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RADIOISOTOPE-FUELED POWER SUPPLIES

Lecture Presented at University of California,
Los Angeles, California

UCLA Short Course
Space Power Systems

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by

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Baltimore, Maryland

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1.0 INTRODUCTION

Investigation and use of the space medium have brought forth the need to develop many new technologies, among which is included the requirement for small, compact, lightweight and reliable power supplies. Toward this end, the Atomic Energy Commission, since 1956, has sponsored the fruitful development of isotopic power. The program has been directed primarily to develop a family of space power systems consistent with the factors mentioned.

The principles underlying radioisotopic power are simple. Heat is generated when the particulate and electromagnetic radiations emitted by a sealed quantity of radioisotopes are absorbed in the surrounding containment material. The heat is then partially converted into electricity by a suitable energy-conversion device, the remaining heat being dissipated to the external environment.

Radioisotopes present sources of energy in relatively high power densities whose natural decay rates determine, in major part, the design lifetime of the power plant. Since these systems have no moving parts, they lend themselves appropriately to the performance of space missions. It will be seen in the ensuing text that these power units can be used most effectively for low powered, long-lived earth-orbit missions, obviating the need for reliance on a battery-charging capability; lunar landings in which power is required during the long lunar night; probes into the dense atmosphere of Venus; and, in general, missions away from the sun. Against these advantages are the considerations that present-day availability of radioisotopes is low and that the practical upper limit of their power-producing capability, based on today's energy-conversion efficiency, is of the order of only several hundred watts. Regarding availability, it is noteworthy that the lack today is not in quantities of radioisotopes, but rather in facilities to separate, purify, and process the active materials into forms useful for power application.

Within the power ranges of interest here, solar cell technology represents the principal competitor for the exploitation of these radionuclide energy sources. Solar cells have been employed in a wide variety of space missions, of which several have resulted in the achievement of solar cell design expectations. We believe the role of nuclear energy in space will soon be established and these power supplies will perform missions hitherto unattainable by present-day technologies.

This chapter will explore first of all those considerations which permit selection of a radioisotope that meets the complex requirements of the ultimate power supply. Methods of separation and purification of useful isotopes will be discussed in representative fashion. Only brief mention will be made of the principles of direct conversion of heat to

electricity, since these are covered adequately by the authors of other chapters. Design approaches for isotopic power units will be discussed extensively, since the use of isotopes as the source of heat and the conditions which prevail in launching the generators into space and ensuring their operation in that environment are not common to more conventional power supplies. A delineation of the approach to the solution of the nuclear safety problem will be made, followed by predictions for the future of isotopic power. Sample problems of the type solved routinely by engineers involved in the SNAP program are given.

I wish to acknowledge my appreciation to all the members of the staff at Martin Marietta Corporation who cooperated in producing this chapter, and particularly to Mr. Justin Bloom, Miss Bonnie Bianchi, and Mrs. Carolyn McQueen.

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(Since preparation of this chapter, a SNAP radioisotope-fueled thermoelectric generator was launched into orbit aboard the Navy's Transit IV-A navigation satellite. Appendix B contains additional details on this operational system.)

J. G. M.

2.0 THE HEAT SOURCE

2.1 Introduction

Heat is generated when the radiations emanating from the source material are absorbed within the source and its containment. Two basic types of heat source materials are considered for space power applications and will be treated in detail in this section. These are the wastes from the controlled fission of uranium in nuclear reactors, referred to as fission products, and those materials produced through irradiation of suitable target materials. This irradiation is normally performed in reactors that generate high thermal-neutron fluxes, such as the Materials Testing Reactor (MTR), the Engineering Test Reactor (ETR), and commercial reactors. Examples of heat source materials produced from irradiation are noted below.

Fission. From the splitting of uranium atoms, two typical and useful radioactive constituents (strontium and cerium) are formed as follows (Ref. 1):

Transmutation. If the target material is not readily fissionable, often the neutron bombardment will result in the formation of a transmuted element as shown in the following equations (Ref. 2):

When examining the problems of a specific mission, a waste fission product is considered first, since a major problem in the nuclear power industry is the disposal of these wastes. In the next 20 years, this might well become the most critical problem. A prediction for the year 1975 states that there will be 12 billion curies of fission products formed that year. In 1958 the radioisotope sales by the U. S. Atomic Energy Commission amounted to only 160,000 curies. Since this 12 billion curies of fission product waste amounts to 50 megawatts of power, it is obvious that some means of using this waste energy should be developed.

The fission process produces more than 200 radioisotopes, but few are suitable for producing power for reasonable periods of time. Fission products with half lives shorter than 100 days are, in general, unsuitable because of processing and handling problems encountered at the production site. In order that the heat source may be kept reasonably small, the radioisotope in its pure state must have a specific power of at least 0.01 watt/gm (Ref. 3). These criteria must be met for efficient conversion of the heat to useful work, and for minimizing the shielding required should personnel have access to the power unit area. Applying these criteria narrows the field of applicable radioisotopes very rapidly.

It is often suggested that the use of mixed rather than separate fission products lowers the cost of separation and handling. However, if mixed fission products are used, a concentration process is needed to increase the specific power of the fuel to an acceptable level. One such process is the Brookhaven National Laboratory calcining process, which results in a fused mass of fission products and process salts. Since these fission products are cooled for at least 100 days before processing, the estimated concentration of the fused salt mixture is 15,000 curies/lb, and its average decay energy is 0.75 Mev. This results in a specific power of 0.01 watt/gm (Ref. 3), but after 100 days the decrease in radioactivity, due to the short-lived fission products, results in much lower specific power. The mixed fission products do not have sufficient power density for space missions, and the rapid decay energy leads to a variable power output, making control problems very difficult. Chemical stability at elevated temperatures is also a factor of major importance in the safe packaging of radioactive fuels. The chemical composition of mixed fission products is unpredictable at this time and, therefore, impossible to control.

Pure fission product heat source materials are clearly preferable, but the choice of fuel is limited by availability and cost considerations. Fission product recovery processes have been under development and study at various national laboratories in the United States for many years. The data used to calculate the specific power of fission products were obtained as a result of work performed at Oak Ridge National

Laboratory (ORNL). The fuel availability is determined by the capacity of a recently completed separation pilot plant. Only those radioisotopes which can be processed by ORNL in sufficient quantities to produce power are considered. As demands for larger quantities of radioisotopes develop, additional separation facilities can be made available at Hanford Laboratory.

The amount of material presently available from reactor wastes has been estimated to be about 2 billion curies of gross fission products (see Fig. 2-1). An estimate of the total heat generation from wastes produced at the Hanford Facilities of the Atomic Energy Commission shows that, with a minimum of refining, the resultant slurry after calcining would be a mixture of sulfates and oxides containing the fission products: Strontium-89, Strontium-90, Yttrium-91, Zirconium-95, Cesium-137, Cerium-144, Promethium-147 and their daughter isotopes.

Provided that a 10-day period were allowed for cooldown, the amount of power generated would decrease over an 18-month period, as shown in Fig. 2-2.

The quantities of gross fission products available in the future can be estimated from Tables 2-1 and 2-2.

The use of fission products is limited for some specific applications; these limitations are a result of the size and weight of the shielding required so that biological specimens may exist close to the sources for long periods and that radiation damage to electronic components or film used in the specific mission may be prevented. The answer to these problems lies in the use of alpha emitters as heat sources. With the proper selection of alpha emitters, a kilowatt of heat can be safely contained in a volume of three cubic inches.

Extensive examination of the alpha-emitting isotopes available, in view of present-day technology, has shown that there are only three isotopes that have suitable potential as alpha heat sources; these isotopes are: Curium-242, Polonium-210 and Plutonium-238. Potential availability and cost, plus the lack of hard gamma radiation, were the deciding factors in selecting the three isotopes. Two of the selected isotopes, Curium-242 and Polonium-210, are in the short life, high power category, while Plutonium-238 has an 86.4-year (Refs. 4 and 5) half life and will meet the requirement for long-lived heat sources.

2.2 Types of Radioisotopes

The two principal types of radioisotopes considered suitable for high thermal energy release are the beta and alpha emitters. The others (i.e., K-capture, L-capture, internal conversion or isomeric transition) are, for the most part, not sufficiently energetic to be suitable.

Fig. 2-1. Estimated Accumulation of Long-lived Fission Product Activities from an Expanding Power Program (for fission products with half lives of 60 days or longer)

Fig. 2-2. Power Generation of Gross Fission Products

TABLE 2-1

Quantities Available for Thermal Power Sources

<u>Waste Fission Products</u>	<u>Curies/year x 10⁶--1960</u>	<u>Predicted Curies/year--1975</u>
Ce-144	2	
Sr-90, Y-90*	2	
Pm-147*	4	
Cs-137	2	
	Total	12 x 10 ⁹

*ORNL Catalog of Radioisotopes

TABLE 2-2

Availability of Power for Isotope-Powered Generators

<u>Fission Product</u>	<u>Specific Power (watts / kilo- curie)</u>	<u>Power (watts/year)</u>			
		<u>1960</u>	<u>1965</u>	<u>1970</u>	<u>1975</u>
Cs-137	(3.44)	4,000		30,000	
Ce-144	7.82	10,000		700,000	
Sr-90	6.52	2,000		60,000	
Pm-147	0.37	250			
Am-241	An intermediate compound				
Total		368 x 10 ³	3211 x 10 ³	15809 x 10 ³	50 x 10 ⁶

Beta-emitting isotopes. Almost all of the radioactive fission products formed from reactor fuels are beta emitters. The attenuation of the beta particle in the fuel and encapsulating material will often result in the generation of heat with a simultaneous release of gamma- or X-radiation, this radiation being known as bremsstrahlung or "braking radiation." When large quantities (kilocuries) of these fission products are concentrated in a capsule, the continuous decay of the isotope results in the release of useful amounts of thermal energy. The heat generation from the oxide of pure Cerium-144 (CeO_2), for example, is theoretically 19.8 watts/gm. Only about 2.18 watts/gm (Ref. 6) have been achieved in practice because of the presence of nonradioactive cerium and other contaminants introduced during chemical processing.

Alpha-emitting isotopes. The alpha-emitting isotopes of practical interest are produced by reactor irradiation of heavy elements such as bismuth, americium, etc. All of the energy of the alpha particles is absorbed in the source, except in the case of alpha-neutron reactions; the neutrons from these reactions may create low-level radiation fields outside the source, but the energy loss is negligible. (Figure 2-3 shows the neutron yields from reactions of Polonium-210 alphas with the elements lithium through argon.) This applies also to gamma rays emitted when alpha decay leads to an excited state of the daughter nuclide. The energy range of alphas from isotopes with practically useful half lives is from 6 to 7 Mev; the energy of betas and gammas, however, is less than 3 Mev. For these reasons, alpha sources permit higher thermal power densities than do other isotopes, and they greatly relax biological shielding requirements as mentioned previously.

An additional characteristic to be considered in comparing various fission products for use as power sources is the level of specific power with respect to time. Figure 2-4 presents this function for separated fission products. The conditions of the particular mission dictate the relative importance of the power level and the decay rate. As the plots show, some isotopes exhibit high specific power and rapid decay, while others maintain a nearly uniform specific power and, therefore, a nearly uniform overall power output. Tables 2-3 and 2-4 and Figs. 2-4 through 2-7 display specific power, decay rate, and decay energy for a number of fission product and reactor-produced isotopes.

2.3 Design Criteria for Isotopic Power Sources

The use of radioisotopes as heat sources has been demonstrated in thermoelectric generators and, more recently, in thermionic units. The selection of suitable radioisotopes for such applications is of utmost importance. Some isotopes require massive shielding from bremsstrahlung or require dilution with inert material to reduce heat generation (specific power). Some isotopes may effect rupture of their

Fig. 2-3. Neutron Yields from Polonium-210 Alphas with Elements from Lithium (Z=3) to Argon (Z=18)⁽¹⁾

(1) SNAP Programs, Quarterly Report No. 3, Tasks 2, 3, 5 and 6 for the period Apr 1 to June 3, MND-P-3011, The Martin Company

Fig. 2-4. Specific Power of Pure Fission Products Versus Time

Fig. 2-5. Specific Power of Product Compounds Versus Time-ORNL

Fig. 2-6. Specific Power of Pure Reactor-Produced Isotopes Versus Time

Fig. 2-7. Isotope Electrical Power (watts) Versus Time

TABLE 2-3
Decay Energies of Fission Product Isotopes

Isotope and Daughter	$\lambda/\text{sec}^{(1)}$	Beta Particles		E avg ⁽²⁾ Factor	E avg ⁽²⁾ (Mev)	Gamma Transitions				Total Usable ⁽²⁾ E (Mev)
		E max (Mev) ⁽¹⁾	Percent ⁽¹⁾			E ⁽¹⁾⁽³⁾ (Mev)	Percent ⁽¹⁾	E total ⁽¹⁾ (Mev)	50% ⁽²⁾ Total	
Ru-106	2.20×10^{-8}	0.0392	100	0.25	0.009	0	0	0	0	
Rh-106	2.31×10^{-2}	1.0 or less	6	0.35	1.358	2.40	0.25	0.245	0.122	1.489
		2.0	3	0.395		1.55	0.5			
		2.44	12	0.41		1.045	2.0			
		3.1	11	0.426		0.87	1.0			
		3.53	68	0.436		0.624	12.0			
						0.513	25.0			
Ce-144	2.76×10^{-8}	0.170	30	0.272	0.046	0.134	~9.0	0.043*	0.021	1.321
		0.300	70	0.286		0.100	~10.0			
						0.0807	~4.0			
						0.054	~10.0			
						0.0337	10.0			
Pr-144	6.60×10^{-4}	0.8	3	0.33	0.264	2.19	0.8	0.08	0.04	
		2.3	2	0.398		1.48	0.25			
		2.98	95	0.41		0.695	1.6			
Ce-141	2.51×10^{-7}	0.442	67	0.301	0.133	0.145	67.0	0.097	0.049	0.198
		0.581	33	0.314						
Sr-90	7.85×10^{-10}	0.61	100	0.359	0.219	0	0	0	0	1.100
Y-90	2.98×10^{-6}	2.18	100	0.403	0.878	1.74	0.4	~0		

*Energy computed from total disintegration or θ value.

