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MASTER

REACTOR - THERMOELECTRIC SYSTEM
FOR NASA SPACE STATION

AEC Research and Development Report

ATOMICS INTERNATIONAL

A DIVISION OF NORTH AMERICAN ROCKWELL CORPORATION

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Printed in the United States of America
Available from

Clearinghouse for Federal Scientific and Technical Information
National Bureau of Standards, U.S. Department of Commerce
Springfield, Virginia 22151

Price: Printed Copy \$3.00; Microfiche \$0.65

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FOR NASA SPACE STATION

By

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CONTRACT: AT(04-3)-701
ISSUED: JULY 1, 1969

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ABSTRACT

The results of a study regarding the design, integration, and operation of nuclear reactor-thermoelectric power systems for Earth-orbiting manned space stations are presented. In this joint study the design and analysis of the nuclear power system was conducted by the Atomics International Division (AI) of North American Rockwell Corporation under contract to the Atomic Energy Commission (AEC), and the space station design, integration, and mission analysis work was performed by the National Aeronautics and Space Administration, Marshall Space Flight Center (NASA-MSFC). This report emphasizes the nuclear power system design and operation, and a separate report to be published by NASA-MSFC will provide more details on the space-station integration and mission-analysis results.

I. INTRODUCTION

Plans for National Aeronautics and Space Administration space missions in the post-Apollo period include the deployment of large semipermanent manned space stations operating in low Earth orbit. Current trends are to increase substantially the overall size, crew complement, lifetime, functions and thus the utility of such space stations relative to early orbiting workshops which will be flown in the early 70's. The space station's electric power requirements are thus expected to increase into the range where nuclear reactor power systems could provide an attractive alternative to solar power systems based on the results of previous studies.^{(1)*}

To help meet potential requirements of this nature the AEC has been developing and improving the technology base for SNAP reactors and key thermoelectric (TE) power-conversion system components for several years. As of this writing the sixth SNAP reactor (S8DR) is operating at Atomics International (AI) in a long-term endurance test at 600 kilowatts (thermal) and 1300°F. Five previous reactors of this same basic type have successfully demonstrated over 35,000 test hours, and more than 10,000 hr endurance have been demonstrated on all key reactor components. Work on TE power-conversion components has also progressed significantly in recent years. The Westinghouse Astronuclear Laboratory (WANL) has developed compact, tubular TE modules using lead telluride which exhibit high efficiency and stable performance at temperatures of interest for a reactor-thermoelectric system. This technology is now sufficiently advanced to permit serious consideration of reactor-TE systems in the 25- to 50-kwe range for manned or unmanned missions in the mid-70's. Consequently, in late 1967 the AEC started a reactor-TE system engineering and component development program at AI with WANL as subcontractor. Two reports^(2,3) were released after the first optimization studies were complete and a system design point and configuration were established. One describes the reactor and the other describes the complete system.

Based on the results of this initial work, then, a joint power system/spacecraft design and integration study was begun in mid-1968 by the AEC and NASA,

*Superscript numbers in parentheses are for References at end of this report.

Marshall Space Flight Center (MSFC). The objectives of this study were to define the nuclear reactor TE power system's design, performance, and operational characteristics, to evolve suitable integration concepts and to identify potential problem areas and development requirements. AI performed the power system design and analysis work under contract to the AEC, and the integration and mission analysis work was performed in-house by NASA-MSFC.

This report gives a summary of the results of the joint study, starting with the mission and power system requirements, as estimated by NASA-MSFC in mid-1968. The design and performance characteristics of a reference 25-kwe reactor-TE power system are then described, after which the integration and operational aspects of the nuclear-powered space station are presented. The general characteristics of reactor-TE power system designs with power outputs up to 50 kwe using the same reactor and technology base as the 25-kwe system, are also included. Finally, the conclusions based on the results of this work are given. A companion report⁽⁴⁾ describes the system technology status and development plan, and a report being prepared by MSFC provides more detail on the integration aspects with the system.

II. POWER SYSTEM REQUIREMENTS

The design and performance characteristics of most space power systems depend strongly on the mission assumptions, constraints, and requirements. The key requirements affecting the power system design for the NASA space station studied are listed in Table 1.

The desired launch date primarily affects the level of technology upon which the power system design can be based. The two alternate launch modes require that the powerplant be configured to fit within the allowable shroud envelopes for both Saturn and Titan class vehicles, weigh less than approximately 32,000 lb, and be fitted with a suitable docking adaptor for attachment and separation from the station.

The crew size has a direct effect on the electrical power requirements, a more detailed breakdown of which is shown on Table 2. The 25-kwe average unconditioned power requirement is near the upper end of the expected range estimated by MSFC for this 6-man station.

The 2- to 5-yr lifetime requirement is based on total mission cost considerations, which obviously favor longer subsystem lifetimes for semipermanent space stations. The reliability goals reflect a desire for high partial-power and full-power reliability, a factor which influences the redundancy and/or modularity incorporated in the design of the power-conversion system.

The assumed allowable crew radiation dose is based on recommendations made in previous studies,⁽⁵⁾ and is roughly consistent with the 55-rem limit set by NASA⁽⁶⁾ for the shorter-duration Apollo mission several years ago. Finally, it was assumed that end-of-life (EOL) reactor disposal provisions would be required to satisfy nuclear safety requirements, since the 7- to 15-yr orbital decay period from the 270-n.mi orbit would not be long enough to permit sufficient decay of long-lived fission products produced during reactor operation.

TABLE 1
MISSION ASSUMPTIONS AND POWER SYSTEM REQUIREMENTS

Initial launch date	1975
Launch mode	Integral with space station on 2-stage Saturn-V (unmanned) or Separate with Saturn-1B or Titan-3M/ Transtage (unmanned)
Operational mode	Zero-g
Nominal orbit	270 n.mi, 50° inclination
Payload limit (for separate launch)	~32,000 lb
Crew size	6 (9 for short intervals)
Average power level	25 kwe (unconditioned)
Voltage	28 volts, dc (minimum); 42 to 84 volts dc desired
Allowable crew radiation dose (total from space and nuclear power system)	80 rem/6 mo; 100 rem/yr
Design lifetime	2-yr minimum; 5-yr goal
Reliability	Full power 0.96; 75% power 0.98; 20% power 0.990
End-of-life reactor disposal	Orbit boost to ~450 n.mi or Deboost into ocean

TABLE 2
ESTIMATED SPACE STATION POWER REQUIREMENTS

	Watts
Life support	5,000 to 7,000
Lighting (general)	1,400
Thermal control	1,200
Attitude control	1,000 to 3,500
Instrumentation and communications	1,000 to 3,000
Maintenance, checkout, repair	1,000
Airlock/multiple docking adaptor	1,000 to 2,000
Command and service module (quiescent) each	1,000 to 2,000
Experiments (on-board)	2,000
Docked remote modules	1,000 to 3,000
Subtotal	15,600 to 26,100
Contingency 10%	1,600 to 2,600
Total	17,200 to 28,700

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III. SPACE STATION DESCRIPTION

The orbital configuration of the space station is shown in Figures 1 and 2. The design of this station evolved from the Intermediate Orbital Workshop design established in a previous study by NASA-MSFC.

The Multiple Docking Adaptor (MDA) is located at the opposite end from the reactor so that the parked spacecraft are farther from the reactor, subtending less of an angle; thus it causes less radiation scattering into the main station.

The inside of the station is so designed that the compartments where the astronauts spend most of their time are farthest from the reactor. The compartment next to the MDA is the main control room from which all operations are conducted. The next compartment contains the ward room and galley, and is where the crew would spend their off-duty waking hours. It is estimated that the men would spend at least 55% of their time in these first two compartments.

The third compartment from the end contains the sleeping quarters, where an astronaut would spend approximately 30% of his time. The fourth, and last one pressurized, is for storage of spare parts and supplies, and for doing minor repair work. The men would spend no more than 15% of their time in this compartment.

An unpressurized storage area for reserve water, etc. is located next to this fourth compartment. Provision is included for docking of two logistics spacecraft at this point, though no more than one spacecraft would be parked there for any significant time. This spacecraft might supply the gas mixture for breathing if a closed-cycle oxygen recovery system is not ready for the first launch, and serves as an alternate escape route from the station.

The nuclear power system is extended out from one end of the space station to maximize the separation distance between the reactor and the station and to minimize the angle subtended by the area requiring primary shielding. The radiator dimensions were established after studying the trade-offs between separation distance, shield weight, launch height, and vehicle side projected area. By going to longer smaller-diameter radiators the separation distance would be increased and shield weight decreased; however, the launch height and

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Figure 1. Space Station in Orbit

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Figure 2. Space Station Orbital Configuration

Figure 3. Launch Configuration

adverse effect of wind on the launch vehicle would be increased. The radiator length selected was somewhat shorter than the length resulting in a minimum-weight system, and was based on the largest reasonable launch height for a replacement system launched on a Titan III-F.

The nuclear power system can be moved in and out of a thermal/structural shroud on a deployment elevator. The system is guided by rollers on the shroud that engage rails in the elevator and radiator. The elevator is moved by drums attached to it that "crawl" along stationary cables fixed to the shroud.

The thermal/structural shroud serves three purposes. First, it is the structure that supports the reactor and shield during launch and protects the radiator from aerodynamic loads. (The launch configuration is shown in Figure 3.) Second, it is a heat shield that prevents the NaK from freezing during prestart and nonoperating periods. A coating on the shroud has a solar-absorptivity to emissivity ratio that, with the NaK circulating at about 5% flow and solar heating on the shroud, will maintain the minimum NaK temperature above about 75°F. Finally, the shroud acts as the extension structure to establish the reactor/station separation distance during nuclear operation.

During launch the shroud is attached to the aft end of the transition cone between the radiator and lower shield. It is separated at this point by firing of a line-charge once orbit is achieved.

