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SPR-9 CONCEPT DEFINITION STUDY
FY-1986 SUMMARY

Carl E. Walter

October 7, 1986

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FOREWORD

This work was performed as part of the Lawrence Livermore National Laboratory effort in support of the Multi-megawatt Space Reactor Program of the Office of Space Reactor Projects, Department of Energy. This report follows the format requested by the Technical Integration Support Office.

Carl E. Walter
MMW Project Leader
Nuclear Systems Safety Program

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SPR-9 CONCEPT DEFINITION STUDY-FY 1986 SUMMARY

1.0 Introduction

In April 1984 Lawrence Livermore National Laboratory proposed two concepts for meeting SDI high power requirements in space (Ref. 1). One concept, SPR-8, was based on a liquid-metal-cooled reactor with Rankine cycle power conversion while the other concept, SPR-9, was based on a gas-cooled reactor with Brayton cycle power conversion. Both of these concepts were based on considerable experience at this Laboratory in the former Space Power Reactor program and the nuclear ramjet (Pluto) program.

The Laboratory was subsequently requested to work toward the advancement of the SPR-9 design (Refs. 2 and 3). The results of that initial work are summarized in Ref. 4, and represent the effort during FY 1985. The present report summarizes the results of our work during FY 1986 under the guidance provided by Refs. 5 and 6.

2.0 General Description

Design Basis

The reactor in the SPR-9 power system concept is based on proven technology developed in the 1950's and 1960's. Development of the technology for beryllium-oxide/uranium-oxide hexagonal fuel tubes, as proposed for SPR-9, was initiated by the General Electric Co. in the Aircraft Nuclear Propulsion program and continued at LLNL in the Pluto (nuclear ramjet) program. A review of the nuclear ramjet program is given in Ref. 7. Results of the Tory II-A reactor tests, which were proof-of-principle and proof-of-material tests, are reported in Ref. 8. The engineering design of a more flight-like reactor in the program series, Tory II-C, is described in Ref. 9 and operational test results are given in Ref. 10. Figure 1 shows these test reactors ready for movement to the test bunker.

Much of the technology developed for the Tory reactors is directly applicable to the SPR-9 MMW Space Reactor Power System. The use of inert gas coolant rather than warm humid air and the probable lower power requirement of the Alert mode application (Tory II-C design power was nearly 500 MWth) ameliorate the conditions of operation for SPR-9. For the Burst mode, it is proposed to use hydrogen coolant and a higher coolant temperature rise through the core to achieve the higher power in the same reactor. (During FY 1986 we concentrated on the design for the Alert mode.)

General Configuration

The power system arrangements that we are considering have a conical configuration. We are investigating four cases. Cases 1 and 2 are indirect cycles with and without a regenerator in the power conversion loop respectively.

Fig. 1 Air-cooled reactors were designed and tested for ramjet application in the early 1960's. Test reactors were placed on rail cars for transportation to the test bunker.

Cases 3 and 4 are direct cycles with and without a regenerator. In all cases heat rejection to space is accomplished by an as yet broadly defined radiator. For the parametric study we included in our computer model a representation of a heat pipe radiator. In the model heat is rejected from the power conversion loop by transfer from the working fluid to the heat pipe evaporator sections. An arrangement of the equipment for Case 1 is shown in Fig. 2.

Fig. 2 Equipment arrangement for Case 1, indirect cycle with regeneration.

Reactor Characteristics

The reactor configuration is shown in Fig. 3. It is a prismatic ceramic core reactor composed of tens of thousand "pencil shaped" tubes made from a homogeneous mixture of moderator and fuel. The tubes are held together by radial compression forces exerted by girdle springs on the cylindrical surface of the core. A layer of zirconia insulation between the fueled core and the girdle springs allows the inlet reactor gas to cool the springs and the reactor pressure vessel before flowing through the core. Spring force requirements are minimal since maneuver loads are very low, and gas pressure gradients are favorable. This allows the springs and pressure vessel to be made from nonrefractory metal. The modest pressure drop through the core is supported at the outlet end by a ceramic dome, designed to experience only compressive stresses. This means of core support was being developed for the Tory III reactor (Ref. 7). Inlet and outlet reflectors are made from the same moderating material as in the core. Several core material choices are possible. Beryllium oxide with uranium dioxide, boron carbide with uranium boride (boron-11 isotope), and carbon with uranium dicarbide are being evaluated. These materials provide thermal or epithermal neutron spectra in the sizes considered.

Fig. 3. Reactor configuration assumed in the system study.

Based on their volatilities, these material systems are estimated to be useful at temperatures up to about 1950, 2000, and 2200 K, respectively. Higher temperatures could be achieved with appropriate coatings. Only a negligible difference in system mass results for identical geometry reactors for the different material systems. Study and experimental work is required to quantify irradiation effects and fission product retention for all three materials.

The beryllium oxide system is an outstanding candidate. This system is highly developed (nuclear ramjet), has good mechanical and physical properties, and results in the lowest critical mass for the same size and porosity reactor. A possible drawback (which needs to be evaluated carefully) is to what extent its toxicity represents a hazard to the public in the event of a launch accident.

The anticipated structural, chemical, and thermal properties of the boron system make this an excellent candidate, particularly for high burnup missions where a burnable poison (boron-10) would be desirable. Sufficiently pure boron-11 can be obtained at a reasonable cost with the desired amount of boron-10 for reactivity control over the design operating life.

As in the case of beryllium oxide, the carbon system is also highly developed (nuclear rocket), has the highest operating temperature capability, and has excellent thermal stress characteristics. Multiple hole fuel elements are more feasible in the carbon system, should that be desirable. The major disadvantage of the carbon system is its relatively high critical mass. All three material systems require much lower fuel inventories, however, than is required for fast reactors.

Of the three, the beryllium and boron systems can indeed be thermal reactors at a reasonable core size (less than 1 m diameter). They have cold clean critical masses of 5 kg compared with 250 kg for the carbon system which has a median fission energy of hundreds of eV. Other results obtained from our neutronic study of these systems are summarized in Appendix 1. In the neutronic analyses, the effect of fuel burnup on criticality is considered by assuming that the reactor operates at 50 MW for 1 y.

Besides the direct advantage of a low fuel inventory as regards the safeguard issue, a low fuel concentration in the moderator material is not expected to alter the properties of the moderator material significantly. Although a high burnup fraction results, potential damage to the fuel tube during irradiation would affect a relatively small volume within the predominantly moderator material.

For these reasons and also since the carbon system is being evaluated by Westinghouse in their "NERVA Derivative Reactor" System concept, we have begun to focus our attention on the beryllium and boron systems.

