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CONCEPTUAL DESIGN OF A 2000 kW NUCLEAR HEAT SOURCE SUITABLE FOR SPACE ELECTRIC POWER

(Title: Unclassified)

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June 20, 1968

AEC RESEARCH AND DEVELOPMENT REPORT

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Abstract

A conceptual design of a 2000 kW_t Rankine system for nuclear-electric space power is described. The system includes a unique reactor-boiler unit operating at 1500°K that utilizes heatpipes in lieu of a conventionally pumped primary loop. An efficient heatpipe radiator rejects waste heat at 1035°K. Overall system efficiency is 18.8% yielding a net electrical output of 375 kW. The system specific mass is 10 g/W including a generous shadow shield for unmanned payloads.

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CONCEPTUAL DESIGN OF A 2000 kW NUCLEAR HEAT SOURCE SUITABLE FOR SPACE ELECTRIC POWER

I. Introduction

In July, 1965, a program to develop the technology for building advanced, high-temperature reactors for nuclear-electric space power systems was initiated at the Lawrence Radiation Laboratory. Reactor concepts using liquid metal Rankine cycle, out-of-pile thermionic, and other conversion systems were examined. Electrical power levels ranged from tens of kilowatts to megawatts. All systems studied were conceptual and intended for application 10 to 20 years in the future.

A technology development program such as this is best placed in proper perspective by examining specific overall power system designs. These designs provide a framework for studies of reactor operating conditions, serve to identify specific problem areas, and indicate the relative importance of the many parameters and system configuration options involved.

This report summarizes the results from one of these design studies—a conceptual design of a 2-MW_t Rankine cycle, nuclear-electric space power system. This system, designated as SPR-4, uses

sodium heatpipes* to transport energy from the reactor heat source to a boiler employing natural potassium as the working fluid. In this design the heatpipes replace the primary loop of a conventional reactor system and eliminate the need for a primary pump. In contrast, a 10 MW_e system, reported by Pitts and Walter¹, used flowing liquid lithium in a conventional primary loop to transfer the heat from the reactor to the boiler.

The design effort for the SPR-4 system was less than originally intended, and the design of some components was not entirely completed. However, by emphasizing work on the reactor-boiler and nuclear shield, and by utilizing existing designs for other components, it was possible to partially optimize the system. The goal of the optimization was to obtain a low power plant specific mass with high system reliability. Further

*Heatpipes are self-contained, thermal conductance devices having high thermal efficiencies.

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details of the work performed, which are not included in this report, have been published in other formal and informal reports.

The SPR-4 system can be applied to ion-propulsion systems for unmanned space missions and to auxiliary electric power systems. Nuclear electric propulsion permits a significantly larger payload for orbital missions to the major planets as compared to other types of

propulsion². Applications to lunar bases and to earth-orbiting power stations also are feasible. The reactor technology developed here is applicable generally to unmanned, long life-time space nuclear power plants. Although this report specifies particular characteristics, the reactor performance and its design should be considered as a general case capable of being adjusted for a variety of uses in future space exploration including auxiliary power for manned missions.

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II. General System Description

The SPR-4 system shown as an artist's conception in Fig. 1 consists of the following major components: an integral reactor-boiler operating at 2 MW and at 1500°K; a nuclear shield composed of alternate layers of lithium hydride (LiH) and tungsten(W); a turbine-alternator unit; two heatpipe radiators, one operating at 1035°K and the other at 770°K; and a single electromagnetic pump for the circulation of the potassium working fluid in the power conversion loop. (See Fig. 2) The system was designed to operate under the principal constraints listed in Table I. Operating conditions of the system are shown in the schematic diagram of Fig. 2 and a breakdown of the system thermal economy is diagrammed in Fig. 3. System efficiency is 18.8%.

An overall length of only ~10 meters allows the system to be launched into orbit as a integral unit. The reactor would

achieve criticality after it was placed either in a stable earth-orbit or positioned on the moon or another interplanetary body.

The SPR-4 reactor design utilizes uranium mononitride (UN) as the fuel, a tungsten alloy containing 25 at. % rhenium (W-25% Re) as the structural material, and sodium as an interstitial thermal bonding agent and heatpipe fluid. The uranium is enriched with the isotope U^{235} in two radial zones. Although the reactor is compact, an even smaller reactor could be designed by introducing the more active U^{233} isotope. The material handling characteristics of U^{233} , however, are more difficult than those of U^{235} . The UN fuel has a high uranium density, good thermal conductivity, a high melting point, and excellent physical properties.

Fig. 1. SPR-4 space nuclear-electric power plant.

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The active core consists of 378 hexagonal fuel elements. Each fuel element has a central hole through which a heatpipe, partially filled with sodium, is inserted.³ Fuel elements are bundled to form an approximate right circular cylinder, 34-cm in diameter, 42-cm long, with a 6-cm-diameter central duct to transport potassium vapor to the power conversion loop. Two potassium boilers are integrated with the reactor, one on

each end of the cylindrical core. Heatpipes extend from the ends of the core into the boilers. The potassium working fluid is boiled on the outside surfaces of the heatpipe tubes protruding into the boilers. These heatpipes replace the primary loop of a conventional conversion system. Thus, heat is transferred under nearly isothermal conditions from the fueled core to the boilers. Each boiler has the approximate shape and

Fig. 2. SPR-4 system schematic.

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Table I. SPR-4 design constraints.

Reactor power	2000 kW
Design lifetime	20,000 hours
Power conversion method	Rankine cycle
Application	Electric power for ion-propulsion or for space station auxiliary power
Design power on-off cycles	10
Payload	
Type	Unmanned
Dose criterion	10^6 R for gamma radiation 10^{13} nvt for neutrons
Distance (from reactor)	~ 10 m
Shield half-angle	~ 14 deg

Fig. 3. SPR-4 thermal economy.

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size of the nuclear core. The potassium flow rate is adjusted to ensure that the fluid leaves the boilers as a saturated vapor.

Reactivity of the reactor is regulated with a dual control system consisting of a molybdenum side reflector and centrally located, in-core, liquid control circuits. The side reflector is 2-cm thick and can be translated axially. Each system is capable of controlling the reactor independently throughout its entire lifetime. This dual control system considerably increases system reliability.

