are associated with the reactor and primary heat transport subsystems, most notably leakage from the subsystem boundaries. The key double-point failure modes are associated with the redundant power conversion, processing, and heat rejection trains; redundant reactor control equipment; and redundant primary loop EM pumps.

The SR-100G System consists of a lithium-cooled fast reactor and primary heat transport loop with two independent and redundant closed-cycle Brayton power conversion, processing, and heat rejection trains. The reactor utilizes cylindrical fuel pin geometry and has an outlet temperature of 1473 K. It is started and controlled by axially translating two (redundant) external reflectors with two (nonredundant) actuator motors. The shield comprises a tungsten alloy gamma shield and a canned, thermally insulated lithium hydride neutron shield. The primary heat transport loop transfers heat around the shield to two lithium-to-gas heat exchangers (one for each power conversion train). The primary heat transport loop has lithium pumped by two EM pumps arranged in series so as to be functionally redundant.

Two independent and redundant closed-cycle Brayton power conversion trains are provided. Each is designed to operate normally at reduced (50%) power but would be increased to full (100%) power if one unit fails. Operating them at low power normally provides design margin and should enhance reliability. The redundancy provided in these trains is maintained throughout the remainder of the power conversion, processing, and heat rejection subsystems.

The working fluid for the power conversion and heat rejection train is a He-Xe gas mixture that picks up its heat from the primary heat transport loop at the lithium-to-gas heat exchangers. Two redundant heat exchangers are provided, one for each power conversion and heat rejection train. The gas leaves this heat exchanger at a temperature of 1400 K and passes sequentially through a turbine, recuperator, radiator, compressor, and recuperator (again) before reentering the lithium-to-gas heat exchanger again. Each turbine and compressor are combined into a single rotating unit, along with a generator that provides power to redundant power processing equipment.

The heat rejection subsystem consists of a plate structure onto which is attached two independent and redundant sets of cooling channels, one for each power conversion and heat rejection train. The hot gas enters these cooling channels at a temperature of 801 K and exits at 514 K.

A battery power supply is included to maintain the reactor and primary heat transport lithium in a liquid state from launch to reactor startup, as well as to actually start up the system. Once startup has occurred, house-keeping power loads are provided from the system power output.

During launch, a double-containment boundary is required for liquid lithium because it reacts with nitrogen, which is the purge gas used in the space shuttle cargo bay. The double-containment cover provided is removed before startup.

Single-point failure modes are those that can cause the entire system to fail due to just one component failure. The most obvious failure mode in this category is leakage from the reactor vessel and primary heat transport loop boundary, including the lithium side of the lithium-to-gas heat exchangers. However, there may also be other single-point failures, such as failure to eject the double-containment cover or the launch subcriticality assembly, and sticking of the sliding reflector segments.

Because the entire power conversion, processing, and heat rejection subsystems are redundant, all failures in them are double-point failure modes. The principal failure modes for the power conversion, processing, and heat rejection subsystems are expected to be (1) gas leakage from the system boundary (from the gas side of the lithium-to-gas heat exchangers, combined rotating units, recuperators, radiators, or interconnecting piping and ductwork) and (2) failure of the combined rotating units due to causes such as bearing failure, metal fatigue, wear or creep, and generator failure.

The EM pumps in the primary heat transport system are another source of potential double-point failures since they are functionally redundant. The reactor control element actuator motors are also potential double-point failures if they fail before reactor startup. The battery power supply for these pumps and the startup equipment for the power conversion trains will be redundant, giving us two additional double-point failure modes.

Failures that are expected to result in slow degradation of performance and possible eventual total system failure include: (1) fuel pin cladding failures, (2) sticking of control reflectors (or failure of actuator devices) after initial full power operation has been achieved, (3) leaks in the lithium hydride neutron shield container or heating of this shield material above its melting temperature of 960 K, (4) internal leaks in the recuperators in the power conversion train(s), and (5) degradation in the performance of the radiator plate structure.

One of the key reliability-related decisions in the conceptual design studies was to select the level of redundancy to be provided in the SR-100G System. This is basically a tradeoff between the potential reliability advantages of redundancy versus the greater mass and complexity inherent in the use of redundancy.

Table IV-16 shows the effect of varying levels of redundancy on required subsystem or component reliability. Most of the credit for redundancy is realized with the first level of redundancy (factor of 10), and a diminishing rate of return is received for each successive level of redundancy. This, coupled with the mass constraint, led to the choice of dual redundancy for most of the SR-100G System, most notably in the power conversion, processing, and heat rejection trains.

The proposed reliability allocation for the SR-100G System is:

● Reactor, primary heat transport, and shielding subsystems	R = 0.98
● Power conversion, processing, and heat rejection subsystems	R = 0.98

This allows a margin (of almost 0.01) to the reliability goal of $R = 0.95$ for the entire system to account for unknowns and failures not attributable to any single subsystem.

TABLE IV-16
EFFECT OF VARYING LEVELS OF REDUNDANCY ON REQUIRED
SUBSYSTEM OR COMPONENT RELIABILITY

Number of Parallel Redundant Subsystems	Allowable Single Subsystem Unreliability ^a	Allowable Increase in Subsystem Unreliability
1 (nonredundant)	0.01	Base
2 (dual redundant)	0.1	X10 (most credit gained here)
3 (etc.)	0.22	X2.2
4	0.32	X1.5
6	0.40	X1.3

^aFor overall reliability allocation of 0.99

It is expected that the 0.02 unreliability allocation for the reactor, primary heat transport, and shielding subsystems will be largely consumed by the system boundaries, startup and control equipment, and EM pumps. Because this is primarily passive equipment, it is reasonable to expect that high reliability can be achieved, even though there are single failure points associated with it.

This 0.02 unreliability allocation is suballocated as follows: 0.01 to the reactor and primary system boundaries and 0.01 to other (redundant) components. To develop an estimate of whether these goals can be achieved, worldwide experience with sodium piping in ground-based applications was evaluated. The latest data available indicated 26 failures in 250,000 m-yr of operation. These data give a reliability estimate of 0.98 (or 0.02 unreliability) for the 100-m (equivalent) SR-100G primary loop operated for 2 yr. Although the data are not directly applicable to space nuclear power systems, they are indicative of potential reliability achievement. The data indicate

that the reliability allocation for the reactor and primary loop boundary is achievable but that considerable reliability growth will be needed.

To satisfy the 0.02 unreliability allocation for the power conversion, processing, and heat rejection subsystems, each redundant train has been sub-allocated an unreliability of 0.1. The joint unreliability of both trains due to independent failures would therefore be 0.01, leaving 0.01 available for potential common cause, dependent, or propagating failures.

Figure IV-51 shows the historical experience with several ground-based types of power generation and conversion equipment, as well as the required reliability for each SR-100G power conversion, processing, and heat rejection train if the reliability allocation of 0.90 is to be achieved. As with the sodium piping data above, the data are clearly not directly applicable to the current situation. Nevertheless, it does tend to suggest that the reliability allocation may be achievable with suitable reliability growth. Based on the foregoing discussion, it is judged that the SR-100G System is capable of achieving good reliability performance. However, actual achievement of the 0.95 reliability goal will require reliability growth.

6. Survivability

The SR-100G system has been configured to survive under multiple threat scenarios. The survivability aspects of the system are classified SECRET and are discussed in the classified addendum to this report (Ref. 10).

7. Dormancy

The SR-100G System has been configured to remain dormant in orbit. In the dormant condition, the lithium loop would be circulated and maintained molten by electric power provided to the EM lithium pumps. The system is capable of restart from the dormant condition. Details of the dormancy control and power requirements during dormancy are classified SECRET and are presented in the classified addendum to this report (Ref. 10).

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Figure IV-51. Reliability Experience and Needs for Power Generation and Conversion Equipment

C. KEY STUDIES

1. Design Point Selection

Selection of the overall system design requirements and baseline design points was predicated on a previous study (the Rockwell subsystem technology assessment report, Ref. 1) that indicated that a lithium-cooled fast reactor combined with a closed-cycle Brayton turboelectric power conversion system would represent the minimum risk mid-term technology combination for space-based nuclear power. The reactor operating temperature was then pushed to the maximum limit compatible with anticipated technological advances in both reactor and PCS materials. This evaluation indicated that the resulting system would be compatible with current shuttle space and mass limitations at a power level of 100 kWe net to the payload. These efforts resulted in an available turbine inlet temperature of approximately 1400 K with which to operate the Brayton cycle.

The incentive for achieving higher operating temperatures is the higher Carnot efficiency and higher system power-to-mass ratio generally associated with these temperatures. This incentive is, however, tempered by the limitations imposed by anticipated materials technology developments, which limit the levels that can be attained. A strong motivation for selecting a dynamic converter, as opposed to static schemes, is that the dynamic PCSs are anticipated to have considerably more growth potential than static converters.

The development of the SR-100G System design point was the result of trade studies performed at Rockwell ESG and at our subcontractor, Garrett Turbine Engine Company (GTEC). The Rockwell studies applied a simplified model of the Brayton and concentrated on the reactor, shield, heat transport, radiator, and power processing subsystems. GTEC applied simplified models of the other equipment elements and concentrated on the Brayton equipment. The system modeling philosophies of the two groups are illustrated in Figure IV-52.

ROCKWELL SYSTEM MODEL

GARRETT SYSTEM MODEL

83-S22-84-209

Figure IV-52. Design Point Selection Tools for the SR-100G System

In practice, the Rockwell model was used to approximate an initial system and to provide a detail characterization of the reactor, shield, heat transport, radiator, and power processing subsystems. These characterizations were then used by GTEC to develop a detailed optimization of the closed cycle gas turbine. The GTEC results were then used in the Rockwell model to further refine the reactor, shield, heat transport, radiator, and power processing subsystems.

For this study, the converter gas pressure level was set at 2240 kPa (325 psia), based on preliminary calculations that evaluated the system pressure over a range of values. The system pressure drop factor (β) was held constant at 0.96, based on previous experience. Detailed evaluation of the heat exchanger and duct masses may ultimately lead to lower β values. A rotor speed was selected and held constant at 45,000 rpm, based on having a reasonable tip speed and L/D for the direct drive, 110-kWe Rice generator.

To reduce the number of permutations for which the complete mass and radiator area computation was performed, a range of compressor pressure ratios (P_2/P_1) was first run for all combinations of the other variables. For each combination of variables, a pressure ratio was selected that would result in maximum system efficiency.

Performance, mass, and radiator areas were then calculated for all combinations of the remaining variables to be permuted:

E_r = recuperator effectiveness

T_1/T_6 = ratio of compressor inlet to turbine inlet temperature

The results were plotted to allow selecting the desired converter design points on the basis of minimum system mass.

An early design decision was to concentrate on the design of a system that featured two identical and completely autonomous Brayton converters. This was done to improve reliability by avoiding single-point failure modes. Several design-point cases were generated to determine the effect of using two redundant converters in the system. The results were generated as functions of specific radiator mass since the radiator design had not yet been fixed. Typical results are given in Figure IV-53. These curves represent minimum mass conditions in which the recuperator effectiveness, compressor inlet temperature, and system efficiency were all variables. The results shown on this figure were developed for the case in which a liquidcooled space radiator was used. These results are of particular interest since they indicate that it would be possible to incorporate both the fully redundant converters and a nondeployable radiator.

Subsequent to the studies leading to the results shown in Figure IV-53, various refinements in the design process were made. These refinements include such things as using a gas-cooled radiator and firming up on individual component masses. This effort resulted in the statepoints shown in Figure IV-54.

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Figure IV-53. Radiator Specific Mass Trade Study Results of the SR-100G System

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Figure IV-54. Design Point Heat Balance Statepoints (Only One Converter Operating)

2. Fuels and Structural Materials

a. Fuel Selection

The SR-100G reactor is designed for a 7-yr life at a power level of 525 kWt. The core contains an array of 121 fuel pins, each 1.9 cm OD, with an active fuel length of about 0.36 m. Lithium is used as the coolant at inlet and outlet temperatures of 1400 and 1473 K, respectively.

The high temperature, long life, and fast neutron irradiation environment makes the selection of fuels and materials critical. The performance of fuels and materials in this operating regime has received little attention over the past 10 yr. Hence, evaluations must be made based on technology developed in the 1960s and early 1970s.

Previous studies on high-temperature advanced reactor designs have concentrated on three basic uranium ceramic fuels: UO_2 , UC, and UN. These fuels possess the required temperature capability as well as the highest uranium density achievable for the compound system. Some of the basic properties of these fuel systems are compared in Table IV-17.

UO_2 has the lowest uranium density and thermal conductivity of the three fuels. Its greatest advantage is in fabrication, where both the technology and the economics are far better than for the other two fuels. The disadvantage of UO_2 for the current reactor design appears to be its incompatibility with lithium coolant. Should there be a breach in the cladding, the UO_2 and lithium would react exothermically to release large amounts of fuel and fission products throughout the primary coolant loop.¹¹

UC has an attractive uranium density and thermal conductivity. Its major drawbacks are irradiation instability, high fission gas release rates, and cladding incompatibility (particularly if excess carbon is present).

TABLE IV-17
FUEL SELECTION FOR REACTOR SUBSYSTEM

Property	UO ₂	UC	UN	Advantage
Uranium density (g/cm ³)	9.5	13.0	13.5	UN
Thermal conductivity (W/cm K)	0.03	0.17	0.25	UN
Compatibility with cladding	Good with Mo and W	Excess carbon reacts	Liner needed for Ta alloys	-
Compatibility with lithium	Incompatible reduced to free U	Good	Good	UC or UN
Fuel swelling at 1600 K $\Delta V/V$ per 10 ²⁰ fissions/cm ³	1.7	3.0	0.7	UN

UN offers the greatest potential for high-temperature conditions. It has the highest uranium density and thermal conductivity of the three fuels. It has good compatibility with high-temperature lithium and with most cladding materials. With tantalum alloy claddings, a tungsten liner is needed to prevent nitride formation.

Because of the attractive behavior of UN fuel in previous technology studies, major emphasis was placed on obtaining material property and behavior data for the UN fuel system. Key issues needed for design were fuel swelling, fission gas release, thermal properties, and compatibility. The literature was reviewed extensively. Additional information was obtained through phone contacts with BMI, ORNL, ANL, and PNL personnel.

The literature on irradiation testing shows that UO₂ has been tested far more than have UC and UN. A recent survey of worldwide experience in LMFBRs shows ~250,000 pins of UO₂ irradiated compared to ~1500 pins of UC and ~100 pins of UN. However, few UO₂ pin irradiations have been done for

high cladding temperatures (1073 to 1973 K), a regime where refractory alloy claddings are mandatory. A survey was made (Table IV-18) to compare the quantities of UO_2 and UN fuel pins that have received high-temperature irradiations and to identify the types of refractory alloys used as cladding. Only three cladding alloys have been used in testing fuel for advanced reactor applications. Also, the fuel pin diameters in all cases ranged from 0.5 to 0.76 cm, a factor of 2 to 3 smaller than the reference SR-100G design (1.9 cm). The UN data presented in Table IV-18 are complete, whereas the UO_2 survey should be considered tentative because the search of the large literature on this fuel has not yet been completed.

TABLE IV-18
REFRACTORY METAL CLADDING USED IN UN and UO_2 IRRADIATIONS
(ADVANCED FAST REACTOR APPLICATIONS)

Cladding Alloy	UN Fuel		UO_2 Fuel		References
	Pins Irradiated	Cladding Temperature (K)	Pins Irradiated	Cladding Temperature (K)	
NB-1Zr	65	1113-1473	3	1708	12, 13, 14
T-111	8	1233-1673	1	1263	15, 13, 14
W-26Re	<u>37</u>	1573-1953	<u>29</u>	1573-2218	15, 16, 17 16, 18, 17, 19, 20, 21
Total	110		33		

Irradiation testing of UN fuel was conducted in the 1960s by PWAC (CANEL), BMI, ORNL, UCRL, and LANL. In all, over 100 fuel pins were irradiated and evaluated for swelling in the temperature range of 1200 to 2000 K. In general, fuel performance seemed to improve in the later tests. Contributing to the amount of scatter observed in the data and to the difficulty in evaluating the swelling behavior was the fact that many of the tests had been designed to study the effects of pellet density, metallic additions, and centerline holes in the fuel pins.