Reactivity Control

Three types of reactivity control of the reactor are being considered: 1) burnable poison and thorium, 2) variable leakage reflector, and 3) internal absorber rods.

A burnable poison incorporated in the moderator material will account for much of the reactivity swing required due to fuel depletion and neutron absorption by fission products. In the case of the boron system the burnable poison is boron-10. Other poisons will be evaluated for use in the beryllium system. The use of thorium will also be evaluated as a means of reducing the required fuel inventory.

A beryllium reflector which parts circumferentially at the middle is located outside the pressure vessel as seen in Fig. 3. The reflector will be connected to actuators which can translate each half away from the core mid-plane. With a 68 cm diameter core, the reflector has a high reactivity worth despite the return coolant annulus surrounding the core and the intervening pressure vessel (see Appendix 1).

We also studied the reactivity worth of internal control rods. The results, given in Appendix 1, show that considerable worth may be controlled with a reasonable number of rods. The internal control rods may be made of hafnium or boron carbide. It is planned to examine the use of pneumatic actuators located at the "forward" end of the reactor.

Shield

A neutron shield of lithium hydride is provided next to the reactor. The shielding value of the power conversion equipment is being assessed. We have also examined the effect of scattered radiation from the radiator if it were located outside the shield cone angle. Results of these shielding analyses are being reviewed at the present time.

Balance of Plant

Our system design has evolved to be more conventional. After considering the characteristics of a charged liquid droplet radiator, we have abandoned its use, at least temporarily. The reason for this is concern about the compatability of such a radiator with the space craft and its orbital environment. It would be difficult to make a conclusive judgement about the merits of a system relying on this type radiator on the time scale being considered in the MMW program.

We also briefly considered a spray-on static membrane radiator, but this design could have serious freeze/thaw problems. Instead our baseline system concept now has a heat pipe radiator. At present, this radiator is modeled in a macroscopic manner to allow an estimate of dimensions and mass of its conical frustrum configuration. Heat is transferred to the heat pipes directly by the power conversion working fluid. Thus there are no liquids (except in the heat pipes) in the system and there are no freeze/thaw issues.

Intermediate heat exchangers and regenerators if used, are counterflow compact designs which we have modeled in sufficient detail to predict their pressure drop, heat transfer, and mass characteristics. The design features are shown in Fig. 4.

Fig. 4 Sketch showing the general design of the compact heat exchangers which might be used as intermediate heat exchangers or regenerators.

Combined rotating units consisting of the turbine, compressor, and alternator in a hermetic case are modeled in terms of specific mass coefficients, lengths and diameters which are dependent on molecular weight of the working fluid, and material strength which is dependent on temperature. Our results agree with specific machines which have been described in the literature. Without a detail design, however, we are unable to determine rotational speed, and frequency and voltage of the output power. The turbine and the compressor would be axial flow machines. A superconducting alternator is not used. This is a refinement which could be introduced in the future if analysis indicates that there is a net benefit in performance and reliability when the complexity of the necessary cryogen refrigeration system is introduced. For the indirect cycles (Cases 1 and 2), a motor driven radial flow turbine is used to circulate the reactor coolant.

System modularity is provided except for the reactor. Our baseline design assumes four parallel flow paths throughout, including four inlet and four outlet pipes to the reactor. All pipes in the system are modeled to permit estimates of their pressure drop and their mass. The hottest pipes in the system including the outlet plenum of the reactor can be insulated internally and cooled externally to allow them to be made of superalloy. At present, as an expedient, the system model assumes cooling by radiation. Convective cooling will be required in certain locations, for example as the pipes pass through the radiation shield.

In order to make mass estimates, it is necessary to establish an allowable material stress for structural materials. In many cases mass is dependent only on the material allowable stress-to-density ratio. We established a continuous stress-to-density curve over the temperature range of interest which we believe appropriately represents creep data and predicted creep data for superalloys, molybdenum alloys, and tungsten alloys.

The SPR-9 system concept has several distinctive attractive attributes:

- High temperature capability
- Lowest fissile inventory
- Reactor concept based on demonstrated design features
- High core and reactor power density
- Low system mass (exclusive of the radiator)
- With an intermediate heat exchanger the reactor can be used with any other chosen means of power conversion (except in-core thermionics).
- No freeze/thaw issues to be addressed
- Pneumatic control actuators (temp, rad resistant)
- Superalloy reactor pressure vessel and pipes

3.0 System Performance

System performance will be predicted on the basis of a System Performance/Mass (SP/M) model. This model consists of equations describing the thermodynamic cycle and component dimensions and masses. The equations are solved simultaneously for a specific set of constraints. The equation solver, Ref. 11, can automatically optimize, by means of trial and error any independent variable by adjusting the values of up to sixty "known" parameters.

Four versions of the SP/M model are being prepared to consider four specific equipment arrangements. The schematics for these cycles are shown in Fig. 5. The models are currently being debugged. Results will be presented in a subsequent report.

A summary of the available performance data in the format requested by Ref. 12 is given in Appendix 3.

Fig. 5 Cycle schematics which are being analyzed to determine optimum configuration for minimum system specific mass. A = Reactor, B = Intermediate Heat Exchanger, C = Reactor Circulator, D = Regenerator, E = Radiator, F = Compressor, G = Turbine, G' = Generator.

4.0 Feasibility Issues

The key feasibility issue facing the SPR-9 concept is the large radiator required. While fission product retention in the fuel tube has not been demonstrated, the prognosis, on a theoretical basis, is good. Two excellent fallback positions are available: 1) coatings on the fuel tubes, and 2) the use of a shielded intermediate heat exchanger.

5.0 Conclusions and Recommendations

Gas-cooled thermal reactors coupled with Brayton cycle power conversion appear feasible and have several attractive features for multi-megawatt space power systems. An advanced radiator design must be developed to reduce system mass. The inherent high temperature capability of the reactor removes the reactor as the limiting component on system performance. The low fuel inventories required, particularly for BeO moderated reactors, make space power systems based on thermal reactors a lesser safeguard risk.

Additional information will be developed during the first half of FY 1987 to better define the SPR-9 concept. It is hoped that sufficient information will be available to formulate reasonable conclusions about the SPR-9 concept at the end of that time and to make recommendations concerning unresolved issues.

A prioritized list of the more important issues which are known at this time is given below.

- o Development of a high performance radiator.
- o Determination that beryllium toxicity is an acceptable risk in the event of a launch pad accident.
- o Evaluation of fission product retention in ceramic fuel tubes.

6.0 References

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11. D. O. Blackketter Analytical Design (AD): An Algebraic and Calculus Equation Solver, to be published
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APPENDIX 2

SPR-9 POWER SYSTEM DATA

September 30, 1986

The data presented in this Appendix is in the format requested by the Technical Integration Support Office of the MMW Space Reactor Program.