The liquid control circuits consist of twelve tubes placed inside the fueled core in an annular region surrounding the potassium vapor duct. The tubes contain a capillary screen structure and are partially filled with the strong neutron-absorbing isotope, lithium-6 (Li^6). Reactivity is varied by adjusting the quantity of lithium in the control tubes.

The reactor-boiler is separated from all other components by a nuclear shadow-shield, which reduces the dose rate of the payload and electronic equipment to acceptable levels. The shield consists of two layers of lithium hydride separated by a layer of tungsten. The shield is contained within a metal container to prevent hydrogen loss and to protect against damage from meteoroids. A slight hydrogen overpressure of less than 50 torr may be required to prevent disassociation of the lithium hydride.⁴ An overall shield thickness of 75.4 cm was calculated, using point-kernel methods, to establish the specific dose at the payload of 10^6 R for gamma rays and 10^{13} nvt for neutrons. Penetrations of piping and other devices through the

shield have helical paths to reduce streaming.

A turbine-alternator similar in size and performance to that reported by Robbins⁵ is used to extract useful energy from the potassium vapor leaving the boilers. Excess heat is removed from the alternator windings through the use of a cylindrical auxiliary heatpipe radiator operating at 770°K.

The vapor leaving the turbine is subsequently condensed in a large, but low-mass, flat radiator containing 2200 heatpipes.⁶ Use of heatpipes in the radiator reduces its mass to about one-half that of a conventional fin-and-tube radiator. The heatpipes operate independently, and in parallel, thereby providing unprecedented redundancy for the radiator.

The system does not require high-temperature pumps. The most critical pump operates at 1035°K in the power conversion loop. An electromagnetic pump, similar to that reported by Diedrich and Gehan,⁷ was chosen for this use because of its reliability. Slight differences in electromagnetic pump performance have a negligible effect on system power output or on specific mass.

The complete system is presented in Fig. 4 and the specific masses of system components are given in Table II. Operating characteristics of the system are tabulated in Table III. The low specific mass of 10 g/W is made possible by utilizing a compact reactor-boiler embodying a high fuel burnup capability, high strength alloys, efficient heatpipe radiators, and an efficient nuclear shield. The smallest practical reactor was selected because reactor size affects not only the mass of the reactor but, more

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Fig. 4. Layout of SPR-4 system.

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Table II. SPR-4 system masses.

	Mass (kg)	Specific mass (g/W)	Relative mass
Shield	1400	3.73	0.373
Reactor-boiler	900	2.40	0.240
Turbine-alternator	400	1.07	0.107
Main radiator	400	1.07	0.107
Pump	200	0.53	0.053
Miscellaneous	<u>450</u>	<u>1.20</u>	<u>0.120</u>
	3750	10.00	1.000
Electrical output power = 375 kW			

Table III. Operating characteristics.

<u>Reactor-boiler</u>	
Reactor power level (kW)	2000
Reactor peak fuel temperature (°K)	1540
Heatpipe fluid temperature (°K)	1500
Boiler potassium outlet temperature (°K)	1485
Average heat pipe radial heat flux (W/cm ²)	46
Maximum heat pipe axial heat flux* (W/cm ²)	7000
Core pressure (bar)	18.4
Heatpipe pressure (bar)	11.0
Boiler potassium pressure (bar)	18.4
Detailed reactor-boiler characteristics	(See Table IV)
<u>Shield</u>	
Maximum temperature (°K)	800
Thickness	
First LiH layer (cm)	11.8
Tungsten layer (cm)	1.2
Second LiH layer (cm)	<u>62.4</u>
	Total, cm
	75.4
<u>Power conversion loop</u>	
Pump inlet pressure (bar)	1.1
Pump outlet pressure (bar)	18.5
Potassium working fluid flow rate (g/sec)	997
Turbine inlet temperature (°K)	1485
Alternator net output power (kW)	375
Auxiliary radiator heat rejection (kW)	53
Auxiliary radiator rejection temperature (°K)	770
Main radiator heat rejection (kW)	1565
Main radiator rejection temperature (°K)	1035

* Define as heatpipe power divided by cross-sectional area of the vapor channel in the heatpipe.

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importantly, the mass of the nuclear shield.

Interactions between individual components were examined using a flexible, simple mathematical representation of the system.⁸ Results of these system studies were used to minimize system specific mass. Calculations were performed in two parts. First, approximate component masses and efficiencies were found by individual component studies and through examination of data available in the literature. Second, the thermodynamic performance of a complete cycle was predicted for an ideal system and then adjusted for the component inefficiencies found from the individual component studies. The specific mass of the system was calculated at a constant reactor power by calculating the net electrical output power from the cycle thermodynamic analysis and totaling the masses

of the individual components. Only a partial optimization of the system was possible inasmuch as detailed performance characteristics and masses were determined only for the reactor, nuclear shield, and main radiator.

It was determined that, for a reactor with an operating temperature of 1500°K, a main radiator rejection temperature of 1035°K produced a minimum system specific mass. The corresponding turbine temperature ratio* was found to be 0.7. A slightly lower optimum turbine temperature ratio than the classic 0.75⁹ is expected because the radiator mass is not predominant in the SPR-4 system.

*Turbine temperature ratio is defined as turbine outlet temperature divided by turbine inlet temperature.

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III. Reactor-Boiler

GENERAL DESCRIPTION

The SPR-4 heatpipe reactor is closely coupled to the boiler of the power conversion loop (See Fig. 5). For this reason it is difficult to discuss their designs separately. The unit may be described as a double-ended, heatpipe-cooled reactor from which the heatpipes extend to form an integral boiler. In the nuclear core, UN fuel is thermally coupled by liquid sodium to the walls of the heatpipes. Sodium heatpipes made with a composite groove and mesh wick structure, penetrate the core from each end and extend to the core mid-plane. These heatpipes are held in a hexagonal array by core end plates and five additional 2-mm-thick transverse plates equally spaced along the core axis. The dodecagon (12 sides) was selected as the most

suitable core cross section consistent with the hexagonal fuel array. Fuel elements consist of hexagonal tubes of UN, slipped over the heatpipes with allowance for differential thermal expansion and fuel swelling.