For the present evaluation, only bulk UN having a pellet density of 90% TD was used. Fuel pins with centerline holes, mixed fuels, and cermetts were also excluded from consideration. This left about 40 UN fuel pins that could be evaluated on a common basis.

Figure IV-55 plots volumetric swelling of the fuel pins as a function of temperature. The line represents an exponential regression analysis of the data; the correlation coefficient is 0.85. The equation for the line is:

$$\frac{\Delta V}{V} = (1396) \exp \left[-1.095 \left(\frac{10,000}{T} \right) \right],$$

where $\Delta V/V$ is the percent swelling per 10^{20} fissions/cm³, and T is expressed in K.

The swelling curves commonly used²² to compare UO₂, UC, and UN are given in Figure IV-56. This clearly shows UN to be the superior to the other two fuels. Superimposed on this figure are the curves presented in Figure IV-55.

The amount of fission gas released from UN during irradiation is important in fuel element design. Accommodation and containment of the fission gas must be handled by gas plenum size, cladding thickness, and cladding alloy strength (assuming design temperature, power, and burnup are fixed).

Information from the literature shows a wide scatter of fission gas release data for UN. The quantity of gas released from a fuel pellet is dependent on such variables as temperature, pellet density, fluence, burnup, and grain size. Initial studies on UN fission gas release were conducted under the SNAP-50 program (CANEL). The results are summarized in Figure IV-57. These data show that for a core average design life burnup of 4×10^{20} fissions/cm³, about 12% fission gas release can be expected with a fuel centerline temperature of 1773 K. Later studies by BMI¹⁷ reported a 15% release

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Figure IV-55. Maximum Growth Correlation Used
 to Size Fuel-Clad Gap

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Figure IV-56. Dependence of Fuel Swelling
for UC, UN, and UO₂ (Ref. 22)

Figure IV-57. UN - UC Fission Gas Release (Taken from Ref. 11)

after a burnup of 1.14×10^{20} fissions/cm³ at 1973 K centerline. However, they also reported on other capsules where the measured gas release was only 5% following a burnup of 2.1×10^{20} fissions/cm³ at temperatures of 1673 K to 1973 K. The role of pellet density on fission gas release is not clear, but the data seem to suggest that fine-grained high-density pellets (>90%) retain more fission gas.

Since data on fission gas release of irradiated UN are sparse, more testing will be required to establish the gas release behavior. For the present design, a 20% fission gas release value is recommended.

In selecting a fuel system, the compatibility of the fuel with the cladding must be considered. UO₂ is compatible at temperatures up to 2143 K with most refractory alloys except Ta, Nb, and Zr base alloys.¹¹ Uranium carbide reacts with most refractory alloys to form a carbide layer at the cladding interface. Tungsten alloys appear to have the best compatibility with stoichiometric UC. However, irradiation upsets the stoichiometric balance, resulting in the availability of excess carbon. UN exhibits good compatibility with W- and Mo-base alloys up to 2143 K.¹¹ With Ta- and Nb-base alloys, nitrogen reacts at the cladding interface to form a nitride layer. This problem has been solved by applying a tungsten barrier between the fuel and cladding.¹¹

Fuel compatibility with high-temperature lithium must also be considered in the event that a breach in the cladding occurs. UO₂ and lithium react exothermically; under reactor operating conditions, a failed fuel pin would result in the dispersal of large amounts of fuel and fission products throughout the coolant system.¹¹ The reaction from a single cladding breach might also interact adversely with adjacent fuel pins. For these reasons, UO₂ is not considered a good choice for high-temperature lithium-cooled reactors.

Lithium does not react strongly with stoichiometric UC.¹¹ However, lithium has a high solubility for the carbon present in hyperstoichiometric UC. UN exhibits good compatibility with lithium under both isothermal and irradiation conditions.¹¹ The only effect detected in UN has been grain boundary attack and dissolution of UO₂ particles. By keeping the oxygen content low, the effect of lithium attack should be small.¹¹

A potential disadvantage of UN is its high vapor pressure relative to UO₂ and UC, as shown in Figure IV-58. In the case of a cladding breach, UN can decompose by giving off nitrogen and leaving free uranium in the UN structure. However, at the peak fuel temperature, the pressure of dissociation is small ($<10^{-9}$ atm at 1650 K peak fuel temperature).

b. Cladding Material Selection

Candidate materials for cladding applications for space nuclear reactors can be grouped into three classes of alloys: (1) stainless steels, (2) nickel-base alloys, and (3) refractory alloys. By far the greatest property data base exists on the austenitic stainless steels, although most of the data apply to the temperature regime of the LMFBR. The irradiation fluence levels expected for space applications should not seriously degrade the stainless steels. These alloys are temperature-limited by strength to about 1000 K, and liquid metal coolant compatibility will reduce this limit by about 200 K. Nickel-base alloys can operate at a temperature several hundred degrees above the limit for the stainless steels, but their mass transfer in liquid metals is greater. Compatibility with the proposed fuel systems needs further study. Nickel alloys tend to be more resistant to irradiation swelling, but because of their more complex microstructures, their fracture ductilities are generally lower. Chromium depletion at temperatures in excess of 1200 K in the space vacuum environment is also a concern.

Because of the high operating temperatures, refractory alloys offer the greatest potential for space reactor applications at temperatures from 1400 to 1700 K. Most of the available property data are for pure Mo, TZM, W-25 Re,

Figure IV-58. Vapor Pressure of Fuels

and molybdenum-rhenium alloys, with some data available on tantalum alloys (ASTAR-811C and T-111).

Computer literature search (e.g., Dialog) and personal contacts with pertinent organizations that performed the work were made on the refractory metals and alloys. Our search for data on the refractory metals and alloys emphasized: (1) high-temperature mechanical properties (e.g., creep at 1473 K), (2) compatibility with the operating environments (e.g., fuels, liquid metals, and impurities), (3) irradiation effects, (4) fabricability, and (5) availability.

The results are summarized in Table IV-19 for seven attractive candidate cladding materials. W-25 Re and ASTAR-811C exhibit the highest creep strength at the design temperature. The data base for the experimental alloy Mo-13 Re is limited and from short-term (~ 1000 h) tests,²³ whereas those for ASTAR-811C, T-111, and W-25 Re alloys are quite extensive.^{15,24-63}

TABLE IV-19
CLADDING SELECTION FOR REACTOR SUBSYSTEM

Alloy	Stress for 1% Creep at 1473 K (In High Vacuum)	1650 K Compatibility With ^a			Performance at Fluence (n/m ²)
		Li	UN	UO ₂	
W-25 Re	117 MPa in 10,000 h	S	S	S	10 ²⁴ at 1575 K, S
ASTAR-811C	104 MPa in 10,000 h	S	S ^c	S ^c	No data
Mo-13 Re	21 MPa in 10,000 h	S	S	S	10 ²⁶ at 1400 K ^b , expect S
T-111	34 MPa in 10,000 h	S	U	U	10 ²⁴ failed at 1573 K
Nb-1Zr	7 MPa in 10,000 h	S	U	U	5 x 10 ²⁵ at 990 K, S
Mo	7 MPa in 7 yr	S	S	S	10 ²⁶ at 1400 K, S
TZM	14-48 MPa in 7 yr	S	S	U	10 ²⁶ at 1400 K, S

^aS = satisfactory, U = unsatisfactory

^bBased Mo-50 Re

^cWith W liner

All the refractory candidate alloys are compatible with liquid lithium.^{24,64-66} However, only W-25 Re, Mo-13 Re, Mo, and TZM are compatible with the fuels.^{11,12,13,15} The tantalum (T-111, ASTAR-811C) and niobium alloys are incompatible with either the UN^{12,15} or UO₂¹¹ fuel at 1643 K. Accordingly, for ASTAR-811C to be used as cladding with UN fuel would require a tungsten liner between it and the fuel. Even under a W liner/ coating, cracks or cavities have been observed in tantalum alloy (T-111) cladding after irradiations with UN fuel; but W-25 Re has performed satisfactorily without any indications of cavities under similar irradiation conditions.¹⁵

The refractory alloys also show good radiation resistances with fluence up to 10^{22} n/cm² at 1400 K. However, no irradiation data are available for ASTAR-811C or Mo-13 Re, although both are expected to be satisfactory with the targeted fluence based on data with similar alloys (T-111, Mo-50 Re).^{15,23}

All the listed candidate refractory alloys except the experimental Mo-13 Re alloy are available and can be fabricated into the desired shapes.²⁵ However, the fabricability of Mo-13 Re is expected to be good since other Mo-Re alloys have good fabricability.²³ ASTAR-811C is readily weldable by the TIG process with good ductility after welding typical of tantalum alloys.^{25,26,28} W-25 Re alloy is weldable using standard high-temperature welding procedures by the standard Heliarc or Argonarc methods.¹⁵ Tubing of W-15 Re has been successfully fabricated by the floating mandrel extrusion and the CVD methods. Since all the refractory alloys oxidize rapidly when heated, the welding and fabrication process(es) must be done in a protective atmosphere with controlled low-O₂, H₂O, CO, and other gaseous contaminants.

The converter materials selected are shown in Table IV-20. The rationale is discussed below.

TABLE IV-20
MATERIALS SELECTION FOR POWER CONVERTER

<u>Hot Leg (>1000 K)</u>	
Heat exchanger	ASTAR-1611C
Piping	ASTAR-1611C
Turbine blades	TZM
Disc	TZM
Housing	C-103, ASTAR-1611C
<u>Cold Leg (<1000 K)</u>	
Piping	Inconel 617
Recuperator	Hastelloy-X
Compressor	
Impeller	17-4PH
Housing	Inconel 625

c. Converter Materials

The highest turbine inlet temperature (1400 K) was selected that is consistent with materials technology projected to be ready for component development by the late 1980s. Ducts and turbine housings in an uncooled spacecraft engine compartment will operate at or near the inlet gas temperature. The turbine rotor and blades will operate at a somewhat lower temperature, and the lithium loop for the lithium-to-gas heat exchanger will operate at a somewhat higher temperature. A turbine with 1400 K will require using advanced refractory metal alloys like the ASTAR series.

The molybdenum alloy TZM (Mo-0.5 Ti-0.1 Zr) has been creep tested. Eleven specimens were tested for temperatures ranging from 1145 K to 1440 K. The time totalled 94,140 h, and the longest single test was 45,354 h (5.17 yr). These data were correlated on the bases of (1) stress versus Larson-Miller parameter and (2) log of stress versus Larson-Miller parameter. However, the

small number of tests of TZM made inadvisable any division of data into two different sets; for each correlation, the entire set of 11 tests was used. Inasmuch as the two correlation coefficients are 0.92 and 0.87, respectively, this factor does not provide a strong incentive to choose one approach over the other. The linear correlation probably underestimates creep strength at the highest temperatures.

For the 2-sigma lines shown, the stresses given in Table IV-21 would produce $\leq 1\%$ creep for TZM in 40,000 h.⁶⁷

TABLE IV-21
STRESS TO PRODUCE 1% CREEP IN 40,000 h (TZM)

Temperature K	Allowed Stress	
	Linear Correlation MPa	Logarithmic Correlation MPa
1169	293	229
1200	249	170
1250	155	106
1300	69	66
1330	18	50

Thus, TZM is sufficiently strong for construction of turbine rotors and blades of Brayton converters at turbine-inlet temperatures up to ~ 1400 K.

ASTAR-811C is a promising potential alloy not only because it has good creep strength properties but also because it is weldable, has a low ductile-to-brittle transition temperature, and is compatible with the Brayton-cycle and reactor heat supply working fluids. Although not as well characterized as ASTAR-811C, ASTAR-1411C (Ta-14 W-1 Re-0.7 Hf-0.025 C) and ASTAR-1611C (Ta-16 W-1 Re-0.7 Hf-0.025 C) are stronger alloys within the same family that benefit from increased solution strengthening from their increased contents of tungsten. A comparison of ultimate strengths is given Table IV-22.⁶⁷

TABLE IV-22
COMPARATIVE STRENGTHS OF ASTAR ALLOYS

Alloy	Temperature (K)	Ultimate Tensile Strength (MPa)
ASTAR-811C	1365	390
ASTAR-1411C	1365	670
ASTAR-1611C	1365	810
ASTAR-811C	1535	290
ASTAR-1411C	1535	500
ASTAR-1611C	1535	630

Similarly, the stresses producing a 1% creep in 10,000 h are estimated in Table IV-23.⁶⁷

TABLE IV-23
COMPARATIVE CREEP STRENGTHS
OF ASTAR ALLOYS

Alloy	Temperature (K)	Creep Strength (MPa)
ASTAR-811C	1365	140
ASTAR-1411C	1365	220
ASTAR-1611C	1365	275
ASTAR-811C	1535	48
ASTAR-1411C	1535	130
ASTAR-1611C	1535	165

In comparison with ASTAR-811C, ASTAR-1611C has roughly three times the long-term creep strength at 1535 K. Alternatively, ASTAR-1611C could provide the same creep strength at 1535 K as ASTAR-811C provides at 1365 K. The balance of the converter system uses well-characterized alloys such as Inconel 617, Hastelloy-X, Inconel 625, and 17-4 PH.

d. Radiator Materials

The materials selected for the radiator are listed in Table IV-24. The reasons for the selections are noted in the table. The rationale is discussed below.

TABLE IV-24
RADIATOR MATERIALS SELECTED

● 300 series stainless steel D-tubes (fabricability)● D-tubes brazed to 300 series stainless steel-clad copper fins (thermal effectiveness)● Calcium titanate emissivity coating applied to stainless steel cladding (stability, bonding, emissivity, absorptivity)

The material for the radiator is generally dependent on the selected converter because of the temperature of the rejected heat. For the most acceptable radiator concept using an He-Xe gas mixture, the maximum heat rejection temperature is about 800 K. In addition to the temperature, the factors that must be considered when selecting materials for the radiator include heat transfer capability (e.g., thermal emissivity, solar absorptivity, thermal conductivity), strength-to-mass ratio, resistance to micrometeoroid impact damage, compatibility with the He-Xe gas, and fabricability. The importance of low mass cannot be overemphasized, since the radiator can become mass controlling for compact space power systems.

The materials considered were beryllium, copper, titanium, vanadium, Ni, TZM, stainless steel tubes with copper fins clad with stainless steel, and metal fiber matrix composites (MMC). Tailored to meet enhanced thermal properties and reduced mass, the MMCs considered were copper-graphite composite, copper-Al-B₄C composite, copper-ion-plated graphite tape composite, boron-Al

composites, boron-Ti-153 composite, graphite-aluminum composite, graphite-magnesium composite, graphite cloth-reinforced copper, graphite cloth-reinforced titanium, chemical vapor infiltration of SiC on braided ceramic cloth, and carbon-carbon composites with different resins. However, the Al-graphite, Al-boron, Mg-graphite, C-C fiber composites were eliminated from further consideration because of the current high-temperature requirements; and boron-Ti-153, graphite cloth-reinforced titanium, chemical vapor infiltration of SiC on braided ceramic cloth were not investigated further because of poor thermal properties and/or inadequate technology maturity.