These data do not represent an optimized system and should only be used for broad perceptions of the SPR-9 system concept.

APPENDIX 2

SPR-9 POWER SYSTEM DATA (September 30, 1986)

I. SUMMARY *

		Design Options	
		<u>Beryllium Moderated</u>	<u>Boron Moderated</u>
A. <u>Power System Summary</u>			
1. <u>Alert Mode System</u>			
a. Type	nuclear		
b. Fuel	BeO-UO ₂		B ₄ C-UB ₄
c. Spectrum	epithermal		
Median Fission Energy, ev	330		714
d. Coolant	helium		
e. Power Conversion	gas turbine		
f. Heat Rejection Mode	heat pipe radiator		
2. <u>Burst Mode System</u>			
a. Type	Nuclear		
b. Fuel	BeO-UO ₂		B ₄ C-UB ₄
c. Spectrum or	epithermal		
Median Fusion Energy, ev	330		714
d. Coolant	hydrogen		
e. Power Conversion	gas turbine		
f. Heat Rejection Mode	open cycle		
g. Power Control	TBD		
3. <u>Secondary Loop</u>		(also under consideration)	
B. <u>Assumptions</u>			
1. Max Alert Power, MWe	}	As specified for the Alert and Burst design points given in SNL letter RS6430/86/133 dated February 17, 1986.	
2. Operating Time, Alert, y			
3. Max Burst Power, MWe			
4. Operating Time Burst, s			
5. Alert Mode (Elect) Energy, MW-y			
6. Burst Mode (Elec) Energy, MW-y			
C. <u>System Masses, Mg</u>			
1. <u>Alert Mode System</u>			
a. Power Source	1.5		
b. Power Conversion	13.0		
c. Heat Rejection Systems	TBD		

* These data represent a snapshot of an evolving design. Current indications are that a larger core has additional benefits, and an update of these data will likely show lower median fission energy and fuel inventory.

I. SUMMARY (cont'd.)

		Design Options	
		<u>Beryllium Moderated</u>	<u>Boron Moderated</u>
C. <u>System Masses (cont'd.)</u>			
2.	Burst Mode System		
a.	Power Source	1.5	
b.	Power Conversion	TBD	
c.	Heat Rejection System	negligible	
d.	Energy or Coolant Storage	TBD	
e.	Power System Controls	TBD	
3.	Power Conditioning	TBD	
4.	Other		
5.	Total System Weight	TBD	
D. <u>Unique Features</u>			
1.	High temperature capability		
2.	Low fissile inventory (safeguards)		
3.	High core and reactor power density		
4.	Reactor concept based on demonstrated design features (nuclear ramjet reactor)		
5.	Low system mass (exclusive of radiator)		
6.	Prismatic core of hexagonal ceramic internally-cooled fuel tubes		
7.	No liquids, no freeze/thaw issues		

II. REACTOR AND SHIELD DATA

		Design Options	
		<u>Beryllium Moderated</u>	<u>Boron Moderated</u>
A. <u>Reactor Data</u>			
1.	Fuel		
a.	Fuel		
-	Composition	BeO-UO ₂	B ₄ C-UB ₄
-	U ²³⁵ mass (kg)	71.5	130.1
-	Total fuel mass (kg)	580	580
-	Enrichment (%)	97	97
-	Density (g/cm ³)	3.34	3.34
b.	Cladding	None	
-	Material		
-	Thickness (mm)		
c.	Configuration	close packed hexagonal fuel tubes about 10 cm long	
d.	Fuel Elements		
-	Dimensions (cm)		
-	- internal diameter	.4	
-	- flat-to-flat	.7	
-	No. of units	~ 63000	
e.	Other		
2.	Moderator		
a.	Material	see above	
b.	Mass (kg)	504	446
c.	Moderator/fuel atom ratio	66	14.4
d.	Configuration	see above	
3.	Axial Reflectors		
a.	Material	BeO	B ₄ C
b.	Mass (kg)	200	170
4.	Radial Reflector		
a.	Material	Be	
b.	Mass (kg)	415	
5.	Active Control		
a.	Type	dual: reflector leakage and core absorption	
b.	No. units	TBD (13 internal control rods likely)	
c.	Mass (kg)	TBD	
6.	Passive Control		
a.	Type	burnable and thorium	
b.	Mass (kg)	TBD	
7.	Core Dimensions *		
a.	Diameter (cm)	68	
b.	Length (cm)	68	

* There are several advantages to increasing the dimensions of the core. Core size is still under consideration.

II. REACTOR AND SHIELD DATA (cont'd.)

	Design Options	
	<u>Beryllium Moderated</u>	<u>Boron Moderated</u>
8. Reactor Dimensions		
a. Diameter (cm)	115	
b. Length (cm)	230	
9. Total Mass		
a. Core (kg)	580	
b. Reactor (kg)	~ 2000	
10. Volume Fractions		
a. Fuel	.70	
b. Coolant	.30	
c. Moderator	(included in "fuel")	
d. Structure	0	
e. Uranium in "Fuel"	.04	.08
11. Parameters		
a. Alert-Max Fuel Temp. (K)	1780	
b. Alert-Max Clad Temp. (K)	None	
c. Burst-Max Fuel Temp. (K)	1900	
d. Burst Clad Temp. (K)	None	
e. Alert-Avg. Fuel Temp. (K)	1470	
f. Burst-Avg. Fuel Temp. (K)	1100	
g. Avg. Prompt Temp. Coef. (\$/K)	TBD	
h. Prompt neutron generation time (s)	TBD	
i. Other Feedback Coef.	TBD	
j. Max n-flux, thermal (n/cm ² s)	~ 5 x 10 ¹³	
k. Max n-flux, total (n/cm ² s)	TBD	
l. Max gamma flux (R/hr)	TBD	
m. Avg. burnup (%)*	40	21
n. Max burnup (%)	TBD	
o. Max Permitted Specific Power (W/g)***	Limit not known**	
p. Avg. Power Density (W/cm ³)***	200	
q. Specific Core Heat Transfer Area (cm ² /g)	1.33	

* includes non fission burnup - assumes reactor power is 50 MW for 1 y.

** these designs require average specific power of about 700 and 385 W/g uranium for Be and B moderated cores, respectively, for assumed power of 50 MW.