Ref. (1) Atomic Waste, Its Nature Use and Disposal, edited by E. Glueckauf, AERE, Harwell, Interscience Publishers Inc., New York (pp 56 to 77) 1961
(2) Hittman, F. "The Feasibility of Producing Power from Radioisotopes," The Martin Company, Nuclear Division, MND-1031 (June 1957)
(3) Reviews of Modern Physics, Volume 30, No. 2, Part II, Amer April 1958 (Am. Physical Soc.)

TABLE 2-3 (continued)

Isotope and Daughter	$\lambda/\text{sec}^{(1)}$	Beta Particles		E avg ⁽²⁾ Factor	E avg ⁽²⁾ (Mev)	Gamma Transitions				Total Usable ⁽²⁾ E (Mev)
		E max (Mev) ⁽¹⁾	Percent ⁽¹⁾			E ⁽¹⁾⁽³⁾ (Mev)	Percent ⁽¹⁾	E Total ⁽¹⁾ (Mev)	50% ⁽²⁾ Total	
Sr-89	1.48×10^{-7}	1.463	100	0.392	0.574	0	0	0	0	0.574
Y-91	1.38×10^{-7}	1.537	100	0.386	0.593	1.22	0.3	~ 0		0.388
Pm-147	8.46×10^{-9}	0.223 μ	100	0.276	0.062	0	0	0	0	0.062
Cs-137	6.66×10^{-10}	0.523	92	0.353	0.225	0	0	0	0	0.529
Ba-137 ^m	4.44×10^{-3}	1.17	8	0.399						(formed in 92% of decay chains)
		0	0	0		0.662	100.0			
Zr-95	1.27×10^{-7}	0.364	57	0.30	0.116	0.757	49.0	0.733	0.366	0.895
		0.396	42	0.305		0.724	49.0			
		0.883	1	0.347						
Nb-95 ^m	2.14×10^{-6}	0	0	0		0.235	100.0	(0.002)	0.001	(formed in 1% of decay chains)
Nb-95	2.29×10^{-7}	0.158	100	0.273	0.044	0.745	100.0	(0.737)	0.368	(formed in 99% of decay chains)
Kr-85	2.14×10^{-9}	0.695	99	0.337	0.234	0.513	0.7	~ 0	~0	0.234

NOTE:

1. The energies in Table 2-3 are the Mev per disintegration. Daughter isotopes that make a significant energy contribution are included here.
2. Maximum beta energies are from existing tables. The average energy factor for each mode of decay varies accordingly with the atomic number of the isotope and the maximum energy.
3. Gamma rays have a long range, and a large volume of absorbing medium is required to capture the total gamma energy. The percentage of a gamma energy captured by a system is a function of atomic number, geometry of the absorbing material and the energy of the ray. An assumed average value of 50% utilization is used here.
4. The beta energy converted to bremsstrahlung is small, and energy thus lost is not considered here.

Ref. (1) Atomic Waste, Its Nature Use and Disposal, edited by E. Glueckauf, AERE, Harwell, Interscience Publishers Inc., New York (pp 56 to 77) 1961
(2) Hittman, F. "The Feasibility of Producing Power from Radioisotopes," The Martin Company, Nuclear Division, MND-1031 (June 1957)
(3) Reviews of Modern Physics, Volume 30, No. 2, Part II, Amer April 1958 (Am. Physical Soc.)

TABLE 2-4
Specific Power Data Derived from ORNL Experiments⁽¹⁾

Parent-Daughter	Stable Nuclides also Present (a)	Active plus Stable Nuclides (elements) (gm ₀ /kilocurie*) (a)		Compound (gm ₀ /kilocurie*) (experimental) (a)	Element (watts ₀ /gm ₀)	Compound (watts ₀ /gm ₀)	Compound (gm ₀ /watts ₀)	Remarks
Ru-Rh-106	Ru-101	~9.55	(b)	15.1 (b)	0.925	0.587	1.701	t = 0
	Ru-102	(~19)	(b)	Metal				
	Ru-104			(30) (b)				t = 1 yr
Ce-Pr-144	Ce-140	0.584	(b)	1.088 (b)	13.48	7.23	0.1383	t = 0
	Ce-142	(~1.5)	(b)	CeO ₂ (2.6) (b)				t = 1 yr
Sr-Y-90	Sr-88	9.5		~70 SrO	0.687	0.0934	10.70	
Y-91		0.0424		0.069 YF ₃ also as YCl ₃ and Y ₂ O ₃	54.4	33.4	0.030	ORNL producing in small quantities
Cs-Ba-137	Cs-133	31.7		39.8	0.099	0.0789	12.70	
	Cs-135			CsCl				
Pm-147	None	1.07		~7 Pm ₂ O ₃	0.344	0.0525	19.05	
Kr-85		2.223		2.223 gas	0.624	0.624	1.603	Produced only in small quantities

NOTE:

(a) These three columns are ORNL data; exceptions in footnote (b). From these values, the data in the remaining three columns are derived.

(b) For Ru and Ce, ORNL supplied specific activities at t = 1 year. These are the values indicated in parentheses. The values at t = 0 for Ru and Ce were calculated from the one year data. All other ORNL data were taken to represent specific activity at t = 0, where t = 0 represents reactor discharge time.

*The curies as determined by ORNL were based on the rate of decay of the parent isotopes.

Reference:

(1) Hittman, F., "The Feasibility of Producing Power from Radioisotope," The Martin Company, Nuclear Division, MND 1031 (June 1957).

containers due to helium pressure buildup from alpha decay. So that radioisotope heat sources may be adequately evaluated, design criteria must be established.

Of the several hundred radioactive isotopes known, relatively few can meet the following design criteria:

- (1) Power--high energy generation, high specific activity (curies/gm), specific power (watts/gm).
- (2) Half life--sufficiently long to accomplish mission.
- (3) Shielding--for gamma rays, neutrons and beta particles.
- (4) Radiological effects--effect on the encapsulation material and effect on animal life exposed because of release of the isotope to air or water.
- (5) Cost--the cost of preparing, isolating, purifying and processing to suitable form.
- (6) Availability--the processing of sufficient quantities for encapsulation.

Each of the above parameters must be individually evaluated, and the relative merits of each must be further considered in determining the conditions and design criteria for any particular mission in space.

Power. The mixed fission products comprise a spectrum of radioisotopes, not all of which can be utilized economically. As shown in Table 2-3, some have relatively short half lives; others emit low energy betas. Selected nuclides such as Strontium-90, Cerium-144, Cesium-137 and Promethium-147 have suitable half lives and power levels in the range that is practical for space systems.

It should be noted that fuel forms in Table 2-5 are considered the most practical from the standpoint of heat source fabrication. At present, those radioisotopes with thermal output greater than 0.05 watt/gm are considered to result in practical, minimum size heat sources.

Specific activity. The activity of any radioisotope or mixture of radioisotopes can be determined by the equations describing radioactive decay; these equations are treated in any text on elementary nuclear physics.

The activity of an unstable daughter is given by:

TABLE 2-5
Half Life, Power-Activity Ratio, Specific Power, Power Density, Density, and Specific Activity of Radioisotope

Radioisotope	Mode of		Half Life	Fuel Form	Power Activity Ratio (watts/kilocurie)	Specific Power		Power Density		Density		Specific Activity (curies/gm)
	Decay					Theoretical (watts/gm)	Practical (watts/gm)	Theoretical (watts/cc)	Practical (watts/cc)	Theoretical (grams/cc)	Practical (grams/cc)	
1. Cs-134	β^-		2.3 yr	Cs		13.9						
2. Cs-137	β^-		30 yr *	Cs	5.1	0.46		0.86		1.873		314(79.5)
				CsCl		0.365	0.079	1.46	0.249		3.97	250
3. Ce-144	β^-		290 days *	Ce	8	24.5	(13.48)			6.9		1708 to 3120
				CeO ₂		24.5	7.23	144.8		19	6.4	918
4. Co-60	β^-		5.3 yr	Co	2×10^{-4}	0.26			3.82	8.9		1120 to 1150
5. Cm-242	α		162.5 days	Cm	36.4	122.2	60	1650		6.77		3310
				Cm ₂ O ₃ -X Ni		109.2						
				Cm ₂ O ₃ -30 Ni				28.5		9.7	9.3	
				Cm-8 Pt					256		15.6	
6. Cm-244	α		17.9 yr	Cm		2.8		36.8	20	13.5		
7. Eu-152	β^-		12.5 yr	EuCl ₃								
Eu-154	β^-		16.0 yr	EuCl ₃								
8. Kr-85	β^-		10.27 yr *	Gas	1.27×10^{-3}	0.57	0.625					448
9. Pu-238	α		89.6 yr	Pu (Metal)	32.5	0.579		9.57	9.3	19.5	16.5	19.8
				PuC		0.516		7.52	6.9		13.6	16.5
				PuO ₂		0.480		5.86	5.51		11.5	14.8
				Pu ₂ C ₃		0.504			6.4		12.7	15.5
10. Po-210	α		138.4 days	Po	31.3	141.3		1325	1188		9.4	4460
11. Pm-147	β^-		2.6 yr *	Pm	0.437	0.411 **	0.344				4.4	932(690)
				Pm ₂ O ₃		0.35	0.525		1.1	6.6	5.0	(400) 141
12. Tl-204	β^-		3.6 yr	Tl		1.04 **				11.86		
13. Ru-106	β^-		1.0 yr *	Ru		25.6	0.525					33 to 40
				Ru cpd:50/1			0.587			12.06		66.2
14. Sr-90	β^-		28 yr *	Sr	6.4	0.921	0.687	2.46		2.66	2.55	141
-Y-90 (daughter)			2.54 days	SrTiO ₃			0.456		2.36	5.11	4.3	19.5
				SrB ₆						3.3		
				SrZrO ₃			0.36			5.48		56.5
				SrO			0.8		3.74	4.7	4.7	143

*Atomic Energy Waste, Its Nature, Use and Disposal, Edited by E. Glueckauf, A. E. R. E., Harwell, Interscience Publishers Inc., New York, 1961
**Dzhelepov, B. S. and Peker, L. K., Decay Schemes of Radionuclei, Translation Ed. D. L. Allan, Pergammon Press, London, 1961

$$N_2^t = \frac{\lambda_1}{\lambda_2 - \lambda_1} N_1^0 \left[\exp(-\lambda_1 t) - \exp(-\lambda_2 t) \right] + N_2^0 \exp(-\lambda_2 t)$$

where

N_2^t = number of daughter atoms present at time t

N_2^0 = number of daughter atoms present at time $t = 0$

N_1^0 = number of parent atoms present at $t = 0$

λ_1, λ_2 = decay constants of parent and daughter, respectively.

If the daughter is stable, the equation becomes

$$N_2^t = N_1^0 \left[1 - \exp(-\lambda_1 t) \right].$$

Different types of equilibrium can result: transient equilibrium, if parent is longer-lived than daughter, or secular equilibrium, if parent is much longer-lived than daughter.

The thermal power density of a radioisotopic source depends on the energy and kind of radiation emitted. In all cases, alpha particles are absorbed within a few microns of the emitting nucleus, except that in the case of high pressure gaseous fuels their ranges are a few millimeters. Even in the latter case, the container walls absorb the alphas.

Beta particles have ranges up to a few centimeters in ordinary materials. Because fuel capsule integrity in the event of launch abort is required, container thicknesses must often be greater than the beta particle ranges in the container materials. As is the case with alpha particles, it is reasonable to assume that the heat generation rate throughout the source is uniform. It is important, however, to determine the mean beta energy rather than the end point energy. Energy distribution curves are available for most of the common beta emitters, which may be integrated to obtain the average energy.

Although bremsstrahlung may be important in dictating shielding requirements, it represents a negligible energy leakage flux from a beta source.