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IV. POWER SYSTEM DESIGN AND PERFORMANCE

A. SYSTEM DESIGN

The power system consists of a zirconium-hydride (ZrH) compact reactor heat source and a lead telluride (PbTe) compact TE converter power-conversion system. The system is designed to produce 25 kw of electrical power. As shown in the schematic flow diagram (Figure 4), the system is divided into four parallel quadrants for redundancy, resulting in high partial-power reliability. Each quadrant consists of two loops, a primary loop that transports heat from the reactor to the hot side of the TE converter, and a heat-rejection loop (HRL) that transports waste heat from the cold side of the TE converter to the radiator. Waste heat is ultimately radiated to space by the radiator. Both heat-transport loops incorporate NaK-78 (a eutectic mixture of sodium and potassium) as the heat-transfer fluid. The fluid loops are pumped by a separate source dc conduction pump. The pump is a dual-throat design providing pumping for both the HRL and primary loop using the same power supply. The power supply consists of three tubular TE modules designed to provide the high-current low-voltage required by the pump. A performance summary of the power system is shown in Table 3.

The design* of the reactor-TE power system is shown in Figure 5. It is 51.7 ft long (excluding the elevator) and the radiator is 11.2 ft in diameter. The weight of the plant is 22,230 lb distributed as shown in Table 4. Most of the weight is concentrated at the forward end of the system in the reactor, shield, and power-conversion equipment. All of the TE converters and pump assemblies are located in the gallery between the two shields; thus the primary loop, which becomes radioactive during operation, is totally shielded from the space station.

B. SUBSYSTEM DESIGN

1. Reactor

The nuclear reactor heat source is based on the technology of several generations of SNAP reactors as described in Reference 4. The latest is the SNAP 8

*The design is presented in further detail by the engineering drawings in the Appendix.

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Figure 4. Reactor Thermoelectric System Schematic (25 kwe)

TABLE 3
REFERENCE ZIRCONIUM HYDRIDE REACTOR-THERMOELECTRIC
POWER SYSTEM PERFORMANCE CHARACTERISTICS

Gross Power Output, kwe			25.2
Voltage, volts			
System			42
Tubular module			14
Thermal-Power Balance, kwt			
Reactor	582.4		
Primary-loop losses	<u>-5.0</u>		
	577.4		
Converters		539.0	
Pump converters		<u>38.4</u>	
		577.4	
Radiator			551.0
Gross electrical power			25.2
Hydraulic power			<u>1.2</u>
			577.4
Temperatures, °F	<u>Inlet</u>	<u>Average</u>	<u>Outlet</u>
Reactor NaK	1044	1145	1246
Converter hot-clad	1225	1125	1025
Converter cold-clad	470	570	670
Radiator NaK	663	563	463
Efficiencies, %			
System			4.33
Converter			4.67
Converter Carnot			35.0
Hydraulics	Total Flowrate (lb/sec)	Pressure Drop (psi)	
Primary loops	13.0	1.87	
Heat-rejection loops	12.4	1.91	
Number of Parallel Flowpaths			
Primary loop			4
Heat-rejection loop			4

TABLE 4
REFERENCE ZIRCONIUM HYDRIDE REACTOR-
THERMOELECTRIC POWER SYSTEM
WEIGHT SUMMARY

	Weight (lb)	
Reactor		1,830
Thermoelectric Converters		1,350
Pumps		410
Pumps and buses	170	
Pump converters	240	
NaK Accumulators		210
Piping		950
Radiator		1,430
Structure		1,210
Radiator	605	
Transition cone	250	
Gallery	105	
Deployment rails	250	
Electrical Wiring		830
Power transmission	410	
Auxiliary pumping bus	285	
Control and Instrumentation	135	
Shields		14,010
Lead shield	1,255	
Lithium-hydride shield	5,535	
Tungsten shield	935	
Uranium shield	6,285	
Total		22,230

Note: 1) All weights of NaK-containing components include the weight of NaK.
2) Shield weights include canning and structure.
3) Integration hardware (thermal shroud, deployment mechanisms, wiring below radiator, etc.) is not included.

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Figure 5. 25-kwe Reactor —
Thermoelectric System

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Developmental Reactor (S8DR) which went on test early in 1969 and is designed to operate at 600 kwt and 1300°F for over 12,000 hr. The Reference Zirconium Hydride Reactor is designed to operate over a wide range of temperature and power levels and can operate completely surrounded by shielding (i.e. a 4π shield). A cutaway view of the reactor is shown in Figure 6 and some of its key design features are given in Table 5.

TABLE 5
REACTOR DESIGN SUMMARY

Thermal Power, kwt	600
Outlet Temperature, °F	1300
Temperature Rise, °F	200
Lifetime, hr (minimum)	20,000
Pressure Drop, psi	0.65
Number of Fuel Elements	295
Peak Fuel Temperature, °F	1420
Fuel Material	U-ZrH
Control-Drum Type	reflector-poison
Number of Control Drums	10
Reflector Material	BeO
Poison Material	Ta-10W
Drum Radius, in.	2.25
Peak Control-Drum Temperature, °F	1520
Peak Drum Bearing Temperature, °F	1350
Coolant	NaK-78
Reactor Weight, lb	1830

The reactor core consists of a triangular pitch array of 295 fuel elements and has an 11.4-in. diameter. The core is surrounded by ten control drums which rotate within drywells. The drywells are surrounded by incoming NaK coolant, thus serving as a sink for the heat generated within the drums. The drums have neutron-reflective material on one side and absorber material on the other, and the reactor is controlled by their rotation which varies the amount of neutron reflection into the core. Since control-drum cooling is provided internally and the neutron-absorber material eliminates the sensitivity of the reactor

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Figure 6. Zirconium Hydride Reactor Reference Design

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Figure 7. Reference Fuel Element

to backscattering from outside the control-drum region, the 4π shield can be located directly adjacent to the reactor.

As in the previous SNAP reactors, the fuel element (Figure 7) consists of an alloy of 10.5 wt % fully enriched uranium in zirconium which is massively hydrided to provide neutron moderation. The hydrogen content of the fuel is 6.3×10^{22} at./cc, which is about the same as in cold water. The fuel rod is contained within a 0.015-in. -thick nickel alloy cladding tube which protects it from the NaK coolant and contains the fission products and hydrogen moderator. Because of the significant permeation rate of hydrogen through the bare cladding, a thin glass barrier is fused to the inside of the cladding tube. Small clearances are provided between the fuel rod and cladding to allow for radiation-induced growth of the fuel.

The control drum (Figure 8) is 4.5 in. in diameter by 18 in. long and consists of BeO reflector material fastened to a neutron-absorbing metal (Ta-10W) which also serves as the main structural member. The drum is supported by self-aligning ball-and-socket-type bearings. The control-drum shaft and the bearing socket are coated with flame-sprayed alumina which provides a low-friction surface when in contact with the solid graphite ball. Drum rotation is produced by a stepper motor operating through an integral 6:1 gear set. In the fully shutdown position the gear teeth are disengaged by a cam lockout device to prevent drum rotation due to launch acceleration.

All parts of the containment vessel and associated structure are 316 SS. The NaK enters the lower vessel plenum at 1044°F through four 1.5-in. inlet lines. It flows upward in the area between the control-drum drywells, turns 180°, flows downward through the core, and exits at 1246°F from the outlet plenum. A minor amount of heat is transferred between inlet and outlet coolant through the flow-dividing cylinder; thus the NaK mixed-mean outlet temperature from the core is a few degrees hotter than the exiting temperature from the vessel.

To obtain optimum fuel-element lifetime, it is necessary to compensate for the core cosine radial power distribution by providing more coolant to the center of the core. This is accomplished by varying the spacing between fuel elements as a function of core radius. The spacing between elements ranges from 0.045 in. in the core center to about 0.010 in. at the edge. The irregular-shaped areas

Figure 8. Reactor Control Drum

between the fuel-element array and the cylindrical core can be filled with solid 316 SS or clad BeO filler pieces to prevent excessive coolant streaming. The inlet-to-outlet pressure drop of the reactor is 0.65 psi under nominal flowrate conditions.

The nuclear characteristics of the reactor are summarized in Tables 6 and 7. A prepoison is used in the core to reduce the reactivity variation during the planned lifetime to less than the \$9.00 control provided by the control drums. Selected isotopes of the rare earths in oxide form are applied by vapor deposition to the outer surface of the fuel rod to provide the necessary poisoning. As nuclear operation proceeds, accompanied by reactivity losses due to uranium burnup, fission-product poisons, and hydrogen loss, the prepoison is consumed and provides reactivity compensation. Tables 6 and 7 are based on Gd-155 prepoison; however, other isotopes such as hafnium-177, europium-151, and samarium-149 could be used and are under consideration. These provide better beginning-of-life (BOL) temperature coefficients or reactivity.

TABLE 6
TYPICAL REACTIVITY SUMMARY
(600 kwt, 1300°F Outlet, 20,000 hr)

Cold Clean Excess Reactivity Available		\$16.00
Gd-155 prepoison loading (cold)	- 8.80	
Tritium converted to He-3 during 2-yr preparation	- 1.30	
Cold Excess Reactivity at Launch		5.90
Beginning-of-Life temperature defect	- 1.20	
Beginning-of-Life power defect	- 0.10	
Xe-135 formation	- 0.70	
Core hydrogen redistribution	- 0.30	
Initial Operating Excess Reactivity		3.60
Hydrogen leakage	- 2.60	
Fuel burnup and fission-product accumulation	- 2.20	
Sm-149 formation	- 1.00	
Additional tritium converted to He-3	- 0.70	
Burnout of Gd-155 (hot)	+ 5.10	
Excess Reactivity After 20,000 hr		2.20
End-of-Life temperature defect	+ 2.20	
End-of-Life power defect	+ 0.20	
Xe-135 decay	+ 0.70	
Cold Excess Reactivity at 20,000 hr		5.30
Remaining tritium converted to He-3	-10.00	
Ultimate Cold Reactivity (~40 yr)		- 4.70

Note: Reactor reactivity, a measure of the energy potential, is expressed in multiples of a dollar, the amount of reactivity that would make the reactor sustain a chain reaction on prompt neutrons alone, if suddenly inserted. Since 0.008 of the fission neutrons are delayed, reactivity (\$) = (potential neutron multiplication factor - 1) ÷ 0.008.