*** Assumes reactor power is 50 MW.

II. REACTOR AND SHIELD DATA (cont'd.)

	Design Options	
	<u>Beryllium Moderated</u>	<u>Boron Moderated</u>
12. Reactivity Defect	TBD	
a. Burnup (\$)		
b. Fission Product (\$)		
c. Xe (\$)		
d. Temp Defect (\$)		
e. Other (\$)		
13. Water Immersion K_{eff}	TBD	
14. Reactivity Control		
a. Active (\$)	TBD	
b. Passive (\$)	TBD	
15. Pressure Vessel		
a. Material	probably superalloy	
b. Thickness (cm)	~ 0.5	
c. Max Temp (K)	1100	
16. Core Radial Support Structure		
a. Material	superalloy	
b. Max Temp (K)	1100	
17. Core Axial Support Structure		
a. Material	BeO	B_4C
b. Max Temp (K)	1600	
B. <u>Shield</u>		
1. Geometry	Conical frustrum	
2. Gamma Shield		
a. Material		
b. Thickness (cm)	TBD (advantage taken of power conversion equipment)	
c. Mass (kg)		
3. Neutron-shield		
a. Material	LiH	
b. Thickness (cm)	~ 100	
c. Mass (kg)	TBD	
4. Transmitted Dose Rate, Max	TBD	
(at shield surface)		
a. neutron (n/cm^2 sec)		
b. gamma (R/hr)		
5. Cooling Requirements	TBD	
6. Payload Separation Distance (m)	TBD	

III. NON-NUCLEAR POWER SOURCES

(NOT APPLICABLE)

IV. HEAT TRANSPORT

		Design Options			
		<u>Direct No Regen</u>	<u>Direct Regen</u>	<u>Indirect No Regen</u>	<u>Direct Regen</u>
		<u>Loop</u>			
A.	Alert Mode				
1.	Cycle type	Brayton			
2.	No. of loops	1	1	2	2
3.	Net efficiency	TBD			
B.	Burst Mode				
1.	Cycle type	Brayton			
2.	No. of loops	open			
3.	Net efficiency	TBD			
C.	Power Source Coolant Data				
1.	Alert Mode				
a.	Coolant	helium	(other gasses being studied)		
b.	Max velocity (m/s)	160			
c.	Pressure (MPa)	2			
d.	Inlet temp. (K)	1000			
e.	Outlet temp. (K)	1600			
f.	Core Pressure drop (MPa)	.075			
D.	Loop Data - (See Supplementary Tables)	Table A.1a	Table A.1a	Table A.1b Table A.1c	Table A.1b Table A.1c

Power Conversion

A.	Alert Mode	Turbo-compressor-alternator
B.	Burst Mode	Turbo-alternator
C.	Power Conversion Data - (See Table A.2)	

Heat Rejection

A.	Alert Mode	
1.	High temp. type	Advanced radiator concept needed *
2.	Low temp. type	Heat pipe radiator
B.	Burst Mode	Open
C.	Heat Rejection Data - (See Table A.3)	

* System studies are being carried out using a heat pipe radiator.

Table A.1a Supplementary Heat Transport Loop Data

		Design Options	
		<u>Direct No Regen</u>	<u>Direct Regen</u>
1.	Loop	single	
2.	Alert/Burst	Alert	
3.	Heat exchanger		
a.	Type	gas to heat pipe evaporator	
b.	Mass (kg)	TBD	
c.	Efficiency	TBD	
d.	Pressure drop (MPa)	TBD	
4.	Recuperator		
a.	Type		counterflow
b.	Mass (kg)		TBD
c.	Efficiency		TBD
d.	Pressure drop (MPa)		TBD
5.	Working fluid		
a.	Material	helium	
b.	Mass (kg)	negligible	
c.	Flow rate (kg/s)	TBD	
d.	Cycle press drop (MPa)	TBD	
6.	Heat Engine		
a.	Type	Brayton, axial flow	
b.	Mass (kg)	TBD	
c.	Materials	TBD (Mo-HfN alloy likely)	
d.	Max. or inlet temp. (K)	1600	
e.	Min. or outlet temp. (K)	1100	
f.	Inlet pressure (MPa)	2	
g.	Compression ratio	-	
h.	Turbine pressure ratio	~ 2.4	
i.	Efficiency		
7.	Pump/Compressor		
a.	Type	Axial flow compressor	
b.	Mass (kg)	TBD	
c.	Compression ratio	-	
d.	Compressor pressure ratio	~ 2.6	
e.	Outlet pressure (MPa)	2	
f.	Pumping power (MW)	TBD	
g.	Efficiency	TBD	
8.	Net Loop Efficiency	TBD	

Table A.1b Supplementary Heat Transport Loop Data

		Design Options	
		<u>Indirect No Regen</u>	<u>Indirect Regen</u>
1.	Loop	primary	
2.	Alert/Burst	Alert	
3.	Heat exchanger		
a.	Type	counterflow	
b.	Mass (kg)	TBD	
c.	Efficiency	TBD	
d.	Pressure drop (MPa)	TBD	
4.	Recuperator	None	
a.	Type		
b.	Mass (kg)		
c.	Efficiency		
d.	Pressure drop (MPa)		
5.	Working fluid		
a.	Material	helium	
b.	Mass (kg)	negligible	
c.	Flow rate (kg/s)	TBD	
d.	Cycle press drop (MPa)	TBD	
6.	Heat Engine	None	
a.	Type		
b.	Mass (kg)		
c.	Materials		
d.	Max. or inlet temp. (K)		
e.	Min. or outlet temp. (K)		
f.	Inlet pressure (MPa)		
g.	Compression ratio		
h.	Turbine pressure ratio		
i.	Efficiency		
7.	Pump/Compressor		
a.	Type	Radial flow compressor	
b.	Mass (kg)	TBD	
c.	Compression ratio		
d.	Compressor pressure ratio	< 1.1	
e.	Outlet pressure (MPa)	~ 2	
f.	Pumping power (MW)	TBD	
g.	Efficiency	TBD	
8.	Net Loop Efficiency	TBD	

Table A.1c Supplementary Heat Transport Loop Data

		Design Options	
		<u>Indirect No Regen</u>	<u>Indirect Regen</u>
1.	Loop	secondary	
2.	Alert/Burst	Alert	
3.	Heat exchanger		
	a. Type	counterflow	
	b. Mass (kg)	TBD	
	c. Efficiency	TBD	
	d. Pressure drop (MPa)	TBD	
4.	Recuperator		
	a. Type		counterflow
	b. Mass (kg)		TBD
	c. Efficiency		TBD
	d. Pressure drop (MPa)		TBD
5.	Working fluid		
	a. Material	helium (argon, He-Xe being considered)	
	b. Mass (kg)	negligible	
	c. Flow rate (kg/s)	TBD	
	d. Cycle press drop (MPa)	TBD	
6.	Heat Engine		
	a. Type	Brayton, axial flow	
	b. Mass (kg)	TBD	
	c. Materials	TBD (Mo-HfN alloy likely)	
	d. Max. or inlet temp. (K)	1600	
	e. Min. or outlet temp. (K)	1100	
	f. Inlet pressure (MPa)	2	
	g. Compression ratio	-	
	h. Turbine pressure ratio	~ 2.4	
	i. Efficiency	TBD	
7.	Pump/Compressor		
	a. Type	Axial flow compressor	
	b. Mass (kg)	TBD	
	c. Compression ratio	-	
	d. Compressor pressure ratio	~2.6	
	e. Outlet pressure (MPa)	~ 2	
	f. Pumping power (MW)	TBD	
	g. Efficiency	TBD	
8.	Net Loop Efficiency	TBD	