For determining the specific power of any radioisotope other than a gamma emitter, provided the daughter product is a stable isotope, the following relation applies:

$$P_{sp} = \frac{W}{M_0} = 7.75 \times 10^3 \left[\frac{E_{ave}}{AT_{1/2}} \right] \exp(-\lambda t)$$

where

P_{sp} = specific power (watts/gm)

W = power, parent (watts)

M_0 = total mass at time zero (gm)

A = radioisotope atomic mass

E_{ave} = average energy per decay (Mev)

t = decay time (days)

λ = radioisotope decay constant = $0.693/T_{1/2}$

$T_{1/2}$ = half life of radioisotope (days).

If the radioisotope daughters are also radioactive, then the cumulative energy release per unit time (or time span) must be determined. Daughters that decay to nuclides of very short half lives (i.e., Yttrium-90 from Strontium-90, Barium-137 from Cesium-137, Promethium-144 from Cerium-144, etc.) assume the decay rate of the parent (secular equilibrium).

In the case of gamma emitters, the energy deposited in the source may usually be determined with ease from the energy flux escaping the container. This is a conventional shielding problem (treated quite thoroughly in standard shielding texts), and energy-dependent energy buildup factors for point isotropic sources in infinite media may be used. It is generally sufficient to assume that the remaining energy is deposited uniformly within the source and container.

Neutron-emitting sources such as spontaneously fissioning Curium-242 and others that contribute to alpha-neutron reactions deposit nearly all their energy from alpha absorption. Neutron absorption is therefore of importance only in regard to shielding design.

Method of energy conversion for radioisotopes. The average beta energy of Strontium-90 and Yttrium-90 in secular equilibrium is 0.894 Mev, or 31.9% of the sum of the end point energies.

The decay of Strontium-90 to Zirconium-90 occurs as follows:

For a 100-watt (thermal) source, the amount of Strontium-90 required is calculated as follows:

$$\begin{aligned}
 A_{\text{sp}} &= \frac{P}{(E)(C)} \\
 &= \frac{100 \text{ watts}}{(0.894 \text{ Mev/dis})(1.6 \times 10^{-13} \text{ watt-sec/Mev})(3.7 \times 10^{10} \text{ dis/sec-curie})} \\
 &= \text{Sr-90 source activity (curies).}
 \end{aligned}$$

Half life. The radioisotopes listed in Table 2-6 are among those most suitable as reliable sources of thermal energy for power generation. Power requirements for space missions varying from a few hours to several years can be met by proper selection from among these radioisotopes.

Shielding. The shielding of radioisotopes of the type selected for power generation varies considerably, depending on the type of decay, the equilibrium condition established between the daughter products, and the levels of energy. In addition to the presence of alpha and beta particles and gamma rays, there is the formation of X-rays and gamma rays as the bremsstrahlung of a beta particle in high-Z materials.

The bremsstrahlung spectrum produced by a continuous beta spectrum of end point energy, E_{β}^0 , in an absorber of mean atomic number Z is:

$$N(E_{\phi})dE_{\phi} = \left[\int_{E_{\phi}}^{E_{\beta}^0} K(Z, E_{\beta}) (E_{\beta} - E_{\phi}) N(E_{\beta}) dE_{\beta} \right] dE_{\phi}$$

TABLE 2-6
Half Lives of Radioisotopes for Power
Generation and Their Energies

<u>Isotopes</u>	<u>Half Life</u>	<u>Energy (Mev)</u>			<u>Daughter</u>
		<u>α</u>	<u>β</u>	<u>γ</u>	
Cs-134	2.3 yr	--	0.683, 0.655	0.203, 0.473	Ba-134
Cs-137	30 yr	--	0.666 (92%)	--	Ba-137
Ce-144	285 days	--	0.300 (70%)	0.080	Pr-144
		--	0.170 (30%)	0.134	
Co-60	5.3 yr	--	0.312	2.505	Ni-60
Cm-242	162.5 days	6.110	--	0.044	Pu-238
Cm-244	17.9 yr	5.801	--	0.043	Pu-240
Cm-243	35.0 yr	5.780	--	0.227	Pu-239
Eu-154	16 yr	--	1.9 (10%) 0.7 (40%) 0.3 (50%)	0.123	Gd-154
Kr-85	10.27 yr	--	0.695	0.517	Rb-85
Pu-238	86.4 yr	5.495	--	0.0434	U-234
Po-210	138.4 days	5.305	--	weak (0.803)	Po-206
Po-208	2.93 yr	5.108	--	weak (0.285)	Pb-204
Pm-147	2.66 yr	--	0.223	0.121	Sm-147
Pm-146	2 yr	--	0.75	--	--
Sr-Y-90	28 yr	--	0.545 (2.26)	weak (1.734)	Zr-90
U-232	74 yr	5.318	--	0.057	Th-232

where

$N(E_\phi) dE_\phi$ = number of photons in the energy interval between E_ϕ and $E_\phi + dE_\phi$.

$N(E_\beta) dE_\beta$ = number of beta particles in the energy interval between E_β and $E_\beta + dE_\beta$.

$K(Z, E_\beta) = 1 \times 10^{-4}$ for $E_\beta < \sqrt{\frac{300}{Z}}$.

$K(Z, E_\beta) = 7 \times 10^{-4}$ for $\frac{300}{Z} \leq E_\beta < \sqrt{\frac{43}{Z}}$.

$K(Z, E_\beta) = 15 \times 10^{-4}$ for $\frac{43}{Z} \leq E_\beta < \sqrt{\frac{20}{Z}}$.

Note: E_β in Mev.

Beta particles have definite end point energies and, therefore, definite ranges in shielding materials. If kilocurie quantities of the beta radiation are concentrated in a small volume, the decay energy can be transformed into thermal energy by the process of attenuation. However, during this process of slowing down (for example the case of Strontium-90 with its daughter Yttrium-90 in a high-Z material), intense X-rays are produced by the bremsstrahlung process as the electrons decelerate. By choice of the proper shielding material, the amount of X-rays produced from the beta particles is minimized. The amount and energy of this radiation is dependent upon the initial beta energy, the absorber and the resultant X-ray spectrum.

The shielding for alpha particles is not an appreciable problem, since a few inches of air or a few mils of a solid will completely shield this radiation. Secondary reactions create a more serious shielding problem, as alpha-neutron reactions are formed with the very low-Z materials such as boron, hydrogen, beryllium, nitrogen and fluorine, as seen in the case of polonium-beryllium neutron sources.

Radiological effects. Two basic biological hazards must be considered in the concentration of a radioisotopic heat source:

- (1) Direct radiation hazards from penetrating radiation associated with alpha and beta decay such as X-rays from the bremsstrahlung process, gamma rays from the parent and its daughter, alpha-neutron reactions and radioisotope impurities (i.e., fission products) present in the material.

- (2) The dispersion of radionuclides in the biosphere, and the subsequent assimilation of the compounds by animals, humans and aquatic life through ingestion, inhalation or absorption.

For alpha emitters, the amount of direct radiation from 100 thermal watts of isotope is shown in Table 2-7, and from a 10-thermal watt unit, in Table 2-8.

If the concept of absolute containment is employed and maintained for the alpha-emitting heat source, that heat source must be encapsulated in a high strength, high temperature and corrosion-resistant material to ensure safe containment under all conceivable conditions of environment.

Cost and availability of fuels. Difficulties are encountered in establishing the cost and availability of radioisotopic fuels since, in most cases, the Atomic Energy Commission has set the cost low to encourage the use of larger quantities of the isotopes.

There are three principal methods of producing radioisotopes in large quantities: (1) by separation of the radioisotope from the radioactive wastes produced in nuclear power reactors (Cerium-144, Cesium-137 and Strontium-90 are presently the most promising of this group); (2) by separation following irradiation of a source material that has a large thermal-neutron, radioactive-capture cross section (purified targets such as bismuth, americium and radium have been used as such source materials); and (3) by separation following natural decay or by separation following neutron capture during the operation of reactors (i.e., from spent fuels).

A fourth method of producing radioisotopes is by cyclotron bombardment with neutrons. However, this method is useful only for very small tracer-type work, and is too expensive. Table 2-9 presents the power density obtained from each of these methods of isotope preparation.

2.4 Effects of Practical Source Preparation

The radioisotope fuel source must be designed to produce a given power (thermal output) based on the design criteria of the mission. Since its rate of decay is fixed, the power density of an isotope must be changed by the diluent used in conjunction with the fuel. Figure 2-8 shows that the capsule will maintain a certain temperature, depending on the method of heat flow. The greater the dilution, the less will be the heat transmitted per unit of exposed surface. In cases where the fuels have very high specific power, it is often necessary to dilute greatly to bring the temperature to a satisfactory operating level. An example of this condition is found with the metal, Curium-242, which has a power output of 122 watts/gm, or with Polonium-210, an output of 141.3 watts/gm. Some dilution occurs in the fuel during processing despite the precautions taken to reduce the amount of dilution by using

TABLE 2-7
Radiation Associated with 100 Watts (thermal)
Unshielded Alpha Emitting Isotope Sources

<u>Isotope</u>	<u>Fuel Form</u>	<u>Gamma Radiation (rem/hr at 1 yd)</u>	<u>Neutron Radiation (rem/hr at 1 yd)</u>	<u>Total Radiation (rem/hr at 1 yd)</u>
Po-210	Po	0.020	0.0065	0.0265
Cm-242	Cm 8 Pt	0.0043	0.0325	0.0368
Cm-244	Cm	0.0083	0.0045	0.0128
Pu-238	PuC	0.00001	0.00078	0.00079
Pu-238	Pu	0.00001	0.00053	0.00054

TABLE 2-8
Comparison of Radioisotopes for a 10-Watt Thermal Heat Source

<u>Isotope</u>	<u>Half-Value Thickness of Lead (in.)</u>	<u>Thickness of Lead to Give 10 mrem/hr at 1 Meter (in.)</u>	<u>Dose Rate at 1 Meter due to Gamma Radiation (mrem/hr)</u>	
			<u>Per Curie</u>	<u>Per 10 Watts</u>
Sr-90	0.45	2.8	12.0	1.85×10^4
Cs-137	0.24	4.1	390	1.24×10^6
Ce-144	0.55	7.5	89	1.20×10^6
Po-210	0.32	None required	0.00453	144
Pu-238	--	None required	*	*
Cm-242	--	None required	*	*
Cm-244**		None required	*	*

* Radioisotopes Pu-238, Cm-242 and Cm-244 emit neutrons from spontaneous fission of the parent and from alpha-neutron reactions if the fuel form contains low-Z materials. Half-value thickness of water to reduce fast neutron dose rates is about 2.6 inches.

** For the pure metal, the neutron dose rate is 15.97 mrem/hr, requiring 1.7 inches of water to reduce the radiation to 10 mrem/hr at 1 meter. With the light elements, the values are 19.23 mrem/hr and 2.5 inches of water.

TABLE 2-9
Radioisotopes of Potential Interest for Space Power

<u>Isotope</u>	<u>Power Density (kg/kw)*</u>	<u>Production Method</u>	<u>Important Decay Radiations (mev)</u>
Po-210	0.0071	Bi (n, γ)	5.3 (α)
Cm-242	0.0080	Pu (3n, γ)	6.0 (α)
Y-91	0.051	Fission	1.54 (β^-)
Ca-45	0.084	Enriched Ca-44 (n, γ)	0.25 (β^-)
Co-60	0.12	Co (n, γ)	0.3 (β^-), 2 to 1.2 (γ)
Cs-134	15	Fission	0.65 (β^-), 6.8 (γ)
Cm-244	3.7	Pu 5 (n, γ)	α , some weak γ
Tl-204	0.86	Tl (n, γ)	0.76 (β^-)
Pm-147	1.76	Fission	0.38 (γ) 0.22 (β^-), weak γ

*Based on assumption that the average beta energy is 45% of maximum, and that 45% of the gamma energy is retained.

TABLE 2-10
Properties of Selected Reactor-Produced Radioisotopes

<u>Radio- active Isotopes</u>	<u>Target Isotope</u>	<u>Target Isotope in Natural Element (%)</u>	<u>Half Life of Product</u>	<u>Specific Power of Pure Isotope (watts/gm)</u>		<u>Cross Section (barns)</u>			<u>Time for Beta Maximum Spectrum Conversion</u>
				<u>Total</u>	<u>Recoverable</u>	<u>Target σ_{act}</u>	<u>Daughter σ_s</u>	<u>σ_b</u>	
Co-60	Co-59	100	5.28 yr	15.72	8.13	36	7	6	1 yr, allowable
Cs-134	Cs-133	100	2.34 yr	12.80	6.89	26	20		1 yr, allowable
Tm-170	Tm-169	100	129 days	11.92	11.58	130		100	1 yr, allowable
Tl-204	Tl-203	29.5	4.1 yr	0.745	0.745	8	14.0		1 yr, allowable
Po-210	Bi-209	100	138.4 days		141.3	0.019	9.0		No beta
Cm-242	Am-241	100	162.5 days		120.0				150 days

Fig. 2-8. Fuel Element Temperature with Capsule Dimensions such that $L/D = 1$

high purity chemicals (low in the elements difficult to separate). As in the case of Curium-242, present methods do not provide a method of separating Americium-241 from the desired product. Americium capsules irradiated and redissolved will contain about 23% americium from about 43% product conversion.