A small concentration of tritium is also added to the fuel to provide an ultimate shutdown mechanism in the event that the control system should fail while the reactor is at power. Tritium decays with a 12.5-yr half-life to He-3 which has a very large (5500-barn) neutron-capture cross-section. During operation, neutron captures in He-3 regenerate the tritium and the rate of poison buildup is

TABLE 7
REACTIVITY CONTROL CHARACTERISTICS

	Drums Out*		Drums In*	
	All	1 Stuck-In	1 Stuck-Out	All
Reactivity Added by Control Drums	\$ 0	\$ 1.00	\$ 7.50	\$ 9.00
Shutdown Margin (Cold)				
Launch pad	- 4.10	Not applicable		
Most reactive point in life	- 1.80	- 0.80		
20,000 hr	- 4.70	- 3.70		
After ultimate He-3 formation (~40 yr)	-13.70	-12.70	-6.20	-4.70
Excess Reactivity (1300°F, 600 kw)				
Beginning-of-life			2.10	3.60
20,000 hr			0.70	2.20
27,500 (defined "end-of-life")			0	1.50
37,500 hr			-	0
Temperature Coefficients of Reactivity	Beginning-of-Life† (¢/°F)		End-of-Life† (¢/°F)	
Fuel temperature	-0.01		-0.10	
Coolant inlet temperature	-0.07		-0.07	
Coolant outlet temperature	-0.03		-0.03	

*The "out" position is least reactive (neutron absorber next to core).

†Change is due to burnout of gadolinium-155.

retarded. As the power level drops at the end of life, the He-3 concentration builds up and provides a positive reactor shutdown within about 10 yr, regardless of the control-drum position. Before operation, the buildup of He-3 constitutes a reactivity penalty which is a function of the storage time.

Table 7 shows that adequate control and energy potential is available to meet the reactor design requirements. Even at the most reactive time during operation, after the period when prepoison burnout overcompensates for reactivity losses, the reactor can be shut down with a comfortable margin with one drum stuck in the most reactive position; yet enough reactivity can be added to

maintain full-power operation for 37,500 or 27,500 hr, if one drum sticks in the least reactive position. It should also be realized that the system power output will fall very slowly even after the control drums are fully rotated, but insufficient reactivity is available for full power. This is shown in Figure 9.

Although reactivity loss is the ultimate life-limiting factor, two other associated life limits have been identified. These are the normal irradiation-induced growth of the fuel, and hydrogen loss sufficient to cause phase transformation of the zirconium hydride. Transformation of the fuel from the Δ to β phase is known to result in accelerated fuel growth. Sufficient clearance is provided between the fuel and cladding to prevent contact due to normal irradiation growth during the reactor lifetime. The larger this clearance, however, the higher the fuel temperature which results in accelerated hydrogen loss and reduces the time to reach fuel-phase transformation. The fuel-to-cladding clearance is therefore optimized to maximize the reactor lifetime. At the power and temperature conditions of this system the fuel-growth/phase-transformation lifetime limit is expected to be in excess of 40,000 hr.

Figure 9.
End-of-Life Power Decay

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The Reference Zirconium Hydride Reactor design offers considerable flexibility in performance over a range of power levels and temperatures. The foregoing reactivity discussion has been based on a nominal reactor design point of 600 kw 1300°F NaK outlet. Figure 10 shows the power/temperature/lifetime trade-off curves for a broad range of interest, including the 25-kwe system design point of 582 kw 1246°F NaK outlet. It should be noted that significant increases in reactor lifetime can be obtained by lowering the operating temperatures by 50° to 100°F.

2. Thermoelectric Converters

Electrical power is produced by the direct conversion of heat to electricity in TE converter assemblies, each consisting of an array of compact tubular modules. Four tubular modules are hydraulically connected to form a converter module or 4-pack. Six 4-packs are then packaged to form a complete converter assembly that produces 6.3 kwe. For the 25-kwe power system, four such converter assemblies are used, all of which are located within the shield gallery. Sketches of the various converter packages are shown in Figure 11.

Figure 10. Reactor Performance Map

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Figure 11. Converter Arrangements

a. Tubular Module

The basic building block for the TE power-conversion system is the compact tubular module. The module design point is the result of a series of trade studies made to minimize power system unshielded weight and minimize radiator area. The module's characteristics are given in Table 8 for the reference condition where the central tube is heated by NaK at an average temperature of 1145°F and the outer cladding is cooled by NaK at an average temperature of 563°F. Clad temperatures differ from these values due to the NaK film drop.

TABLE 8
TUBULAR MODULE CHARACTERISTICS

Electrical		
Output power, watts		262
Voltage at matched load, volts		14
Current, amp		18.7
Temperatures, °F		
Mean hot-clad		1125
Mean cold-clad		570
Efficiency, %		4.67
Materials		
Thermoelectric N-washer		GE-NL PbTe
Thermoelectric P-washer		2P PbTe
Insulation (between washers)		Mica
Inner and outer insulation sleeve		Boron nitride
Outer clad		Stainless steel
Inner clad		Inconel-718
Dimensions		
OD, in.		1.5
ID, in.		0.75
Active length, in.		15
Number of couples		208

Each tubular module consists of an assembly of PbTe washers fabricated from 2P-doped material and 2N-doped stoichiometric material arranged alternately along a central tube, separated by mica washers which act as insulators. Conductor rings span adjacent washers on both their ID and OD to provide a serpentine electrical path through the module. Because of a problem (discussed in Reference 4) of migration of 2P-doped PbTe around the mica at its interface with the hot conductor ring, this ring is split to accept mechanical insertion of the mica. The conductor rings are electrically isolated from the central tube and an outer stainless-steel cladding by thin sleeves of boron nitride which also provide good thermal contact between the cladding and rings. This complete assembly, with appropriate end closures containing electrical feedthroughs, is highly compacted using an autoclaving process into a rugged essentially void-free sealed unit.

A plug with a helical ridge machined onto its OD is inserted in the central tube to increase the NaK velocity and, therefore, improve the heat transfer to the tube. The plug also provides sufficient hydraulic impedance to insure good flow distribution among the tubular modules.

b. Converter Assembly

In assembling a converter, four tubular modules are first manifolded together as shown in Figure 11. Each tubular module is enclosed within a larger tube that provides an annular passage for HRL NaK to cool the outer cladding of the module. A wire wrap is included in this annular to again improve heat transfer and increase the hydraulic impedance of a single tubular module. Differential thermal expansion between the hot and cold loops is accommodated by flexibility in the elbows between the ends of the tubular modules and the headers.

Six 4-packs are manifolded together as shown in Figure 11 to form a converter assembly with the characteristics shown in Table 9. The assembly is mounted within a box structure that serves to isolate potential electrical shorting effects of a NaK leak to the quadrant where the leak occurs. Structural attachment to the gallery structure is at the base plate of the converter assembly with loads coming to the base plate from the individual converter modules by a series of angle structures. This structure is assembled so as to allow thermal differential expansions to occur.

TABLE 9
CONVERTER ASSEMBLY CHARACTERISTICS

Power output, kwe	6.3
Efficiency, %	4.67
Matched-load voltage, volts	42
Number of tubular modules	24
Number of tubular modules electrically in series	3
Wet weight, lb	337
Temperatures, °F	
Mean hot-clad	1125
Mean cold-clad	570
Flowrates, lb/sec	
Primary loop	3.03
Heat-rejection loop	2.90
Pressure drop, psi	
Primary loop	0.5
Heat-rejection loop	0.5

Electrically, three 14-volt tubular modules are connected in series to provide 42-volt output. Eight parallel electrical circuits exit from each converter assembly to deliver power to the space station. As the conductors for these circuits are quite large and stiff, and are only needed when the system is extended, the wires are mounted to the inside of the thermal shroud and radiator with contact being made only when the elevator is fully extended.

3. Pumps

a. Pump Converter

Three special tubular modules are used for the power supply for each of the four pump assemblies. These special modules have only four TE couples each to provide the high current at low voltage required by the pump. The three modules are connected electrically and hydraulically in parallel. The module assembly is also connected hydraulically in parallel to a main power-converter assembly, and consequently operates at the same temperature conditions. Characteristics of the pump converter are given in Table 10.

TABLE 10
PUMP CONVERTER CHARACTERISTICS

	Tubular Module	Pump Converter Assembly (3 modules)
Electrical		
Output power, watts	100	300
Voltage at load conditions, volts	0.166	0.166
Current, amps	600	1800
Temperatures, °F		
Mean hot-clad	← 1125 →	
Mean cold-clad	← 570 →	
Efficiency, %	← 3.1 →	
Materials	Same as power modules (Table 9)	
Dimensions		
OD, in.	2.2	-
ID, in.	0.75	-
Active length, in.	7.0	-
Number of couples	4	-
Wet weight, lb	-	60
Flowrates, lb/sec		
Primary loop	0.073	0.22
Heat-rejection loop	0.067	0.20

b. Pump Assemblies

The pumps used for both the primary loop and HRL's are dc conduction pumps powered by the special TE converters just described. As with the electrical generating converters, the pumping packages have no moving parts and their hydraulic power increases automatically as the reactor operating temperature increases. Four such pumps are used in the 25-kwe power system, one for each quadrant.