Thermal conductivity of the radiator material is an important consideration. The ratios of thermal conductivity to density of several candidate materials are compared in Figure IV-59. Pyrolytic graphite has the highest conductivity relative to its density, with the copper, beryllium, and Ti alloys coming next in decreasing order. TZM, Ni, low-carbon steel (Fe-0.2 C), vanadium, and the titanium alloy Ti-8 Al-1 Mo-1 V are closely grouped together, and 304 stainless steel and Inconel 718 (IN-718) have the poorest ratio of thermal conductivity to density.⁶⁸

Because graphite and copper have high ratios of thermal conductivity to density, the use of metal-matrix composites for the fins of the radiator appears to be attractive. The thermal and physical properties of two candidate metal matrix composites are estimated in Table IV-25.

Because of the low coefficient of thermal expansion of the metal-matrix composites, titanium ($11 \times 10^{-6}/K$) appears to be the preferred tube material to minimize any excessive thermal stresses induced by thermal fluctuation during operation.

One alternative to copper composite is to use 304 stainless steel tubing with copper fins. Both sides of the copper fins would then be clad with

Figure IV-59. Thermal Conductivity/Density Ratios of Some Candidate Materials

TABLE IV-25
THERMAL AND PHYSICAL PROPERTIES OF TWO METAL MATRIX COMPONENTS

	Cu-40% C	Cu-8 Al-50% B ₄ C
Density (g/cm ³)	6.3	5
Thermal conductivity (W/cm-K)	> 2.61	> 1.95
Coefficient of thermal expansion (cm/cm-K x 10 ⁻⁶)	12.1	10.8

304 stainless steel for coating with calcium titanate for enhanced emissivity. This alternative materials combination is a well-established technology with good experience but at a penalty of higher mass. This concept was selected for the SR-100G System.

Because most metals have low thermal emittances, which is contrary to the requirements of a space radiator, high-emissivity coatings must be applied to the radiator. Another requirement for an earth-orbiting power system is that the radiator surfaces have a low solar absorptance, α_s . (Most bare metals have high values of α_s .) However, because the solar input is small compared with the energy output, low solar absorptance is not as important as high emissivity.⁶⁸ Criteria for selecting an emissivity coating include its thermal conductivity, specific heat, thermal diffusivity, and Z value (for survivability). A list of emittance coatings is shown in Table IV-26 with many inorganic coatings listed; these emittance coatings possess good emissivity, but most are not stable in the vacuum of space at the operating temperature of the radiator. SP-40MA-I, a ceramic coating with an emissivity of 0.85 developed by Solar Turbines International, has been applied to the beryllium radiators on SNAP-27 thermoelectric space power systems located on the moon, but it is limited to <700 K for continuous operation.⁶⁸

The work conducted to date indicates that calcium titanate and iron titanate are suitable materials for high-temperature space radiator applications.

TABLE IV-26
EMISSIONITY (ϵ) AND ABSORPTANCE (α) OF COATINGS

Coating	Temperature K	Emissionity, ϵ	Absorptance, α	α/ϵ
Z-93 ($\text{ZnO-K}_2\text{SiO}_3$)	422	-	-	0.15
	450	-	-	0.28
	533	-	-	0.32
Iron titanate	1005	0.89 to 0.87	-	-
Black chrome	673	-	0.97 0.08 (reflectance)	-
	573	0.05 to 1	0.94 to 0.96	-
	700	0.88	-	-
Chromate	-	0.06	0.90	-
Black coating	-	-	-	-
Calcium titanate	1005	0.91 to 0.88	-	-
Zinc orthotitanate	-	0.95	0.11	-
Zinc oxide	297	0.90	-	-
SP-40MA-I	700	0.85	-	-

Other coatings have been suggested for high-temperature use. The most promising of these use the CVD of tungsten and silicon carbide on molybdenum and tungsten alloys. These coatings, however, still require in-vacuum testing to verify their emissive and absorptive properties. Calcium titanate was selected for the SR-100G System.

In space, the radiator (and the rest of the spacecraft) could be struck by micrometeoroids. The response of material to high-velocity impact has been studied for many years but there is no reliable mathematical formulation with which to predict damage to a complex target by a hypervelocity fragment. Consequently, several empirical relationships have been proposed to approximate the micrometeoroid impact damage that might occur under limited conditions. But, because these empirical relationships are based on simulation experiments where impact particle velocities and densities differ significantly from those of the average micrometeoroid, the accuracy of the resultant

performance predictions is questionable. Light gas gun impact experiments using flat-pate material samples have provided most of the data from which predictions have been made; however, it has been shown experimentally for some materials that flat-pate impact data are slightly conservative when applied to tubular shapes.⁶⁹

e. Materials Compatibility

The refractory metal Brayton power converter contains high-pressure He-Xe gas as a working fluid, and it has been hypothesized that the moving gas could transport impurities around the loop and embrittle the refractory metal piping. To preclude this from happening, all possible sources of tramp gases have been excluded from the system.

The lithium-cooled reactor is not a source of tramp gases. On the other hand, a direct coupled, gas-cooled reactor could be a source. This is one reason that a liquid metal reactor was selected over a gas-cooled one for the proposed system.

Electrical materials in the alternator and pumps could also contribute to any tramp contamination. The approach is to have a hermetically sealed Brayton unit. In the proposed concept, these areas are isolated from the working fluid gas with ceramic or metal bore seals. These seals hermetically separate these tramp gas sources from the rest of the Brayton loop. This approach carries through to the gas supply and charging equipment, which must be of welded construction and use only bellows-sealed valves. The use of mechanical joints should be avoided in the auxiliary ground test equipment. Gas bearings were selected over oil-lubricated bearings to avoid complicated rotating seals and potential oil leakage into the Brayton loop.

The Brayton converter will be heated and evacuated to outgas the various surfaces before it is charged with the working fluid gas. The unit could be outgassed and sealed for an extended period of time before charging. The working fluid gas will contain less than 1 ppm oxygen, which can be gettered.

Oxygen is considered one of the worst of the tramp impurities that could cause embrittlement. All of the refractory metal alloys that would be used in the converter contain appreciable amounts of hafnium or zirconium. These two elements are effective getters of oxygen, and the alloys are annealed at 1365 K to 1590 K to assure that all oxygen in the alloys is in the stable HfO_2 and ZrO_2 forms. This condition gives the refractory alloys columbium and tantalum compability with the liquid metals. Any oxygen transport within the closed Brayton converter will therefore be dependent on the vapor pressure of HfO_2 or ZrO_2 . As shown in Figure IV-60, the vapor pressures of these two oxides are very low. No significant transport of material would occur for vapor pressures below 10^{-8} mm Hg.

Ground tests of the refractory metal converter will have to be conducted in ultra-high vacuums ($\sim 10^{-9}$ mm Hg). This test environment assures that external contamination of the refractory metals will be very low. As a backup to the above procedures, a special gettering unit containing pure zirconium at the optimum gettering temperature could be incorporated into the ground test or flight converter design at a small mass penalty.

The approaches used to prevent oxygen embrittlement of the refractory alloys are summarized below:

- Tramp Gas Sources Minimized

- Liquid-metal-cooled reactor
 - Electrical bore seals
 - Gas bearings

- Operating Procedures

- Outgassed evacuated system
 - <1 ppm O_2 in gas working fluid
 - Hermetically sealed gas supply equipment
 - Ultra-high-vacuum test environment

Figure IV-60. Vapor Pressure-Temperature Relationships for Refractory Metals and Oxides

- Metallurgy

Alloys heat treated to form HfO_2 -type oxides

- Special getter unit.

There is also concern about the use of superalloys in the converter in a vacuum environment. When the vapor pressure of a metal or other substance approaches 10^{-9} Torr, an appreciable amount of surface may be vaporized if exposed to a space vacuum. The vapor pressure of selected metals used in superalloys is shown in Figure IV-61. The rate of surface sublimation from one side of a sheet can be calculated from:

$$Y = 2.15 \times 10^5 (P/d) (M/T)^{1/2},$$

where

Y = loss in thickness (cm/1000 h)

d = density of solid material (g/cm^3)

P = vapor pressure (mm Hg)

M = molecular weight, vapor phase

T = temperature (K)

Surface sublimation for metals exposed to an ultra-high vacuum is plotted versus temperature in Figure IV-62. For this worst case, to assure that surface sublimation is held below 0.025 mm per 10,000 h, these metals should be used at temperatures below 1300 K when the bare surface is exposed to high vacuum.

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Figure IV-61. Vapor Pressure-Temperature Relationships for Superalloy Constituents

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Figure IV-62. Maximum Loss in Thickness for 10,000-h Exposure

3. Reactor Core and Reflector Configuration

Nuclear analysis performed to determine the size and configurational parameters of the core and the reflector are discussed in this section. The reasons for selecting sliding reflector control are also discussed.

The methodology used in the criticality calculations is based on solutions to the neutron transport equation using the APC computer code. This code performs a leakage iteration between two one-dimensional calculations to synthesize a two-dimensional solution. The code uses the ANISN discrete ordinate code to perform the one-dimensional transport theory solutions. Comparison of APC results for compact reactors with two-dimensional DOT calculations have shown sufficient accuracy for conceptual design parametric analysis. The cross-section library used in the analysis was the 42-group library described in Ref. 70. This library was processed through the IDX code to perform the necessary resonance self-shielding of the infinitely dilute cross sections using the core composition of the SR-100G reactor. No spatial self-shielding was done due to the lack of internal moderation. The 42-group format was retained for the APC calculations.

Many nuclear parametric analyses were performed in selecting the reactor configuration. Certain of the analyses used reactor dimensions that differ from the final dimensions of the reactor. However, the results of the analyses are applicable at the final reactor dimensions.

The cold, BOL excess reactivity of the reactor as a function of fuel element loading is shown in Figure IV-63. The reactivity is plotted as a function of total UN fuel mass. A scale showing the number of fuel elements is also provided. These data were calculated for a core height of 30 cm (the final core height was about 36 cm), a Be reflector thickness of 10 cm (the selected value), and a fuel-clad gap of 0.13 mm (the selected gap was 0.76 mm). The fuel mass in each fuel element is 1.03 kg. Even though a smooth curve is shown, there are unique numbers of fuel elements on a triangular lattice that

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Figure IV-63. Cold BOL Reactivity as a Function of Fuel Element Loading

best fit within a cylindrical vessel. Over the range shown, these unique numbers are 121, 139, and 151. The estimated required excess reactivity for 7-yr operation at the 525 kWt design point of the reactor is 5% $\Delta k/k$. A 121-fuel-element core is the minimum size, which meets the reactivity requirement.

Figure IV-64 shows the BOL reactivity as a function of core height. The calculations were made using 121 fuel elements, a reflector thickness of 10 cm, and a fuel-clad gap of 0.13 mm. The total fuel mass in the core for each core height is also shown in the figure. The results indicate that a core height of 30 cm is sufficient; however, further calculations relative to fuel swelling led to an increase in gap size (to 0.76 mm). To compensate for the increased gap, the core height was changed to ~36 cm.

The reactivity change due to a variation in reflector thickness at core mid-height is shown in Figure IV-65. The core size was held constant for this calculation and had 121 fuel elements with a 0.13-mm fuel-to-cladding gap in a 30-cm core height. The reactivity gain beyond a thickness of 10 cm is very small. This shows at what point the "neutronic" worth of the reflector diminishes and allows the minimum reflector thickness to be set. Because reflector thickness affects reactor diameter and therefore shield diameter and mass, this is an important quantity. The calculations were extended to zero reflector thickness to obtain the total reflector worth. This quantity is used in the calculation of control segment worth. For this size core, the reflector has a total reactivity worth of 17.4% $\Delta k/k$.

During the time the parametric analyses described above were being performed, a parallel effort was under way to evaluate the available data on UN fuel growth under irradiation. To evaluate the effect of a potential adjustment in the fuel-to-cladding gap, the reactivity of the core as a function of fuel fraction was calculated. The number of fuel elements was held constant at 121, as was the reflector thickness at 10 cm. The core height was increased to 36 cm and held constant at this value as the fuel fraction was decreased. The fuel element OD and cladding thickness were also kept constant

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Figure IV-64. Cold B0L Reactivity as a Function of Core Height

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Figure IV-65. Cold BOL Reactivity as a Function of Be Reflector Thickness

at 19 mm and 0.5 mm, respectively. The results, given in Figure IV-66, show the reactivity to be strongly dependent on the size of the fuel-to-cladding gap. The variation in total fuel mass is also shown. As points of reference, the core fuel fraction is 0.76 with a 0.13-mm gap and 0.57 with a 1.5-mm gap. The fuel masses at these points are 146 kg and 124 kg, respectively. Once the available UN fuel swelling data had been compiled, a fuel performance analysis indicated that a diametral gap of 0.76 mm would be required to assure no contact between fuel and cladding in the peak-fuel element at the end of 7 yr. The cold, BOL reactivity at this condition is 5.9% $\Delta k/k$.

Reactivity calculations were also made on the effects of variations in cladding and coolant volume fractions. For these analyses, the core size was held constant and only the atom densities of the cladding or coolant were varied. This is similar to the fuel fraction analysis in that the reductions in the material of interest are compensated for by void. The results indicated that the reactivity is very weakly dependent on cladding or coolant fraction over the ranges considered. The results also showed a slight positive reactivity effect for both cladding and coolant. This means that the scattering effect is more important than the poisoning effect of these materials in this high-leakage core.

Sliding reflector control was chosen for the system from the three potential types of reflector control that can be used in a compact space reactor. The three types of control are shown in Figure IV-67. In the first type, void-backed rotating drums are used, similar to those employed in the SNAP-10A and SNAP-8 reactors. As the drums swing out, they open windows for neutron leakage at the periphery of the reactor. However, as they swing out, they also increase the effective diameter at the top of the reactor. Because of neutron scatter, this top diameter is the one that must be used as the starting point for the system cone (i.e., the cone subtended to the base diameter of the power system). Since the neutron shield must have large enough diameter for the shield to extend to this system cone to attenuate scattered neutrons, the larger effective reactor diameter when using void-backed

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Figure IV-66. Cold BOL Reactivity as a Function of Fuel-Cladding Gap

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Figure IV-67. Sliding Segment Control Selected Because Reactor and Shield Are Smaller

rotating drums causes the shield diameter to be larger and thus the shield mass to be higher.

To reduce this effect, the second type of control can be considered: smaller control drums that have back-side poison to absorb leaking neutrons. Poisoned-back rotating drums can reduce the effect of diameter of the reactor, but one must be concerned with nuclear heating in the poison and removal of this heat from the reflector surface. In the past, these poison-backed drums have tended to be used only with manned systems. In these systems, the reactor was shielded on all sides (4π shielding), and the poison backing on the drum reduced the amount of neutrons leaking through the reflector into the 4π shield.

The third type of control uses sliding reflectors. This type of reflector can be relatively thin because only the portion of reflector material that really has worth "neutronically" must be provided. The reflector segments slide down, opening "windows" for neutron leakage. The reflector can be tapered from top to bottom to reduce the diameter at the top of the reactor, and this has little penalty on control worth. The principal advantage of sliding segment control is that it minimizes the diameter at the top of the reactor and allows the neutron shield to be smaller than with any other control configuration. This control configuration was used for the 5-kWe reactor thermoelectric system and is reported in Ref. 71. The sliding reflector control windows on the periphery of the reactor (when the control segments are in the down position) can be filled with poison blocks to provide water subcriticality.

4. Brayton Turboelectric Converter

The Brayton power converter was designed to provide 100 kW (EOL) net dc power to the user for 7 full-power years. Total system lifetime is 10 yr.