Table A.2 Supplementary Power Conversion Data

	Alert	Design Options	Burst
1. Type	Gas Turbine		
2. Mass (kg)	TBD		
3. Volume (cm ³)	TBD		
4. Materials	Mo-HfN alloy		
5. No. of units	4-10		
6. Shaft speed (rpm)	TBD		
7. Cooling method	TBD		
8. Coupling method	single shaft CRUs *		single shaft turbo-alternator
9. Oper. temp. (K)	1600		TBD
10. Voltage (V)	TBD		
11. Current (A)	TBD		
12. AC/DC	AC		
13. No. of phases	TBD		
14. Frequency (Hz)	TBD		
15. Specific power (kW/kg)	TBD		TBD
16. Efficiency	TBD		
17. Other	TBD		

Table A.3 Supplementary Heat Rejection Data

1. Type	
2. Mass (kg)	}
3. Surface area (m ²)	
4. Materials	
5. Working fluid	
6. Pump	}
a. type	
b. power (MW)	
7. Armor material	
8. Inlet temp. (K)	
9. Outlet temp. (K)	
10. Inlet press (MPa)	
11. Outlet press (MPa)	
12. Efficiency	
	TBD

* Combined Rotating Unit (Turbine, Compressor, Alternator)

V. ENERGY STORAGE

Design Options

(NOT APPLICABLE)

VI. POWER CONDITIONING

Design Options

Direct
No Regen

Direct
Regen

Indirect
No Regen

Direct
Regen

(TO BE DETERMINED)

APPENDIX 1

CRITICALITY STUDY FOR GAS-COOLED REACTORS FOR THE MMW SPACE POWER PROGRAM -INTERIM RESULTS-

J. S. PEARSON

September 30, 1986

(Reference Internal Report No. CSM338)

INTRODUCTION

This interim report summarizes the current results of a criticality computer study of a gas-cooled reactor power system for use in space. The work is in support of the concept definition study of SPR-9 MMW space power system being conducted at LLNL. Three fuels have been addressed: B_4C+UB_4 , $BeO+UO_2$ and $C+UC_2$. Some studies have been done over a range of reactor porosity (0.20 to 0.35) and core size (68 cm to 100 cm, where core diameter equals core height). A more detailed study has been done on the 68 cm core reactor with B_4C+UB_4 and $BeO+UO_2$ fuels. This study includes thermal effects on reactivity, reflector worth and control rod worth.

COMPUTER CODE

The ALICE computer code (Ref. 1-2) was used for the calculations described in this report. ALICE is a Monte Carlo neutron/photon transport code which uses the probability table method for handling multiband cross sections (Ref.3). This method properly treats resonance self-shielding effects. The code has been compared over a range of uranium enrichments against the Monte Carlo code MCNP which uses continuous energy cross sections. (Ref.4) Excellent agreement was found between the results of these two codes. The cross section sets used for this study were derived from the Lawrence Livermore National Laboratory Evaluated Nuclear Data Library, ENDL-86. One set of cross sections was used for "cold" calculations at room temperature (291K). Another cross section set with doppler broadening was used for "hot" calculations at 1400 K. (A considerably greater effort will be required to generate the temperature cross sections which apply to the expected core temperature distribution of about 1000 to about 1800 K.)

MODEL GEOMETRY

The standard model used for these calculations is shown in Figure 1. The fuel core consists of a right circular cylinder with height equal to diameter. This report primarily studies the case where the core dimension is 68 cm. The figure includes the region dimensions for this case. A few cases were run at other dimensions for comparison. Unless otherwise specified in the text, the core size (height equal to diameter) for these cases was changed but the thickness of all other regions was kept the same.

In addition to the standard model, several variations were also used. These will be denoted in the text and tables by a letter designator. The standard model just described, is designated as model A. Model B is similar to model A, except that holes for thirteen control rods of 1 cm diameter are included. The control rod holes pass through the fuel and axial reflector. However, when control rods are inserted in

Figure 1. Reactor model A shown for the 68cm core (all dimensions in cm). (a) Cross section through center, parallel to cylinder centerline. (b) Cross section through center, perpendicular to cylinder center line.

the model, they extend only through the fuel itself. One control rod hole was located through the center of the core lengthwise. The other twelve holes were located equidistant from one another on a radius equal to one-third the fuel radius (e.g. on a radius of 11.33 cm for the 68 cm diameter core). For the study, control rods were inserted in combinations of one, seven or thirteen. The center rod was inserted first, followed by six alternating rod positions on the circle, and finally by the remaining rod positions. Model C refers to an early study wherein the axial reflector was not porous and the He gas density used was five times larger (0.01g/cm^3).

In addition to these models, cases were also studied where the thickness of either the radial or axial reflector was varied.

MATERIAL COMPOSITIONS

Three fuels were considered: $\text{B}_4\text{C}+\text{UB}_4$, $\text{BeO}+\text{UO}_2$, and $\text{C}+\text{UC}_2$. In each case the fuel number densities were calculated as a function of the core porosity (the fraction of the core dedicated to the gas flow) and the volume fraction of the fissile compound (either UB_4 , UO_2 , or UC_2) in the fuel.