The pump assembly is shown in Figure 12 and its characteristics are summarized in Table 11. The assembly contains two throats, one for pumping the

PRINCIPLE OF OPERATION

Figure 12. Pump Assembly

TABLE 11
PUMP CHARACTERISTICS

	Primary Loop	Heat-Rejection Loop
<u>Throats</u>		
Flowrate, lb/sec	3.25	3.10
Pressure rise, psi	2.25	2.10
NaK temperature, °F	1.044	663
Current, amp	1,800	1,800
Throat dimensions, in.		
Height	0.5	0.5
Width	0.98	0.98
Length	2.50	2.50
Wall thickness	0.025	0.025
Magnetic flux, lines/in. ²	15,000	15,000
NaK resistivity, μ ohm-in.	29.0	21.6
Wall resistivity, μ ohm-in.	43.0	37.5
<u>Assembly</u>		
Thermal shunting between loops, watts thermal		500
Weight, lb		
Pumps and buses		42
Pump converters		60
Total		102
Dimensions, in.		
Length		10
Width		11
Height		8

primary loop and one for the HRL. The two identical throats are connected electrically in series. The primary-loop throat is in the cold part of that loop and the HRL throat is in the hot part of its loop to minimize thermal stresses as well as thermal-shunt losses between throats. The copper bus between the throats and connecting the throats to the pump converter is made of space flat strips for flexibility to accommodate differential expansions. This bus has an average cross-sectional area of about 2 in.²

The primary-loop pump can produce a net pressure rise of 2.25 psi at the design flowrate of 3.25 lb/sec, and the HRL pump 2.10 psi at 3.10 lb/sec. The calculated pressure drop in the primary loop is 1.87 psi and in the HRL it is 1.91 psi, resulting in an adequate design margin. Although the pump assembly components are relatively inefficient, only 6.6% of the reactor power is required for all system pumping.

Electrical taps are included on the pump bus, as shown schematically in quadrant No. 2 of Figure 4, so that current can be supplied for auxiliary pumping during nonoperating periods. Before startup or any time the system is not operating, 70 amp will be supplied to the taps on either side of each HRL throat resulting in 5% flow in that loop. Electrical current, shunting through the pump converters and primary-loop throats, results in about 2% reverse flow in the primary loop. These flowrates are sufficient to eliminate unacceptably cold regions in the system when only solar heating through the thermal shroud is present. The power required, which is nearly all consumed in the conductors between the station and pumps, is 330 watts at 1.2 volts.

During the startup period the current is switched to the primary-loop throats so the flow conditions are reversed. The 5% forward flow in the primary loop is important in limiting the first reactor temperature transient, as it quickly causes the negative coolant-inlet temperature coefficient to be affected by the increasing temperature. Since the auxiliary pumping is only needed when the system is retracted, the circuits are designed to open when system extension begins and no extendable wiring is required.

4. Radiator

The radiator provides the necessary area to reject the waste heat from the power system to space. Figure 13 depicts the radiator and gives its key dimensions.

The basic unit of the radiator is the diffusion-bonded aluminum-fin/stainless-steel tube combination. The AI also provides meteoroid armor for the tube. Fifteen of these finned tubes are assembled and hydraulically connected into panels approximately 4 ft by 18 ft. The complete radiator is made of 16 of these identical panels, four in each quadrant. The hydraulic arrangement is such that each quadrant feeds two panels on one side of the radiator and then two on the

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Figure 13. Radiator

TABLE 12
RADIATOR CHARACTERISTICS

Radiated Power, kw	
Prime radiator	461
Transition cone	40
Deployment rails	40
NaK headers	10
	<hr/> 551
Temperatures, °F	
NaK inlet	663
NaK outlet	463
NaK film drop	5
Hydraulics	
Flowrate per quadrant, lb/sec	3.1
Pressure drop (NaK tubes only), psi	0.28
Meteoroid Protection	
Earth orbit, n. mi	270
Mission life, hr	20,000
Nonpuncture probability	0.99
Fin Effectiveness	0.85
Areas, ft ²	
Prime radiator	1,160
Total effective	1,390
Environmental Parameters	
Radiator orientation	Normal to sun and earth
Solar absorptivity	0.3
Surface emissivity	0.9
Earth absorptivity	0.9
Earth albedo	0.35
Weights, lb	
Prime radiator	1,430
NaK and tubes	515
Armor	360
Fins	555
Structure	605
Total	<hr/> 2,035

Figure 14. Radiator Weight vs Area

opposite side so that all quadrants receive roughly the same heat input during nonoperating periods regardless of the sun orientation.

The panels are connected to the radiator structure by flexible brackets that allow for differential thermal expansion. The basic structure is of the semi-monocoque type with a stressed skin, frames, and longitudinal stiffeners, and is fabricated from a titanium alloy. It is designed to carry its own weight and to support the reactor and shield during operation in space including the firing of the EOL disposal rockets. It does not support the reactor and shield during launch; however, the loads are carried through the thermal shroud surrounding the radiator.

A summary of the radiator characteristics is given in Table 12. Note that the radiator with its headers only rejects 471 kwt since 80 kwt of the total heat-rejection requirement is rejected through the shield-support transition cone and the deployment rails. For comparison purposes the total effective area (the area required if all heat were rejected through the radiator) is also shown. The effective specific area is 55 ft²/kwe.

The radiator design includes a trade-off between weight and area. Such a trade-off is shown in Figure 14 which shows the minimum radiator weight as a function of area for cases with and without the supporting structure. The curves include the results of a separate trade study to determine optimum tube spacing for minimum weight (a trade-off of the tube, NaK, fin, and armor weight) and include the effect of heat being lost from the fins to the deployment rails and the transition cone. The 1160-ft² area was chosen at a penalty of 200 lb over the minimum weight point to save 240 ft².

5. NaK Accumulator

The primary loop and HRL's are designed to be void-free at all times. Accumulators are used in the loops to accommodate the expansion of the NaK from the "cold" shutdown condition to the "hot" operating condition, and to maintain the system at the nominal 20-psi operating pressure. A total of eight identical accumulators are used, four in the primary loops and four in the HRL's.

The principle of operation of the accumulator, shown in Figure 15, involves two gas-backed bellows assembled in series. Two sets of series-stacked bellows are mutually opposed. The inner surface of the primary bellows and the

Figure 15. NaK Accumulator

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bellows end closures provide containment for the NaK while the space between the bellows and the cylindrical outer shell is evacuated and constitutes the secondary containment in event of a NaK leak through the primary bellows. The inner volume of the secondary bellows contains argon gas for system pressurization. The opposed bellow design provides partial system pressurization in the event of loss of pressure from one gas chamber.

Table 13 gives characteristics of the NaK accumulator. With increase of system temperature, the NaK expansion displaces the primary bellows and the pressure within the secondary bellows is increased by the pressure-volume-temperature relationship. The units are designed so that even with loss of a

TABLE 13
NaK ACCUMULATOR CHARACTERISTICS

Condition	Primary Loop		Heat-Rejection Loop	
	NaK Volume (in. ³)	Pressure (psi)	NaK Volume (in. ³)	Pressure (psi)
Isothermal at 20°F	38	4.0	14	4.0
Isothermal at 100°F	52	4.8	50	4.8
Normal operating temperatures	260	20.0	282	20.0
Maximum expected conditions	284	22.5	314	23.8
Maximum NaK in accumulator	328	29.0	328	26.0
Design operating temperature, °F			750	
Materials				
Bellows			AM-350 (SCT 850)	
Outer cylinder, end caps, tubes			304 stainless steel	
Dimensions, in.				
Outside diameter			10.0	
Height			12.3	
Dry weight, lb			26

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Figure 16. Shield

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Figure 17. Shield Weight vs Separation Distance
(Constant Side Doserate, 200R/hr at 100 ft)

primary bellows or of gas from one chamber, enough pressure is maintained in the cold shutdown condition to keep the loops void-free. Because of the commonality of the four primary loops, at least four of this type failure could occur before a pressurization problem would occur. Since the maximum NaK vapor pressure in the reactor core is only about 10 psi during operation, failures are less critical for the hot condition than the cold.

6. Shield

Approximately two-thirds of the system weight is in the radiation shield assembly, a combination of heavy-material gamma shields and cast LiH shields for neutron attenuation. The shield assembly itself will be described here and the discussion of the effects of the shielded radiation environment on the astronauts and photographic film will be presented in Section VI. As shown in that section, the reference 14,010-lb shield (Figure 16) meets all system requirements by limiting the direct-doserate in the space station to 1.85 mrem/hr and the doserate to the side of the reactor to 200 rem/hr at 100 ft.

While a specific shield design is described in this report, large changes in requirements could be tolerated without severely modifying the weight picture. Figure 17 shows the shield weight sensitivity to the direct-doserate and to the separation distance selected. Note that the doserate could be reduced by a factor of four for less than 2000 lb or by increasing separation distance by about 25 ft.

The two significant types of radiation emitted by the core are neutrons and gamma rays. In addition, there is significant gamma radiation from the radioactive primary-loop NaK and from neutron capture in structure and shielding. The latter is referred to as secondary gamma radiation.

Because of the relationship between neutron levels and secondary gamma production, the optimum selection and placement of the shielding materials to attain a shield of minimum weight constitute a rather complex trade-off. The configuration resulting from optimization studies for the space station application is that shown in Figure 16. The necessity of shielding the space station from the primary-loop NaK results in the placement of all primary-loop components in a "gallery" region. The massive uranium gamma shielding immediately following the gallery region reduces the space station gamma dose levels to

those consistent with requirements for continuous occupancy. No shielding of gallery components is needed in the radial direction because of their relatively small contribution to the higher doses allowable during rendezvous maneuvers.

The gamma shield immediately below the reactor is lead, molten during operation, contained in a Croloy 2-1/4 container. The use of Pb proves to be advantageous over other gamma-shield materials such as W and depleted U because of its low secondary-gamma production in the high neutron-flux environment near the reactor. The gamma shield is followed by a LiH neutron shield which is thermally insulated from the gamma shield to maintain temperatures less than 1100°F (the LiH melting point is 1270°F). Shielding thickness and contours are optimized to produce the minimum shield weight to attain the required dose levels. In the radial and forward directions, the higher allowable doserates permit the use of W and steel gamma shielding. The reduced neutron flux and temperature levels at the gamma shield below the equipment gallery permit the use of depleted U. This gamma shield provides the added shielding required for the reactor source plus that for the induced radioactivity in the primary-loop NaK coolant. The final shielding region is LiH.