The converter and its combined rotating unit (CRU) described here are similar in many respects to those of three previous NASA and/or DOE programs. One of these, the 2-kW mini-Brayton rotating unit (BRU) in the 1.3-kW Brayton isotope power system (BIPS) was built and tested by GTEC for NASA and DOE in the 1970s. The BIPS system was the smaller of the two and designed for a 1144 K radioisotope heat source. It is nevertheless similar enough to provide valuable background for the study of larger systems. The BIPS ground test program successfully completed 1000 h of testing in 1978. Another CRU, the turboalternator-compressor (TAC) Brayton turbogenerator, was designed and components built and tested by GTEC for NASA in the early 1970s. This program provided design background for the current converter design study. Also in the 1970s and late 1960s, the original BRU CRUs were built by Garrett and extensively tested by NASA. Oil-lubricated ball bearings and gas-lubricated foil bearing versions were built and tested, but most of the test experience (41,000 h) was accumulated on the initial gas-lubricated pad-bearing version.

The final performance of the SR-100G Brayton converter was calculated by an optimization procedure that selects the system having minimum mass for a particular set of design parameters. The parameters assumed for the study are listed in Table IV-27.

The closed-cycle converter was optimized to define the best practical system with the maximum cycle efficiency, minimum system mass, minimum size radiator, or a desirable combination of the above. Cycle efficiency is readily defined; the remaining variables require extensive modeling of the major components. The parameteric design and optimization was conducted with the aid of a unique series of computer programs for design, off-design, and transient analyses. The accuracy of these computer programs has been proven through correlation with component and system test data. These programs provide the capability to explore design questions quickly and in detail.

TABLE IV-27
DESIGN PARAMETERS USED TO SET CONVERTER DESIGN POINT

<u>System</u>	
Number of converters in system	2
Number of converters operating	1
Output power – net to user (EOL)	100 kWe, dc
DC output voltage	TBD (100 to 400 Vdc)
Output power – generator terminals (BOL)	110 kWe, 1500 Hz
<u>Reactor Subsystem</u>	
Shield mass	424 kg
Heat source loop and coolant mass	289 kg
<u>Radiator Subsystem</u>	
Specific mass ^a	8.8 kg/m ²
Emissivity	0.90
Fin effectiveness	0.90
Sink temperature	200 K
Type	Gas-cooled
<u>Power Conversion Subsystem</u>	
Li-gas heat exchanger mass	191 kg
Turbine inlet temperature	1400 K
System maximum pressure	2241 kPa
System pressure drop parameter, β	0.96
Heat loss	Neglected
<u>Combined Rotating Unit (CRU)</u>	
Rotor speed, N	45,000 rpm
Bearing loss	3.0 kW
Generator rotor tip speed	335 m/s
Generator electrical efficiency	0.947
Generator windage loss	7.9 kW

^aEarlier studies used this parameter as a variable

The basic design program matches empirical turbine and compressor performance data to the thermodynamics of the Brayton cycle and defines the complete statepoints throughout the closed loop. Cycle modifiers that can be considered include compressor intercooling and turbine reheat on single-shaft or free-turbine configurations. Both radial and axial turbomachinery configurations are included. Component and duct pressure drops throughout the cycle are considered in the analysis as well as bearing losses, cooling flows, and the variation in working fluid transport properties with temperature. Various working fluids can also be investigated.

The second part of the design program includes design routines for the plate-fin pure counterflow recuperator, and both pure counterflow and cross-counterflow cooler and intercooler configurations employing finned tube or plate-fin construction. These heat exchanger design programs define the recuperator and cooler(s) using empirically determined heat transfer and pressure drop data for each selected heat exchanger core configuration. Heat exchangers are actually designed for each of the cycles defined by the cycle portion of the computer program. For the SR-100G System, coolers and intercoolers were not used; consequently, these subroutines were not necessary.

The last portion of the design program consists of a summary routine that compiles selected data from the first two parts, prints the summary information, and automatically plots the system mass and volume versus the cycle efficiency. A second version of the program includes a design program for a gas or liquid space radiator.

The design program is used in three operating modes. Simultaneous evaluation of the major system parameters can be made and the results plotted. These "shotgun" plots are generated by varying the major design parameters over wide ranges.

The result is the definition of the minimum system mass (or volume) line as a function of the cycle efficiency. It is common to evaluate hundreds or

even thousands of complete systems during the "shotgun" performance evaluations. Note that each point on the shotgun plot represents the major influences of a complete power converter design.

The second program operating mode is defined as the "single value parametric study." Several converter designs along the minimum line shown on the shotgun plots are selected for in-depth review. Each of the major parameters used to define the design is then systematically varied over a small range and the results plotted. This analysis is used to fine-tune the converter defined during the shotgun analysis. The final step is to obtain detailed computer output that precisely lists all of the temperature and pressure statepoints around the cycle, component sizes, efficiencies, masses, windage losses, heat transfer core geometries, surface finish requirements, and other data. From these data, layouts can be made to satisfy packaging constraints. The final result is the selection of an optimum design that meets all of the requirements.

Values of design parameters selected by the optimization process are as follows:

Recuperator effectiveness, E_r	0.767
Compressor inlet temperature, T_1	514K
Compressor pressure ratio, $r_c = P_2/P_1$	1.92

The selected cycle statepoints are defined on Figure IV-68. The following estimates of transient performance are also provided for preliminary use only:

Starting energy (per unit)	0.075 kW h
CRU start to load capability interval	20 s
CRU shutdown to stop interval	60 min

A breakout of estimated component mass is shown in Table IV-28.

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Figure IV-68. Design Point Heat Balance and Statepoints (Only One Converter Operating)

TABLE IV-28
MASS BREAKDOWN

Power Conversion Subsystem	Quantity	Mass (kg)
Li-gas heat exchanger (HSHX)	2	191
Combined rotating unit (CRU)	2	337
Recuperator	2	146
Gas ducts and insulation	2	150
Gas management equipment	2	25

The converter (Figure IV-69) consists of a centrifugal compressor, a four-pole 1500-Hz Rice generator, and a radial inflow turbine, all linked by a single shaft. The shaft is supported by gas-lubricated journal and thrust bearings.

The aerodynamic performance of the compressor and turbine is consistent with current GTEC production units adjusted for the specific speed, size, pressure ratio, and working fluid of the selected design point.

The journal bearings are of the foil type; these inherently have hydrodynamic start, steady-state, and stop capabilities. The foil thrust bearing is a double-acting configuration with the two thrust foils mounted face-to-face and separated by a gauged spacer. Both sides of the thrust bearing incorporate hydrodynamic start, steady-state, and stop capabilities. The converter design can readily be adapted to incorporate pad-type gas-lubricated bearings, if desired.

A labyrinth seal is located at the compressor and turbine end of the CRU to minimize leakage of the bearing cavity pressurization gas. Compressor discharge bleed gas is ducted to the thrust bearing and exits from the journal bearings to provide bearing cooling.

ROTOR SPEED (rpm)	45,000
COMPRESSOR ROTOR DIAMETER (cm)	15.6
COMPRESSOR BLADE TIP WIDTH (cm)	0.622
TURBINE ROTOR DIAMETER (cm)	17.1
GENERATOR ROTOR DIAMETER (cm)	13.7
GAS JOURNAL BEARING DIAMETER (cm)	5.84
GAS JOURNAL BEARING LENGTH (cm)	5.84
GAS THRUST BEARING DIAMETER (cm)	14.0
NUMBER OF POLES	4
FREQUENCY (Hz)	1500

MATERIALS

TURBINE	TZM
TURBINE HOUSING	ASTAR-1611C
COMPRESSOR	17-4 pH
COMPRESSOR HOUSING	INCONEL 625

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Figure IV-69. Design Characteristics for Brayton Combined Rotating Unit

The generator stator assembly, end bells, and heat exchanger are sandwiched between two frame assemblies to which all structural ties are made. The wound stator assembly contains no organic material and is canned to preclude containation of the working fluid. The CRU is designed to use the working fluid to cool the generator directly.

A four-pole Rice generator was selected for the SR-100G because, being a wound-field type, it permits control of generator voltage and because preliminary design characteristics for it are available from earlier work done for NASA on the BRU and BIPS turbogenerator programs. The major design characteristics of the Rice generator selected for the SR-100G are given in Table IV-29.

TABLE IV-29
SR-100G GENERATOR DESIGN CHARACTERISTICS

Parameters	Design Selection
Type	Rice
Design power (kWe)	110
Speed (rpm)	45,000
Number of poles	4
Frequency (Hz)	1500
Rotor diameter (cm)	13.7
Efficiency (%)	95
Voltage control	Active field
Maximum field power (kWe)	4
Cooling	Full compressor discharge
Cooling temperature (K)	695
Startup	Starter winding

The recuperator is a compact, pure-counterflow, plate-fin heat exchanger of all brazed and welded construction. It is fabricted of Hastalloy X. The design is shown in Figure IV-70. The internal construction is typical of that

Figure IV-70. Recuperator Concept for Brayton Turboelectric Converter

used on counterflow heat exchangers. Heat transfer takes place predominately in the rectangular counterflow section, where high-performance rectangular offset fins are used. Triangular end sections are used to distribute and collect gas flow to and from the counterflow section. End sections incorporate plain rectangular fins to distribute gas flow uniformly with minimum pressure drop. In the pure-counterflow section of the core, fins in the hot and cold passages are 8 fins/cm, rectangular, 2.5 mm high, 3.2 mm offset, and 0.1 mm thick. In transition sections, plain rectangular fins (6 fins/cm) are used.

The major differences between the current 100-kWe recuperator and the Mini-Brayton recuperator are in temperature effectiveness (0.767 versus 0.975) and the associated difference in the length of the pure-counterflow section. Temperature effectiveness derives from the mass minimization of the optimization process rather than from physical limitations.

The Li-gas heat exchanger consists of a multipass finned-tubed cross-counterflow heat exchanger arranged and connected in the unique manner shown in Figure IV-71. This arrangement was used to meet two requirements: total allowable lithium side pressure drop of 4.8 kPa and a total separation of the two working fluid loops. Any gas leak must be to the vacuum of space so that no single gas leak can cause both engines to be shut down.

The net thermal heat transfer capacity of either shell container is 501 kW, which is adequate for operation of a single converter at its maximum rated power. The shell material, ASTAR-1611C, was selected for its high-strength properties at elevated temperatures. The resulting heat exchanger mass is 191 kg (both units).

The selected Brayton converter concept utilizes existing technology except for components subjected to high turbine inlet temperature and high-temperature generator coolants. The system concept is thus predicated on the timely continuation of developments in these high-temperature technology areas.

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Figure IV-71. Li-to-Gas Heat Exchanger for the SR-100G System

All technologies other than these two are considered to be sufficiently established to permit starting full-scale component development at any time. These include:

- Aerodynamic components

High performance radial and axial designs are state of the art.

Radial configurations were selected.
- Gas film bearings

Foil and solid geometry pad gas bearing designs have recently been developed.

Foil configurations are selected.
- Brushless high-speed generators

Rice, Lundell, Lorentz, and permanent magnet generator designs are state of the art.

A Rice configuration was selected.
- Inert gas mixtures

He-Xe was selected.

5. Alternate Free Piston Stirling Converter

The SR-100G is a preconceptual design for a space-based nuclear power system capable of producing 100 kWe using a lithium-cooled reactor with a Brayton turboelectric converter. Constraints on the size and mass of the system heavily influenced the design decision to develop a high-temperature system. This design features a turbine inlet temperatures of 1400 K. Concern over potential problems with operating at such temperatures prompted the selection of a Stirling engine as an alternate power converter. But the technology of Brayton converters is better developed than that of Stirling engines and hence the Brayton remains the preferred option for the power converter.

Using information developed by Rockwell on Stirling engines for space power applications, approximate heat input and heat rejection temperatures for a system of minimum mass were determined. Technology assessments and projections indicated that a heat input temperature of 1050 K was achievable. This suggested that, for this heat input temperature, a system of minimum mass would reject waste heat at a temperature of about 517 K. The mass of the Stirling engine in such a system was expected to be under 930 kg. Mechanical Technology Incorporated (MTI) was contracted to conduct preconceptual design studies of a Stirling engine using these temperatures and this mass limit.⁷²

The system configured for the fallback converter uses an all-aluminum radiator with Dowtherm-A coolant in a tube-and-fin design. The Stirling engine head temperature was limited to a value of 1050 K to permit the use of superalloy materials in the design of the reactor and primary heat transport subsystems. The reactor coolant is sodium, and its peak temperature is limited to 1085 K. The two primary sodium pumps are linear EM pumps (dc) that are cooled with Dowtherm-A coolant from the primary radiator. The reactor produces 360 kWt. The sodium flow rate in the primary heat transport system is estimated to be 3.7 kg/s. The engine operation at 32% net efficiency provides 110-kWe gross electrical output at its design point. Approximately 8.5 kWe of this power is used to power the system's primary pumps and radiator pumps and to account for transmission line and electrical power processing losses. The user is supplied with a minimum of 100 kWe at 400 Vdc during steady-state operation.

The design point for the radiator was based on a maximum area of 110 m². This value was estimated to be the maximum radiator area available from a single deployment radiator design.^a Figure IV-72 shows the variation of total system mass and radiator area with average heat rejection temperature. The design point selected was at an average heat rejection temperature of 506 K. The radiator consists of 98 m² operating at an average heat

^aSelection of a higher head temperature could result in using a nondeployable radiator.

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Figure IV-72. Fallback Converter Preliminary Design Point

rejection temperature of 506 K plus 12 m^2 at a 365 K operating temperature. The lower temperature section is for a power conditioning radiator to accommodate the waste heat generated in the tuned rectifier circuits, which convert the ac output of the linear alternators to 400-Vdc user power. For the primary radiator, the total heat rejected is 250 kWt. A 70 K temperature drop across the radiator was required to limit the coolant pressure drop to less than 620 kPa. This design provided an average Stirling engine cooler wall temperature of 522 K, which yields an overall net efficiency near 32%. A schematic indicating the fallback system's key design points is provided in Figure IV-73.

A partial mass breakdown by subsystems and components of the Stirling system is presented in Table IV-30.

MTI proposed a Stirling engine with eight cylinders in a Rinia configuration. In this configuration there is one piston per cylinder, and each piston acts both as a displacer to transfer the working gas to and from the next cylinder and as a power piston. The hot space of each cylinder is connected through a heater, regenerator, and cooler, in that order, to the cold space of the next cylinder, and the cylinders are arranged in a circle. MTI arranged adjacent cylinders in opposite directions. A schematic of the arrangement is given in Figure IV-74. Each piston drives a linear alternator with four alternators on one end of the engine and four on the other. The heater sections are placed adjacent to one another in the middle of the engine to minimize the volume of liquid metal in the heater. The cooler sections are in two groups of four, one group on each side of the engine. The overall arrangement is as shown in Figure IV-75. The engine, which operates at a pressure of 15 MPa, is enclosed in a pressure vessel; most of the components therefore see no pressure differential across their boundaries, and stresses are minimized. The pressure vessel is expected to remain at a temperature of 525 to 600 K.

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Figure IV-73. Fallback Stirling Converter Statepoint Diagram

TABLE IV-30
MASS BREAKDOWN OF THE STIRLING SYSTEM
USING A STIRLING ENGINE

	Mass (kg)
Reactor subsystem	410
Shielding subsystem	245
Primary heat transport subsystem	196
Power conversion subsystem (Stirling engine):	
Engines	236
Alternators	286
Mounting plates	73
Tuning capacitor	45
Contained sodium	9
Miscellaneous tubes and fittings	32
Pressure vessel	<u>90</u>
Subtotal	771
Radiator heat rejection subsystem	640
Radiator deployment mechanism	192
Power processing and control subsystem	246
Miscellaneous	<u>180</u>
Total	2,880

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Figure IV-74. Stirling Engine Schematic of the "Daisy Chain" Cylinder Arrangement

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Figure IV-75. MTI's Perspective of 100-kWe Stirling Engine Concept

The engine is constructed as follows. The eight hot-cylinder assemblies are welded between two base plates forming an octagonal array. The cold power piston cylinders are bolted to the base plates, and the alternator stators are bolted to the cold cylinders. The assembly is then mounted within a pressure vessel, which is sealed by welding and reinforced by filament winding.