For the $\text{B}_4\text{C}+\text{UB}_4$ fuel, the following equations were used:

$$\begin{aligned}\rho &= f (1-p) 9.31 + (1-f) (1-p) 2.71 \\ C &= (1-f) (1-p) \text{Avo } 2.71/56.036 \\ B &= 4 (1-f) (1-p) \text{Avo } 2.71/56.036 \\ &\quad + 4 f (1-p) \text{Avo } 9.31/279.079 \\ U &= f (1-p) \text{Avo } 9.31/279.079\end{aligned}$$

$$\text{U235} = \text{EU}$$

$$\text{U238} = (1-\text{E}) \text{U}$$

where

$$\begin{aligned}\rho &= \text{fuel density (g/cm}^3\text{)} \\ C &= \text{number density of carbon (atom/cm.b)} \\ B &= \text{number density of boron (atom/cm . b), B}^{11} \text{ assumed} \\ U &= \text{number density of uranium (atom/cm.b)} \\ \text{U235} &= \text{number density of U}^{235} \text{ (atom/cm.b)} \\ \text{U238} &= \text{number density of U}^{238} \text{ (atom/cm.b)} \\ f &= \text{volume fraction of fissile compound} \\ p &= \text{core porosity} \\ E &= \text{uranium enrichment (assumed to be 97\%)} \\ \text{Avo} &= \text{Avogadro's number}\end{aligned}$$

Similarly for the $\text{BeO} + \text{UO}_2$ fuel, number densities were calculated:

$$\begin{aligned}\rho &= f (1-p) 11.0 + (1-f) (1-p) 3.0 \\ \text{Be} &= (1-f) (1-p) \text{Avo } 3.0/25.012 \\ \text{O} &= \text{Be} + 2 f (1-p) \text{Avo } 11.0/267.132 \\ \text{U} &= f (1-p) \text{Avo } 11.0/267.132 \\ \text{U235} &= \text{EU} \\ \text{U238} &= (1-E) u\end{aligned}$$

For the $\text{C} + \text{UC}_2$ fuel, the number densities were calculated:

$$\begin{aligned}\rho &= f (1-p) 11.5 + (1-f) (1-p) 1.8 \\ \text{C} &= (1-f) (1-p) \text{Avo } 1.8/12. + 2 f (1-p) \text{Avo } 11.5/259.04 \\ \text{U} &= f (1-p) \text{Avo } 11.5/259.04 \\ \text{U235} &= \text{EU} \\ \text{U238} &= (1-E) U\end{aligned}$$

Several studies were made using fuel loaded with additional uranium compound to approximate the fuel needed for consumption at a thermal power of 50 MW for one year. The approximate amount of U^{235} to be consumed beyond the cold critical value was calculated from the equation (ref.6):

$$\text{consumption (g)} = 0.895 (1+\alpha) \frac{P A T}{E_r}$$

where

$$\begin{aligned}P &= \text{power} = 50 \text{ MW} \\ T &= \text{time} = 365 \text{ days} \\ E_r &= \text{recoverable energy} = 200 \text{ MeV} \\ A &= \text{atomic weight} = 235 \\ \alpha &= \text{capture-to-fission ratio}\end{aligned}$$

The capture-to-fission ratio was approximated from the computer output by dividing the total number of (n, γ) reactions by the total number of $(n, \text{fission})$ reactions. For both the $\text{BeO} + \text{UO}_2$ and $\text{B}_4\text{C} + \text{UB}_4$ fuels in the standard model with 68 cm core and porosity of 0.30, the value of α was about 0.44 (actually 0.442 ± 0.003 and 0.437 ± 0.003 , respectively). Notice that this is higher than the value of $\alpha = 0.175$ that one expects for a thermal reactor. However, the higher result is quite reasonable since these reactors are much faster than thermal with median fission energies in the tens to hundreds of eV. Further, it turned out that α did not vary much with the addition of the consumption fuel in the 68 cm reactor of porosity 0.30, perhaps because the range of fission energies is rather broad so that α is not very sensitive to small changes in the median value. Using these values, the consumption mass for the standard model 68 cm core was calculated to be 27.6 kg for both UO_2 and UB_4 .

The axial reflector is composed of the same moderator as used in the fuel (e.g. B_4C , BeO or C) with the same porosity (except for model C where the axial reflector was not porous, as previously stated). The effective reflector density is thus

$$\rho_{eff} = \rho (1-p)$$

where p is the porosity and the densities (ρ) were assumed to be 1.8 g/cm^3 for graphite, 2.52 g/cm^3 for B_4C and 3.01 g/cm^3 for BeO .

The fuel is surrounded radially by concentric cylinders of various materials. The density and elemental number densities used for each are listed in Table 1. The insulator is composed of zirconium oxide (ZrO_2) plus eight weight percent yttrium oxide (Y_2O_3). The helium gas was calculated for 2 MPa at 1000 K. The pressure vessel is composed of molybdenum, the radial reflector of beryllium. The control rods are assumed to be pure B^{10} , so they represent an ideal. Because of self-shielding, natural boron or smaller diameter rods might be nearly as effective, but this has not yet been investigated.

ANALYSIS

The first study conducted (Table 2) was a criticality survey of the three fuels in a 68 cm core as a function of porosity between 0.20 and 0.35. A set of points was also determined for the $C+UC_2$ fuel in a 100 cm core. Model C was used which maintained a constant axial reflector density rather than accounting for porosity in that reflector. None-the-less, the study shows several significant effects. Varying the porosity in the 68 cm core has a dramatic effect on the mass of U^{235} fuel required in all three cases. Increasing the porosity requires a much larger quantity of fuel to compensate and yield criticality. (This effect would be even greater if the axial reflector porosity were also varied with core porosity.) The more porous reactors which require a higher fuel loading also tend to be much faster reactors with a large increase in the median fission energy, E_f . Nearly all the cases shown in Table 2 are faster than thermal, and most are in the intermediate reactor range between a few eV to 10 KeV. The graphite reactor requires much more uranium than the other two fuels for a corresponding core size and porosity. The graphite reactor also is the fastest of the three fuels (highest median fission energy). For the subsequent studies, attention was focused on the B_4C+UB_4 and $BeO+UO_2$ fuels.

Several critical points are listed in Table 3 for porosity 0.30 using model A which includes a porous axial reflector. Comparison of corresponding points between Table 2 and Table 3 indicates the relative effect of treating the axial

Table 1. Compositions of various materials used in the reactor model.

Component	Density (g/cm ³)	Element	Number Density (atom/cm.b)
Insulator ZrO ₂ +Y ₂ O ₃ (8%)	1.44	Zr	0.0061384
		Y	0.0009822
		O	0.0137494
Coolant	0.002	Hc	0.0003009
Radial Reflector	1.848	Be	0.123488
Control Rods	2.16	B ¹⁰	0.130086

Table 2. Preliminary survey of critical points as a function of porosity.