The shield provides the structural support for the reactor. Launch loads are carried through aerospace-type shell-and-stringer structures contained within the cast LiH shields to rings that mate with the reactor, gallery structure, and structural support cone. The cast LiH is supported within the shield structures by a combination of mechanical bonding to the internal structural members and by transferring loads through the shield casings. Gamma shielding is supported directly on the primary structural members.

The highest gamma-shield temperatures are in the Pb shield immediately below the reactor. As the heat generated within this shield must be thermally radiated to the reactor outlet plenum, the temperature stabilizes at about 1350°F. Sheets of glass cloth sandwiched between thin sheets of stainless steel provide insulation between the Pb shield and the LiH. Approximately 5 kw of energy is deposited in the radial gamma shield adjacent to the reactor; however it can radiate to the 1044°F vessel wall, transferring the heat to the NaK coolant, so the peak temperature is 1190°F. This shield is insulated from the radial LiH shield.

The maximum LiH temperature, 1100°F, occurs in the radial shield. The thinner adjacent gamma shielding results in considerably higher heat-generation rates in this LiH. The outer shield surface has a high emissivity coating to aid heat dissipation. In the final design, LiH hydride temperatures may be reduced by using the low heat-generation characteristics of isotopic lithium-7 hydride. A full scale irradiation of this material is currently in progress in the S8DR.

The heat-generation rates in the shield below the gallery are very low and the shield temperature is governed primarily by the thermal environment of the radiator and gallery components. The maximum temperature is about 600°F.

C. SYSTEM OPERATION

The TE system is simple to operate since only movement of the reactor control drums is required. Pumping and electrical power generation automatically increase as the reactor temperature and power respond to the control-drum insertion.

1. Control System Description

The system can be primarily controlled on reactor temperature or electrical power output, depending partially on the power-conditioning scheme employed. In either case it is desirable to use the TE converter output as the control signal (the converter is, in fact, a large thermocouple) because: (1) it is sensitive to changes in temperature; (2) it has a large amount of redundancy; (3) it will obviously be supplying the signal as long as the plant is still useful; and (4) no special high-temperature circuits are required. If the converter output voltage is regulated at a constant value, power (current) output can be used directly for control. If, however, the output voltage and current are allowed to vary at fixed operating conditions, the plant must be controlled on primary-loop temperature. The converter characteristics must be understood to see how it can be used in this case.

Electrically the converter looks like a battery and resistor in series. The capacitance and inductance terms in an electrical analog of the converter are very small. Its output for a fixed operating point is as shown in Figure 18. The power-conditioning scheme selected by NASA uses load-following units that accept variable voltage and deliver 28 volts dc. The power is delivered at

EQUATION FOR OPEN-CIRCUIT VOLTAGE:

$$V_{oc} = V + IR_{INTERNAL}$$

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Figure 18. Thermoelectric Converter Output

42 volts dc when operating at converter matched load, but, as less power is demanded from the converter, its putput voltage increases. As the load approaches zero, the delivered voltage approaches the converter open-circuit voltage (V_{oc}) of 84 volts.

The thermal/hydraulic parameters of the system are little affected by changes in power demand on the converter over the entire voltage range, so if the control system maintains a fixed V_{oc} , temperatures and reactivity requirements will not significantly vary. With the selected power conditioners, then, a simple circuit that measures output voltage and current and computes V_{oc} is used for prime control. Any one of the power delivery circuits can be used for this control without taking it off the line.

The control system is depicted schematically in Figure 19. It uses three input signals in addition to the V_{oc} signal for control. First, a drum step-counter maintains an indication of drum position using the current pulses to the drum actuators as a basis. Past experience has shown that if the actuators draw current a step will actually take place. The drum position indicator does maintain a count of missed steps if it senses that an actuator receives a voltage pulse but does not draw current. Second, a signal is received from a resistance temperature detector located on the primary loop. This is used in a bridge circuit for gross temperature control during the startup and as an over-temperature cutoff signal. Third, signals showing whether the power plant is extended or retracted are used.

The control logic is provided by a 800 in.³ electronic package, weighing about 40 lb, located in the space station. The total power requirement for the control system, including auxiliary pumping, is given in Figure 19. The electronic package consists of nine (five different kinds) replaceable modules. These include: (1) two logic, timing, and sequencing units (one active, one standby); (2) a selector unit to select between them; (3) three independent power switches, each capable of handling the full power load to the actuators; (4) two drum-position indicators (one active, one standby); and (5) a checkout unit used to check the other units and provide telemetry signals.

Further detail is given in the wiring and control-system schematics in the Appendix. The wiring used in all high-temperature areas is woven fiberglass insulated wiring with woven stainless-steel armor both for EMI shielding and mechanical protection.

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Figure 19. Control System

The operation of the control system is best understood by going through a normal sequence.

2. Control Sequence

Throughout the launch and prestart period the nuclear power system is dormant except for NaK flowing at 5% in the HRL's and 2% in the primary loops. This flow is maintained by electric current supplied to taps on the main pumps (described in Section IV-B) to preclude local NaK freezing.

Before startup, settings are established for the desired operating point and the drum position at which the second reactivity insertion rate is to be started. This latter setting varies a few degrees over the life of the reactor and any one setting is good for a few months. Startup is initiated by a single command that energizes the control system.(see Figure 20).

During the first phase of startup the drums are stepped in sequence at 3-sec intervals until they reach the preset drum position, approximately 105° from full-in. This phase reduces the shutdown margin to about 25¢ of reactivity

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Figure 20. Reactor - Thermoelectric System Startup

and takes about 30 min. When the insertion rate is changed the auxiliary pumping current is also switched to the primary-loop throats so that they provide 5% forward flow and the HRL flow drops to 2% in the reverse direction.

During the second phase of startup, the approach to sensible heat, a 150-sec drum-stepping rate is used. The reactor goes critical at about 90 min into the startup; however, the power level does not reach the thermal kilowatt range for approximately 30 more minutes. As excess reactivity is inserted throughout this time with no negative temperature feedback, a power transient occurs. It is the allowable magnitude of this transient that dictates the stepping rate during the approach to sensible heat. Higher primary-loop flowrate or an increase in the negative fuel temperature coefficient of reactivity could allow an increase in the reactivity insertion rate during this period, which takes about 2 hr.

When the signal from the resistance temperature detector on the primary loop shows the temperature is above 310°F, reactivity insertion and auxiliary pumping are stopped. At this time the control system signals the plant extension system that deployment can be started. It is assumed that deployment takes 15 min.

Once extended, the 310°F temperature cutoff limit is raised to 1325°F and the stepping period is changed to 30 sec. When the present operating temperature is reached, as indicated by the V_{oc} computer, reactivity insertion is again stopped and the plant is maintained at this condition within a small deadband. The total startup time to the full 25-kwe operating point is approximately 4 hr.

During operation, only very limited monitoring of the plant is required and this can be done either from the space station or on the ground. Since the electronic control package is modular in design, has built-in redundancy, and incorporates self-checking circuits, a periodic check can be made to determine if redundancy has been lost; if so, it can be restored by substitution of spare modules. It is important to note that, even if the control system were turned off or lost power, the output of the power system would not change significantly for a period of weeks (Figure 9).

If a power setting change is desired, only a readjustment of the open-circuit voltage setting is required. The control system will automatically cause the output to seek the new level at the 30-sec stepping rate. It is not expected that setting changes will be made more often than every few weeks.

A single shutdown command causes the drums to step outward at the 3-sec stepping rate. The neutron radiation drops to about 1% in 3 min at this rate and in about an hour the reactor-outlet temperature will cool to 290°F. At this time the control system will restart the auxiliary pumping and send a signal to start power plant retraction. These are automatic safety features to prevent NaK freezing.

Restart is simply a repeat of the initial startup and is accomplished by repeating the single startup command. The setting for the drum position at which the reactivity insertion-rate change is made can be easily determined using the position of the drums just before previous operation was terminated.

D. SYSTEM CHARACTERISTICS

1. Reliability

The system reliability is shown as a function of power level in Figure 21 for a mission time of 20,000 hr. Although the system is rated at 25 kwe, the full power capability shown is 105% of the nominal rating. This is the initial power attainable at a reactor outlet temperature of 1270°F. (The 15% additional power available at the maximum reactor outlet temperature of 1325°F is allocated 5% for random failures and 10% for converter degradation.)

The reliability evaluation technique has been used as a tool to define critical areas in the system design and to establish when redundancy, etc., should be incorporated. The results of the evaluation have not been demonstrated by test and the values are not considered extremely accurate. They are based, however, on reasonable failure rates for similar components or welds and should be good order-of-magnitude estimates.

The incremental power losses to a level of 75% capacity are attributed to module failures in the converter. A large incremental change in reliability occurs below 75% power due to the allowance of one quadrant failure in the heat-rejection system. These individual effects are shown in the figure for the three main subsystems.

It can be seen that the limiting reliability for the system is the reactor and primary loop. The reactor itself is estimated to have a high reliability of 0.9978. This is attainable largely through high margins on fuel performance

SYSTEM RELIABILITY ESTIMATE

Figure 21. System Reliability

and redundancy of control drums. The primary-loop reliability is based on the leaktight integrity probability estimates for critical areas such as piping welds.

The principal reliability consideration for the heat-rejection system is also the leaktight integrity of the loops. Each of the four loops has a reliability of 0.990.

The reliability of the converter electrical system is calculated on the basis of various combinations of converter module conditions. Each module is considered to have three potential failure modes consisting of open-circuit, short-to-ground, or internal short. The influence of a module failure on system power depends on the particular location of the module in a three-module series. Each step in the reliability curve for the converter represents the power loss associated with a module failure.

The reactor control subsystem is not included in the system reliability evaluation. This portion of the system is located within the manned portion of the space station and is readily accessible for maintenance.