Presented in Fig. 2-9 is the rate of conversion of americium to curium, based on the number of days in the reactor.

Specific power of pure reactor-produced radioisotopes. A sizable source of radioisotopes for space power is the conversion of available target materials in sealed tubes by irradiation in the high thermal-neutron flux reactor. Shown in Figs. 2-10 and 2-11 are the process purification equipment and sealed capsules. Such reactors as the MTR and the ETR have been used to produce large quantities for industrial uses. Increasing the length of the irradiation of the target material does not always give maximum conversion, due to the decay or further neutron capture of the product. Therefore, selection of the minimum time required to obtain optimum production of the selected isotope must be made. Isotopes included in this group are: Cobalt-60, Cesium-134, Thulium-170, Tellurium-204 and Polonium-210. The properties of selected reactor-produced radioisotopes are listed in Table 2-10.

Since in most cases the irradiation does not convert all of the target isotope to the desired material, pure isotopes must be obtained through chemical separation and purification. This type of separation is feasible for both Polonium-210 and Curium-242.

Thermal-neutron fluxes of 10^{14} n/cm²-sec or greater can be obtained in the MTR, the ETR and others. Figure 2-12 presents the conversion ratio of reactor-produced isotopes based on an assumed flux of 5×10^{13} n/cm²-sec. Approximately 100 days at a perturbed flux of 4×10^{14} n/cm²-sec are required. The intense slow neutron irradiation of Americium-241 results in the ground state of Americium-242, with a half life of 100 years, and the metastable state, Americium-242m, with a half life of 16 hours. The metastable form will decay primarily by beta emission (80% approx) to Curium-242 or by electron capture (20% approx) to Plutonium-242. A small percentage of gamma radiation is emitted to produce the 100-year Americium-242.

Calculations of the burnup rate of Americium-241 and percent conversion, on the basis of constant effective flux values of 2, 3 and 4×10^{14} n/cm²-sec, to produce, respectively, 37.3, 41.3 and 45.2% curium in 141, 120 and 99 days are shown in Fig. 2-9.

Fig. 2-9. Rate of Conversion of Americium to Curium Versus Time in Reactor⁽¹⁾

⁽¹⁾ SNAP Programs Quarterly Progress Report No. 3, Task 2, 3, 5 and 6, 1 April through 30 June 1960, MND-P-3011, The Martin Company

Courtesy of Martin Marietta Corp.

Fig. 2-10. Split Die Block with Aluminum Tube Loaded in Position

Courtesy of Martin Marietta Corp.

Fig. 2-11. Aluminum Tubes Showing How Cap Is Mounted for Fusion Welding

Fig. 2-12. Conversion Ratio of Reactor-Produced Radioisotopes

Similar rates of radioisotope conversion by irradiation for Cobalt-60, Cesium-134, Polonium-210 and Thallium-204 are compared with Curium-242 in Fig. 2-12 and in Table 2-10.

The determination of the percent saturation of a particular isotope versus its production time in the reactor is shown in Fig. 2-13. The data were developed at ORNL.

2.5 Mixed Fission Products as a Source of Isotopic Power

As stated earlier, the operation of power reactors will produce large quantities of fission wastes; the removal of these fission products is required to recover unspent fuel. The waste products from the operation of these reactors, unless an adequate means of disposal is found, can overburden the storage capacities of all the reactor's processing and recovery facilities within the next 10 years. One of the methods suggested to alleviate this enormous problem is to utilize the gross fission products as heat sources. With elimination of such materials as Strontium-90 and Cesium-137, the activity in a fission product waste will drop sharply. Figure 2-14 shows the order of magnitude of waste activity over a 25-year period with removal of Cesium-137 and Strontium-90 as compared to the initial waste activity. The separated Cesium-137 and Strontium-90 may be further purified and processed into heat sources.

Table 2-11 gives the characteristics of some reactor-produced isotopes. Figure 2-15 gives the conversion ratios of reactor-produced isotopes. Figures 2-16 and 2-17, respectively, show the amount of natural element (at the start of irradiation) required to produce one watt at the desired time after removal from the reactor.

2.6 Cost of Radioisotope Preparation

Several surveys have been reported on the cost of radioisotopes. The cost per curie and cost per watt are extremely important to the growth of radioisotope application during the next 10 years. Figures 2-18 and 2-19 illustrate the relative rates of reduction of the cost of cesium and strontium when produced in large quantities.

Table 2-12 shows the projected availability of certain isotopes in thermal watts per year in 1960, and in typical future years during the following decade. In the case of transuranium isotopes, this information is classified.

Fig. 2-13. Percent of Saturation Versus Production Time of Radioisotopes

Fig. 2-14. Reduction In Activity from the Removal of Cs-137 and Sr-90 in Fission Product Wastes

TABLE 2-11
Reactor-Produced Isotopes

Isotope	β -Particle			E_{avg} Disinte- gration	γ -Ray			Energy per Disintegration	
	E_{max}	Percent	Average Factor		E_{mev}	Percent	Total	Total $E_{\beta} + E_{\gamma}$	Assumed Recoverable, $E_{\beta} + 0.5E_{\gamma}$
$^{60}_{27}\text{Co}$	0.308	100	0.287	0.0884	1.17 1.33	100 100	1.17 1.33	2.588	1.338
						Sum	2.50		
$^{134}_{55}\text{Cs}$	0.683	13	0.324	0.0287	0.473	2	0.00946		
	0.655	50	0.321	0.1051	0.563	9	0.05067		
	0.215	5	0.278	0.003	0.569	13	0.07397		
	0.083	32	0.260	0.0069	0.605	100	0.60500		
					0.796	91	0.72436		
					0.801	18	0.14418		
					1.038	0.9	0.00934		
					1.168	3	0.03504		
					1.367	4.6	0.06288		
			Sum	0.1437		Sum	1.71490	1.8586	1.0011
$^{170}_{69}\text{Tm}$	0.886	22	0.331	0.0645	0.084	22	0.0185		
	0.970	78	0.386	0.2542					
			Sum	0.3187			0.0185	0.3372	0.3279

NOTE: Wide variation in total gamma energy per disintegration in reported experimental results. The energies agree closely, but the fraction per disintegration for a given energy varies widely. The gamma energy available, according to different experiments, are:

Shpinel and Forafontov: Zkur. Ekspt'l. i Theoret. Frz <u>21</u> , 1376(1951)	E/Dis. = 1.6713
Wolfson: Proc. Roy Soc. Canada <u>44</u> 193A (1950)	E/Dis. = 1.3581
Muller, Hoyt, Klein, DuMond, Phys. Rev. <u>88</u> , 775 (1952)	E/Dis. = 0.7215
Jaffee, UCRL 2537 (1954)	E/Dis. = 0.819
Johns and Nablo, Phys. Rev. <u>96</u> , 1599 (1954)	E/Dis. = 0.8756
Baggerly, Marmier, Boehm, DuMond, Phys. Rev. <u>100</u> , 1364 (1955)	E/Dis. = 1.0476

TABLE 2-11 (continued)

Isotope	β -Particle			E_{avg} Disinte- gration	γ -Ray			Energy per Disintegration	
	E_{max}	Percent	Average Factor		E_{mev}	Percent	Total	Total $E_{\beta} + E_{\gamma}$	Assumed Recoverable, $E_{\beta} + 0.5E_{\gamma}$
$^{192}_{77}\text{Ir}$	0.672	100	0.313	0.2103	0.1362	2.6	0.0035	1.0859	0.6481
					0.1740	0.8	0.0014		
					0.2012	0.8	0.0016		
					0.2054	3.5	0.0072		
					0.2815	1	0.0028		
					0.2958	26	0.0769		
					0.3084	28	0.0863		
					0.3165	77	0.2437		
					0.374	0.5	0.0018		
					0.440	0.5	0.0022		
					0.4678	57	0.2666		
					0.4844	5.2	0.0252		
					0.5887	6.4	0.0377		
					0.6045	10	0.0604		
					0.6127	7.7	0.0471		
					0.745	0.05	0.0004		
					0.783	0.05	0.0004		
					0.8854	0.9	0.0079		
					1.065	0.15	0.0016		
					1.157	0.08	0.0009		
$^{204}_{81}\text{Tl}$	0.765	100	Sum	0.2103		0.8756			
			0.384*	0.2937		No gamma	0.2937		
			*First forbidden spectrum $\Delta l = 2$ years						
$^{210}_{84}\text{Po}$	No β 5.298 α				0.784	0.001	0.008	5.2988	5.2984
$^{242}_{96}\text{Cm}$	No β 6.110 α				neg.		neg.	6.098 ave	6.098 ave

Fig. 2-15. Conversion Ratios of Reactor-Produced Isotopes

Fig. 2-16. Grams Natural Element Required at End of Irradiation Period to Produce One Watt at the Desired Time After Removal ⁽¹⁾

⁽¹⁾Hittman, F., "The Feasibility of Producing Power from Radioisotope," The Martin Company Nuclear Division, MND-1031 (June 1957)

Fig. 2-17. Grams of Target Which Must Be Irradiated to Produce One Watt at the Desired Time After Removal from Reactor ⁽¹⁾

⁽¹⁾Hittman, F., "The Feasibility of Producing Power from Radioisotope," The Martin Company Nuclear Division, MND-1031 (June 1957)

Fig. 2-18. Cesium-137 Cost Estimate

Fig. 2-19. Strontium-90 Cost Estimate

TABLE 2-12
Projected Availability of Isotopes

<u>Isotope</u>	<u>Power-Activity Ratio (watts/kilocurie)*</u>	<u>Isotope Availability (watts/yr)</u>	
		<u>1960</u>	<u>Future</u>
Cs-137	5.1	3,760	31,400
Ce-144	8	10,100	700,000
Sr-90	6.4	1,630	60,000
Pm-147	0.437	260	?

*See Table 2-5 for power activity ratio of other isotopes.

3. RADIOISOTOPE PROCESSING

3.1 Americium-241 Wet Chemical Processing

In The Martin Company's Heavy Elements Processing Facility, crude Americium-241 is being processed for irradiation in a high flux reactor. The method is a modified form of the procedures used by the Lawrence Radiation Laboratories at Livermore, California.

Although the alpha particles from americium can be readily shielded, the neutron flux from spontaneous fission and alpha-neutron reactions with low-Z materials is sufficiently high to warrant special handling precautions. Chemical purification dry boxes fitted for processing 100-gram quantities of americium have been designed and fabricated as the primary containers of the radioactive material to be handled (see Fig. 3-1). Heavy duty lead-impregnated rubber gloves are required to shield workers against the gamma rays present in the crude feed, and a special console permits remote operation of the equipment.

Fig. 3-1. Chemical Purification Boxes with Attached Consoles

The system has the capability of:

- (1) Purifying 100 grams of Americium-241 solution to a purity of less than 0.01% plutonium (as determined by the pulse height analysis), and less than 1.0% of other contaminants.
- (2) Converting the purified solution to americium dioxide (AmO_2).

The flow sheet for chemical purification and mechanical processing is displayed in Table 3-1.

TABLE 3-1
Americium Processes

Wet Chemistry

Mechanical processing. The fully loaded americium capsule weighs approximately 160 grams and contains about 4 grams of AmO_2 encased in a Type 5052 aluminum (H 38) jacket with the following dimensions: wall thickness, 1/10 inch; diameter, 1.00 inch; length, 5.00 inches. Special seal-welding techniques are applied to produce high integrity targets for irradiation in a high flux reactor (i.e., MTR or ETR).

Irradiation of Americium-241. After 100 days or more of irradiation, the Americium-241 is transmuted to Curium-242 through the beta decay of Americium-242 (16-hour half life). Yields of up to about 35% can be achieved by this method of irradiation.

3.2 Curium Processing

At the Hot Cell Facility, Quehanna, Pennsylvania, Curium-242 will be isolated from the irradiated aluminum targets after about a 30-day cooldown period to allow the short-lived fission products to decay. The curium and other transuranics are first separated from the aluminum matrix by a hot sodium hydroxide solution which dissolves the aluminum, leaving the curium, plutonium and americium hydroxides, plus any fission product impurities, as precipitates. After washing of the precipitated materials, these insolubles are dissolved in a mixture of 6N HCl with a trace of nitric acid. The solution from this step is refiltered and then boiled to reconcentrate the chloride ion. Lithium chloride is also added to the remaining solution to provide the feed mixture for the ion exchange purification of the curium. In all of the purification steps, the americium present will remain with the curium as they are so similar in nature. However, by methods of ion exchange separation similar to the americium purification system, curium can be separated from iron, monovalent ions, plutonium and fission products.

The steps in the purification process are illustrated in Table 3-2.