This integral configuration of the SPR-4 reactor-boiler results in a temperature difference of less than 12°K in the reactor fluid loop. Because of the nearly isothermal characteristics of heatpipes and the good thermal conductivity of UN, liquid sodium, and W-25% Re, the total difference between the peak fuel temperature and that of the potassium working fluid leaving the boiler, under normal operating conditions, is only 48°K. Operating characteristics of the reactor-boiler are tabulated in Table IV.

The double-ended boiler configuration was chosen for two reasons. First, the reactor radius is smaller for a

Fig. 5. Integral reactor-boiler.

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Table IV. Characteristics of the SPR-4 reactor-boiler.

<u>Nuclear parameters</u>	
Power level (kW_t)	2,000
Lifetime (hr)	20,000
Average fuel burnup (a/o)	0.68
Maximum fuel burnup (a/o)	1.08
Fuel density (fraction of theoretical)	0.97
Average fuel enrichment (fraction of U^{235})	0.93
Maximum fuel enrichment (fraction of U^{235})	0.99
Average fuel power density (W/cm^3)	100.
Maximum fuel power density (W/cm^3)	159.
Radial peak-to-average power density ratio (at most severe transverse cross section)	1.18
Axial peak-to-average power density ratio	1.35
Overall peak-to-average power density ratio	1.59
Median neutron energy causing fission (MeV)	0.53
Maximum neutron flux in core above 1 MeV ($\text{n}/\text{cm}^2\text{sec}$)	5.1×10^{13}
Average neutron energy (MeV)	0.39
Prompt neutron lifetime, ℓ^* (sec)	2.4×10^{-8}
Delayed neutron fraction, β_{eff}	0.0066
Worth of 1 cm increase in radius of fueled core (Δk)	0.043
Worth of 1 cm increase in length of fueled core (Δk)	0.006
<u>General dimensions</u>	
Fueled core length (cm)	41.9
Mean fueled core outer diameter (cm)	33.9
Mean liquid lithium control zone outer diameter (cm)	12.8
Potassium vapor return duct inside diameter (cm)	6.3
Pressure vessel inside diameter (cm)	36.2
Side reflector inside diameter (cm)	37.2
Side reflector outside diameter (cm)	41.2
<u>Fuel element and heatpipe dimensions</u>	
Fuel inside radius (cm)	0.466
Fuel flat-to-flat half dimension (cm)	0.773
Heatpipe vapor channel radius (cm)	0.313
Heatpipe inside radius (cm)	0.417
Heatpipe outside radius (cm)	0.463
Heatpipe capillary radius (cm)	0.025
Number of fuel elements	378
Number of heatpipes	756

Table IV. (Continued)

<u>Fueled core volume fractions</u>	
Fuel volume fraction	0.616
Heatpipe metal volume fraction	0.062
Intermediate tube spacer volume fraction	0.016
Heatpipe liquid volume fraction	0.117
Heatpipe vapor volume fraction	0.150
Swelling allowance volume fraction	0.039
Total fueled core volume (ℓ)	32.4
<u>Masses</u>	
Mass of uranium required (kg)	250
Mass of U^{235} required (kg)	232
<u>Control worth</u>	
Minimum reactivity state, k_{eff} (Reflector Off, 300°K, beginning of life, 100% Li-6)	0.943
Operational shutdown (Δk)	1.50%
Maximum temperature effect (300°K to 1500°K, 35% Li-6) (Δk)	1.04%
Operational losses due to fuel burnup, etc. (Δk)	0.54%
Total control reactivity required (Δk)	1.58%
Total worth of reflector (Δk)	8.0%
Total worth of liquid control circuits (Δk)	3.4%
<u>Thermal-hydraulic parameters</u>	
Total heatpipe fluid flow rate (kg/sec)	0.566
Heatpipe fluid saturation temperature (°K)	1500
Heatpipe fluid saturation pressure (bar)	11.0
Maximum heatpipe liquid velocity (cm/sec)	7.8
Maximum heatpipe vapor Mach number	0.02
Average heatpipe radial heat flux in (W/cm^2)	46.4
Average heatpipe radial heat flux out (W/cm^2)	23.2
Maximum heatpipe axial heat flux based on vapor volume (kW/cm^2)	7.0
Nominal peak fuel temperature (°K)	1533

double-ended configuration with a central vapor return duct than it would be for a single-ended configuration without a return duct because of limitations in heat transfer from the core faces. Second, the double-ended configuration facilitates fabrication, testing, and assembly of heatpipes into the reactor. Each of the

two heatpipe assemblies could each be pretested before surrounding them with fuel. The two reactor-boiler halves would be joined by a single girth weld. A light pin at one end locates the core axially. There is only a small net force acting on the core due to vapor flow in the central duct.

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The reactor heatpipes extend from the nuclear core to make available twice the heat transfer area of the core for boiling the potassium in the power conversion loop. Boiling data for flow inside tubes¹⁰ indicate that high quality vapor is obtained with average heat flux rates of 25 to 50 W/cm². Thus, it appears that the limiting heat flux is governed by boiling of potassium on the outer surface of the heatpipes. Recognizably, boiling would be enhanced by wrapping a capillary screen on the outside of the tubes. Swirling tapes also may be desirable.

Figure 6 shows the potassium flow passage inside the reactor-boiler. Liquid potassium at 1035°K is admitted to the reactor at the core mid-plane. The flow divides and continues axially along the pressure vessel toward the end faces of the reactor-boiler. This flow arrangement, together with HfO₂ tile insulation, maintains the maximum reactor vessel

temperature below 1250°K. This maximum temperature allows the use of a 0.4-cm thick tantalum alloy for the vessel wall. After reaching the core end faces, the flow is distributed radially and then continues axially through the boiler. The potassium leaves the boiler as a saturated liquid. Fluid vaporized in the outboard boiler is returned through a 6-cm-diameter central return duct. Total pressure drop in the potassium, as it flows through the reactor-boiler, is only 0.1 bar.

The natural shape of the fuel for a heatpipe reactor is a hollow hexagonal washer. The hexagonally-shaped fuel can be placed over the previously tested subassembly of heatpipes. Final dimensions chosen for the heatpipe-fuel elements are shown in Fig. 7.

With this geometry, heat is removed along the center of the fuel pellets and the temperature at the periphery of

Fig. 6. Schematic diagram of integral reactor-boiler flow passage cross section.