2. Off-Design Performance

As previously discussed the power setting can be adjusted to any level below the 25-kwe design level. Figure 22 shows how the thermal and electrical power decrease with decreasing reactor-outlet temperature. Reducing temperature and power has a marked effect on reactor life, also shown in Figure 22. At 25 kwe the predicted reactor life is approximately 4 yr. Reducing the power to 20 kwe, for instance, lengthens the lifetime to over 6 yr. These lifetimes are based on the reactor performance data shown in Figure 10.

This characteristic of the system is extremely important if the required power level at the beginning of the space station mission is less than the power requirement later in life for which the system is designed. An increase in total lifetime to well over 5 yr is possible if, for instance, the power level during that period gradually increased from 15 kwe to the 25-kwe design point.

Figure 22. Thermoelectric Power System Performance at Constant Voltage

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V. MISSION OPERATIONS

A. FACTORY-TO-FLIGHT ACTIVITIES

A series of tests, including nuclear critical experiments, will be performed on the power system before launch; however, since the nuclear power level will be limited to less than about 100 watts thermal, no significant radioactivity will result. Access to the reactor area and integration of the system will then be as with any other non-nuclear payload. Some detail on the preflight testing is given in Reference 4.

Shipment of the system from California may be by air or surface transportation. This operation will be affected by whether the complete radiator is assembled and attached before or after the shipment. During transportation and possibly during storage or certain handling operations the reactor control drums will be replaced by neutron poison material to prevent any possibility of accidental criticality. This prevention of accidental criticality during all pre-launch activities, both by design and by administrative procedure, is the basic safety approach and is based on standard procedures used in the nuclear industry.

The only other special precaution required is to maintain the system in a warm, dry, relatively clean environment. This is to prevent the high-temperature wire insulation from absorbing moisture and reducing its insulation to ground, and to prevent corrosion of some of the high-temperature materials that may be sensitized. The wire insulation is not damaged by moisture but problems or delays during checkout will result if the humidity is too high. On the launch pad a source of warm dry filtered air will be required to blow over the system.

B. LAUNCH

The space station and nuclear power system are designed to be launched together, unmanned, on a two-stage Saturn-V vehicle. Since no significant fission-product inventory is present, no severe safety problem exists. Should the reactor fall in water on an aborted launch, the worst possible result has been determined by test to be an excursion about equal to one pound of TNT

exploding. The release of fission products is small and the effect is localized. The required exclusion distance for the SNAP 10A reactor, for instance, was less than that required around the Atlas-Agena launch vehicle.

In the launch configuration, with the space station replacing the S-IVB stage, the total stack is several feet shorter than the standard Apollo vehicle with the launch escape tower. The side-projected profile of the stack is such that it could be launched during 95% of the month of March, the windiest month at the Cape. Subtracting a conservative space station weight estimate of 100,000 lb from the vehicle payload capability of 186,000 lb leaves at least 86,000 lb for the power system. There is obviously no weight problem for this launch mode.

A method of replacing a complete power system in orbit was also defined and is shown in Figure 23. Should replacement of the powerplant be necessary, either because of a failure or as a planned operation to extend the station life, the spent power system would be separated from the elevator at the fully extended position. The replacement system would be launched inverted on a Titan III-F (a Titan III-M with a Transtage) as shown in Figure 23. The height of this stack is approximately the same as the reference Titan III-M/MOL/Gemini vehicle. An analysis of the configuration by the Martin Marietta Corporation showed that 32,200 lb could be placed in the 270-n.mi 50°-inclination orbit. In this analysis the maximum dynamic pressure was constrained because of the size of the payload fairing. This is about the actual weight of the payload, including a shroud, nose cone, structural adaptor, docking mechanism, and the complete EOL disposal package that weighs over 4000 lb.

The transtage would be used to achieve an unmanned rendezvous and the new power system would be docked into the extended elevator. Athermal/aerodynamic shroud surrounding the radiator would be jettisoned during the initial startup of the new power system, but for subsequent shutdowns the system would be retracted into the original shroud.

C. ATTITUDE STABILIZATION

Once in orbit, the space station is rotated to a vertical or gravity gradient orientation and the reactor is started. The gravity gradient attitude mode was found to require the smallest and simplest attitude control system, and is

Figure 23. Nuclear Power System Replacement Launch Mode
Unmanned Rendezvous and Docking

compatible with the communication system. The directional antennas aimed at the ground, relay satellites, or the remote experiment modules can be controlled more simply when the station maintains a constant Earth-oriented attitude. It is anticipated that the remote modules will have short-range omnidirectional transmitters and their main link to the ground will be through the station communication system.

The gravity gradient mode is also compatible with a typical experimental program. Four types of experiments are projected: Earth-oriented, sun-oriented, stellar-oriented, and unoriented. The stellar-oriented experiments require very accurate pointing and could be disturbed by even small oscillations of the space station. These are probably best performed from a stable remote module. The sun-oriented experiments are inherently suited for remote operation since they can be efficiently powered by a small solar cell system without an additional orientation requirement. The gravity-gradient-stabilized station fits the program nicely, then, as a stable base for the Earth-oriented experiments. The unoriented experiments can, of course, be carried out on the station regardless of attitude.

Though the gravity gradient mode was selected by NASA, attitude-control system requirements for all of the modes shown in Figure 24 were analyzed. While the nuclear power system is compatible with an artificial-g rotating-station approach, only zero-g approaches were considered.

The gravity gradient torque tending to restore the station to the local vertical is shown in Figure 25. With restoration torques up to 18 ft-lb it is clear that maintaining any attitude other than near the local vertical or horizontal will require a large attitude-control system.

If the station's principal longitudinal axis is maintained on the local vertical or with this axis perpendicular to the orbital plane (horizontal), only 3000 ft-lb-sec angular momentum storage capability is required. This is only to provide adequate oscillation damping and can be accomplished with one set of Control Moment Gyro's (CMG's) like those used on the Apollo Telescope Module (ATM). These weigh approximately 1200 lb and provide 6000 ft-lb-sec storage.

Figure 24. Orbital Modes

Figure 25. Gravity Gradient Restoration Torque vs Offset Angle

If the Earth-resources experiments are not gimballed and the station longitudinal axis must accordingly be on the local vertical during the daylight part of the orbit, 7500 ft-lb-sec of angular momentum storage is required. This added momentum storage is to resist torques resulting from the station longitudinal axis being up to 3.5° off the principal axis with Command and Service Modules (CSM's) parked unsymmetrically on the station. The most severe requirements result if the station must be held in an attitude that makes an angle of 45° with the orbital plane. If this is required only during the day half of the orbit, 38,000 ft-lb-sec of momentum storage or seven sets of ATM-size CMG's would be needed. In the foregoing cases, only about 1000 lb of Reaction Control System (RCS) propellant per year would be required as the torques on the CMG system can be reversed for half the orbit to eliminate any significant net torque requirements. If the 45° case is maintained continually and angular momentum is compensated by the RCS jets once each orbit, 11 sets of ATM-size CMG's would be required and propellant would be expended at the rate of 40,000 lb/yr.

This analysis confirmed the conclusion that the most practical zero-g space station would be designed to operate in the gravity gradient mode. It further showed a significant reliability and safety advantage for operation in this mode. Even with the CMG system inoperative, the station is extremely stable and the maximum postulated disturbances can only cause small oscillations.

D. NORMAL SPACE STATION OPERATIONS

On the second day of the mission, after it is verified that all systems are operating and the station is habitable, the first three men would be launched to rendezvous with the station. These men would be in an Apollo CSM.

Additional groups of three men would be launched at approximately 3-mo intervals. Within a few days after the third group arrived, the first group would return to Earth. Other than during the brief overlap periods, then, the station would be occupied by six men. Each three-man crew would bring a logistics module or a new experiment module with it. After docking this module they would disconnect their CSM and move to another docking port so they could enter the station.

The experiment modules would be checked while attached to the station and then moved to a position in the same orbit as the space station but preceding or following it by a few miles. This might be accomplished unmanned or it could be done using the CSM as a tug. Throughout the mission the remote modules would be periodically returned to the station for such things as retrieval of film, checkout, and minor maintenance.

The remote module maneuvers are shown schematically in Figure 26. The modules will actually be placed in a slightly higher orbit, about 2 mi behind the station; therefore they will have a lower velocity and will slowly move away from the station. Having a higher drag coefficient, however, will cause them eventually to fall to a lower orbit and begin to gain on the station. When they have closed to two miles, they will be boosted to their starting position to repeat the cycle. A typical close-in cycle was used for radiation dose calculations. In this cycle the module moved from 2 mi out to 11 mi and back in a three-day period.

Figure 26. Remote Module Maneuvers

The remote module will be returned to the station about every 60 days for retrieval of film, etc. With up to four modules this type operation will occur three to twelve times in a 6-mo period. In the mission model used for radiation dose calculations, twelve times was assumed and, to be conservative, it was assumed that each module deployment and retrieval operation was done with a manned CSM. It was assumed that all rendezvous with the station were accomplished by approaching at 5 ft/sec to within 50 yd and then 1.5 ft/sec to contact. Departures were assumed to be at twice these rates.

E. END-OF-LIFE DISPOSAL

The major safety problem associated with the use of a nuclear reactor power system for the Space Station mission is the satisfactory disposal of the reactor after sustained operation in orbit. If the nuclear power system were merely abandoned in orbit at the end of its useful lifetime, random reentry would occur within approximately 5 to 15 yr. Although the activity of relatively short-lived fission products in the core would decay significantly during this period, the longer-lived fission products (notably Sr-90 and Cs-137) would still constitute a potential radiation hazard.

Over the past several years this problem has been studied extensively under the AEC's Aerospace Safety Program, using analysis and experimental tests to quantify the potential hazards and risks and identify plausible safeguards. This work has included reentry and arc-jet tests on bare (unshielded) reactors to determine the degree of burnup or disassembly which would occur, destructive reactor transient tests to determine the results of sudden immersion in water, long-term fuel-element corrosion tests in water, etc.