Curium fuel form and encapsulation. After the curium (americium) oxide has been purified, the preparation of the fuel form (i.e., the fuel compound and matrix material) is perhaps the most difficult to handle, since the fuel alone has a theoretical power output of 1169 watts/cc (as Cm_2O_3). The use of special chill blocks and auxiliary cooling is mandatory even with a practical power density of 50 to 100 watts/cc. The curium oxide must be blended with a metal matrix, compacted and sintered into the desired fuel form, and shaped to meet the auxiliary power system requirements. In addition, it is important to the safety of the unit that a void space be provided in the container in which the fuel is encased. The planned void space is an added safeguard to accommodate the internal pressure from the helium produced during the decay of Curium-242 to Plutonium-238. The pressure buildup over the life of any unit is accurately calculated, and establishes the maximum pressure based on any size void space. From the allowable stresses at operating temperature in the capsule container material, a safe unit can be designed to give "silent power" for remote, unattended or inaccessible areas.

TABLE 3-2
Curium Processing
by Ion Exchange

4. ENERGY CONVERSION SYSTEMS

The heat obtained from radioactive decay may be converted into electricity by rotary, thermoelectric or thermionic devices. With efficiency and reliability as guidelines, only the latter two schemes need be considered at power levels of a few hundred watts. Early missions will use thermoelectric converters, but thermionic systems, with such significant advantages as higher heat rejection temperatures, lower weight and higher efficiencies, are being extensively studied for future missions.

Since the principles, theory and methods of direct conversion of thermal energy to electrical energy are treated in detail by other lecturers, no attempt will be made here to cover this subject except as regards those problems unique to the use of radioisotopes.

5. ENGINEERING DESIGN CONSIDERATIONS

This section will consider the design of isotope-fueled generators with particular reference to the heat source properties which affect design.

5.1 Physical Properties of Radioisotopes and Their Effect on Heat Source Design

In the preparation of a suitable heat source, factors other than the decay of the radionuclide must be considered. The form of radioisotope and its matrix (if required) should be stable under the environmental conditions to be encountered (i.e., insolubility of the compound formed when the isotope is exposed to air or water, safe thermal characteristics of the fuel to prevent rupture of the capsule due to expansion, or hot spots within if the fuel material contracts). The mating of the container to the fuel and matrix is also extremely important.

The fuel materials shown in Table 2-3 are, in some cases, diluted considerably prior to encapsulation. Some dilution derives from the formation of stable fission products, and a considerable amount is caused by the reagents used to purify or separate the waste products. ORNL has been investigating methods for obtaining higher specific activity in cerium present in the chemicals used. Heretofore, where gross fission products were being discarded, it did not matter, but in the future, if the demand for large quantities of Cerium-144 increases, the purity of the isotope should also be increased. From this point on, it becomes a matter of economics; the removal of other cerium isotopes which are formed during fission is much more difficult and costly. Gaseous diffusion separation methods, although technically feasible, are extremely costly.

Certain fuels are best combined as cermets (i.e., ceramic with a metal matrix) which have been sintered at high temperature.

Some artificially produced radioisotopes such as Polonium-210 and Curium-242 have very high specific power. If the material is used in the metal state, it is difficult to maintain it in solid form without external cooling. For instance, curium metal with a thermal power density of 122 watts/gm would vaporize if allowed to stand in air. For this reason, it is often advantageous to dilute the metal with an inert matrix having a high conductivity. Gold, nickel and molybdenum are some of

the diluents used for these fuels (see Table 5-1). The thermal conductivity of the diluted fuel is a very important factor.

Thermal conductivity. The dissipation of heat from the centerline of a fueled capsule is rigorously analyzed to prevent excessive heat buildup during the operation of an isotope-fueled generator. Consequently, accurate thermal conductivity measurements are required.

Several methods for increasing the thermal conductivity have been successfully applied:

- (1) Adding a small amount of a high conductivity metal to the mixture.
- (2) Increasing the compact density.
- (3) Laminating the fuel with high conductivity material and bonding it to the outer shell.

Table 5-2 lists typical thermal conductivity values for various heat sources.

TABLE 5-2
Thermal Conductivity of Radioisotope Heat Sources

	<u>Theoretical Density (%)</u>	<u>Thermal Conductivity (cal/sec-cm-°C)</u>
Ce (metal)	100	0.026
Po (metal)	100	0.034
Pu (metal)	100	0.055
CeO ₂	95	0.01
Cm (metal)	100	0.0309

Radioisotope containment. For the handling of large quantities of radioisotopes, it is paramount that the isotope be encased in a material which will withstand all conceivable conditions of abort, corrosion, external and internal pressure, and burnup upon re-entry from an orbital flight (see Section 7--Space Radiological Safety).

A number of high temperature, high strength materials, including Inconel X, Hastelloy B, Rene 41, A-286 and Haynes-25, are available.

TABLE 5-1
Curium Fuel-Matrix Materials

	<u>Z</u>	<u>Melting Point (°F)</u>	<u>Thermal Conductivity (Btu/hr/ft²/ft/°F)</u>
Graphite	6		78.8
Molybdenum	42	4760	84.6
Rhodium	45	3571	50
Zirconium	40	3355	9.6
Platinum	78	3224	42
Titanium	22	3140	8.1
Vanadium	23	3100	8.1
Palladium	46	2829	41
Molybdenum, Nickel Curium Steel	26	2735	21.7
Nickel	28	2620	35
Gold	79	1945	172

Nichrome, titanium, palladium, platinum, vanadium and stainless steel--Types 316, 410 and 430--are under consideration for use in a thermoelectric device where expected temperatures range from 1000° to 1500° F. The refractory metals, molybdenum, tungsten, tantalum, rhenium and columbium, are being considered for use in a thermionic device where temperatures may range from 2000° to 3700° F.

The magnitude of the mechanical, chemical and thermal stresses which the containers enclosing the radionuclide fuel can withstand is illustrated in the following relationships.

a. Internal pressure

For alpha-emitting radioisotopes (Polonium-210, Plutonium-238, Curium-242, Americium-241), the containment material must be capable of withstanding internal pressure due to helium buildup; this pressure is a function of time, void volume, ambient temperature, operating temperature and the quantity of fuel.

For beta-emitting radioisotopes, the internal pressure problem is not nearly so acute, but must nevertheless be considered; this pressure is a function of temperature, fuel form, void volume and quantity of fuel.

The internal pressure which a cylindrical vessel can withstand is given by the equation:

$$P = \frac{b^2 - a^2}{b^2 + a^2} \eta$$

where

P = pressure (psi)

a = inside radius (in.)

b = outside radius (in.)

η = yield strength (psi).

b. Impact

Shock is a potential hazard to the fuel capsule; therefore, the material must be capable of withstanding extreme shock without failure.

Impact must be considered in relation to the kind of surface, the velocity and orientation of the impacting body. This makes the analytical approach nearly impossible. However, it is possible to evaluate

materials by using full-scale models of these containers and impacting them at high velocities against such targets as granite, sand, soil and water.

An estimate of the impact resistance of a capsule can be determined by applying an equation which relates the resistance of a cylinder to a static load resting on a flat plate.

$$P = \frac{2b\sigma^2}{0.349 E}$$

where

P = resistance (lb/linear in.)

E = tension (psi)

σ = yield strength (psi)

b = outside radius (in.).

c. External pressure

External pressure on the container can result from accidental dropping of the unit or from a re-entry impact. The containment material must be capable of withstanding any such pressure.

The external pressure at which a capsule would fail is calculated by the use of the equation:

$$P = \left(\frac{b^2 - a^2}{2b^2} \right) \sigma.$$

d. Thermal shock

The container must maintain thermal integrity at the operating temperature, in the event of internal temperature excursion or external temperature excursion such as a fire.

For a cylindrical capsule with the inside and outside walls at different temperatures, it can be shown that the level of stress that is produced is given by

$$\sigma = \frac{\alpha E}{1 - \nu} \left[\frac{2}{b^2 - a^2} \int_a^b T_b r dr - T_a \right].$$

If a uniform temperature gradient through the wall is assumed, this reduces to

$$T_b - T_a = \pm \frac{\sigma (1 - \nu)}{E} \cdot \frac{3 (b + a)}{2 b + a}$$

where

T_b = temperature of the outer walls

T_a = temperature of the inner wall

ν = Poisson's ratio

α = temperature coefficient of expansion.

e. Burnup

In the event that re-entry burnup is desired, the container must be designed so as to burn up at the expected re-entry temperature.

f. Chemical stresses

The metals used in the fabrication of the container must possess chemical properties favorable to normal operations and for many extreme chemical conditions.

Alloys from which outer containers are fabricated must undergo extensive corrosion tests over a period of several years. For instance, in the case of Hastelloy C tested in sea water for 10 years, the corrosion rate was determined to be 0.0001 in./yr.

Similar tests must also be conducted in an oxygen environment at high temperatures and in the presence of such oxidizing chemicals as nitric acid, aqueous solutions of ferric and cuprous salts, or wet and dry chlorine.

5.2 Generator Design Aspects

Power flattening. Among the primary design factors in a space power system is a constant power output for the design lifetime of the vehicle. A radioisotope decays with a known half life and, consequently, the exponentially decreasing thermal input must be balanced with the constant electrical output required. The amount of radioisotope initially loaded into the fuel core must be adjusted so that, at the end of the design life, sufficient thermal power is available to drive the thermoelectric system. Constant power can be achieved by dumping the excess

power as either heat or electricity during the initial phase of life. The type of thermoelectric material used in the conversion, in general, dictates which method must be used. In the case of lead telluride thermoelements, a limitation of 1000° F at the hot junction must be observed because of the tendency of this material to sublime at higher temperatures. Since maximum power output is achieved when the hot junction-cold junction temperature differential is maximum, it is desirable to operate with as high a hot junction temperature as possible. The use of lead telluride, therefore, dictates that heat rather than electrical energy must be dumped during the initial phase of life.

Lead telluride is considered to be the only truly operational thermoelectric material available today; there are, however, other very promising thermoelectrics coming along in the near future. Among these, cadmium sulfide, cobalt silicide and gadolinium selenide offer high temperature characteristics with Seebeck coefficients comparable to that of lead telluride. The use of one of these materials would eliminate the requirement of dumping excess heat and would enable the generator system to yield much higher power outputs during the initial phase of life. If constant power output were required, the excess power could be dumped electrically, allowing a far lighter and more efficient system.

Heat balance. The thermal balance of the unit is a design consideration that must be investigated during the conceptual design of an isotopic generator. Power output as a function of hot and cold junction temperatures must be optimum. The heat loss associated with the number and size of penetrations in the insulation shell between the hot and cold junction, and the actual sizing of the thermoelectric elements themselves are also extremely important and must be considered. With lead telluride, the hot junction limitation is generally 1000° F; the temperature of the cold junction, which is also used as a radiator, is determined by the size of the surface area available for heat rejection. Maximum efficiency of the unit is achieved by reducing heat losses to a minimum and adjusting the radiator dimensions so that the cold junction temperature is minimized. An IBM code has been written for the optimization of a generator system.

The optimization is accomplished through three different procedures:

- (1) Considers a thermoelement of given cross-sectional area and determines element length and number.
- (2) Requires a given length input and calculates the element diameter resulting in maximum efficiency.
- (3) Determines the efficiency of a thermoelement of any shape; there is no optimization procedure in this calculation.

The output in this case merely gives the electrical output of the generator being analyzed. The input data for this code are determined by the physical limitations imposed by the generator and the thermoelectric material used. From a space and weight standpoint, it is usually desirable to make the cold junction and, hence, the radiator surface, as small as possible, and still give cold junction temperatures of reasonable values.

Environmental considerations. After the conceptual design and heat balance have been determined, detailed design is then performed. One of the most important aspects of the detailed design is the structural integrity of the unit. Here, two problems must be considered; the first of these is the ability of the generator to withstand the severe environmental (acceleration, vibration and shock) conditions that are experienced during the launch of a space vehicle. The most susceptible portion of the generator to such an environment is the thermoelectric system. It must be designed carefully so that the thermoelements, which are extremely fragile, do not fracture during launch.

The second environmental condition that must be considered in the detailed design concerns the integrity of the isotope core itself during an aborted mission. The conditions that can exist during abort are discussed in another section

The electrical characteristics of the space power system must be matched with the electrical requirements of the vehicle involved. The power output is one of the design features of the generator itself, but the voltage out of the power supply, generally, has to be altered by a dc-to-dc converter to match the system requirements of the vehicle. The dc-to-dc converter is a completely static solid-state system using a flip-flop circuit, a transformer and a rectifier. Converter efficiencies of 70% are easily achieved at the voltages in question (between 3 and 30 volts). Converters have been designed with up to 85% efficiency.

6. NEAR-OPERATIONAL DEVICES

Research and development work on radioisotope-fueled generators for use in space has been in progress since 1951. A well diversified group of generator designs have resulted. SNAP 3, a 2.8-watt, six-month thermoelectric generator is now operational; SNAP 1A, a 125-watt, one-year thermoelectric generator is in the advanced hardware stage; SNAP-TIP, a Curium-242 fueled thermionic generator, is also in the testing stage. Design configurations have been established for additional thermoelectric generators with power capabilities from 10 to 100 watts and operational lives up to ten years.