Each hot cylinder assembly is composed of an expansion cylinder, heater section, regenerator section, and cooler. Internal components are brazed into subassemblies, and these modules are joined by electron-beam welding. Heater and cooler both incorporate bundles of straight tubes that extend through a helical hot or cold liquid heat transfer path. The layout is such that each end of a cylinder assembly is isolated from the hot zone and thus can be welded to a support plate. The particular shape of the heater was chosen to even out head temperature and minimize the internal liquid volumes required. The modules can be individually tested as subassemblies to assure leak-tight construction.

The two support plates are identical and provide the load paths that allow forces from the various reciprocating masses to be cancelled internally. The support plates also provide the mounting points to attach the individual power piston cylinder and alternator assemblies and means to attach the pressure vessel to the engine assembly. The hot sections expand and contract with variations in hot-side temperature, making it possible to rigidly mount only one base plate to the pressure vessel while constraining the other radially. The two halves of the pressure vessel are joined and sealed by welding. The vessel is reinforced as required by filament winding.

The engine operates with a heat input temperature of 1050 K and a heat rejection temperature of 522 K. The operating pressure is 15 MPa. Hydrogen is the working fluid, but helium could be substituted with little degradation in performance and a slight increase in the engine mass.

The system behaves as a thermally driven free oscillator. At full power, the pistons oscillate at a frequency of 120 Hz, with a phase difference of 90° from one piston to the next. Together, the linear alternators can put out 110 kWe at 400 Vdc. At all power levels, the engine is in complete dynamic balance. The estimated overall efficiency of this power converter, using hydrogen as the working fluid, is 32%. A breakdown of this efficiency is given in Table IV-31.

TABLE IV-31
EFFICIENCY BREAKDOWN FOR MTI's
110-kWe STIRLING ENGINE CONCEPT

Estimated overall efficiency (hydrogen)	0.32 ^a
Efficiency breakdown	
Carnot	0.50
Device	0.80
Mechanical ^b	0.90
Alternator ^b	0.90
Rectifier	0.98

^aDC electric out/heat in

^bIncludes mechanical cooling power

The eight-cylinder Rinia configuration of the engine has the characteristic that the dynamic forces and force couples acting within the engine cancel. The proposed engine design is nominally balanced over the entire range of operation. For identical cycle-to-cycle dynamics, thermodynamics, and electrical parameters, the dynamic elements are balanced, and the engine has zero net vibration. Some variation in cycle parameters is to be expected, and some degree of dynamic imbalance and system vibration will be experienced. This vibration results in a degree of force transmission to the space power system. The magnitude of the force transmission depends on the magnitude of vibration and the stiffness of the engine mounts. This vibration force level can be reduced by several methods:

- Improving the cycle-to-cycle parameter match
- Reducing the engine-to-ground stiffness
- Dynamically structuring the heating and cooling piping to cancel the forces to ground at the design frequency
- Incorporating an active vibration absorber within the engine structure.

Active force cancellation has the advantage of eliminating fundamental vibrations without significant mass or efficiency penalties. The mass of an active force cancellation system will not be known until its configuration and performance have been specified; however, the mass is expected to be small (on the order of several kilograms).

The electrical equipment comprises two major subsystems: the alternator and the rectifier circuit. A linear alternator design was chosen for its compatibility with the engine design and its simplicity in providing direct mechanical coupling between the alternator and the power pistons in the engine. In a linear alternator machine, a sinusoidally varying voltage is induced in a stator coil by the field associated with an oscillating plunger.

The alternator in the proposed design is a moving field machine that uses plunger-mounted samarium cobalt magnets to generate the plunger flux. This design is used because it has both high specific power and low inductance. The alternators used in the present design have not been optimized from a system point of view: rather, they have been designed at what appears to be a reasonable design point (92% electrical efficiency at 2-3 kg/kWe). The breakdown of losses in the machines is as shown in Table IV-32.

The linear alternators are connected to the dc load through a simple tuned bridge rectifier circuit. The electrical characteristics of this circuit and the electromechanical parameters of its components were chosen to provide stability and controllability of the system by determining the shape of the load line seen by the engine. The tuned rectifier circuit provides a

TABLE IV-32
ALTERNATOR LOSSES

I^2R loss in coil ^a	5.5%
Iron loss (eddy current)	1.5%
Mechanical loss (windage and friction)	Negligible
Cooling parasitic (fan power)	1.0%
Total loss	8.0%
Alternator efficiency	92%

^aThis loss includes the effect of harmonic distortions of the current waveform by the rectifier circuit.

wide design margin for selecting the alternator inductance value in the design point configuration selected for this design. The parameters for the alternator/rectifier design used in this engine concept are given in Table IV-33.

The proposed system is life-limited by creep in the hot sections. All reciprocating components are supported on gas bearings, and all seals are clearance seals with no material contact under normal operation. Creep affects the regenerator and heater tube wall structure and indirectly affects engine efficiency. Heater tube wall thickness affects the temperature drop across the tube wall and the effective Carnot temperature in the expansion space. Regenerator wall thickness affects the magnitude of the direct conduction from heater to cooler through this wall. The final choice of hot-side materials depends on temperature, weldability and brazeability, hydrogen compatibility, and hot-sodium capability (for the heater tubes). For the present design, the pressure vessel and the expansion and regenerator cylinders will be constructed of stainless steel, and the heater tubes will be designed with Udimet 500 or 700. Udimet is a nickel-based superalloy with excellent creep properties at the cylinder head temperature for the engine concept.

TABLE IV-33
TUNED RECTIFIER DESIGN POINT OPERATION

Alternator stroke	2.032 cm
Inductance	0.015 H
Resistance	0.50 Ω
Capacitance	116 μ F
Generator Voltage	539 V
Average current	34.4 A
RMS current	39.2 A
Coil loss	768 W
Battery voltage	400 V
Battery power	13,750 W
Efficiency	0.947
Capacitor voltage	645 V
Current duty cycle	1.0
Current Harmonics	
1st 55.1 A	-10.5°
3rd 5.3 A	92.8°
5th 1.8 A	93.6°
7th 0.9 A	94.6°

6. Radiator Configuration Selection

Two key trade studies were performed to assist in the selection of a radiator concept for the SR-100G system:

- Fixed versus deployed radiator
- Radiator coolant (gas, liquid, or heat pipe).

a. Fixed Versus Deployed Radiator

The envelope constraints imposed by the program design goals and requirements limit the size of fixed radiators. Figure IV-76 shows the fixed

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Figure IV-76. Variation of Radiator Area
with Shielded Cone Half Angle

radiator area as a function of core half-angle. Fixed radiator area is also available as a disc on the bottom side of a conical radiator. As the core half-angle is increased, more fixed radiator area can be obtained. There is a mass penalty associated with increasing core half-angle since the shield size increases with increasing half-angle. This effect is shown in Figure IV-77.

If a radiator area in excess of $\sim 64 \text{ m}^2$ is required, a deployable radiator must be used to meet the system envelope constraint. We chose to use a "folding panel" type deployable radiator (Figure IV-78) instead of a nested cylinder arrangement because of the simpler deployment mechanism, which can be used for the panel concept. The deployment mechanism mass was estimated to be slightly in excess of 190 kg (Table IV-34).

The mass penalty associated with increasing core half-angle for the fixed radiator is somewhat offset, therefore, by the mass of the deployment mechanism required for the deployable radiator.

The fixed-versus-deployed radiator study showed that system mass trade-offs need to be performed to determine the optimum radiator configuration. For the SP-100G System, it was found that by using a gas-cooled radiator (eliminating the cooler), performance could be improved such that a fixed radiator was the minimum mass system.

b. Radiator Coolant

In parallel with the above trade study, a second study addressed the selection of the radiator coolant. Coolants considered in the study were:

- NaK-78
- Dowtherm A
- Brayton converter working fluid (i.e., He-Xe).

Figure IV-77. Shield Mass versus Cone Half-Angle

(a) NUCLEAR POWER SYSTEM

(b) CUBIC EQUIPMENT SPACE
AVAILABILITY

9467-261

Figure IV-78. "Folding Panel"
Deployable Radiator

TABLE IV-34
RADIATOR DEPLOYMENT MECHANISM
MASS SUMMARY^a

Item	Mass (kg)/ Number	Subtotal
Extra manifolds	8.2/8	65.5
Elbow piping	1.1/4	4.4
Bellows	10/8	80
Bellows guide tube	1.7/8	13.6
Bellows rotational guide	0.9/8	7.2
Rotational manifold supports	0.9/8	7.2
Manifold interconnect	0.7/8	5.6
Rotational drive gear	0.6/4	2.4
Worm gear and drive motor	1.1/4	4.4
Miscellaneous hardware	0.5/4	<u>2.0</u>
TOTAL		192.3

^aBased on using stainless steel materials

Performance studies indicated that both NaK-78 and Dowtherm A would require a deployable radiator. The gas coolant (He-Xe) could use a fixed radiator.

Three radiator concepts (one for each coolant) were analyzed separately to determine the minimum area radiator that was compatible with the Brayton converter conditions. For the two liquid systems, this required sizing the converter cooler as well as the radiator itself. Since it was necessary to minimize the temperature differences in the cooler between the converter gas and the radiator fluid, the Reynolds number of the radiator fluid tended to approach laminar conditions. For the NaK fluid analyses, this was not a problem; but for the Dowtherm A fluid analyses, laminar flow was found to limit the allowable conditions in the cooler and radiator. This, then, required a larger radiator fluid flow rate and decreased the effective radiator temperature by over 100 K. This led to a radiator mass two and one-half times as great as the gas-cooled radiator concept.

For the trade study, the following conditions were selected for making a mass comparison:

- Heat rejected = 396 kWt
- Recuperator outlet temperature = 801 K
- Compressor inlet temperature = 514 K
- Gas flow rate = 2.6 kg/s

Masses included the cooler (for the organic and NaK radiators), fluid inventory, tubes, fins, armor, emissivity coating, structure, and deployment mechanism. The results of the study indicated the gas-cooled radiator to be the lightest. The organic-cooled radiator was 2.5 times as heavy, and the NaK-cooled radiator was 1.35 times as heavy. Based on this study, a gas-cooled fixed radiator was selected.

During the trade study, we also considered a heat-pipe radiator concept. A heat-pipe radiator concept would be much less susceptible to fatal damage from micrometeorites than any of the pumped fluid concepts considered in this study. However, there are two major thermodynamic drawbacks to a heat-pipe radiator concept for a Brayton-cycle converter.

The first drawback is caused by the heat pipe use of vapor transport of energy. This requires the heat pipe to operate at a temperature that gives a reasonable vapor density. The fluid thermodynamic properties therefore require that several different working fluids be used in the radiator because of the temperature range of the coolant (rubidium, mercury, and Dowtherm A were considered in the previous study).

The second drawback to the use of a heat pipe radiator is the small heat transfer area of the heat pipe evaporator. This small area will increase the fluid-to-wall heat flux by an order of magnitude. This either requires a much larger fluid velocity (and pressure loss) or results in a much larger temperature drop between the fluid and radiating surface.

After a preliminary look at heat pipes for the deployable portion of the radiators (which would allow a single working fluid to be used), the concept was discarded in favor of conventional transport of energy by a pumped fluid.

c. Description of Selected Concept

The SR-100G radiator concept is shown in Figure IV-79 as part of the complete power system. The radiator has 61 m^2 of area and consists of dual D-tubes attached to a stainless-steel-clad copper fin. The tube-fin concept is shown in Figure IV-80, while the radiator manifolding arrangement for the two Brayton systems is shown in Figure IV-81. The radiator makes use of 10 m^2 of area in the form of a disc that covers the large open end. Up to 13.8 m^2 of area would be available if needed for design margin.

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Figure IV-79. Cutaway View of SR-100G System

Figure IV-80. Radiator Dual Tube-Fin Configuration of the SR-100G System

Figure IV-81. Radiator-Manifold Arrangement

83-S17-84-104

The equation used to determine the amount of armor required to protect the SR-100G radiator tubing from penetration by micrometeorites was:⁸

$$\delta = \gamma_a \left(\frac{6}{\pi \rho_p} \right)^{1/3} \left(\frac{\rho_p}{\rho_a} \right)^{1/2} \left(\frac{V_p}{\sqrt{\frac{E_a g}{\rho_a}}} \right)^{2/3} \left(\frac{\alpha A_v \tau}{-\ln P(0)} \right)^{1/3\beta} \left(\frac{1}{\beta + 1} \right)^{1/3\beta}$$

This equation was used in conjunction with the following meteorite flux coefficients for the near-earth environment:⁹

$$\text{LOG}_{10} N_t = -14.37 = 1.213 \text{ LOG}_{10} m$$

The analyses of the armor requirement indicated that the outer tube thickness required was 0.3 cm for a 0.99 probability of nonpuncture during the 10-yr mission. The radiator thermal design was optimized based on a tube spacing to utilize the 61 m² area available. The design is such that each Brayton converter uses half of the outer tubes and half of the interior tubes. This gives the same nonpuncture probability for each of the two converter trains.

A review of the available literature for secondary damage effects from micrometeorites led to the conclusion that the nonpuncture probability for the interior tubes was probably above 0.99 for meteorites penetrating the fins. This could not be fully quantified because the data are sparse. In a like manner, the nonpuncture probability for meteorites entering through the 2.4-m-diameter opening in the back of the SR-100G System is also in excess of 0.99. These are both worst-case estimates. The overall nonpuncture probability for each of the independent radiator gas flow paths will be somewhat below 0.99 because of these effects.

The analysis resulted in the mass summary in Table IV-35.

TABLE IV-35
MASS SUMMARY

Item	Mass (kg)
Fins	248
D-tubes	114
D-tube armor	301
Emissivity coating	10
Radiator structure	86
Manifolds	38
Payload thermal shield	15
Power processing radiator	<u>22</u>
Total	834

Although they were not utilized in the analysis, there are at least two options that could reduce this mass:

- Accepting an 0.98 nonpuncture probability for the exterior tubes would reduce the mass increase from 89 kg to 19 kg.
- Increasing the Brayton converter operating pressure should also reduce the mass.

V. TECHNOLOGY DEMONSTRATION PLAN FOR THE MOST ATTRACTIVE SYSTEM

This section presents a proposed technology demonstration plan (or, more descriptively, a technology risk reduction plan) for the system. The plan addresses the technology issues and concerns that we currently foresee for the system.

A space reactor power system generic development plan (from Ref. 1) is shown in Figure V-1. The approach is a step-by-step process involving scoping development tests followed by ground subsystem tests, a ground system test, and a culminating flight system test. The individual test phases overlap, with data from the preceding test phase becoming available just before the subsequent test phase is started.

A variation on this generic approach is to proceed more directly to the ground demonstration system (GDS) test, conducting only those feature tests or subsystem tests judged critical to proceeding with the GDS test. For example, a reactor subsystem test or power conversion subsystem test could be deleted if it were decided that sufficient data existed to proceed directly to the GDS test without the intermediate test. Rockwell used this approach in the SNAP-10A program, foregoing both the reactor and power conversion subsystem tests in favor of a direct GDS test. That approach was selected based on the favorable results from the prior SNAP-2 reactor subsystem test and individual thermoelectric module tests.