Fuel	Core (cm)	Cross Section (K)	Model	Volume Fraction	Porosity	k	$\bar{E}_f$ (eV)	Mass (kg)				
								U235	Be	O	C	B
UB ₄	68	291	C	0.01	0.20	0.997±0.008	0.23	15.02			113.5	419
UB ₄	68	291	C	0.04	0.25	1.006±0.010	56.7	56.34			103.2	390
UB ₄	68	291	C	0.063	0.30	1.006±0.008	200	82.82			94.0	361
UB ₄	68	291	C	0.093	0.35	1.007±0.008	407	113.5			84.5	332
UO ₂	68	291	C	0.004	0.25	1.006±0.008	0.037	6.96	199.4	355		
UO ₂	68	291	C	0.015	0.30	1.006±0.007	6.75	24.4	184.0	330		
UO ₂	68	291	C	0.038	0.35	1.013±0.008	83.9	57.3	167.0	304		
UC ₂	68	291	C	0.10	0.20	1.016±0.010	5130	200			341	
UC ₂	68	291	C	0.12	0.25	0.987±0.011	8170	225			317	
UC ₂	68	291	C	0.15	0.30	1.013±0.012	18300	262			292	
UC ₂	100	291	C	0.025	0.20	0.996±0.009	180	159			1119	
UC ₂	100	291	C	0.033	0.25	1.003±0.008	300	197			1046	
UC ₂	100	291	C	0.045	0.30	0.995±0.010	640	250			971	
UC ₂	100	291	C	0.057	0.35	0.993±0.006	1240	295			897	

Table 3. Comparison of several critical cases at a porosity of 0.30 using Model A.

Fuel	Core (cm)	Cross- Section (K)	Model	Volume Fraction	Porosity	k	$\bar{E}_f$ (eV)	α	Mass (kg)				
									U235	Be	O	C	B
UB ₄	68	291	A	0.078	0.30	0.999±0.005	340	0.437±0.003	102.5			192.5	360
UO ₂	68	291	A	0.027	0.30	0.998±0.003	42.8	0.443±0.003	43.8	181.8	329		
UC ₂	68	291	A	0.154	0.30	0.999±0.005	20000		269			292	
UB ₄	90	291	A	0.0015	0.30	1.005±0.008		0.016	4.57			232	353
UO ₂	90	291	A	0.0008	0.30	1.009±0.003		0.015	3.01	433	768		
UO ₂	90	291	A	0.00076	0.30	0.996±0.003		0.017	2.86	433	769		

reflector porosity. Comparing critical masses the difference is small for the graphite reactor (3%), moderate for the B_4C+UB_4 reactor (24%) and quite large for the $BeO+UO_2$ reactor (80%). Table 3 shows a comparison between a 68 cm core and a 90 cm core for both the B_4C+UB_4 and $BeO+UO_2$ reactors. The size of the reactor has a very significant effect. The larger reactor has a much lower median fission energy (in the thermal rather than intermediate energy range). The critical mass of the larger reactor is also much lower. Finally, the neutron capture-to-fission ratio, α , is much lower in the larger reactor. This is the ratio of the relative probabilities that the compound nucleus will decay by emission of capture gamma rays versus the probability of decay by fission. Since useful fuel consumption is proportional to $1+\alpha$ (ref.6), the smaller value of α means that less fuel burn-up is required per unit of power produced in the 90 cm core as compared to the 68 cm core.

Compared in Table 4 are the effects of loading the 90 cm reactor of porosity 0.30 with various fuel loadings. Both the B_4C+UB_4 and $BeO+UO_2$ fuels are considered. As the volume fraction of fuel (and hence the U^{235} mass) is increased from the critical value, the median fission energy and the capture-to-fission ratio, α , increase. In Figure 2, α is plotted for both fuels as a function of the mass of U^{235} in the core.

Shown in Table 5 and Fig. 3 is the relative effect on multiplication factor, k , of varying either the axial or radial reflector thickness. Results are shown for both the $BeO+UO_2$ and B_4C+UB_4 fuels at porosity of 0.30 in a 68 cm reactor. As can be seen, the reflector thickness has a significant worth in these reactors. The radial reflector, which could be used as a control on the reactivity, is worth about 5-7% on k . The axial reflector is worth about 6-8%.

The thermal effects on k were studied for the 68 cm and 90 cm core with the $BeO+UO_2$ and B_4C+UB_4 fuels. The nuclear effect of doppler broadening was tested using a set of cross sections doppler broadened for 1400 K (Ref.7). The geometrical effect was tested in a conservative way by thermally expanding the core and axial reflector model, but not the other reactor materials. For B_4C a linear expansion factor ($\Delta L/L$) from room temperature to 1400 K was determined from Ref. 8 to be 0.60%. For BeO , the corresponding linear expansion fraction was determined from Ref. 9 to be 1.07%. The results of the study are summarized in Table 6. For the 68 cm core a small reduction in k may be seen from the material expansion, especially in the $BeO+UO_2$ core which has the larger linear expansion fraction. However, no clear conclusion can be drawn for the cross section effect of doppler broadening within the statistical uncertainties. Since both of these reactors studied are rather fast with median fission energies of tens to hundreds of eV, one

Table 4. Comparison of the effects of loading the 90 cm core at porosity 0.30 with various fuel loadings for both B_4C+UB_4 and $BeO+UO_2$.

Fuel	Core (cm)	Cross-Section (K)	Model	Volume Fraction	Porosity	k	$\bar{E}_f$ (eV)	α	Mass (kg)				
									U235	Be	O	C	B
UB_4	90	291	A	0.078	0.30	1.213 ± 0.010	273	0.444 ± 0.006	237.8			214.4	833
UB_4	90	291	A	0.040	0.30	1.146 ± 0.007	57.5	0.453 ± 0.005	121.9			223.2	843
UB_4	90	291	A	0.015	0.30	1.111 ± 0.007	0.75	0.366 ± 0.006	45.7			229.0	850
UB_4	90	291	A	0.002	0.30	1.040 ± 0.008	0.017	0.213 ± 0.003	6.10			232.2	853
UB_4	90	291	A	0.0015	0.30	1.005 ± 0.008	0.016	0.204 ± 0.003	4.57			232.3	853
UB_4	90	291	A	0.001	0.30	0.934 ± 0.007	0.015	0.194 ± 0.003	3.05			232.4	853
UO_2	90	291	A	0.027	0.30	1.198 ± 0.010	42.9	0.446 ± 0.006	101.7	422	763		
UO_2	90	291	A	0.010	0.30	1.185 ± 0.006	0.44	0.352 ± 0.004	37.7	429	767		
UO_2	90	291	A	0.002	0.30	1.149 ± 0.009	0.018	0.220 ± 0.003	7.53	432	769		
UO_2	90	291	A	0.0008	0.30	1.009 ± 0.003	0.015	0.193 ± 0.003	3.01	433	769		

Table 5. Effect of varying axial or radial reflector thickness on the
 $\text{BeO}+\text{UO}_2$ and $\text{B}_4\text{C}+\text{UB}_4$ reactors.