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Fig. 7. Thirty-degree segment of hexagonal heatpipe—fuel element assembly.

the fuel is higher. Based on thermodynamic arguments, therefore, the fission gas bubbles would tend to move away from the heatpipe and be released at the outer free surface of the fuel pellets. At present, it is difficult to predict the

amount of fission gas which would be released under these conditions. Accordingly, provisions must be made to remove all the fission gas which would be generated during the life of the reactor. At the peak fuel operating temperature of

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1533°K, nitrogen over-pressure for the uranium nitride fuel may not be required. If an over-pressure of nitrogen is required, however, it can be easily provided.¹

Sodium was chosen for the heatpipe working fluid at a nominal temperature of 1500°K. Its use results in higher heatpipe performance and hence in a smaller core size compared to the core sizes necessary when lithium and potassium working fluids are used. The primary limitation on heatpipe performance arises from the internal vapor pressure drop which depends strongly upon the volumetric flow rate. Lithium has a higher latent heat of vaporization than sodium, but its low vapor pressure at 1500°K requires a larger volumetric flow rate per unit power. On the other hand, potassium has a higher vapor pressure than sodium, but in this case its lower latent heat of vaporization produces a similar effect. At a higher temperature lithium would be better than sodium. A disadvantage to using sodium in the heatpipes instead of potassium is the adverse pressure gradient of ~7 bar on the heatpipes in the boiler region. The potassium power-conversion-loop fluid must be at its saturation pressure (1485°K, 18.4 bar) and the heatpipe working fluid at its saturation pressure (1500°K, 11 bar) in the boiler region. Furthermore, the bonding fluid used in the core could not be potassium because, unless its pressure were considerably in excess of 18 bar, it would not remain a liquid under high power conditions. Consequently, this would result in a low tolerance to hot spots. Preliminary calculations indicate that the pressure

differential with sodium and potassium would be tolerable from the standpoint of creep buckling in the heatpipes.

Natural potassium was chosen as the power-conversion-loop fluid mainly because information on potassium turbines was most readily available. Its use also produces considerably fewer neutron activation problems than sodium. Either natural potassium or natural sodium are acceptable for the dose criterion chosen for SPR-4 even though both sodium-23 and potassium-41 are activated by neutrons. Natural potassium contains only 7% potassium-41. This concentration can be reduced economically to less than 0.1% if necessary.¹¹ Natural sodium contains the single isotope sodium-23. A manned payload application requiring a reduced dose rate could be accommodated only if the potassium isotope, potassium-39, were used.

The extent of corrosion in the heatpipes is not well known and experimental verification of both heatpipe performance and corrosion is desirable. Fortunately such tests would not be difficult and could lead to a less conservative choice of capillary mesh size shown in Table IV. The temperature difference in the heatpipe fluid loop is about one-fifth that which usually results in a convective fluid loop. And the mass flow rate in the heatpipe fluid loop is about one-tenth of that normally considered in a convective fluid loop. This tends to reduce corrosion. A disturbing factor is that complete vaporization of the fluid occurs at the heated end of the heatpipe.

The tungsten heat pipes are clad on the outside with tantalum where they extend into the boilers to avoid dissimilar

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metal difficulties in the potassium loop. The core is contained in a sealed can made from 0.15 cm W-25% Re sheet. This can also is clad externally with tantalum.

The outside reflector radius for the double-ended heatpipe reactor is 20.6 cm compared with 17.2 cm for a conventionally cooled reactor. The outside reflector radius could be reduced to 18.5 cm if more active U^{233} were used in the fueled core. In either case, however, the increased radius and additional length of the integrated reactor-boiler would require a larger shield. Whether or not the heatpipe reactor is justified depends on factors such as reliability and development potential of the heatpipe and conventional systems.

An additional advantage to heatpipe cooling is that it prevents overheating of the reactor after shutdown because heat from decay products is automatically removed from the reactor.

DESIGN METHODS

The SPR-4 reactor was designed using a composite neutron transport-mechanical-fluid dynamic computer code similar to that described by Brown and McCauley.¹² This computer code considered nuclear performance, structural component sizes, expansion due to fuel swelling and thermal effects, plus production of daughter fissile species in a single pass. By means of this code complete design of the reactor could be computed in a few minutes rather than after weeks of iterative calculations. The final reactor design was internally consistent and all structural considerations

were incorporated a priori. Additional items such as neutron streaming from the core faces and water safety were considered separately for the reactor design but later integrated into the final version of the computer code.

Numerical values remain in the computer throughout the design and analysis. This feature permits small perturbations to be made with a high degree of numerical consistency. Complex interactions also can be studied in detail rather than as a series of approximate partial derivatives. Thus interactions that are not anticipated frequently can be uncovered.

The reactor code required about two years of development. A previous reactor design study¹ utilized an earlier version of the code. Basically the neutron calculations were performed with a modified one-dimensional neutron transport code, DTF¹³, with an S_4 approximation. The reactor is described in eight radial zones, each having five sub-regions. The thermodynamic portion of the program permitted a variety of thermo-mechanical constraints. Neutron cross sections were derived from the Howerton evaluated cross section library¹⁴ with the resulting data collapsed into a 16-group structure by the MESNE code.¹⁵ Monte Carlo calculations were made to appraise the effects of neutron streaming using the O5R Computer Program.¹⁶

Computation of the reactor begins with a statement of the boundary conditions of thermal power, lifetime, operating temperature, core aspect ratio, materials, limits on thermal stress, fuel wall temperature difference and fuel

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surface heat flux (See Fig. 8). From this information, temperature and time-dependent material properties can be determined.

For small, criticality-limited cores such as that of the SPR-4, the maximum allowed fuel enrichment is specified and fuel burnup is an independent design variable. The reactor design subroutine is called and a minimum porosity core is designed within the assumed constraints. An iteration scheme assures that the number of fuel elements chosen is consistent with the fuel element limitations and the core configuration selected. This is an important point for small reactors having a small number of rows of fuel elements.

Once the core is designed, the code adds thermal insulators, the pressure vessel, and the external side reflector. Whenever desired, boundaries of the fuel element rows can be examined to determine optimum power-flattening zones. The dimensions and volume fractions of this trial mechanical design are translated into a dimensional and an atom density array which is then fed automatically into the neutronic subroutine which calculates the multiplication factor. Core dimensions then are reduced to room temperature dimensions and the atom density array is manipulated to appraise reactivity effects in water.