The results of previous nuclear safety work are currently being re-evaluated and the safety aspects of the EOL phase of the Space Station mission are being analyzed in depth under a separate AEC study being conducted at AI. A preliminary safety evaluation report will be completed in January 1970 for review and evaluation by the AEC and possibly other government agencies involved in the approval chain. A more complete preliminary safety analysis report which covers all phases of the mission is scheduled for release by June 1971. Since this work is currently in progress as a separate study effort, a detailed discussion of the rather complex safety problem will not be attempted here. Tentative conclusions which can be made from the results obtained to date, however, are as follows:

- 1) The potential hazards due to random reentry of the reactor after sustained operation in orbit are significantly reduced by the presence of the large, 4π shield. Burnup of the reactor and/or disposal of the fuel elements during reentry cannot occur due to the protection offered by the LiH neutron shield.

- 2) The maximum potential hazard due to random reentry appears to be that of direct radiation exposure upon approaching the spent core if it has impacted on concrete or hard rock. The area that would be potentially unsafe (i.e. could result in a dose of ~25 rem in one hour) is of the order of 100-ft radius. Impact on soil would result in deep burial, due to the high terminal velocity.
- 3) Corrosion rates of the fuel in water are extremely small, and potential contamination of reservoirs would thus be orders of magnitude less than that assumed in previous safety analyses for the SNAP 10A system.

As a result of the above observations it is currently believed that the risks associated with random reentry after sustained operation in orbit are acceptably small. In this study, however, it has been assumed that an EOL reactor disposal system would be required for the Space Station mission. Two approaches are being considered; controlled deboost into deep ocean areas, and boost into a 400- to 450-mi orbit where the orbital decay time is sufficient to allow the fission-product activity to fall to a safe level. The required ΔV 's for these two disposal modes are roughly comparable from the nominal 270-n.mi operational orbit; the orbital boost maneuver could be provided by the Command and Service Module.

The disposal package shown in Figure 27 was incorporated into the design to provide a controlled de-orbit capability. Disposal is accomplished by separating the system from the deployment elevator, orienting for the retrofire, spinning it to about 20 rpm for stability, and firing two of the three solid rocket motors attached to the radiator. One of the rockets is redundant. During the life of the power system these rockets will be maintained below 150°F and they will receive less than 1000 rads of gamma radiation. Though little information is available on long-term storage of solid rockets in space, it is expected these conditions will be satisfactory.

The rockets will impart a velocity change of 600 ft/sec to the power system, sufficient to reduce the possible error in splashdown point to less than ± 60 n.mi. This is extremely conservative since almost any spot in the oceans, off the continental shelves, is considered to be an acceptable disposal site.

Figure 27. Reactor Deorbit System Concept

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VI. RADIATION ENVIRONMENT AND EFFECTS

The radiation environment created by the shielded nuclear reactor is shown in Figure 28. The doserate inside the space station at the effective separation distance, considering the occupancy times for the different compartments, is 2.8 mrem/hr or 25 rem/hr. The two significant sources are the direct radiation and that scattered off the CSM parked at the forward port. During a rendezvous a spacecraft will receive an integrated dose of 0.4 rem. These radiation levels, coupled with the natural space radiation, were evaluated for their effect on both astronauts and film.

A. ASTRONAUT DOSE EVALUATION

The dose evaluation for astronauts is summarized on Table 14. Two types of astronauts were considered, the astronaut-pilots and the astronaut-scientists. The astronaut-pilot may depart and return to the station as many as twelve times while taking part in remote module deployment and retrieval operations during his 6-mo stay on the station. As shown on the table he receives slightly less radiation from the reactor than he does from normal space radiation. His total radiation exposure is about one-half the allowable dose used for planning (see Section II). Of the second type, the astronaut-scientist, at least one might remain on the station for a full year. He would not be assisting in the remote module operations, however, so he would make only one rendezvous with the station. Only 37% of his dose would be from the reactor, 63% coming from ambient space radiation. As the allowable planning dose is increased to 100 rem for a year, he would receive only about two-thirds of the allowable.

B. FILM DOSE EVALUATION

The effect of the reactor radiation on the most sensitive film expected to be used on the station, Panatomic-X, is summarized on Table 15. Only the gamma dose is listed since the effect of the neutrons is very small by comparison. The sequence considered involves the film's coming to the space station, being stored in a vault surrounded by 1.8 in. of Pb during 120 of its 180 days in space, staying on an unshielded remote module for 60 days, returning to the space station, and finally departing for earth. The effective separation distance while

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Figure 28. Space Station Radiation Environment From Nuclear Reactor

TABLE 14
ASTRONAUT RADIATION DOSE EVALUATION

	Radiation (rem)			
	Reactor	Space	Total	
<u>Astronaut-Pilot</u>				
13 Rendezvous	5.2	2.0	7.2	
1 Arrival				
12 Returns from remote module station				
13 Departures (twice rendez- vous velocities assumed)	2.6		2.6	
180 Days on space station	<u>12.5</u>	<u>22.0</u>	<u>34.5</u>	<u>"Planning"</u>
Total	20.3	24.0	44.3	80
<u>Astronaut-Scientist</u>				
1 Arrival rendezvous	0.4		0.4	
1 Departure	0.2		0.2	
360 Days on space station	<u>25.0</u>	<u>44.0</u>	<u>69.0</u>	<u>"Planning"</u>
Total	25.6	44.0	69.6	100

TABLE 15
FILM RADIATION DOSE EVALUATION,
(Most sensitive film programmed, Panatomic-X)

Radiation from Reactor	Gamma Dose from Reactor (rads)
2 Rendezvous	0.5
1 Arrival	
1 Return from remote module station	
2 Departures (twice rendezvous velocities assumed)	0.25
60 Days at remote module station (5.5 mi effective)	1.5
120 Days on space station shadow-shielded by 1.8 in. lead (Most film will not require this shielding)	0.25
Total	<u>2.5 → 0.1 fogging</u>

on the remote module is 5.5 mi for the close-in three-day cycle described previously in Section V-D. Experimental data show that the 2.5 rads received during the above sequence will increase the film's fog density to 0.1 (density = $\log$ of $1/[\text{fraction of light passed}]$).

Since the film must be shielded against natural space radiation, the increase in this shielding was taken as a measure of the effect of having a reactor on the space station. Two typical allowable densities were considered and the results are summarized in Table 16. The required shielding with and without a reactor is shown for the two fogging levels. The Al shielding required on the space station with the reactor present is reduced by taking credit for the 1.8 in. of Pb. Review of the table shows that the engineering problems with incorporating Al shielding to protect against space radiation will not be appreciably different with or without a reactor on the space station.

The above discussion has been for Panatomic-X. Most other films, particularly those for photographing the earth, will be much less sensitive and will not require the local Pb shielding on the station.

TABLE 16
EFFECT ON FILM

Allowable Fog Level	0.2		0.6	
	Reactor	No Reactor	Reactor	No Reactor
Required Shielding on Space Station				
Aluminum thickness, in.	1.17	2.34	0	0.53
Weight, gm/cm ²	8.0	16.0	0	3.6
Required Shielding on Remote Module				
Aluminum thickness, in.	1.73	1.0	0.35	0.35
Weight, gm/cm ²	12.0	7.0	2.4	2.4

VII. THERMOELECTRIC SYSTEM POWER RANGE CAPABILITY

The foregoing discussions have all been related to a 25-kwe TE system. This power level was chosen to allow a detailed investigation of a specific plant and was based on an early 1968 estimate of the space station power requirement. It is by no means a unique or limiting power level.

A. CURRENT TECHNOLOGY

Five specific plant designs covering the power range up to 50 kwe were investigated, using exactly the same technology base and ground rules as with the 25-kwe system. These plants use the same reactor, are divided into four power-conversion system quadrants, and have discrete TE module arrays that fit the gallery design. In each case an optimized shield design was established. The plant characteristics are indicated by the 1250°F curves in Figure 29, and the information is listed in Table 17 under the 1250°F reactor-outlet temperature. The profiles of the five plants are shown in Figure 30 and the smallest and largest are compared in Figure 31 with the standard Apollo configuration and two payload envelopes that had been evaluated in a previous study by NASA-MSFC. The latter comparison shows the annual launch probability of the 50-kwe system to be about 95%. The probability would be less during the windiest month, March, and higher during the least windy month, August.

By dropping the primary-loop temperature 100°F the system lifetime is significantly extended at some penalty in system performance. This is shown by the 1150°F information on Figure 29 and in Table 17. The electrical power output of a given plant size is reduced about 14% but its lifetime is almost doubled. At a 50-kwe power level, for instance, derating the temperature increases weight by about 10% and radiator area about 15%, but lifetime is increased from 2 to 3.5 yr. If this plant were run at a gradually increasing power level, reaching 50 kwe only during the last year or so of operation, its life could be near 5 yr (see Off-Design Performance, Section IV-D-2).