The proposed technology demonstration plan for the SR-100G System is based on this variant of the generic development program. The plan has a parallel structure: tests to resolve the currently foreseen technology concerns are conducted in parallel with the design and fabrication of the GDS test. The results of the tests are fed into the design and fabrication activities, forming "decision gates." These "decision gates" provide the program with the opportunity to assess the test results and either continue or

Figure V-1. Generic Development Schedule

redirect (if required) the design and fabrication activities accordingly. The plan is a dynamic balance between design/fabrication of the GDS and the test data which is constantly being iterated against the design/fabrication progress to provide maximum assurance of success for the GDS test.

We have selected this parallel approach over a "series" approach. The series approach involves conducting tests to resolve the currently foreseen technology concerns before finalizing the design and proceeding with fabrication. Although the series approach would minimize risk, it is neither schedule nor cost effective.

We believe the system as we have configured it represents one with acceptable risks – risks commensurate with adopting a parallel approach technology demonstration plan. In our judgment, the cost and schedule impact of adopting a series approach plan are not justified for the lithium-cooled fast reactor/Brayton power converter system (the SR-100G).

A. TECHNOLOGY ISSUES AND CONCERNS

A technology issue, for purposes of this discussion, is a technical factor or a technical unknown that has the potential for becoming a "show stopper." In other words, a technology issue has the potential to jeopardize the launch of the system. A technology issue will also have significant effects on the program's cost and schedule.

A technology concern is a technical factor or technical unknown that will require a design "fix" to resolve the concern. Resolving a technology concern will have some effect on cost and schedule, but less than that associated with resolving a technology issue.

One technology issue has been identified: the safety requirement for reentry dispersal of the system. Our aerospace safety experience from the SNAP program, specifically with both SNAP-8 and SNAP-10A, suggests that it may

be difficult to demonstrate conclusively that reentry dispersal of the system will be achievable. This reservation is strengthened by the fact that this is a high-temperature system (i.e., the alloys and fuels may burn up more slowly). If this issue cannot be favorably resolved, the launch of the system could be jeopardized.

Several technology concerns have been identified. These concerns revolve primarily around the system being a very high-temperature one. The system uses materials that have not yet been extensively characterized. The technology issues and concerns for each subsystem and for the overall system are discussed below.

1. Reactor Subsystem

The currently foreseen technology concerns for the high-temperature, lithium-cooled, refractory-metal-clad, UN-fueled fast reactor subsystem are associated with the fuel element design and long-term performance, containment and structural materials behavior, fuel-coolant compatibility, accommodation of gases formed in or released to the coolant, selfwelding, actuator performance, foil insulation effectiveness, and — most important — safety.

Fuel element technology concerns are associated with fuel growth, fission gas release, fuel-cladding compatibility, degradation of cladding mechanical properties with irradiation, and irradiation-induced swelling and enhanced creep of the cladding. Compatibility between the fuel and the coolant is a concern in the event a fuel element is breached during operation.

Some data exist describing the fuel growth characteristics of UN fuel. Figure V-2 shows these data with an upper bound fuel growth line indicated. The data, although somewhat sparse, do span the range of operating conditions anticipated. The peak burnup of the fuel for the 7-yr operating lifetime is relatively low, only 16,800 MWD/MTU (or approximately 5.9×10^{20} fissions cm^3). Relatively high, however, is the operating temperature of the fuel;

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Figure V-2. Maximum Growth Correlation Used to Size
 Fuel-Clad Gap in SR-100G System

this is the result of the high coolant outlet temperatures required to obtain 100-kWe system power output within the mass and size constraints.

Figure V-3 analyzes fuel growth as a function of time. Fuel growth characteristics are shown relative to the closure of the diametral gap between the fuel and the cladding. Diametral expansion of the cladding caused by irradiation-induced swelling is not accounted for. Under these conditions, imminent gap closure is predicted for design point conditions at the end of the 7-yr reactor operating lifetime. However, the fuel growth characteristics of UN fuel show a "knee" in the curve near the design point. Because of the sparse data base, additional testing should be performed to verify fuel growth in the vicinity of the design point in order to confirm the sizing of the fuel-to-cladding gap.

The amount of fission gas released by the fuel during operation determines how thick the cladding must be. A 20% fission gas release in the peak burnup element is the design value. The existing data tend to indicate that less than 12% fission gas release should be expected. We have used 20% in our current conceptual design calculations and have sized the cladding accordingly. Despite this conservative approach, the magnitude of fission gas release is a technology concern. This concern is amenable to a design fix, such as increasing the clad thickness; there would, of course, be resultant penalties in system mass and reactor size.

Uranium nitride fuel is not compatible per se with ASTAR 811C. Therefore, a tungsten barrier has been incorporated on the interior of the cladding to prevent contact between the fuel and the ASTAR 811C cladding. The effectiveness of this barrier is a technology concern. It is amenable to a design fix, such as thickening the barrier.

Irradiation causing the mechanical properties of the cladding to degrade is always a concern in reactor design. No irradiation data exist for ASTAR 811C, although there are some data on T-111, an alloy very similar to

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Figure V-3. Fuel-to-Clad Gap Closure With Operating Time

ASTAR 811C. The technology concern this presents can be ameliorated by reducing the loads placed on the cladding during operation, but data for irradiated ASTAR 811C are required. We have chosen to size the diametral gap between the fuel and cladding so that the fuel will not strain the cladding, even at the end of the operating lifetime. Nevertheless, the degradation of the mechanical properties of the cladding is a technology concern that needs to be resolved to verify the design.

The compatibility of the coolant with the fuel would be important should the fuel element cladding be breached during operation. Of particular concern would be the accommodation of gas released from the ruptured fuel element and its subsequent fate within the reactor coolant system.

A corollary to this concern is the accommodation of any noncondensable gases such as helium formed in situ during operation of the reactor. The reaction of the neutrons with the lithium coolant will produce helium, and some provisions may need to be incorporated to accommodate this helium during the 7-yr operating lifetime. A potential fix for both technology concerns is a gas filter to remove the gases from the coolant during operation. Proof of such a design concept, if required, would be needed.

Technology concerns regarding containment and support structures for the reactor relate to their long-term physical and mechanical properties and to the effects of irradiation on the materials. As with the fuel element, the effects of irradiation on the long-term mechanical properties of metals are always a concern in reactor design. For the ground demonstration system test, a conservative design approach, where stresses are kept to low levels and strains are minimized, will reduce these concerns. But data on irradiated materials* will be required to verify the design.

*A technology concern associated with the materials test program is the use of prototypic materials for testing. Only materials produced to program-approved materials specifications should be used. This will assure, to the extent possible, the applicability of the resulting data to the design verification.

Self-welding of materials, particularly in the control mechanism drive-line, is a technology concern. In our SNAP-8 and SNAP-10 designs, this concern required a significant effort to resolve. Antiseize coatings will require development and subsequent verification testing.

Verification testing of the control element drive actuators will also be necessary. But it is not a major technical concern because the operating conditions should approximate those used for the SNAP-8 actuators, (because of the location of the actuators in the SR-100G).

The effectiveness of the foil insulation around the reactor pressure vessel must be demonstrated. The beryllium reflectors and control elements must be maintained at temperatures below 1033 K. The reactor vessel will be operating at approximately 1470 K. The foil insulation must be demonstrated to be effective to maintain these limits without restricting the movement of the control elements.

The most critical issue – the only technology issue identified for the system – is the reentry dispersal of the reactor. The safety requirement of reentry dispersal may be very difficult to meet. Based on the SNAP aerospace safety program conducted by Rockwell in the 1960s, it is questionable whether the SNAP-8 and SNAP-10A systems could meet this requirement. And those were relatively low operating temperature systems. For the current high-temperature system, the problem may very well be worse. Meeting the reentry dispersal safety requirement will be a substantial challenge.

2. Shield Subsystem

No major technology concerns are associated with either the gamma or neutron shield subsystems. For the neutron shield, it will be necessary to demonstrate fabrication of the shield. Some performance testing of the shield, particularly demonstration of the effectiveness of the gamma shield and the patch shields, would be desirable although probably not mandatory.

The GDS test will produce these results. The shield subsystem should present no overriding technology concerns relative to either the design or operation of the system.

3. Primary Heat Transport Subsystem

The primary heat transport subsystem is constructed of ASTAR 811C and ASTAR 1611C. Technology concerns relative to the long-term mechanical properties of ASTAR exist for the primary heat transport subsystem, although design stresses are low and should not present any major stumbling blocks.

The helical induction electromagnetic (EM) pump contains no moving parts. It has been tested by General Electric at temperatures of approximately 1475 K for 3000 h. Although 3000 h is considerably less than 7 yr, it does offer proof of the operating principle of the pump design. Verification testing of the pump should be adequate to demonstrate its design.

Similarly, expansion compensator units were designed and tested by Rockwell during the SNAP program. To verify the design of the expansion compensator for this system, a cyclic life test would be reasonable. There is, however, no major technology concern associated with either the design or operation of the compensator units.

The lithium-to-gas heat exchanger is a finned tube-in-shell design. Verification proof testing of the design would be accomplished during the prototype Brayton converter test (see below).

4. Power Converter and Process

The prime technical concern for the power conversion subsystem is the operation of a 1400 K inlet Brayton turboelectric power converter for 7 yr. Specific concerns are related both to materials and to certain design features. The materials concern relates to potential oxygen contamination of the

refractory metal, which causes embrittlement of the metal. How this concern is dealt with in the current system is briefly discussed below.

The refractory metal Brayton power converter would contain high-pressure He-Xe gas as a working fluid, and it has been hypothesized that the moving gas could transport impurities around the loop and embrittle the refractory metal piping. To preclude this, all possible sources of tramp gases are excluded from the system. The lithium-cooled reactor is not a source of tramp gases. On the other hand, a direct-coupled, gas-cooled reactor would be a source of tramp gases. This is one reason that a liquid metal reactor was selected over the gas-cooled concept for the proposed system.

Electrical materials in the alternator and pumps could also contribute to any tramp contamination. The approach is to have a hermetically sealed Brayton unit. In the proposed concept, these areas are isolated from the working fluid gas with ceramic or metal bore seals. These seals hermetically separate these tramp gas sources from the rest of the Brayton loop. This approach carries through to the gas supply and charging equipment, which must be of welded construction and use only bellows-sealed valves. The use of mechanical joints should be avoided in the auxiliary ground test equipment.

The Brayton converter will be heated and evacuated to outgas the various surfaces before it is charged with the working fluid gas. The unit could be outgassed and sealed for an extended period before charging. The working fluid gas will contain less than 1 ppm oxygen, which can be gettered. All of the refractory metal alloys that would be used in the converter contain appreciable amounts of hafnium or zirconium. These two elements are effective getters of oxygen, and the alloys are annealed at 1365 K to 1590 K to assure that all oxygen in the alloys is in the stable HfO_2 and ZrO_2 forms. This condition gives the refractory alloys columbium and tantalum compability with the liquid metals. Any oxygen transport within the closed Brayton converter will therefore be dependent on the vapor pressure of HfO_2 or ZrO_2 . As shown in Figure V-4, the vapor pressures of these two oxides are very low. No

significant transport of material would occur for vapor pressures below 10^{-8} mm Hg.

Ground testing of the refractory metal converter will have to be conducted in ultra-high vacuum ($\sim 10^{-9}$ mm Hg). This test environment assures that external contamination of the refractory metals will be very low. As a backup to the above procedures, a special gettering unit containing pure zirconium at the optimum gettering temperature could be incorporated into the ground test or flight converter design at a small mass penalty.

The approaches used to prevent oxygen embrittlement of the refractory alloys are summarized below:

- Tramp gas sources minimized
 - . Liquid-metal-cooled reactor
 - . Electrical bore seals
 - . Gas bearings
- Operating procedures
 - . Outgassed evacuated system
 - . <1 ppm O_2 in gas working fluid
 - . Hermetically sealed gas supply equipment
 - . Ultra-high-vacuum test environment
- Metallurgy
 - . Alloys heat treated to form HfO_2 - and ZrO_2 -type oxides
- Special getter unit (if required).

There could also be concern about the use of superalloys in the converter in a vacuum environment. When the vapor pressure of a metal or other substance approaches 1×10^{-9} Torr, an appreciable amount of surface may be vaporized if exposed to a space vacuum. The vapor pressure of selected metals used in superalloys is shown in Figure V-5. The rate of surface erosion from the side of a sheet can be calculated by the following formula:

Figure V-4. Vapor Pressure-Temperature Relationships for Refractory Metals and Oxides

$$Y = 2.15 \times 10^5 (P/d) (M/T)^{1/2},$$

where:

Y = loss in thickness (cm/1000 h)

d = density of solid material (g/cm³)

p = vapor pressure (mm Hg)

M = molecular weight gas phase

T = temperature (K).

A plot of surface erosion versus temperature is shown in Figure V-6. As can be seen for this worst case, to assure that surface erosion is held below 0.025 mm per 10,000 h, these metals should be applied at temperatures below 1300 K when the bare surface is exposed to high vacuums.

The concern expressed relative to the compatibility of the refractory metals in the Brayton loop with the inert gas working fluid is acknowledged but, as discussed above, it is not received with any degree of alarm. A simple materials test should provide sufficient verification that it is not a problem. Final verification would be obtained during testing of the prototype Brayton power converter.

Specific design concerns are associated with the performance of the bearings, the alternator, and the turbine wheel and blading arrangement. The design items are judged to be the critical design features of the Brayton turboelectric converter that could give rise to premature failure. To resolve these concerns, we believe it prudent to conduct feature tests of bearings, alternators, and turbine wheel and blading arrangements to discover whether there are any basic design flaws. These feature tests would precede fabrication and testing of a full-up Brayton power converter (including the heat exchanger, recuperator, and associated ancillary equipment). We believe that a 10,000-h test of the full-up system should be adequate to identify and correct any lingering design problems before the Brayton converter is used in the ground demonstration system test.

Figure V-5. Vapor Pressure-Temperature Relationships for Superalloy Constituents

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Figure V-6. Maximum Loss in Thickness for 10,000-h Exposure

One area that has often proven troublesome in previous terrestrial systems and may do so again is the design and operation of transition joints. In our system, the heat rejection subsystem is constructed of stainless steel, whereas the Brayton turboelectric converter is constructed of refractory metal, Hastelloy, and Inconel. The coefficients of thermal expansion of the metals differ. This difference must be accommodated by the joints that make the transitions from one metal to another. Our experience suggests that a substantial development and testing effort will be needed to define the fabrication, testing, and acceptance procedures necessary to assure adequate transition joints.

For the power processor, we see technology concerns associated with the use of ballast resistors and the performance of the power processor. Again, feature tests of the ballast resistors and the power processor should be performed early in the program to define any technical problems and allow technical fixes to be applied. We do not foresee major technology concerns in this area.

5. Heat Rejection Subsystem

Our design calls for a gas-cooled radiator using a double d-tube arrangement with bonded fin and armor. Although not a major technology concern, a demonstration of the fabrication techniques for making this geometry and for verifying that no long-term compatibility problems exist needs to be accomplished. In addition, the development, application, and proof testing of a calcium titanate emissivity coating capable of delivering ~ 0.9 emissivity/ ~ 0.15 absorptivity levels during the 7-yr operating lifetime of the system will need to be accomplished.

6. Overall System

Major technology concerns exist relative to assuring that system performance can be achieved over the 7-yr operating lifetime at the high

operating temperatures required. Data simply do not now exist to support unequivocally the design of a system to achieve these requirements. The reliability data necessary to support 7-yr reliability projections for the system do not exist. These technical concerns can only be addressed through substantial testing and, in particular, through design and operation of the ground demonstration system test.

Regarding survivability requirements, we believe that the satisfaction of these requirements can best be demonstrated through analyses. Verification that survivability requirements can be met could conceivably be demonstrated through test of a system mockup. However, the cost of such a test probably makes it desirable to demonstrate compliance through analysis. The dormancy/restart verification can, of course, be accomplished during the ground demonstration system test.