This amount of detail is necessary for conceptual reactor design because of the number of interactions between heat transfer, thermal expansion, fuel swelling, and nuclear effects. It is necessary to know how these effects combine to evaluate the reactivity states of the

reactor under all conditions. Simple calculation of the reactor reactivity state at various constant temperatures shows that the SPR-4 reactor has negative temperature coefficients. Power and temperature coefficients of reactivity are computed, but a kinetic analysis has not yet been incorporated into the codes.

PERFORMANCE CHARACTERISTICS

Axial heat flux in the reactor-boiler heatpipe was based on the mean core power and the factors listed in Table V for assumed deviations from the mean core power. The resulting combined factor is 3.4 which is applied as a multiplier to the total power expected for an average heatpipe. The maximum probable local radial heat flux into the heatpipes is 1.85 times the average value quoted in Table IV. This value is calculated by multiplying the maximum-to-average core power density by a factor that accounts for increased load due to the failure of an adjacent heatpipe. Pool boiling correlations¹⁷ indicate that this maximum probable value is at least a factor of 2 below burnout.

Redundancy of control is provided by two fully independent reactivity control systems.¹⁸ Each of these control systems is located at its optimum efficiency position in the core and covers only the range of the adjustable reactivity required for the duration of the mission. The alternative control system provides for unforeseen reactivity requirements and for individual control-element failures. This principle enables each system to be physically small and highly efficient. The

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Fig. 8. Heatpipe reactor calculation schematic.

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Table V. Factors applied to the mean core power.

Radial peak-to-average power density ratio (at most severe transverse cross section)	1.1
Heatpipe working fluid property uncertainty	1.2
Increased load in the event of a failure of an adjacent heatpipe	1.17
Core power fluctuations	1.1
Safety margin	2.0

overall result is that the size of a reactor with dual controls is generally smaller than the size of an identical reactor governed by a single control scheme. In other words, a single control system necessarily requires a greater control range because reserves and redundancy provisions have to be included.

The controls of the SPR-4 reactor provide for adjustments to the neutron leakage external to the core and also the neutron capture inside the core. Mechanical reactivity adjustments are made by separating the two halves of the axially-translating, molybdenum side reflector. Liquid reactivity adjustments are made through the use of in-core control circuits, with lithium-6, the neutron absorbing isotope, as the liquid control agent. Each of these systems is capable of carrying the mission to completion without using the other system (see Fig. 9).^{*} With a dual control system, a particular reactivity state can be approached by combining the possible control states of the individual controls, as long as the combined effect matches that of the desired reactivity requirement. This is

illustrated in Fig. 9 by the shaded area for achieving criticality of the beginning of life and at 600°K. The dotted arrows indicate a suggested path.

Ground safety of 6.5% Δk is achieved by opening the side reflector (the least reflective position) and filling the liquid control circuits completely with lithium-6. Once the reactor is launched into space, the reflector halves are partially closed and the lithium-6 is removed to reach criticality. Because the reflector halves are fairly close together throughout most of the core lifetime, a long-time, large-motion actuator is not needed.

The total adjustable reactivity requirement for the reactor, 3.08% Δk , is small because backup controls supply reactivity reserves and provide for failure of individual segments of each system. The total reactivity requirement consists of:

Temperature effect (300°K to 1500°K, 35% Li-6) *	1.04% Δk
Losses due to fuel burnup, fuel swelling, and fission products	0.54% Δk
Operational shutdown	1.50% Δk
Total adjustable reactivity required	3.08% Δk

^{*}Because the mechanical and the liquid control systems act on the periphery and inner boundary of the fueled core, respectively, reactivity effects of the two systems are decoupled.

^{*}The isothermal temperature effect varies slightly depending upon the amount of lithium-6 present in the controls.

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Fig. 9. Range of the dual control system for SPR-4 reactor. CEOL designates critical at end of life; CBOL indicates critical at beginning of life.

Molybdenum was chosen as the material for the side reflectors over MgO , Al_2O_3 , Ni, and BeO (See Fig. 10), even though these materials are somewhat more effective, because it can be constructed simply and is compatible with high-temperature operating conditions in the radiation environment. The oxides would require a metal supporting structure. Nickel is an excellent candidate material and would have been selected had a detailed analysis of reflector heating justified its use. The lithium-6 isotope was chosen as a fluid for the liquid control circuits because of its strong neutron absorption characteristics.

The reactor is self-regulating on temperature and power because of several features. Its self-regulating temperature characteristics arise from negative void coefficients for the sodium in

the core. Total loss of sodium from the fueled core results in $\Delta k = -0.62\%$. Its self-regulating characteristics on power are possible as the result of changing effective thickness of the potassium liquid in the boilers adjacent to each end of the core. Here, the liquid potassium affects the end leakage of neutrons and renders a maximum effect of $\Delta k = 0.23\%$. As reactor power increases, the effective potassium thickness decreases, causing a small loss of reactivity which effectively reduces power. As reactor power decreases, the reverse occurs.

Neutron streaming out of the core end-faces through the voids in the heat-pipes and through the central vapor duct was estimated using the 05R Monte Carlo code.¹⁶ This estimate of the reactivity effect, as shown in Table VI, indicates

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that neutron streaming is significant in small reactors with a hard neutron energy spectrum.

Radial power flattening was computed automatically by the reactor design code for the SPR-4 reactor. Results are shown in Fig. 11. Reactor design was limited to the use of uranium-235 and uranium-238 and was designed without absolute water safety as an imposed constraint. No appreciable advantage was found in having more than two fuel zones.

The radial power profile can be adjusted slightly by judicious use of the dual controls. Reactivity can be increased by

removing lithium-6 from the central liquid control circuits. This tends to increase the power generated near the inner radius of the fueled core. On the other hand, reactivity also can be increased by adding side reflector; however, this tends to increase the power generated near the outer radius of the fueled core. Appropriate adjustments of both control systems can flatten the power profile more than the degree shown in Fig. 11. This additional flattening amounted to only 2 - 3% for the SPR-4 reactor and it was not considered essential because some of the redundancy inherent with the dual controls would have been lost.