B. PROJECTED IMPROVEMENTS

Conservative assumptions made on projected technology improvements upon which to base systems in the 1980's, and the characteristics of such systems are shown in Figure 29 and Table 17.

a

b

c

Figure 29. Thermoelectric System Characteristics vs Power Level

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TABLE 17
REFERENCE ZIRCONIUM-HYDRIDE REACTOR-THERMOELECTRIC
SYSTEM CHARACTERISTICS
(25 to 50 kwe)

Reference Conditions Number of Thermoelectric Modules	96 4 (4 x 6)	120 4 (5 x 6)	140 4 (5 x 7)	160 4 (5 x 8)	192 4 (6 x 8)
Reactor Outlet Temperature, °F	← 1250 →				
dc Power, kwe	25.2	31.4	36.7	42.0	50.4
Reactor power, kwt	583	730	850	970	1160
Radiator area, ft ²	1390	1730	2030	2320	2780
Total height, ft	56.3	55.9	60.2	63.9	69.6
Diameter, ft	11.2	11.9	12.9	14.0	15.5
Effective separation distance, ft	117	122	128	134	143
Weight, lb					
Unshielded	8300	9900	11,500	12,900	15,400
Shield	14,000	14,600	15,200	15,900	16,400
Total	22,300	24,500	26,700	28,800	31,800
Specific weight, lb/kwe	885	780	728	685	631
Lifetime, hr	35,000	29,000	25,000	22,000	17,000
Derated Performance					
Reactor-outlet temperature, °F	← 1150 →				
dc Power, kwe	21.7	27.0	31.6	36.2	43.5
Reactor power, kwt	507	641	740	825	1020
Specific weight, lb/kwe	1040	905	843	800	723
Lifetime, hr	60,000	50,000	44,000	41,000	34,000
1980 Performance					
Reactor-outlet temperature, °F	← 1300 →				
dc Power, kwe	28.6	35.7	41.7	48.0	57.41
Reactor power, kwt	578	714	830	945	1126
Radiator area, ft ²	1230	1540	1790	2060	2470
Shielded weight, lb	20,900	22,400	24,400	26,600	29,000
Specific weight, lb/kwe	731	628	585	555	505
Lifetime, hr	70,000	58,000	50,000	45,000	38,000

Figure 30. Reactor Thermoelectric System Sizes

Figure 31. Saturn-V Payload Envelope

The area and weight improvements result from operating at 50°F higher primary-loop temperature and by taking credit for the best current experimental data (Table 18). The current technology assumptions are conservative in that they use the low side of current data. The converter efficiency is 5.3% in these systems rather than 4.67%. These changes result in roughly 25% more electrical power from a plant of given area and weight.

The projected improvement in lifetime results mainly from two of the factors listed in Table 19. First, a lower hydrogen leakrate based on out-of-pile tests of S8DR production-run fuel elements is used rather than the data from earlier elements, and second, the reactor core length is increased from 17 to 24 in. to add reactivity and reduce power density and fuel temperature. No change is projected in core diameter or control-drum construction. With these changes the projected lifetime of a 50-kwe system could be extended to approximately 5 yr at 1300°F.

TABLE 18
IMPROVEMENTS IN A 1980 REACTOR-THERMOELECTRIC SYSTEM

Parameter	Reference 1975 System	1980 System	Basis
Reactor-outlet temperature, °F	1246	1300	Basic reactor design temperature
Average thermoelectric hot-clad temperature, °F	1125	1175	Expect experimental backup of stable performance at this temperature
Average thermoelectric cold-clad temperature, °F	570	590	Optimum temperature with increased hot-clad temperature
Thermoelectric converter efficiency, %	4.67	5.3	A) Increased Carnot efficiency at higher temperature B) Assumes high side of current experimental data
Pump converter efficiency, %	3.1	3.5	Same as above
Pump-throat efficiency, %	25	40	Possible with larger higher current pumps (good for larger 1975 systems also)

TABLE 19
IMPROVEMENTS IN A 1980 ZIRCONIUM-HYDRIDE REACTOR

Parameter	Reference Reactor	1980 Reactor	Basis
Hydrogen leakage factor, cc/hr/atm ^{1/2} -cm ²	1.0 to 7.5 (increasing with time)	1.0 (constant)	Consistent with data being generated on fuel elements from S8DR production run. Improvements expected in end leakage and coatability with Incoloy.
Maximum cladding strain, %	0.4	1.0	Better ductility with Incoloy, strain limit based on glass barrier experiments.
Fuel-cladding material	Hastelloy-N	Incoloy	Experiments show improved coatability and ductility, and simplified fabrica- tion sequence.
Fuel-element length, in.	17	24	Within capability of fabrication process. Added experience will eliminate geometry change uncertainties.

VIII. CONCLUSION

The major conclusions derived from this study and from previous and current development programs are as follows:

- 1) Average power requirements for future space stations will be higher than those previously estimated and will probably be at least 25 kwe.
- 2) Reactor-thermoelectric systems can meet the power demands in the 25- to 50-kwe range, based on current or near-term technology.
- 3) The general performance characteristics of 25- to 50-kwe reactor-thermoelectric systems designed for manned space stations of the type studied are in the range of: 600 to 800 lb/kwe (shielded), 280 to 325 lb/kwe (unshielded), 50 to 60 ft²/kwe, and 3 to 5 yr lifetime. These performance characteristics are in general competitive with alternate power system candidates in this power range.
- 4) The integration of a reactor-thermoelectric power system has an important effect on the space station configuration and operation; however, no significant problems have been uncovered which would prohibit a satisfactory integration.
- 5) The major areas of uncertainty which must be resolved are the establishment of allowable mission radiation doses and official approval of nuclear safety provisions regarding end-of-life disposal of the reactor.

APPENDIX
DESIGN DRAWINGS

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Figure A1. 25-kwe Nuclear
Thermoelectric Power
Supply
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Figure A2. Reactor, Shield, and
Gallery Components, 25-kwe
Nuclear Thermoelectric
Power Supply
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Figure A3. Reference Zirconium Hydride Reactor

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REVISIONS			
LTN	DESCRIPTION	DATE	APPROVED

CONNECTOR & WIRE TABULATION				
CONN	NO. OF PINS	SIZE OF WIRE	LENGTH EACH CABLE FT	WEIGHT OF WIRE LBS
P1	12	12/C #20	65	7.5
P2	27	7/C #18, 3 #16, 3 #14, 3 #12, 3 #10, 3 #8, 3 #6, 3 #4	65	11.7
P3	17	4 #16, 2 #14, 2 #12, 2 #10, 2 #8, 2 #6, 2 #4	65	15.6
P4	11	1 #16, 1 #14, 1 #12, 1 #10, 1 #8, 1 #6, 1 #4	65	3.0
P5	17	2 #16, 2 #14, 2 #12, 2 #10, 2 #8, 2 #6, 2 #4	65	7.35
P6	17	2 #16, 2 #14, 2 #12, 2 #10, 2 #8, 2 #6, 2 #4	65	14.7
P7	17	2 #16, 2 #14, 2 #12, 2 #10, 2 #8, 2 #6, 2 #4	65	14.7
P8	17	2 #16, 2 #14, 2 #12, 2 #10, 2 #8, 2 #6, 2 #4	65	14.7
P9	17	2 #16, 2 #14, 2 #12, 2 #10, 2 #8, 2 #6, 2 #4	65	14.7
P10	17	2 #16, 2 #14, 2 #12, 2 #10, 2 #8, 2 #6, 2 #4	65	14.7
P11	3	3 #16	60	7.1
P12	3	3 #16	60	7.1
P13	3	3 #16	60	7.1
P14	3	3 #16	60	7.1
P15	2	2 #16	60	7.1
P16				
P17				
P18				
P19				
P20				
P21				
P22				
P23				
P24				
P25				
P26				
P27				
P28				
P29				
P30	4	4 #16	60	7.1
TOTAL WEIGHT				410 LBS.

Figure A4. Thermoelectric System Control and Power Cable Diagram

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* SHIELDING = 3 LBS/100FT
** 1/C #20 300°F WIRE = 7.4 LBS/100FT

DR BY	F. J. LODAS	4 6 65	
CHK BY			
			THERMO-ELECTRIC SYSTEM CONTROL AND POWER CABLE DIAGRAM
			SIZE: 11x17 IN. (254x432 MM) E 09974 N69626112
			SCALE: _____ SHEET: _____

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TELEMETRY SIGNALS

TEM Voc COMPUTER REDUNDANCY LOSS
 TEM Voc COMPUTER OUTPUT SIGNALS
 TEM Voc REFERENCE SETTING
 TEMPERATURE SENSOR REDUNDANCY LOSS
 TEMPERATURE SENSOR OUTPUT SIGNALS
 PLANT STATUS SIGNALS
 RATE CHANGE SETPOINT
 RATE CHANGE SIGNAL
 SYSTEM SUPPLY VOLTAGE LEVELS
 CONTROL SYSTEM TEMPERATURE
 DRUM POSITION
 DRUM DEVIATION
 POWER SWITCH REDUNDANCY LOSS
 POWER SWITCH MALFUNCTION
 ACTIVE CONTROLLER STATUS
 STANDBY CONTROLLER STATUS
 CONTROL SYSTEM - TIMING RATES

- DRUM IN-OUT SIGNALS
- DRUM STEP SIGNALS
- CONTROL LOGIC MALFUNCTION
- TIME BASE ERROR
- PHASE PULSING SIGNAL
- ACTUATOR PULSING SIGNAL
- SEQUENCER MALFUNCTION

CONTROL AND CHECKOUT SYSTEM

REVISIONS			
SYM	DESCRIPTION	DATE	APPROVED

DISPLAY PANEL

Figure A6. Thermoelectric System Control System Modules and Display Panel

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WEIGHT AND VOLUME
 (NOT INCLUDING INPUT-OUTPUT CABLING)
 < 40 LBS AND ~ 800 IN³

DISPLAY AREA
 75 TO 90 IN²

POWER REQ.
 FOR STARTUP FOR COMPLETE OPERATION
 50 W 75 W
 35 W 60 W
 ± 28VDC OR 56VDC @ 30 AMPERES MAX FOR ACTUATORS
 (NOMINAL 400 WATT-HOURS FOR A STARTUP)

NOTES UNLESS OTHERWISE SPECIFIED

UNLESS OTHERWISE SPECIFIED DIMENSIONS ARE IN INCHES TOLERANCES ON DECIMALS ANGLES FRACTIONS XX = ± .03 ±30' ± 1/32 XXX = ± .010				DR BY		CHK BY		APPROVED BY		ATOMIC INTERNATIONAL A Division of North American Aviation, Inc. CANOGA PARK, CALIFORNIA	
HOLES NOTED "DRILL"				013 THRU 040 + .001 - .001		041 THRU 130 + .002 - .001		131 THRU 229 + .003 - .001		230 THRU 500 + .004 - .001	
501 THRU 750 + .005 - .001				751 THRU 1000 + .007 - .001		1001 THRU 2000 + .010 - .001		CODE IDENT NO		SIZE	
APPLICATION (USAGE) DATA				DO NOT SCALE PRINT		09974		D		R-1169-12.55	
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