Fuel	Core (cm)	Cross- Section (K)	Model	Volume Fraction	Porosity	k	$\bar{E}_f$ (eV)	Axial Reflector Thickness (cm)	Radial Reflector Thickness (cm)
UO_2	68	291	A	0.027	0.30	0.998 ± 0.003	42.8	10	10
UO_2	68	291	A	0.027	0.30	0.983 ± 0.010	47.9	6	10
						0.970 ± 0.009		5	10
						0.967 ± 0.008	49.6	2.5	10
						0.924 ± 0.007	49.7	0	10
						0.994 ± 0.006	50.0	10	5
						0.967 ± 0.007	49.2	10	2.5
						0.950 ± 0.006	52.2	10	0
UB_4	68	291	A	0.078	0.30	0.999 ± 0.005	340	10	10
						0.976 ± 0.004	330	5	10
						0.955 ± 0.004	335	2.5	10
						0.935 ± 0.005	335	0	10
						0.980 ± 0.005	362	10	5
						0.957 ± 0.006	366	10	2.5
						0.934 ± 0.003	368	10	0

Figure 2. Capture-to-fission ratio plotted as a function of the U^{235} mass for the 90cm core at porosity 0.30, cold cross sections.

Figure 3 (a). Multiplication factor, k , as a function of the axial reflector thickness.

Figure 3 (b). Multiplication factor, k , as a function of the radial reflector thickness.

Table 6. Effects of using 1400K cross sections and thermal expansion at the core either separately or together on 68cm core.

Fuel	Core (cm)	Cross- Section (K)	Model	Volume Fraction	Porosity	k	$\bar{E}_f$ (eV)	Comment
UO ₂	68	291	A	0.027	0.30	0.998±0.003	42.8	Cold
UO ₂	68	1400	A	0.027	0.30	1.003±0.003	42.8	Use 1400 K cross sections, but no thermal expansion.
UO ₂	68	291	A	0.027	0.30	0.992±0.003	42.9	Thermal expansion of core, use cold cross sections.
UO ₂	68	1400	A	0.027	0.30	0.985±0.002	42.4	Thermal expansion of core and 1400K cross sections.
UB ₄	68	291	A	0.078	0.30	0.999±0.005	340	Cold
UB ₄	68	1400	A	0.078	0.30	1.004±0.003	307	Use 1400 K cross sections, but no thermal expansion.
UB ₄	68	291	A	0.078	0.30	0.994±0.004	302	Thermal expansion of core, use cold cross sections.
UB ₄	68	1400	A	0.078	0.30	0.998±0.002	306	Thermal expansion of core and 1400 K cross sections.
UB ₄	90	291	A	0.0015	0.30	1.009±0.003	0.016	Cold
UB ₄	90	1400	A	0.0015	0.30	0.888±0.003	0.070	Use 1400 K cross section, but no thermal expansion.
UO ₂	90	291	A	0.00076	0.30	0.996±0.003	0.017	Cold
UO ₂	90	1400	A	0.00076	0.30	0.863±0.004	0.072	Use 1400 K cross sections, but no thermal expansion

probably doesn't expect to see much effect from doppler broadening. However, the larger 90 cm cores for both the B_4C+UB_4 and $BeO+UO_2$ fuels, which have median fission energies which are thermal, show a very significant decrease (about 12-13%) in reactivity with rising temperature as a result of the doppler broadened cross sections at 1400 K.

The relative effectiveness of 1 cm diameter control rods modeled of pure B^{10} was studied. Results are listed in Table 7 for the 68 cm core with either B_4C+UB_4 or $BeO+UO_2$ fuels. The reactor was modeled in these cases including the excess consumption fuel to provide 50 MW of power (thermal) for one year. The geometrical model and fuel burnup requirements were discussed earlier. The rods are quite effective. The central rod has a very significant worth, and the addition of six more rods on a radius one-third the distance out from the center of the core is sufficient to reduce the multiplication factor several percent below critical.

Table 7. Study of effectiveness of 1-cm-diameter B¹⁰ control rods.

Fuel	Core (cm)	Cross- Section (K)	Model	Volume Fraction	Porosity	B ¹⁰ Rods	k	$\bar{E}_f$ (eV)	α	Mass (kg)				
										U235	Be	O	C	B
UB ₄	68	1400	B	0.09926	0.30	0	1.057±0.009	559	0.422±0.005	130.12			90.1	355.8
UB ₄	68	1400	B	0.09926	0.30	1	1.025±0.006	613	0.423±0.008	130.12			90.1	355.8
UB ₄	68	1400	B	0.09926	0.30	7	0.970±0.001	631	0.415±0.006	130.12			90.1	355.8
UB ₄	68	1400	B	0.09926	0.30	13	0.899±0.003	714	0.413±0.005	130.12			90.1	355.8
UO ₂	68	1400	B	0.04417	0.30	0	1.044±0.005	132	0.455±0.003	71.53	178.12	326.0		
UO ₂	68	1400	B	0.04417	0.30	1	1.011±0.011	139	0.454±0.007	71.53	178.12	326.0		
UO ₂	68	1400	B	0.04417	0.30	7	0.941±0.009	166	0.450±0.009	71.53	178.12	326.0		
UO ₂	68	1400	B	0.04417	0.30	13	0.870±0.008	330	0.448±0.010	71.53	178.12	326.0		

CONCLUSIONS

1. In the range of core size (core height and diameter of 68 to 100 cm) and porosity (0.20 to 0.35) the graphite reactor operates far above thermal neutron temperatures and requires a large quantity of U_{235} fuel (several hundred kg).
2. For both the B_4C+UB_4 and $BeO+UO_2$ fuels, the core size has a very significant effect. Larger reactor cores (e.g. 90 cm height and diameter) operate in the thermal neutron temperatures, have a low capture-to-fission ratio (e.g. about 0.20), and require comparatively less critical mass. Smaller reactors (e.g. 68 cm) operate at much higher neutron temperatures, have a significantly higher capture-to-fission ratio, and require a substantially larger uranium critical mass.
3. Changes in porosity also have a strong effect. Higher porosity cores tend to have higher median fission energy and require a larger critical mass.
4. For the 68 cm core with UB_4 and UO_2 fuels, there appears to be a small negative temperature coefficient of reactivity resulting from expansion of the core. However, the neutronic temperature effect from doppler broadened cross sections is too small to be separated from the statistical uncertainties.
5. The 90 cm core with either the UB_4 or UO_2 fuels, which have thermal fission energies, displays a very significant negative temperature coefficient of reactivity resulting from doppler broadened cross sections.
6. For the 68 cm core with UB_4 and UO_2 fuels, both the radial and axial reflectors contribute significantly to the reactivity: 5-7% in k for the radial reflector; 6-8% for the axial reflector.
7. For the 68 cm core with UB_4 and UO_2 fuels, a small number of control rods has been shown to be capable of controlling the reactivity.

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