Table VI. Effects of neutron streaming from core end-faces as calculated by the 05R Monte Carlo code.

Δk (neutron streaming along heatpipes)	-0.005 ± 0.005
Δk (neutron streaming through central duct)	-0.018 ± 0.005
Δk (total estimate of neutron streaming)	-0.023 ± 0.007

Fig. 10. Reactivity worth of different side reflector materials.

Fig. 11. SPR-4 power profile and fuel loading.

The axial power profile assumes a natural cosine-like shape with a peak to average power density ratio of 1.35. The reduction of maximum burnup fraction and radial heat flux, which could be achieved by flattening the axial power profile, was not considered essential to the core design.

The nuclear heating rates within the reactor-boiler were calculated using the two-dimensional transport code,

DOT.¹⁹ These heating rates include fission, elastic and inelastic neutron scattering, and (n, α) events as well as the effects caused by gamma ray absorption. The volume-integrated heat generation for each component within the reactor is given in Table VII. The maximum integrated fast neutron flux in the reactor-boiler is 10^{27} nvt. The maximum heating rate in the lithium hydride nuclear shield is approximately 0.2 W/cm^3 .

Table VII. Nuclear heating in the reactor-boiler.

Component	Heating (kW)		Total
	Neutron	Gamma	
Lithium-6 control region	2,388	6,186	8,574
Core	1878.	121.	2000.
Tungsten core can	0,020	4,010	4,030
Liquid potassium coolant	0,014	0,106	0,120
Thermal insulation	0,228	1,988	2,216
Pressure vessel	0,016	2,724	2,740
Reflector	0,216	5,456	5,672
Boiler heatpipes	0,048	1,460	1,508

IV. Nuclear Shield

DESIGN CRITERION AND GENERAL DESCRIPTION

The SPR-4 shield was designed for an unmanned mission using payload dose levels of 10^{13} nvt for neutrons and 10^6 R for gamma rays. The payload dose plane was taken to be 10 m from the nearest reactor face. These dose levels represent a reasonable compromise between the cost of hardening the electronic components and the mass savings in the nuclear shield. If presently available electronics are used without regard to radiation effects, a maximum integrated fast neutron flux of 10^{11} neutrons/cm²* and an integrated gamma dose of 10^4 R could be tolerated. However, if care is used in selecting the proper electrical components, designing redundant circuits, and providing local shielding, these values may be increased to 10^{14} nvt and 10^7 R.²⁰ By incorporating a shielding half-angle of 14 deg, all turbo-machinery and the main radiator can be placed between the nuclear shield and the payload (See Fig. 4). Attenuation of radiation at

*Equivalent 1 MeV neutron flux, based upon damage in silicon.

the payload due to components other than the shield was neglected.

The shield was designed for minimum mass which was found to occur when materials were alternated between lithium hydride and tungsten as shown in Table VIII. Lithium hydride was chosen to attenuate neutrons and tungsten to attenuate gammas because of their well-known material and shielding properties and because their use resulted in a shield which was lighter than shields fabricated from the other materials considered.

The nuclear shield was shaped in the form of a truncated cone, 75.4-cm long, with its axis of symmetry oriented to correspond to the system centerline between the reactor-boiler and the main radiator as shown in Fig. 4. The shield diameter varies from 105.2 cm to 142.9 cm so that the shield outer boundary lies along the 14-deg shielding half-angle.

The shield is contained inside a stainless steel vessel that prevents hydrogen leakage. This vessel is protected against meteoroids by beryllium 1 cm thick. Necessary penetrations through the shield for plumbing and other devices should have about 3/4-turn along a helix path to

Table VIII. SPR-4 nuclear shield dimensions.

Region	Thickness (cm)	Material	Shield relative mass	Maximum diameter (cm)
1	11.8	LiH	0.10	111.1
2	1.2	W	0.19	111.7
3	<u>62.4</u>	LiH	<u>0.71</u>	142.9
	75.4		1.00	

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prevent streaming. For calculational purposes, the amount of shield material removed by the penetrations is assumed to be added to the rear of the shield so that the total shield volume remains constant. The internal heating rate in the shield is high enough to cause excessive lithium-hydride temperatures if internal cooling is not utilized. Total heat generation inside the shield is expected to be about 5 kW. This heat can be dissipated by adding heatpipes to conduct heat from the center to the surface of the shield where it can be radiated.

ANALYTICAL DESIGN METHOD

An adequate shield design includes not only the neutrons and gammas emitted by the fission process by also the gammas generated by inelastic neutron scattering, neutron capture, and activation and subsequent transport of material throughout the system. To include all of these processes, a detailed calculation was performed on the reactor which included all the neutron, gamma, and secondary gamma reactions. This work made use of the two-dimensional transport code DOT and a set of cross sections containing 16 neutron groups and 7 gamma groups. The geometry included all the components of the heatpipe reactor-boiler. Only the neutron and gamma currents hitting the shield surface were calculated because cross sections used in the DOT code are not adequate for deep penetrations through the shield. These currents are given in Table IX. As far as the shield is concerned, the reactor is simulated by an equivalent

point source emitting 0.73 neutron/fission and 1.41 MeV of gamma energy/fission.

The shield was designed using point-kernel techniques. The usual removal theory kernel was used for neutrons using the Albert-Welton kernel for hydrogen and the effective reactor point source obtained from the DOT problem. Seven energy groups of gamma rays, corresponding to those given above, were used with the appropriate build-up factors. The production of inelastic gamma rays was related to the removal of fast neutrons while the capture rate of thermal neutrons was given by solving the diffusion equation in one-dimensional slab geometry with a source given by the fast removal kernel. The thickness of the laminated regions of the shield were varied to obtain minimum shield mass using a steepest descent method.

Calculations indicated that 4.6×10^{13} n/sec were absorbed in the potassium coolant. Part of these are absorbed in the potassium-41 isotope which becomes activated and produces delayed gamma radiation. Because the potassium circulates throughout the system and behind the shield, this activation produces a fixed dose rate at the payload. Assuming that potassium-41 has the same fast neutron absorption cross section as natural potassium, that the radiator can be considered as a flat trapezoidal source of uniform strength per unit area, that half the potassium is behind the shield, and neglecting the attenuation of the radiator, it was found that the use of natural potassium (6.91% K^{41}) caused an integrated gamma dose at the payload of 2100 R. This

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Table IX. Neutron and gamma currents impinging upon the shield surface.