6.1 SNAP 1A Generator

SNAP 1A consists, structurally, of a central fuel container supported by tubular truss members within two concentric shells. Load members between the shells provide strong points for ground handling and flight vehicle attachment. A cylindrical block of Inconel X, 3-3/4 inches in diameter and 11 inches long, contains the radioisotope. Seven thin-walled stainless steel tubes loaded with ceric oxide pellets are inserted into holes drilled in the block. Threaded closure caps are screwed into place and later fusion welded to provide a sealed container. The loaded block may be transported separately in its own shipping cask and installed with portable ground handling equipment through a port in the generator at the launch site. Physical relationships are illustrated in Fig. 6-1.

Heat generated in the central source is radiated to the inner shell of stainless steel. The inner shell serves as a hot junction for the thermoelectric elements installed between the shells. Unconverted thermal energy conducted through the elements and escaping through the insulation is radiated from the outer aluminum shell. Excess energy, resulting from the excess fuel loaded initially to compensate for isotope decay, escapes from an exposed area of the inner shell. Insulated shutters, actuated by a mercury vapor bellows, control the amount of exposed area as required to maintain a constant hot junction temperature.

Approximately 6000 thermal watts will be produced by an initial load of 880,000 curies of Cerium-144. Over a period of one year, the output will decay to 2700 watts, the generator design condition. The outer shell, 24 inches in diameter and 34 inches long, must possess sufficient area to radiate this energy to the space sink. At initial loading 3300 watts will be dumped from the exposed inner shell with the shutters completely open. A fibrous insulation, Min-K 1301, is used between the shells to reduce heat losses to a minimum.

Fig. 6-1. SNAP 1A Thermoelectric Generator

The electrical features of SNAP 1A are as follows. A total of 277 thermoelectric couples are connected in series to produce 125 watts at 28 volts. Each couple consists of two lead telluride cylinders doped with either lead iodide, to form the N element, or sodium, to form the P element. An iron shoe connects the elements at the hot junction while copper caps are soldered to the cold junction ends. The elements are 0.375 inch in diameter, with lengths of 0.78 inch for the N and 0.66 inch for the P. Operating at a hot junction temperature of 1000° F and a cold junction temperature of 335° F, these elements attain a thermoelectric efficiency of 6.75% which, because of the thermal losses through the generator insulation and structural members, decreases to an overall efficiency of 4.7%. Significant design characteristics are summarized in Table 6-1.

Cerium-144 was selected as the radioisotope heat source for this generator. The basis for selection was its availability; it is a fission product separated from reactor wastes at Oak Ridge National Laboratory. It decays through beta emission, with a half life of 285 days, to Praseodymium-144 which, in turn, decays through beta emission with a half life of 17.5 minutes. The third element in the decay scheme, Neodymium-144, decays with a half life of 1.5×10^{15} years through emission of an alpha particle. Because of its long half life, neodymium is considered to be stable. Although some of the thermal power originates from the initial cerium decay, the major energy source is derived from the energetic beta decay of praseodymium. In addition to the decay gammas, a large portion of the photon radiation results from the slowing down of these energetic beta particles (bremsstrahlung).

To reduce the radiation level to a reasonable amount for ground handling purposes, sufficient volume exists between the heat source and the inner shell to contain approximately 4000 pounds of mercury. Because of the source and shell geometry, shield thickness and dose rates vary from point to point around the unit. With the mercury in place, radiation levels one meter from the generator surface vary from 2 to 95 mr/hr and 11 to 750 mr/hr on the generator surface. The radiation level 100 feet from the bare core is 22 r/hr.

Two SNAP 1A units have been constructed for test purposes using an electrical heater in lieu of a radioisotope heat source. A third unit currently under construction will be fueled and subsequently ground tested.

TABLE 6-1
SNAP 1A Design Characteristics

Heat source		
Radioisotope		Cerium-144
Quantity (curies)		880,000
Power output (electrical watts)		125
Operational life at constant power (days)		365
Voltage (volts)		28
Current (amps)		4.46
External load (ohms)		6.28
Efficiency (%)		
Thermoelectric		6.75
Overall		4.7
Thermal energy (watts)		
At initial loading		6000
At end of operational life		2700
Operating temperature (°F)		
Hot junction		1050
Cold junction		335
Radiator surface (ft ²)		
Low temperature		14.5
High temperature		3.5
Weight (lb)		
Hg shield (removable)		4000
Flyaway		200
Thermoelements	<u>N Element</u>	<u>P Element</u>
Diameter (in.)	0.375	0.375
Length (in.)	0.78	0.66
Doping	Lead iodide	Sodium

6.2 SNAP 3 Generator

This device was the first to produce a practical amount of electrical power by direct conversion of the heat resulting from radioisotope decay. It was demonstrated at the White House on January 16, 1959. The characteristics of this unit are tabulated as follows:

Physical dimensions (in.)	4.75, diameter; 5.5, height
Weight (lb)	5
Source (curies)	1495, Po-210
Half life (days)	138
Thermal power (watts)	48
Electrical power (watts)	2.4
Efficiency (%)	5
Thermoelectric material	Doped lead telluride
Hot junction temperature (°F)	720
Cold junction temperature (°F) (outer skin)	175
Dose rate at surface (mr/hr)	350
Dose rate at 1 meter (mr/hr)	3.5

Polonium-210 with a half life of 138 days was selected as the radioisotope heat source for this unit on the bases of ready availability and high power density. Its thermal energy results almost entirely from the emission of a 5.3-Mev alpha particle. A low gamma emission coincident with a 4.5-Mev alpha particle occurs in 0.0012% of the disintegrations. With no shielding other than the fuel-encapsulating material, the SNAP 3 dose rate is only 3.5 mr/hr at one meter. Mound Laboratory has provided the polonium for all generators fueled to date. Double stainless steel capsules are inserted in a heavy-walled heat source cylinder. This cylinder is then closed with a threaded plug and sealed by fusion welding. In this unit the thermoelements support the central source.

The Minnesota Mining and Manufacturing Company designed and fabricated the thermal-to-electrical conversion section. The lead telluride elements, approximately one inch in length, have diameters of 0.210 inch (N type) and 0.225 inch (P type). These are paired to form 27 couples which are mounted in six vertical rows with a radial arrangement of nine elements per row. Each element is loaded axially against a hot shoe by a two-pound force provided by a spring and adjustment screw located in an aluminum cold junction ring. The hot shoe is of

iron which has been flame-sprayed with aluminum oxide. The shoe makes the electrical connection between element pairs at the hot junction, while the oxide coat prevents current flow through the supporting hot junction cylinder. The couples are connected in series by copper wires at the cold junction and insulated electrically from the cold junction ring. Insulation is provided by an oxide coating (Martin Hardcoat) on the aluminum bearing caps located between the element end cap and the pressure spring. The hot junction cylinder supports the elements through the hot shoes and contains a tapered internal hole to receive the fuel capsule previously described. Mica sleeves are placed on the hot junctions of the elements to provide additional protection and insulation.

The remaining void inside the generator shell is filled with Johns-Manville Min-K 1301 insulation. This material is formed into solid pieces in the hemispherical ends and is in powdered form in the thermoelectric area. This internal arrangement is illustrated by the cut-away artist's concept in Fig. 6-2. After the first unit was constructed, a spun copper shell was used instead of the brass case illustrated. Other improvements in construction details decreased the weight to four pounds.

Three additional units have been fueled with various amounts of polonium ranging upward to 2320 curies. A maximum output of four watts at four volts has been obtained with an efficiency of 5.75%. The last two units were subjected to over 2000 hours of severe environmental testing prior to being fueled; their operation has been satisfactory to date.

It may be noted that the output of SNAP 3 was previously listed as 2.8 watts, versus the four watts given above. This is the result of two effects. First, in a space environment the isotope load must be reduced slightly to avoid overheating. Since radiation is the only method of heat transfer from the cold junction, its temperature increases slightly over that in air, causing a corresponding increase in hot junction temperature if the same thermal input is provided.

Secondly, the electrical output would fall off to approximately 1.8 watts over a 90-day period without some method of power flattening. A method of obtaining a constant output has been devised through tests in Martin laboratories. Conductivity of the Min-K insulation decreases rapidly at pressures less than one atmosphere. Thus, by introducing a conductive gas mixture of helium and hydrogen into the shell, the conductivity may be increased initially and then decreased by allowing the gas to leak into space. This conductivity change may be controlled by proper selection of a fixed orifice so as to be inversely proportional to thermal energy decrease. Initially, with a gas charge of one atmosphere,

Fig. 6-2. SNAP 3 Thermoelectric Generator

heat losses through the insulation are high and generator output is decreased to approximately 2.8 watts. As this gas charge leaks out, generator efficiency improves such that a constant output is produced for a 90-day period.

6.3 Satellite Generator

The configuration of this unit is, in general, similar to that of SNAP 3. This approach was adopted in order to reduce the development time required and to increase reliability. An operational life of 5 to 10 years in orbit was the design objective (see Fig. 6-3).

Plutonium-238, in carbide form, was selected as the radioisotope heat source since its half life of 86 years obviates the need for a heat dump or power flattening mechanism. Because of the relatively low power density (6.90 watts/cc) of plutonium carbide, the central heat source (a right circular cylinder 2.8 inches in diameter) is large in comparison to the overall generator size.

Conversion of the thermal energy will be accomplished at an efficiency of 6.54%. Lead telluride elements arranged in 36 couples will produce 14.5 watts at 3 volts. A hot junction temperature of 900° F was selected to assure satisfactory element operation over the operational lifetime. In order to maintain an efficient temperature differential, the cold junction temperature was lowered to 235° F through extended radiator surfaces. Six fins extending 3-1/2 inches from the outer shell surface increase the effective radiation area.

Heat losses through the insulation at operating temperatures were calculated to be 40 watts, placing the total energy required at 262 watts. Combining this thermal efficiency of 85% with the thermoelectric efficiency gives an overall efficiency of 5.5%.

The estimated weight of this unit is 10.4 pounds with a specific power of 1.4 watts per pound.

This unit is particularly attractive where extended exposure to Van Allen radiation occurs. The performance of solar cells degenerates significantly after 200-hour exposure to proton activity at altitudes between 2000 and 5000 miles.

Dose rates for the unshielded plutonium-fueled generator are as follows:

Fig. 6-3. Satellite Generator

	Generator Surface (mrem/hr)	3.5 Inches from Generator Surface (mrem/hr)	1 Meter from Generator (mrem/hr)
From fission neutrons	870	51	1.2
From decay gammas	<u>87</u>	<u>10</u>	<u>0.3</u>
Total	957	61	1.5

6.4 Soft Lunar Landing Generator

Development of the generator was coordinated with the Jet Propulsion Laboratory of the California Institute of Technology so that the unit would meet the general requirements of space probe packages under concurrent development at CIT. The criteria for the conceptual design proved to be of the utmost significance in the work that ensued. These criteria were:

- (1) Electrical power--the generator was to deliver 13 watts at 3 volts dc of continuous, unregulated electrical power for a period of at least six months following launching into space.
- (2) Weight--not to exceed 15 pounds.
- (3) Environment--the unit was to operate in normal fashion in an undefined space trajectory (i.e., a circumlunar probe), following subjection to missile-induced conditions typical of the Vega launch vehicle. Development of the Vega vehicle was cancelled during the course of the program; these conditions were then held to be illustrative only.
- (4) External radiation--because of the presence of radiation-sensitive equipment in the JPL payloads, external radiation emanating from the generator was specified not to exceed 7 gamma photons/cm²/sec above 100 kv at a point 10 cm from the surface. In the case of an anisotropic radiation field, the generator could be oriented with respect to the rest of the payload so as to minimize the radiation level contacted by the payload. No restriction was placed on neutron fluxes external to the generator during its operational life. For safe ground handling, external radiation levels were to be reduced to a maximum of 60 mrem/hr at a distance of one meter.
- (5) Fuel form--Curium-242 was specified as the radioisotope to be evaluated in the study. An extensive program aimed at

developing this isotope as a heat source was being undertaken concurrently under Contract AT(30-3)-217. The chemical form or geometry to be employed was not specified and was to be determined by the state of curium technology and the demands of the generator design. Its half life of 163 days and specific thermal power of 122 watts/gram make Curium-242 a suitable heat source for minimum size generators having moderate operational lives of three to six months. Biological shielding problems are minimized with Curium-242 due to the relatively low level of ionizing radiation emanating from the sealed source of the isotope.

The configuration of this unit is apparent in Fig. 6-4. In general, it resembles SNAP 1A in that a central heat source radiates its energy to the hot junction and a movable thermal shutter controls hot junction temperature. Two novel features were used in the design. First, the high temperature heat dump radiator forms an integral part of the heat source. This reduces radiator size to a minimum since its temperature will be 1300° to 1500° F. Second, a block of tungsten shield material was incorporated as part of the heat source. Generator characteristics are as listed in Table 6-2.