The major technology issue associated with the system is the reentry dispersal safety requirement. We believe that ground-handling safety, shuttle safety, and even water-subcriticality safety can be demonstrated through appropriate analyses supported by appropriate test data.

However, we believe that meeting the requirement for reentry dispersal will be difficult. This is a high-temperature system constructed of high-temperature materials. Based on our SNAP experience, it is improbable that either the SNAP-8 or the SNAP-10 reactors could meet the reactor dispersal requirements imposed in today's world. In our judgment, if there is a single technical factor that could prevent launch of the system, it is the requirement for reentry dispersal. A detailed program plan defining the various steps necessary to verify that this requirement has been met will be necessary.

7. Test Facilities

Although not strictly germane to the system, we have technology concerns relative to providing test facilities to support the program. Test facilities

for long-term materials properties testing, feature and component tests (such as self-welding tests, feature tests of bearings, alternators, and the power conversion and processing subsystem prototype test), and certainly the test of the Ground Demonstration System itself will require extremely sophisticated test facilities. These test facilities will be required to provide very hard vacuums at extremely high temperatures for very long lifetimes. Periodic access to the interior of these test facilities for retrieval of test specimens will complicate the design and operation of such facilities. We currently see the schedule for constructing (or even modifying existing) facilities as potentially being the program critical path.

B. TECHNOLOGY DEMONSTRATION PLAN

1. Objectives, Goals, and Overall Approach

The goal of the technology demonstration plan is to provide reasonable assurance that the ground demonstration system will perform "as advertised." It is a technology risk reduction plan. To accomplish this goal, the objective of the plan is to obtain verification test data aimed at reducing the technology risks associated with the technical concerns listed in the previous section.

An integrated overall plan for technology demonstration of the lithium-cooled fast reactor/Brayton turboelectric converter system, leading to system demonstration at the earliest possible time, is presented in this section. The plan is system oriented and addresses the technology concerns discussed in the previous section.

Technology demonstration can be approached in two ways. The first – or parallel – approach is to perform the design and fabricate the ground demonstration system in parallel with conducting appropriate technology demonstration tests to verify that the design will meet requirements. The second – or series – approach is to conduct the technology demonstration

tests and obtain appropriate design data to allow finalization of the design before fabrication of the ground demonstration system starts. We chose the parallel approach for our plan.

Figure V-7 illustrates a summary schedule for a parallel approach technology demonstration plan which we selected for the SR-100G System. Technology demonstration tests are conducted in parallel with the design and fabrication of the ground demonstration system test. The test results form a series of "gates" (i.e., decision points) that will allow the program to assess the ground demonstration systems testing is initiated, the technical risk associated with achieving the GDS test objective.

In most cases, results of the technology demonstration tests will not be obtained quickly enough to affect the design activities. In this sense, therefore, there is a program risk associated with proceeding with design and fabrication of the GDS. However, if an unexpected "show stopper" were to be uncovered as a result of a technology demonstration test, the results would be achieved in time to modify the GDS — although with substantial cost and schedule impact to the program — before the ground demonstration test was initiated.

If the second type of approach — series — is employed, it would be necessary to delay the initiation of the ground demonstration system test fabrication activity for approximately 3 to 3-1/2 yr. This would allow technology demonstration test data to be made available and incorporated into the design before fabrication of the GDS. There obviously would be a sizable cost and schedule penalty associated with this test, although the technical risk for proceeding could be substantially reduced.

A review of the system design and the technical data base supporting that design indicates to us that the appropriate technical approach is to choose the parallel technology demonstration plan. We believe that there is a sufficient data base to support design of the reactor, including the critical

Figure V-7. Technology Demonstration Plan Summary Schedule for Key Elements

fuel elements, and the power conversion subsystem on a schedule consistent with the parallel approach. The key tests, indicated in Figure V-7, should verify that the design of the two critical subsystems (i.e., the reactor and power conversion and processing subsystems) will meet advertised performance.

Four conditions are key to the plan:

- Condition 1 A detailed, in-depth, cost- and schedule-effective test program plan for the technology demonstration activities of the program. This test program plan should be in place before testing is begun.
- Condition 2 A well-defined baseline system design meeting the program-approved system specification. This must be established first to define the required test conditions/ranges to be investigated in the test program.
- Condition 3 A well-defined set of system materials specifications. The materials used in the tests should conform to program approved system materials specifications.
- Condition 4 Adequate test facilities. These must be capable of imposing test conditions that duplicate (as closely as practicable) the anticipated component, subsystem, or system operating conditions.

We expect that a firm conceptual baseline system design (condition 2) could be generated in the first 8 months of the program. During this 8-month time frame, a detailed technology demonstration test program plan (condition 1) would also be generated.

Once the baseline system design and test program plan are in place and have been approved by the program, fabrication of test articles using materials procured to program-approved materials specifications (condition 3) would begin. In parallel, the necessary test facilities would be readied and checked out (condition 4). We believe these activities would – depending on the test article and test facility complexities – require at least 1 yr (perhaps slightly less for simple tests).

Therefore, approximately 1-1/2 to 2 yr (after start of conceptual system design) would be required to initiate the first technology demonstration tests. To shorten the estimated 1-1/2- and 2-yr time frame, one or more of the four conditions noted above would have to be modified. For example, "mini" test plans for specific high-payoff tests could be substituted for the detailed, in-depth technology demonstration test program plan (i.e., condition 1). Or, test conditions for a specific test could be defined based on a preconceptual system design, e.g., the type of designs currently being evolved on the program) rather than awaiting a firm conceptual system design (i.e., condition 2). Or, materials tests could be conducted using representative, although not prototypic, materials that may, or may not, reflect the final program-approved materials specifications (i.e., condition 3).

Such approaches obviously entail a certain degree of risk. For the SR-100G System, we do not foresee any technology concerns that are of sufficient magnitude to justify adopting these shortcut approaches. The plan presented below is an integrated, logical plan for achieving technology demonstrations for the lithium-cooled fast reactor/Brayton turboelectric converter at the earliest point in time.

2. The Plan

The proposed technology demonstration plan is shown in detail in Figure V-8. The plan addresses the five major subsystems, the overall system, generic materials testing, and test facility requirements for the ground demonstration system test. The technology demonstration plan activities for each of these elements are discussed below.

a. Overall System

The technology concerns associated with the overall system relate to safety, performance, lifetime, reliability, dormancy/restart, and survivability requirements. As noted above, we foresee the safety requirement for

Figure V-8. Technology Demonstration Plan of the SK-100G System

reentry dispersal as being the single technology issue associated with the system. Assuring that the requirements for reentry dispersal can be met will be a formidable task. Our experience with the SNAP-8 and SNAP-10A indicates that meeting this requirement with those systems would be improbable. Meeting the requirement for a high-temperature system such as the current system may prove even more difficult. The technology demonstration plan is aimed basically at demonstrating compliance with the reentry dispersal requirement by analysis with supporting test data and associated probability analyses for dose rates depending on the orbital reentry characteristics. If there is indeed a show stopper associated with the program, we firmly believe it is the reentry dispersal safety requirement.

We believe that the requirements for ground-handling safety, water subcriticality, and shuttle safety can be met rather straightforwardly through appropriate analyses and procedures. The water subcriticality requirement will of necessity be met by engineered features in the system. "Crit" tests will be performed to verify the effectiveness of these engineered safety features, and compliance with the requirement will be demonstrated by analyses supported by these data.

Demonstration of compliance to performance requirements must, of necessity, be accomplished by the ground demonstration system test. Demonstration that lifetime requirements can be met will be based on the supporting technology demonstration tests shown in Figure V-8 coupled with the results of the ground demonstration system test. A proof test for the full 7-yr lifetime seems unrealistic. Therefore, extrapolation of the result of the ground demonstration system test will probably be necessary to show compliance with the lifetime requirement (supplemented by specific component or subsystem "life" tests).

Compliance with the reliability requirements will also be, of necessity, performed by analyses. It is simply impractical to perform the number of tests necessary to demonstrate compliance with the reliability requirement in

any reasonable cost or time frame. Therefore, existing data bases, coupled with results of the component and ground demonstration system tests, will form the basis for analytical predictions that reliability requirements can be satisfied.

The requirements for dormancy/restart can be straightforwardly addressed during the ground demonstration system test. We do not perceive this requirement as being of major technical concern.

Demonstration that survivability requirements have been met will, as with most system requirements, of necessity, be demonstrated by analyses. It is conceivable that a system mockup could be built and subjected to appropriate testing to demonstrate that survivability can be achieved. However, this type of testing would be quite expensive, and we judge that demonstration by analyses should be adequate to show compliance with the survivability requirement.

b. Reactor Subsystem

Technology demonstration for the reactor subsystem can be approached in several different ways. A fuel/materials irradiation, followed (or perhaps overlapped) with a reactor subsystem test, followed by the ground demonstration test, is one approach. A second approach would be a fuel/materials irradiation test followed directly by a ground demonstration test of the full system (not just the reactor subsystem). A third approach would be to proceed directly to a ground demonstration system test, foregoing both the irradiation test and the reactor subsystem test.

The choice of approach involves considering the data base available to support the reactor design and programmatic issues such as cost, schedule, and acceptable levels of risk. It is our judgment that either the first or second option listed above is the reasonable course of action. The third option – proceeding directly with the ground demonstration test – appears to involve excessive risk given the current data base on refractory-clad UN fuel.

The choice between the first and second options is difficult. Option 1 minimizes risk but involves a longer schedule and more cost than option 2. Option 2 is a greater risk, but can cut as much as 3 yr off the schedule for the ground demonstration test (with consequent cost reduction).

We have selected option 2 for purposes of the technology demonstration plan presented herein. The risk is reduced somewhat by performing an irradiation test in parallel with the design and fabrication of the ground demonstration test. Information from the irradiation test would be available in time to form the last "gate" (i.e., decision point) for proceeding with the system test. The situation would be the same if a reactor subsystem test were to be substituted for the system test. There seems to be little gained by adopting option 1 in this regard.

The delay of the system test (probably of the order of 3 yr) if option 1 is chosen, coupled with the potential need for two very expensive test facilities (one for the reactor test and one for the system test), appears to favor the selection of option 2. But the selection is certainly not black and white. A detailed study of the options will be required by the SP-100 Program before the selection of an approach can be firmed up.

As previously noted, we have selected option 2. For comparative purposes, however, we have also shown a potential reactor subsystem test (option 1) in Figure V-8. In the event option 1 is selected, it should be clearly noted that there would be a delay of about 3 yr in starting the ground demonstration system test.

Based on the selection of option 2, an irradiation test of the fuel element would be performed to address the technical concerns associated with the reactor subsystem. Long-term materials properties for both the cladding and containment and internal structural materials utilizing both unirradiated and irradiated material (irradiated with the fuel element test) would be performed. A development and demonstration self-welding test for the drive

mechanisms would be accomplished, along with verification testing of the control mechanism actuators (which are anticipated to be of the SNAP-8 variety). In addition, development and testing of an in-line gas filter for removing noncondensable gases formed in situ might be necessary and would be performed early in the program if found to be required.

The fuel element irradiation test is anticipated to require approximately 4-1/2 to 5 yr to complete. The purpose of the test is to verify the fuel growth characteristics shown in Figure V-2 in the vicinity of the design point. In addition, the test would define the fission gas release, verify fuel cladding compatibility associated with the tungsten barrier applied to the inside diameter of the ASTAR 811C cladding, and provide a materials irradiation that could subsequently be used for defining the degradation of the cladding, structural, and containment materials. The test would also define the amount of irradiation-induced cladding swelling that could be expected.

Long-term mechanical properties of the cladding and structural and containment materials would be conducted on both unirradiated and irradiated materials. The unirradiated material data base would form the baseline for design of the primary heat transport coolant boundary, while the irradiated mechanical properties data would form a verification of the design basis for the cladding, reactor pressure vessel, and internal support structures.

As shown in Figure V-8, most of the test data will be available just before the start of the ground demonstration system test. In this respect, the data are expected to be confirmatory in nature, verifying the adequacy of the design. In the event the data indicate that a certain performance or lifetime requirement of the GDS cannot be met, the resultant data can be used to appropriately qualify the GDS test; these data can also be used to make appropriate corrections for the design and fabrication of the first flight system.

c. Shield Subsystem

The technology demonstration plan calls for a shield fabrication demonstration run followed by a shield performance test with particular emphasis on the effectiveness of the gamma shield and the patch shields. Data from these demonstrations and tests will be available in plenty of time for any important results to be incorporated in the ground demonstration system fabrication. We anticipate that there will be no substantial technology concerns associated with the shield subsystem.

d. Primary Heat Transport Subsystem

The technology demonstration plan calls for a verification test of the EM helical induction pump and a cyclic life test of the expansion compensator. Both of these tests are expected to be of the design confirmation type, and any results to the contrary can be readily incorporated in the ground demonstration system fabrication without substantially affecting either the program's schedule or cost.

e. Power Conversion and Processing Subsystems

The technology demonstration plan calls for certain feature tests of critical subcomponents for both the power conversion and the power processing subsystems. Specifically, feature tests of the bearings, alternator (with specific emphasis on alternator cooling), and the turbine wheel/blading arrangement would be performed early in the program to provide baseline data in support of the design and fabrication of a prototype power converter. In addition, feature tests of the ballast resistors and the power processor equipment would also be performed in support of the prototype test design and fabrication.

A straightforward materials test (included in Figure V-8 under the Generic Materials Testing category) would confirm that the compatibility of

the refractory metal with the inert gas working fluid is not a major technology concern. The prototype power conversion and processing subsystem would be designed, fabricated, and tested for 10,000 h to verify that its performance is adequate. The results of this test would be available approximately 1 yr before completion of the ground demonstration system fabrication, thereby allowing adequate time for modifications to the power conversion and processing subsystem for GDS in the event it were necessary.

The key feature of the power conversion and processing subsystem will be the necessity for using transition joints to join the refractory and super-alloy metals of the power conversion system to the stainless steel metal of the heat rejection subsystem. There is a substantial difference in the coefficient of expansion for the metals, and it is expected that development of the transition joint will be a major activity during the technology demonstration. The development, fabrication, and testing of the transition joints can proceed in parallel with the prototype test of the power conversion and processing subsystem, and it does not now appear that the development of the transition joints will be schedule limiting. However, it has been our experience on several programs that the development and proof-testing of transition joints generally is a major technical hurdle.

f. Heat Rejection Subsystem

The heat rejection subsystem is a relatively straightforward double D-tube arrangement with bonded fin and armor. The emissivity/absorbtivity coating is applied to the heat transfer surfaces of the heat rejection subsystem. We believe it is necessary to perform a fabrication demonstration and long-term compatibility demonstration test, but we would anticipate that the results of this test would not have any substantial effect on the ground demonstration system itself. This test should provide 20,000-h data well in time for incorporation in the ground demonstration system fabrication effort, if required.

g. Generic Materials Testing

Fabrication development, weld development, development of bonding and forming techniques, long-term compatibility testing, and corrosion and mass transfer tests would be conducted under this activity. We anticipate that no major technical breakthroughs will be required. Activities under this generic materials testing element are merely confirmatory in nature and would not be expected to affect the overall ground demonstration system test activity.

h. Test Facility for the Ground Demonstration System Test

The design and construction of new test facilities (or the modification of existing ones) for conducting the ground demonstration system test is expected to be a very large activity. The test facility will require very high vacuums in order to conduct the ground demonstration system test. The time available for obtaining the necessary facilities and supporting test equipment is relatively short (on the order of approximately 5 yr), based on the technology demonstration plan shown in Figure V-8. We anticipate that this element of work may, indeed, be the critical path on the program.