Group	Type	Energy range	Neutron or gamma/sec
1	Neutron	10 - 3 MeV	2.97×10^{14}
2	Neutron	3 - 1.4 MeV	5.58×10^{14}
3	Neutron	1.4 - 0.9 MeV	4.22×10^{14}
4	Neutron	0.9 - 0.4 MeV	1.20×10^{15}
5	Neutron	0.4 - 0.1 MeV	2.18×10^{15}
6	Neutron	0.1 - 0.017 MeV	5.04×10^{14}
7	Neutron	17 - 3 keV	1.02×10^{13}
8	Neutron	3 - 0.55 keV	3.45×10^{10}
9	Neutron	0.55 - 0.1 keV	4.74×10^7
10	Neutron	100 - 30 eV	4.33×10^6
11	Neutron	30 - 10 eV	5.99×10^5
12	Neutron	10 - 3 eV	2.79×10^5
13	Neutron	3 - 1 eV	8.76×10^5
14	Neutron	1 - 0.4 eV	2.17×10^6
15	Neutron	0.4 - 0.1 eV	2.39×10^6
16	Neutron	0.1 - 0.0 eV	1.22×10^6
17	Gamma	15 - 9 MeV	2.17×10^{10}
18	Gamma	9 - 7 MeV	1.32×10^{12}
19	Gamma	7 - 5 MeV	1.37×10^{13}
20	Gamma	5 - 3 MeV	7.02×10^{13}
21	Gamma	3 - 2 MeV	9.99×10^{13}
22	Gamma	2 - 1 MeV	1.63×10^{14}
23	Gamma	1 - 0 MeV	3.31×10^{14}

value is negligible when compared to the gamma dose criterion of 10^6 R.

It is possible for activated materials, such as sodium from a ruptured heatpipe, to get into the potassium working fluid and raise the dose rate at the payload. If one heatpipe ruptures at the beginning of life and all of its sodium is discharged into the power conversion loop, an integrated dose of about 2.5 R at the payload results. Rupture of 5% of all the heatpipes within the core would result in a dose of 95 R. This dose also is negligible. If a heatpipe does rupture, its fluid would only diffuse slowly into

the potassium loop because there would not be a driving pressure force. A breakdown of the gamma radiation sources is shown in Table X.

The mass of the shield can be reduced by varying the payload-to-shield

Table X. Breakdown of gamma radiation sources.

Reactor sources	6.5×10^5 R
Shield secondary sources	3.5×10^5 R
Potassium coolant sources	2.1×10^3 R
Total	1.0×10^6 R

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Fig. 12. SPR-4 shield mass versus shield-payload separation.

Fig. 13. SPR-4 shield mass versus shield half-angle.

Table XI. Dose rates present outside the shield shadow.

Distance from reactor	Fast neutron (n/cm ² sec)	Gamma (R/hr)	Total (Rem/hr)
10 m	1.8×10^7	1.3×10^4	2.5×10^5
100 m	1.8×10^7	1.3×10^2	2.5×10^3
1000 m	1.8×10^5	1.3	2.5×10^1

separation distance or the dose criterion as shown in Fig. 12. A position of 10 m was chosen because this distance permits a very compact system. Figure 13 indicates the dependence of the shield mass upon the half-angle of the shield.

The shield provides a reduced dose rate within the shadow only. Outside the shadow, it is possible to see the unshielded reactor package. The approximate dose rates outside of the shadow are tabulated in Table XI.

V. Power Conversion

HEATPIPE RADIATORS

The heatpipe radiator (See Figs. 1 and 4) provides a reliable method of rejecting waste heat. At the same time the heatpipe radiator has low specific mass. The advantages of redundancy and low mass are achieved through the use of about 2200 highly efficient, independent heatpipes in the main radiator. Radiator specific mass is about 0.25 g/W of rejected power.

The selected heatpipe design was that of Werner & Carlson.⁶ The main radiator is constructed of a series of manifolds, positioned in a rectangular array, with a large number of heatpipes penetrating the manifolds. The manifolds and other ducting are protected against meteoroids in the same way as other radiator designs. Because of their low presented area, however, they contribute only slightly to the total radiator mass.

An excess number of heatpipes is installed at the beginning of life. The pipe wall thickness is determined so that a certain fraction (approximately 10%) of the heatpipes fail from meteoroid penetration during their lifetimes. The number of surviving heatpipes at the end of the mission must be sufficient to reject the required heat. This number is determined by approximating the distribution of heatpipe failures with a normal distribution. The effect of the puncture of a heatpipe is only that it will cool and not contribute significantly as a heat exchange member. The ducting must have the desired degree of meteoroid protection to ensure that no punctures occur. Some characteristics of the main radiator are shown in Table XII.

The effects of various parameters on the SPR-4 main radiator mass were investigated. While the parameter variation was not exhaustive, it was complete

Table XII. Heatpipe radiator characteristics.

	Main radiator
Total heat rejection required (kW)	1565
Total projected area including piping and manifolds (m ²)	19.2
Total radiator mass (kg)	400
Number of heatpipes	2200
Heatpipe fluid	Potassium
Heatpipe wall material	Niobium
Meteoroid barrier material (where required)	Beryllium
Heatpipe rejection temperature (°K)	1025
Heatpipe inside diameter (cm)	0.75
Heatpipe radial input flux (W/cm ²)	50
Radiator survival probability	0.99

enough to specify a near-optimum main radiator design. The radiator survival probability was found to be the major parameter affecting the radiator mass. Heatpipe diameter also was found to be a strong influence on radiator mass; the smaller diameters producing smaller masses. Size selection was based upon fabricability.

Figure 14 shows the mass penalty necessary for radiator survival probabilities higher than 0.99.

Design of the auxiliary heatpipe radiator, which is necessary to cool the alternator windings, is similar to the main radiator except that the heatpipes were inserted directly into the area around the windings. Heatpipes utilizing sodium as the working fluid conduct the heat away from the windings. The auxiliary radiator rejection temperature is 770°K.