TABLE 6-2
Generator Characteristics

Operational life (months)	6
Size	
Diameter (in.)	7.5
Height (in.)	8.38 (shutter closed)
Efficiency	
Thermoelectric (%)	6.51
Thermal (%)	73.5
Overall (%)	4.8
Temperatures	
Hot junction (°F)	1000
Cold junction (°F)	370 (in space)
Heat loss (watts)	36
Insulation thickness (in.)	1.25

TABLE 6-2 (continued)

Thermoelectric elements

Output voltage (volts)	3
Number of pairs	30
Cross-sectional area	
P-type (in. ²)	0.1300
N-type (in. ²)	0.1089
Doping	
P-type (% Na)	1.0
N-type (% PbI)	0.03
Length (in.)	0.75

Fuel

Isotope	Curium-242
Purity (%)	45
Dilution (Au to Cm by weight)	5 to 1
Mass of isotope (gm)	6.3
Thermal power	
At encapsulation (watts)	752
At launch (watts)	582
At end of life (watts)	270
Estimated weight (lb)	16.64

Because of the low radiation dose rate of the Curium-242 source, safe ground handling of the generator is achieved with a shield equivalent to 1.6 inches of water. This shielding reduces the radiation level to the prescribed 60 mrem/hr at one meter. The relative amounts of various types of radiation and the effect of the water shield upon them are as follows:

Fig. 6-4. Generator for Soft Lunar Landing

<u>Type of Radiation</u>	<u>Dose Rate (mrem/hr at 1 meter)</u>	
	<u>No Shield</u>	<u>1.6-in. Water Shield</u>
Neutron	110.0	48.6
Decay gamma	12.6	8.8
Fission gamma	2.7	2.2
Capture gamma	0.03	0.026
Inelastic gamma	<u>0.015</u>	<u>0.013</u>
Total	125.3	59.64

6.5 Hard Lunar Impact Generator

The conceptual design of the lunar impact generator was to be based on the following specifications:

- (1) A maximum weight of 18 pounds.
- (2) A continuous output of 13 watts at 3 volts dc for a period of 60 days after impact.
- (3) Ability to withstand the decelerative forces resulting from lunar impact at 500 ft/sec.
- (4) Maximum reliability and efficiency consistent with the other specifications.
- (5) Use of Curium-242 as the radioisotope fuel.
- (6) Assurance of fuel capsule integrity under adverse conditions of launch vehicle abort.
- (7) Reduction of external radiation level to 60 mrem/hr at one meter for ground handling purposes and to the following levels in operation for protection of radiation-sensitive equipment in the instrument capsule; on a circular area of 100 sq cm located five meters from the generator, radiation was not to exceed:
 - (a) $1 \text{ photon/cm}^2\text{-sec}$ in the energy range 0.4 to 3.0 mev.
 - (b) $0.5 \text{ photon/cm}^2\text{-sec}$ in the energy range 3.0 to 10.0 mev.

- (8) Loading of the fuel capsule occurs 60 days (maximum) prior to launch.

Since details of the launch vehicle, impact vehicle and instrument capsule were not available during the course of this study, it was necessary to make some arbitrary assumptions concerning them in order that the generator design would at least reflect representative conditions for launch, abort or lunar impact.

The impact vehicle was assumed to consist of a retrorocket attached to a spherical instrument capsule roughly 3 feet in diameter, being primarily, a crushable structure to cushion the impact forces on the instruments. Since the crushable material might possess thermal insulation properties, the generator was assumed to be mounted on the exterior of the capsule at a point farthest removed from the retrorocket. In a successful mission, the retrorocket case would probably impact first. Uniform deceleration of the payload was assumed to occur over a distance of one foot, leading to a figure of about 4000 earth g deceleration. To be conservative, the generator design was based on an impact angle varying to a maximum of 15 degrees from the axis of the vehicle. The design that evolved was symmetrical about a linear axis which coincided with the axis of the impact vehicle, and the maximum stresses were taken to be compressive along the axial direction.

The resulting generator configuration, shown in Fig. 6-5, consists of a fuel block in the shape of a rectangular parallelepiped, containing the curium fuel in four sealed canisters. Two modular arrays of thermoelectric elements are mounted on opposite sides of the fuel block, and the remaining internal volume of the generator is filled with solid thermoelectrical insulation. The rigid outer shell, in the general shape of an ellipsoid of revolution, conducts rejected heat to radiator surfaces and is attached to a shock-absorbing bellows structure which, in turn, is fastened to the impact vehicle. Simplicity of the design and the lack of moving parts are apparent.

Results of the design analysis are tabulated in Table 6-3. A specific power of 2.1 watts/lb may be attained after 60 days of operation while specific power at launch would be 2.9 watts/lb. Although this unit and the soft lunar landing unit described previously were designed for different missions, their basic differences are due to the thermoelectric material characteristics (lead telluride versus a higher temperature material). Nearly all of the advantages of the impact design are due to the ability of its thermoelements to operate in a higher temperature region, the exception being the improved shock resistance due to superior physical properties.

Fig. 6-5. Generator for Hard Lunar Impact

TABLE 6-3

Results of Lunar Generator Design Analysis

End of Life Parameters in 250° F Ambient--Lunar Environment

Efficiency (%)

Thermoelectric	5.36
Thermal	95
Overall	5.2

Temperature (°F)

Hot junction	1400
Cold junction	458
Fuel centerline	1500

Heat loss (watts)	17
-------------------	----

Voltage, end of life (volts)	3
------------------------------	---

Power, end of life (watts)	13
----------------------------	----

Number of elements	127
--------------------	-----

Weight (lb)	6.2
-------------	-----

60 Days Prior to End of Life (at launch)

Maximum temperature (°F)

Hot junction	1660
Cold junction	476
Fuel centerline	1800

Power (watts)	18
---------------	----

Performance characteristics as previously described are entirely feasible through development efforts in the following areas:

- (1) Reproduction of current laboratory materials on a production scale.
- (2) Generator and couple fabrication techniques to utilize these materials with low internal circuit resistances.
- (3) Determination of the life and stability factors of recently discovered materials.
- (4) Improvement of the figure of merit and physical properties of current material combinations.

A generator utilizing a high temperature material could operate more efficiently without a variable heat dump, provided the load cycle of the system had been properly programmed. A constantly decreasing power output from the generator would increase voltage control difficulties, but would make a more reliable system as the thermal shutter mechanism could be eliminated.

6.6 Space Probe Generator

This design exercise was conducted to establish the characteristics of a larger generator using Curium-242 as its heat source. No specific application was assumed, and the integration of this unit with a vehicle or power system was not investigated. A power output of 100 watts for an operational life of six months was arbitrarily chosen.

Several alternate configurations were studied with results indicating that an arrangement similar to that of SNAP 1A offered the highest specific power. The characteristics of this design are listed in Table 6-4. With a weight of 77 pounds, a specific power of 1.3 watts/lb may be attained. Although no significant technological advances were used to develop this design, its specific power represents a significant improvement over that of SNAP 1A (0.625 watts/lb). Two factors contribute to this improvement: the higher specific power of Curium-242 versus Cerium-144, and a reduction in the mass of structural members since a mercury biological shield need not be supported.

TABLE 6-4

Characteristics of the 100-Watt Cm-242 Generator

Outer shell (cold junction)--aluminum	
Thickness (in.)	0.040
Height (in.)	21.5
Width, diameter (in.)	19.5
Inner shell (hot junction)--stainless steel	
Thickness (in.)	0.050
Height (in.)	19.5
Diameter (in.)	17.5
Thermoelectrics--pairs of PbTe elements	169
Length (in.)	0.3937
Diameter	
P element (in.)	0.500
N element (in.)	0.455
Doping	
P element (%Na)	1.0
N element (%PbI ₂)	0.03
Heat source	
Height (in.)	4.5
Effective diameter (in.)	4.5
Number fuel tubes	5
Diameter (in.)	0.956
Height (in.)	3
Height of fuel pellet (in.)	2.4 (20% void space)
Thermal power at encapsulation (watts)	7500
Fuel mixture, by weight	20:1 Ni-Cm mixture
Encapsulating material	Hastelloy C
Thermal shutter	
Area (ft ²)	1.72

TABLE 6-4 (continued)

Insulation--Min-K 1301

Thermal conductivity (Btu-in./hr-ft ² -°F) (vacuum)	0.1
Thickness (in.)	0.9
Density (lb/ft ³)	20

Temperature

Hot junction (°F)	1000
Cold junction (°F)	490
Heat source at launch (°F)	1800
Heat source at end of life (°F)	1470

Weight (lb)	77
-------------	----

Efficiency

Thermoelectric (%)	5.20
Overall (%)	4.95

6.7 SNAP-TIP System

The SNAP Thermionic Isotope Power System (SNAP-TIP System) consists of two vacuum diode thermionic converters located at the opposite ends of the disc-shaped Curium-242 heat source (see Fig. 6-6). The generator is approximately 2 inches in diameter and 2 inches in length. The generator, when fueled, will weigh approximately 14 ounces and produce 4 to 5 watts of dc power. The other characteristics of the generator are listed in Table 6-5.

The isotope heat source is Curium-242, and the fuel form consists of Cm₂O₃ in a platinum matrix. This fuel form will be pressed and coined into a cylindrical disc, the outer surface of which will be pure platinum. The fuel pellet will be encapsulated in arc-cast tungsten. The capsule will be welded to form a leaktight structure capable of containing the helium gas pressure which builds up due to alpha decay. The surface of the capsule is provided with grooves to increase thermal emissivity. The heat is radiated from the fuel capsule to the back of the two thermionic emitters.

The fuel capsule is supported by a set of concentric tantalum cylinders which are fastened directly to the outer titanium casing (see Fig.

Fig. 6-6. Assembly of Thermionic Generator

6-7). Between the 0.005-inch cylinders are several 0.0005-inch foils which act as radiation shields. This design provides rigid support of the isotope container, yet the total heat losses due to conduction and radiation are quite low.

The two diodes are mounted separately on opposite halves of the titanium casing and are completely independent of the isotope heat source. This feature greatly simplifies the fueling of the generator. The details of each diode are as shown in Figs. 6-8 and 6-9. The emitters are Type B Philips cathodes*, brazed to a dense tungsten backup plate with a molybdenum braze. The collector is molybdenum and is mounted on the emitter by means of a tantalum spacer. An insulating braze is used to fasten the collector to both the tantalum ring and the diode support piece to prevent electrical shorting.

Four electrical leadthroughs are provided, two for each diode. There is no internal electrical connection between the two diodes.

Electrically heated versions of the TIP generator have been life tested for more than 2100 hours and have undergone environmental tests simulating missile-launch conditions. Present plans call for the fueling of a generator with a Curium-242 heat source in March 1962. Following this fueling, operational missions for SNAP-TIP systems can be selected.

TABLE 6-5
Characteristics of SNAP-TIP Systems

Heat source	
Radioisotope	Curium-242
Quantity (gm)	1 (3318 curies)
Thermal power output (watts)	120
Operating temperature (capsule surface) (°F)	2510
Thermionic diodes (2)	
Emitter	Impregnated Tungsten**
Collector	Molybdenum
Spacing (in.)	0.0004 to 0.0005
Collector work function (ev)	1.85 to 1.95
Emitter temperature (°F)	2240
Collector temperature (°F)	1110

*Impregnated porous tungsten.

** Philips Type B Cathode.

Fig. 6-7. Heater Section of Generator

Fig. 6-8. Diode Assembly

A. Back view of emitter

B. Back view of collector

Fig. 6-9. Diode Assembly for Thermionic Generator

TABLE 6-5 (continued)

Generator performance

Power output (nominal)(watts)	5
Efficiency (%)	4.2
Voltage (max. power per diode)(volts)	0.7 approx
Weight (oz.)	14
Specific power (watts/lb)	5.7
Surface temperature (radiating surface) (°F)	640

Generator physical characteristics

Diameter (in.)	2
Height (in.)	2
Casing	Titanium
Leadthroughs	4

6.8 Summary

In summary, auxiliary power systems with radioisotope sources capable of fulfilling the demands of a variety of space missions are indicated in the following tabulation.

<u>Designation</u>	<u>Electrical Power (watts)</u>	<u>Operational Life</u>	<u>Weight (lb)</u>	<u>Isotope Source</u>
SNAP 1A	125	1 year	200	Ce-144
SNAP 3	2.8	90 days	4	Po-210
Satellite	14.5	5 to 10 years	10.4	Pu-238
Soft lunar landing	13	6 months	16	Cm-242
Hard lunar impact	13	2 months	6	Cm-242
Space probe	100	6 months	77	Cm-242
SNAP-TIP	Thermionic	90 dyas	0.9	Cm-242

The three SNAP generators have already reached the hardware stage, while designs for the others are complete, and component tests which will further the development of the final units are under way.