However, the importance of adequate test facilities cannot be over-emphasized. Indeed, we recommend designing the facility such that interim test data from the ground demonstration system test can be obtained in support of the design and fabrication of the first flight system. Of particular importance would be the ability to remove and replace components such as fuel elements during the test period in order to monitor the time variation of such variables as fuel growth, fission gas release, corrosion, etc. This approach would be extremely valuable in support of the flight system design but would, at the same time, have a significant impact on the design, cost, and schedule for the test facilities. It is an approach that is highly desirable but is not without drawbacks.

3. Expected Output

The major output of the technology demonstration plan will be, of course, the demonstration of the system's ability to achieve as-advertised performance for a sufficiently long period that projections can be made indicating that lifetime requirements can be achieved. In addition, results of critical safety analyses, supported by appropriate test data, will be necessary to verify that the reentry dispersal safety requirement can be satisfied.

Component and materials testing will be performed in parallel with the ground demonstration system design and fabrication activities. Although some of these data may not be available in time to affect the ground demonstration system itself, they can form the basis for design improvements in the first flight system. Should a significant technical problem arise, as indicated by the results from the tests, the ground demonstration system could be delayed until necessary design modifications or provisions for handling this technical problem have been incorporated.

4. Estimated Cost and Schedule

As shown in Figure V-8, it is estimated that approximately 5-1/2 to 6 yr will be required from initiation of conceptual design to initiation of the ground demonstration system test. This assumes the use of the parallel activity approach, deleting the reactor subsystem test in favor of proceeding directly to the ground demonstration system test. If the reactor subsystem test is conducted, the ground demonstration system test will be delayed by approximately 3 yr.

The estimated costs to achieve technology readiness (see Figure V-7) are ~\$200M. To advance through proven development of the system will require an additional ~\$900M. These cost estimates are based on the data presented in Ref. 1. The costs do not include facilities costs, aerospace safety costs, or system integration costs.

VI. BACKUP SYSTEM

The backup system is defined in Part I, Item 7, of our workscope to be "...the one which has the best combination of performance and technical maturity." In considering "best combinations," there is a dilemma. The reactor with the most technical maturity is one that operates in the 975 K (1300°F) outlet range (or lower)(i.e., an LMFBR-type reactor at SNAP-reactor temperatures). The power converter with the most technical maturity for space power applications is the thermoelectric converter. But to achieve, within the imposed mass and envelope requirements, reasonable performance (i.e., power output) with thermoelectrics appears to require reactor outlet temperatures well in excess of the 975 K of the technically mature reactor. In other words, the technically most mature reactor and the technically most mature power converter are incompatible.

For a backup system, we originally chose a high-temperature reactor coupled to thermoelectrics. The reactor, a lithium-cooled, UN-fueled fast reactor, operates at conditions essentially identical to those at which the reactor chosen for the most attractive system operates (e.g., 1470 K design point outlet temperature). Although a 1470 K reactor cannot be classified as technically mature, it does:

- Obtain reasonable performance from the thermoelectrics
- Maximize technology carrythrough from the most attractive system
- Provide for potential future growth in system power output.

Were a technically mature reactor selected (outlet temperature of 975 K or less), the potential power converters would be the Rankine, Stirling, and Brayton engines (i.e., dynamic machines) and in-core thermionics. In-core thermionics are attractive but suffer from limited lifetimes due to insulator degradation and fuel swelling. Rankine machines face the questions of condensing and vapor-liquid interface control in zero-g environments. The Stirling is attractive but cannot yet be classified as technically mature.

That leaves the low-temperature Brayton, which we do regard as a relatively mature technology at temperatures of 1150 K (1600°F) and below. We have included some very preliminary parametric data (net power output versus reactor outlet temperature) in Section VI.B for purposes of comparison with the high-temperature reactor/thermoelectric backup system.

A. DESCRIPTION OF A REACTOR-THERMOELECTRIC BACKUP SYSTEM

Our initial selection for a backup system used a lithium-cooled fast reactor (the same as the most attractive system, thereby maximizing technology carrythrough between the most attractive system and the backup system). Thermoelectric power converters, coupled directly to the coolant piping,* were selected.

Initial scoping analyses indicated that this type of backup system would produce 50 to 60 kWe. During the final activities in FY 1983, Rockwell re-optimized the design point selection for the backup system and performed a preliminary characterization of this system. This re-optimization and characterization indicated that the beginning-of-life (BOL) power of the backup system is 40 kWe given the mass and envelope constraints of the SP-100 Program.

The overall system configuration is shown in Figure VI-1. The characteristics of the system are provided in Table VI-1. The system has an overall thermal-to-electrical efficiency of 5% at BOL. Degradation of the thermoelectric converters would cause this efficiency to drop over the 7-yr operating life of the system, resulting in an end-of-life (EOL) net power output of 35 kWe.

*The direct-coupled thermoelectric converters will require some development. "Off-the-shelf" converters would be of the radiative-coupled type, the type currently used in RTGs.

Figure VI-1. Thermoelectric System of the Backup Reactor

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TABLE VI-1
CHARACTERISTICS OF A REACTOR-THERMOELECTRIC BACKUP SYSTEM

Net electrical power (BOL)(kWe)	40
Net electrical power (EOL)(kWe)	35
Full power life (yr)	7
Length (m)	6.1
Diameter (maximum)(m)	4.3
Overall efficiency (BOL)(%)	5
Radiator area (m ²)	52
Mass (kg)	2980
Specific power (BOL)(We/kg)	13
Specific mass (BOL)(kg/kWe)	75
Reactor type	Pumped Li-cooled fast
Heat transport type	Pumped Li D-tube
Converter type	Ternary-doped Si-Ge
Radiator type	C-C composite direct fin

The reactor employed in the backup system is the same type as that used for the heat source of the most attractive system (described in Section IV). The reactor characteristics are shown in Table VI-2. The reactor is UN fueled and Li cooled. The reactor thermal power is 800 kWt with a reactor outlet temperature of 1470 K.

The reflector, control features, and the gamma and neutron shield are similar to those used in the most attractive system. Because of the higher reactor power of this system as compared with the 525 kWt of the most attractive system, the shield mass is higher, as shown in Table VI-3.

TABLE VI-2
CHARACTERISTICS OF REACTOR SUBSYSTEM FOR BACKUP SYSTEM

Thermal power (kWt)	800
Reactor coolant	Li
Reactor outlet temperature (K)	1470
Reactor temperature rise (K)	60
Reactor fuel	UN
Fuel pin OD (cm)	1.9
Peak linear power/fuel pin (kWt/m)	23
Core diameter (cm)	30
Core height (cm)	35.5

TABLE VI-3
MASS SUMMARY FOR A REACTOR-THERMOELECTRIC BACKUP SYSTEM

Component	Mass (kg)
Reactor subsystem	550
Shielding subsystem	480
Primary heat transport subsystem	576
Li pumps (2)	95
Volume accumulator	65
D-tubes	316
Manifolds	51
Pump(s)-heat rejection subsystem	49
Converter/radiator subsystem	910
Thermoelectric converters/ radiator fins	544
Thermal barriers and insulation	356
Interconnecting circuitry	10
Internal support structure	220
Power control equipment	244
Startup/processor	224
Transmission line	20
Total mass	<hr/> 2980

The lithium coolant in the reactor is pumped by redundant, helical induction EM pumps (Figure VI-1). Pumped fluid enters a manifold and is transferred to individual D-shaped tubes, which could be called the thermal power distribution subsystem. This subsystem supplies the individual thermal paths to the thermoelectric converters and also provides support for the converters and the radiator.

The helical induction EM pumps, which provide hydraulic power for the lithium, also provide heating of the lithium before reactor startup. The pumps and their cooling equipment are located inside an internal thermal barrier for the system. This conical-cylindrical barrier separates the high-temperature D-tubes from the relatively cooler interior of the system. The radiators for the EM pumps are located at the back of the cone-cylinder section and reject their heat to a deployable thermal reflector, which also serves as part of the secondary lithium containment barrier during the shuttle ascent stage for the system.

Startup is accomplished by deploying the rear thermal reflector for the system and then ejecting two halves of a thermal barrier/lithium secondary containment canister that surrounds the thermoelectric arrays.

The thermoelectric converters are piggybacked onto the D-tubes. The lithium flowing through the D-tubes provides thermal input to the converter. Each converter is equipped with its own integral radiator fin. There are 28,800 thermoelectric converter units. The characteristics of the primary heat transport subsystem are provided in Table VI-4. The conceptual configuration for the converters and fin is shown in Figure VI-2.

TABLE VI-4
CHARACTERISTICS OF THE PRIMARY HEAT TRANSPORT SUBSYSTEM FOR
A REACTOR-THERMOELECTRIC BACKUP SYSTEM

D-tube material	ASTAR 811C
Tube length (m)	5
Tube inside radius (cm)	0.5
Tube wall thickness (cm)	0.05
Tube surface coating material	CVD tungsten
Tube coating thickness (cm)	0.0025
Number of converters per tube	117
Total number of tubes	246
Mass of tube and Li per converter (g)	11
Total tube and Li mass for 28,800 converters (kg)	316

Each converter requires 27 W of thermal power and can produce 1.55 W of electric power. The characteristics of the converter are shown in Table VI-5. The converter material chosen for the backup system was silicon germanium with a 5 wt. % addition of gallium phosphide. This material has a figure of merit (Z) of $0.85 \times 10^{-3} \text{ K}^{-1}$. The total system electric power is provided by connecting the thermoelectric converters in parallel and series to perform 20 parallel circuits of 187 Vdc each.

Each thermoelectric converter has its own carbon-carbon composite fin. The fin characteristics are shown in Table VI-6. The total finned radiator area is 52 m^2 .

Figure VI-2. D-Tube and Converter Stack Concept of a Reactor-Thermoelectric Backup System

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TABLE VI-5

THERMOELECTRIC CONVERTER CHARACTERISTICS OF A REACTOR-THERMOELECTRIC BACKUP SYSTEM

AVERAGE HOT JUNCTION TEMPERATURE (K)	1325	AVERAGE HOT SHOE TEMPERATURE (K)	1370
AVERAGE COLD JUNCTION TEMPERATURE (K)	825	HOT SHOE ABSORPTIVITY	0.85
AVERAGE FIGURE OF MERIT (K^{-1})	0.85×10^{-3}	CURRENT STRAP WIDTH (cm)	1
MATERIAL EFFICIENCY (%)	7.2	CURRENT STRAP THICKNESS (cm)	0.05
OPEN CIRCUIT VOLTAGE/COUPLE (V)	0.22	ELECTRICAL INSULATOR MATERIAL	Al_2O_3
LOAD VOLTAGE/COUPLE (V)	0.13	TEMPERATURE AT RADIATOR FIN JUNCTION (K)	810
CURRENT/COUPLE (A)	12	COMPONENT MASSES (g)	18.88
POWER/COUPLE (W_e)	1.55	HOT SHOE	2.37
HEAT FLUX (W/cm^2)	27	HOT STACK	0.98
COUPLE LENGTH (cm)	1.57	HOT STRAP (Mo)	0.52
COUPLE AREA, n + p (cm^2)	0.80	THERMOELECTRIC MATERIAL (SiGe GaP)	4.1
CONVERTER LENGTH (cm)	4.25	COLD STACK	1.12
HOT SHOE AREA (cm^2)	13.5	COLD STRAP (Cu)	1.84
HOT SHOE THICKNESS (cm)	0.1	RADIATOR FIN (C-C COMPOSITE)	4.35
HOT SHOE MATERIAL	(EXPERIMENTAL GRADE) GRAPHITE	INSULATION (ASTROQUARTZ)	3.60
SUBLIMATION BARRIER	SiO_2	SINGLE CONVERTER/FIN, MASS (g)	18.9
		TOTAL MASS FOR 28,800 CONVERTERS (kg)	544

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TABLE VI-6
RADIATOR FIN CHARACTERISTICS FOR A REACTOR-THERMOELECTRIC
BACKUP SYSTEM

Radiator fin material	Carbon/carbon composite
Surface emissivity	0.85
Area/converter (cm ²)	16.4
Shape	Triangular
Fin root thickness (cm)	0.25
Fin effectiveness	0.83
Average operating temperature (K)	795
Heat rejection capacity/converter (W)	27

This radiator area was used in optimizing (Figure VI-3) the D-tube and thermoelectric converter design point. The specific mass of the converter and D-tube assembly is plotted versus the heat flux through the converter. Two parabolas, showing minimums, are provided on the figure for two different radii of the D-tubes. Also shown on the figure is the gross electrical output that could be provided with a 52-m² radiator, again for each of the D-tube radii. The design point was selected as slightly off the optimal minimum for the 0.5-cm D-tube radii. Although the smaller D-tube radii designs provided lower specific masses, they could not supply the maximum power at the indicated heat flux that the largest D-tube array could supply. The design point for the D-tube and converter was chosen for its maximum power capabilities rather than for minimum mass. Such a minimum mass system would have optimized at 33 W/cm² but at a gross electrical output of only 38 kWe. Based on the selected design point, the characteristics of the converter/radiator subsystem are presented in Table VI-7.

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Figure VI-3. Selection of the Converter Design Point for the Backup System

TABLE VI-7

CONVERTER/RADIATOR SUBSYSTEM CHARACTERISTICS OF THE BACKUP SYSTEM

ELECTRICAL POWER DELIVERED (BOL) (kWe)	40	FRUSTUM HALF ANGLE ($^{\circ}$)	22.5
ELECTRICAL POWER DELIVERED (EOL) (kWe)	35	RADIATOR AREA (m^2)	52
FULL POWER LIFE (yr)	7	MINOR DIAMETER OF FRUSTUM (m)	1.36
OPERATIONAL LIFE (yr)	10	MAJOR DIAMETER OF FRUSTUM AND CYLINDER (m)	4.3
NUMBER OF PARALLEL CIRCUITS/VOLTAGE PER CIRCUIT (V dc)	20/187	AXIAL SUBSYSTEM LENGTH (m)	4.87
THERMAL LOSSES (%)		COMPONENT MASSES (kg)	
REACTOR AND PHTS	3	THERMAL BARRIERS	
THERMOELECTRIC ARRAY	20	INTERNAL	112
ELECTRICAL LOSSES (kWe)		EXTERNAL-SIDES (EJECTED)	124
LITHIUM PUMP(S) REQUIREMENT	1	END (DEPLOYED ON BOOM)	44
CONDITIONING AND TRANSMISSION	3.8	INTERCONNECTING CIRCUITRY	10
RADIATOR FIN ARRAY HEAT REJECTION CAPACITY (kWt)	777	THERMOELECTRIC CONVERTER ARRAYS	544
SUBSYSTEM EFFICIENCY (%)	5.1	MULTIFOIL INSULATION BETWEEN CONVERTERS	76
NUMBER OF THERMOELECTRIC CONVERTERS	28,800	TOTAL CONVERTER/RADIATOR SUBSYSTEM MASS (kg)	910

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The remaining components of the system include such equipment as the volume accumulators for the primary heat transport subsystem, the internal support structure, and the power control equipment. The masses of all the subsystems and of this additional equipment are shown in Table VI-3. The total mass of the backup system is just under the 3000-kg limit for the SP-100 Program.

B. A LOW-TEMPERATURE BRAYTON BACKUP SYSTEM

As discussed in Section VI.A, a low-temperature Brayton system might be considered as an attractive backup system. We have made some very preliminary parametric calculations to scope the potential of the low-temperature Brayton. Figure VI-4 presents the results.

In the range of "technically mature" reactors (≤ 975 K), this system may have promise. More detailed investigation of the low-temperature Brayton might be a fruitful effort for the SP-100 Program.

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Figure VI-4. Electric Power versus Temperature Characteristics for a Low-Temperature Reactor — Brayton Turboelectric Backup System

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