TURBINE-ALTERNATOR

Since development work on high temperature turbine-alternator assemblies for Rankine cycle space power systems has been conducted by a number of organizations,²¹⁻²³ it was decided to select a previously designed unit having operating characteristics similar to those required by SPR-4. Minor adaptations are expected to be necessary; however, the performance characteristics should not change appreciably.

The turbine-alternator characteristics shown in Table XIII were based generally on a General Electric Co. 400 kW, potassium turbine-alternator unit.⁵ The turbine portion is a six-stage counterflow unit using the potassium of

Fig. 14. Effect of meteoroid penetration survival probability on the main radiator mass.

the power conversion loop as a fluid. Interstage moisture extraction was considered unnecessary as long as adequate consideration was given to turbine bucket corrosion through appropriate aerodynamic design.

The alternator is a radial-gap, homopolar-inductor alternator. Both the stator and field coil windings of the unit are cooled heatpipes. The evaporator region of each heatpipe is embedded in the outer region of the stator casing. The condenser region extends outward from the end of the alternator to form a cylindrical ring which operates as auxiliary heatpipe radiator surface.

The turbine-alternator assembly is a standard four-bearing configuration with a flexible coupling located between the turbine and alternator. Bearings are lubricated with potassium. Hydrodynamic seals are used on the turbine rotor side of each bearing and rotor disc to preclude leakage of lubricant into turbine flow passages.

ELECTROMAGNETIC PUMP

A helical-induction electromagnetic pump, based on the design of Diedrich

Table XIII. SPR-4 turbine-alternator operating characteristics.

Speed (rpm)	24,000
Turbine	
Inlet temperature (°K)	1485
Inlet pressure (bar)	18.4
Outlet temperature (°K)	1040
Outlet pressure (bar)	1.11
Exit quality (%)	82
Flow rate (g/sec)	997
Overall efficiency (%)	80
Alternator	
Frequency (Hertz)	2400
Voltage (volts)	2400, 3-phase
Power factor	0.8
Output power (kW)	390.5
Efficiency (%)	88
Auxiliary radiator rejection temperature (°K)	770

Table XIV. Condensate pump characteristics.

Potassium flow rate (g/sec)	997
Inlet pressure (bar)	1.06
Outlet pressure (bar)	18.5
Inlet temperature (°K)	1035
Overall efficiency including cyclo-converter and transformer (%)	19.6
Total mass including cyclo-converter and transformer (kg)	200

and Gehan,⁷ was selected as the condensate pump in the power conversion loop. Pump performance characteristics are given in Table XIV. Other miscellane-

ous low-temperature pumps, which are necessary for the reactor-liquid control circuits, are not considered to be a major design problem.

VI. Alternative Systems

A more nearly optimum reactor which utilizes U^{233} along with the U^{235} and U^{238} isotopes was designed as a companion reactor to the reference SPR-4 double-ended heatpipe reactor-boiler system. Coupling of power flattening with a water immersion safety requirement allowed a 27% reduction in reactor core diameter and also a 54% decrease in fuel inventory. Beyond the obvious physical improvement in configuration, the design approach demonstrates the relative simplicity of incorporating more involved and realistic

design constraints with no effective increase in computation time. Relative characteristics of this comparison design and the SPR-4 reactor-boiler are shown in Table XV. It should be noted that the use of U^{233} together with increased allowable fuel burnup and allowable heatpipe fluxes account for most of the comparison reactor size reduction.

Possibly more important than the reactor-boiler size reduction is the development of an integrated reactor design method that includes power flattening and water immersion safety criteria.

Table XV. Comparison of SPR-4 and companion reactor characteristics.

Item	SPR-4	Companion reactor
Power (kW_t)	2000	2000
Life (hr)	2×10^4	2×10^4
Average fuel burnup	0.68	1.50
Radial peak-to-average power density ratio (at most severe transverse cross section)	1.18	1.06
Axial peak-to-average power density ratio (at most severe longitudinal cross section)	1.35	1.35
Overall peak-to-average power density ratio	1.59	1.43
Fueled core length (cm)	41.9	35.5
Mean fueled core outer diameter (cm)	33.9	24.7
Mean liquid lithium control zone outer diameter (cm)	12.8	8.4
Side reflector thickness (cm)	2.0	4.7
Number of heatpipes	756	648
Average heatpipe radial heat flux in (W/cm^2)	46.4	99.0
Average heatpipe radial heat flux out (W/cm^2)	23.2	44.5
Maximum heatpipe axial heat flux (based on vapor volume) (kW/cm^2)	7.0	21.4
Estimated fuel costs (dollars)	2.7×10^6	1.2×10^6
Associated shielding mass (kg)	1400	1100

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This design method, reported by Brown and McCauley,¹² uses mechanical-nuclear analyses that are consistent for a particular reactor design.

The SPR-4 system also can be compared to other types of conversion systems²⁴ as shown in Fig. 15. The greatest potential for improvement in this rough comparison is shown by the lithium heatpipe out-of-core thermionic diode system,²⁵ in which the specific mass can be reduced to 4.3 g/W_e at a reactor temperature of 2000°K, assuming optimistic diode efficiencies and low thermal contact resistance between the diode plates.

The Rankine cycle system with potassium as a working fluid and a sodium heatpipe reactor could achieve a minimum specific mass of 9 g/W_e by reducing the shield angle by one-half and

Fig. 15. System specific mass comparison.

increasing the reactor temperature to 1650°K. Turbines tend to become overstressed when the reactor temperature exceeds 1800°K.

VII. Conclusions

Analysis of an integral heatpipe reactor-boiler confirms its attractiveness as a space nuclear heat source. Computer techniques that incorporate a composite neutron transport — mechanical-fluid dynamic analysis now exist for the detailed design of heatpipe-cooled reactors.

The fast spectrum heatpipe reactor-boiler and the heatpipe radiator provide considerable system redundancy and permit pump operation at temperatures no higher than is required for heat rejection. High reliability is predicted.

A dual reactivity control system provides an efficient method of achieving a redundant reactivity reserve. The control is provided by centrally located lithium-6 control circuits along with an axially translating molybdenum side reflector.

Mass of the nuclear shield is minimized through the use of alternate layers of lithium hydride and tungsten. Even so, 56% of the total system specific mass of 10 g/W_e is contributed by the nuclear shield.

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M-3679, C-93a Reactor Experiments

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