7. RADIOLOGICAL SAFETY

One of the most significant factors in establishing the practicability of isotopic power for space applications is the assurance that can be provided for the radiological safety of operational systems employing such devices. The normal safety considerations of fuel core loading, core transportation and ground handling procedures at the launch site have been studied intensively. The problems here, however, are not particularly unusual, but rather, represent normal safe handling procedures for radioactive materials. The environments to which an isotope-powered generator can be subjected during missile launch do present a very significant departure from normal handling techniques, and it is these aspects of the safety problem which will be reviewed.

During the ascent of a space vehicle, a great many abort conditions can take place. These may be defined in detail by analyzing a typical trajectory (see Fig. 7-1), which may be divided into four fundamental phases:

- (1) Failure at launch.
- (2) Ascent failures.
- (3) Final stage or injection stage failures.
- (4) Orbital decay and re-entry before the design lifetime of the satellite in question.

The first phase involves those failures that can occur on the launch pad or shortly after launch, and are usually of the type where fuel tank rupture occurs or guidance system failures require command destruct of the vehicle. In either case, two severe environmental conditions can occur. These are explosion of the unused fuel and fire. The explosion forces for typical missiles in operational use today create blast overpressures of the order of 10,000 psi. Launch pad fires reach temperatures of the order of 6000° F for microsecond periods but rapidly decay to temperatures between 1600° and 3000° F for burning times up to 30 minutes. The most severe fire or explosion occurs while the vehicle is actually on the launch pad with none of its fuel expended in flight. SNAP 1A and SNAP 3 cores have been subjected to tests simulating these environments and have completely contained the fuel (see Figs. 7-2 and 7-3).

The second type of failure is that which occurs during ascent of the vehicle to orbital altitudes. This range is considered to be above 10,000 feet up to the design orbital altitude of the vehicle in question. The most severe environmental condition that can occur during this phase of the trajectory is that of impact of the isotope core. The most

Fig. 7-1. SNAP 3 Aborted Mission

Fig. 7-2. Simulated Launch Pad Explosion with 1600 lb of TNT

Fig. 7-3. Simulated Missile Fire with Hypergolic Fuel and Aluminum Missile Structure

stringent impact conditions occur if the injection stage fails to ignite. Maximum impact velocities are determined by the drag factor associated with the core in question but are generally of the order of 500 fps. The core impacts at its operating temperature, and impact can conceivably occur on any type of surface. With our present missile ranges, Vandenberg Air Force Base or Cape Canaveral, it is generally the practice to fire the missile system over large expanses of ocean area. Hence, the most probable impact location is the sea. Should this occur, the problem of salt water corrosion on the isotope core would be a major consideration. Impact on hard media such as granite can occur in the case of a misaligned trajectory where the command destruct system fails. The probability of this occurrence, however, is extremely low. The SNAP 1A and SNAP 3 cores have been tested in impact at Aberdeen Proving Ground's Supersonic Ballistic Sled Track. Target media ranged in hardness from water to granite. Impact velocities up to 500 fps at core operating temperatures (1600° and 1000° F, respectively) were investigated. The SNAP 1A and the SNAP 3 cores successfully survived these experiments (see Figs. 7-4 and 7-5). Sea water corrosion studies on fuel cores conclusively prove that the SNAP 1A and SNAP 3 cores will not release the fuel until the activity level is well below one curie.

Final-stage failures occur when the injection or transit phase of the flight vehicle undergoes a propulsion or guidance malfunction. This results in impact of the core far downrange. The area in which this impact can occur can be defined only by the missile system in question and the azimuth under which it is flown. Detailed studies of the SNAP 1A system and the SNAP 3 systems used in a polar launch from Vandenberg Air Force Base have been performed; these studies include aerodynamic heating occurring during re-entry, and determine the percentage of core burnup and the point of impact for propulsion failures and guidance failures.

The fourth condition involves orbital decay and re-entry of the SNAP device. This problem has been studied in detail by the SNAP Hazards Committee and the Aero Space Nuclear Safety Committee of the Atomic Energy Commission. The basic design philosophy adopted by these committees is that where impact can be controlled, it is desirable to contain the isotope. If, however, an orbital status is achieved and it is no longer possible to reliably control the point of impact, the fuel core should burn up to very fine particle size at very high altitudes--specifically, particles averaging less than 10 microns at altitudes greater than 100,000 feet. Detailed aerodynamic heating studies coupled with plasma arc tests have established that both the SNAP 3 and SNAP 1A systems meet these criteria (see Fig. 7-6).

When the fuel burns up at altitudes greater than 100,000 feet, the fuel debris behaves very much like weapons fallout. It is widely

Fig. 7-4. Preliminary SNAP 1A Cores After Unsuccessful Impact on Granite

Fig. 7-5. SNAP 1A Cores After Successful Impact on Granite

Fig. 7-6. SNAP 3 Post-Orbital Re-entry

dispersed, and, when finally it does descend, it constitutes a very minor increase in the radiation levels already present on the earth's surface. In the case of SNAP 1A, these levels are approximately 5 millicuries/sq mi, in comparison with 2000 millicuries/sq mi already present on the ground. The SNAP 3 condition is significantly less than this.

The test programs described briefly above have conclusively demonstrated the operational safety of the SNAP systems in question. Such tests will be repeated for all space systems to assure that they, too, can meet these requirements.

8. FUTURE

It follows from the preceding discussions that systems such as SNAP 3 are now ready for operational use. Based upon state-of-the-art technology, SNAP 3 has been successfully ground operated for more than one year. Improvements in the technology of direct conversion will lead to higher temperature, more efficient thermoelectric materials allowing smaller fuel inventories and lighter-weight systems. Thermionic technology, which is also developing at a rapid rate, offers the above improvements plus the added factor of higher heat rejection temperatures. Within a dollar framework, these growth factors are reflected in Table 8-1 below. The fuel cost is considerably greater than the cost of the remainder of the power system.

TABLE 8-1
Cost of 100-Watt (E) Ce-144 Power Supply (1-year life)

<u>System</u>	<u>Efficiency (%)</u>	<u>Thermal Power (watts)</u>	<u>Curies</u>	<u>Fuel Cost* (\$)</u>
Thermoelectric (available today)	5	5000	675,000	70,000
Thermionic (1965)	13	2000	270,000	28,000

*Production cost.

The relative position of isotopic power with respect to chemical reactions, fusion and fission, is shown in Fig. 8-1. The ever increasing source of power from the decay of radioisotopes will become more important in the next ten years.

Recovery of the usable waste products from spent reactor fuels will serve to repurify the unused fuel and will provide isotopes for use as power sources. In addition, it will greatly aid the growing problem of waste disposal since many of the noxious radioisotopes will be removed for commercial and military applications.

Basic to the use of nuclear energy in space is the technical assurance that established safety criteria have been met. A considerable portion of the AEC-SNAP Program has been directed toward establishing such criteria and subsequent testing of core, containment and generator configuration for each system against these criteria. Other factors, which are nontechnical, but nonetheless important, also merit detailed consideration; these are beyond the intended scope of this text.

Fig. 8-1. Energy Available in Radioisotopes Compared to Chemical Reactions, Fission and Fusion

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APPENDICES

APPENDIX A

REPRESENTATIVE MATHEMATICAL PROBLEMS

Problem 1

The half life of an isotope is defined as that period of time in which the isotope will decay to a level of activity one half its original value.

Polonium-210 has a half life of 138 days. What is its radioactive decay constant? What percentage of the polonium atoms originally present will decay in a period of 60 days?

Problem 2

The isotope Curium-242 has a half life of 162.5 days and a specific power of 120 watts/gram. If you wish to build a thermoelectric generator to produce 25 watts of electricity for a period of 90 days at an overall efficiency of 5%, how many grams of Curium-242 are required at loading?

Problem 3

The E-I characteristic of a typical thermoelectric generator, with minor approximations, can be shown as follows:

- (1) Determine the maximum output of the generator to four significant figures.
- (2) What would be the load resistance, the operating current and the voltage across the load at the maximum power point?
- (3) What is the open-circuit voltage of the generator?

Problem 4

If you are in possession of a facility that will produce 0.1 gram of Curium-242 per day on a continuous basis, how long will it take to accumulate the 6.109 grams of fuel required for the mission in Problem 2? What is the maximum quantity of Curium-242 that this facility could ever accumulate?

Problem 5

Curium-242 is an alpha-emitting isotope with a half life of 162.5 days. Assume that 2 grams of active Curium-242 in the form Cm_2O_3 (in a platinum matrix with a total volume of 2 cc) are placed in a sealed container with 3 cc of interior volume. The power density of the Cm_2O_3 -Pt matrix is 50 watts/cc. Assume that the temperature of the container decreases linearly from 1000° F at the time of loading to 500° F in 100 days.

- (1) Derive an analytical expression for the pressure buildup in the container versus time.
- (2) Evaluate the above expression at the 100-day period.

NOTE: Helium obeys the perfect gas law to extreme pressures.

Problem 6

The open-circuit voltage of a thermoelement is given by:

$$dE = \alpha(T)dT$$

where

- E = open-circuit voltage (volts)
 α = Seebeck coefficient (volts/°R)
 T = temperature (°R).

If we have a thermoelement with T and $\alpha(T)$ related as shown in the following diagram, what will be the open-circuit voltage produced if the element is operating between 1100°F and 100°F ?

Problem 7

From a statistical study of radioactive decay, it has been shown that the number of disintegrations occurring in unit time is proportional to the total number of radioactive atoms present, or

$$-\frac{dP}{dt} = \lambda P$$

where

P = number of radioactive atoms present

λ = radioactive decay constant

t = time.

As the power of a radioisotope is directly proportional to the number of disintegrations, P may also be considered to be the power produced by the isotope.

Consider that we have an isotope decay chain as follows:

Derive an expression for the power at any time, t , if, at $t = 0$, only the isotope P is present with a power level of P_0 .

Problem 8

A source of 100 thermal watts is required for a certain application; Cobalt-60 is to supply the heat. The specific activity of the cobalt is to be 30 curies/gm.

- (1) Find the weight of cobalt (active and inactive combined) and the number of curies of active cobalt. Assume 90% of the gamma ray energy is recovered. This can be done by having the self-absorption of the pellets and the absorption of the cladding equal to a tenth-value thickness. Data required:

Half life-- 5.3 years

$$\text{Beta rays--} E_{\max} = 0.312 \text{ Mev} \frac{E_{\text{avg}}}{E_{\max}} = 0.287$$

Gamma

rays-- $\gamma_1 = 1.17 \text{ Mev}$, 100% of disintegrations

$\gamma_2 = 1.33 \text{ Mev}$, 100% of disintegrations

$\gamma_3 = 2.16 \text{ Mev}$, $1.2 \times 10^{-3}\%$ of disintegrations.

- (2) Find the dose rate at one meter from the source without additional shielding (include, however, the 90% absorption). Flux-to-dose rate conversion factors are:

For 1.17 Mev-- $2.18 \times 10^{-3} \text{ mr/hr/photon/cm}^2/\text{sec}$

For 1.33 Mev -- $2.42 \times 10^{-3} \text{ mr/hr/photon/cm}^2/\text{sec}$

For 2.16 Mev-- $3.42 \times 10^{-3} \text{ mr/hr/photon/cm}^2/\text{sec}$

- (3) Find the dose rate at one meter with 20 cm (7-7/8 in.) of lead shielding. Include buildup factor in calculations.

$$\text{Buildup factor, } B = A_1 \exp \left[-\alpha_1 (\mu t) \right] + A_2 \exp \left[-\alpha_2 (\mu t) \right] .$$

The following constants apply for lead:

<u>E</u>	<u>μ</u>	<u>A₁</u>	<u>α_1</u>	<u>A₂</u>	<u>α_2</u>
1.17	0.69	1.715	-0.0421	-1.715	0.1406
1.33	0.63	2.758	-0.0451	-1.758	0.1402
2.16	0.51	2.603	-0.0598	-1.603	0.1374

Problem 9

Thorium-228 has a half life of 1.9 years, and is a potentially useful power-producing isotope. Its first decay product (or daughter) is Radium-224 which has a half life of 3.6 days. The remaining decay products (6 in all) have very short half lives and reach secular equilibrium in less than one day. The unshielded dose rate at one meter from one gram of pure Thorium-228 is 1.16×10^3 mr/hr. The dose rate from one gram of Radium-224 is 1.55×10^8 mr/hr. (This latter figure includes the contribution from all subsequent daughters after Radium-224.) Find the unshielded dose rate from an originally pure one-gram sample of Thorium-228 after 1, 10 and 100 days of decay.

Problem 10

The half life for spontaneous fissioning of Curium-242 is 7.2 million years.

- (1) Calculate the neutron production rate (neutrons/sec-gm) for pure curium if 2.65 neutrons are released per fission.
- (2) The fast neutron removal cross section for polyethylene is $\Sigma_r = 0.111 \text{ cm}^{-1}$. Use this cross section as an absorption coefficient in the point source equation to find the thickness of polyethylene required to give a neutron flux of 100^1 neutrons/cm²-sec at one meter. This flux will give approximately 10 mrem/hr.

IV-F-2

Problem 11

The decay of Polonium-210 has a 0.8-Mev gamma associated with it. The ratio of gamma rays to alphas is $1.2:10^5$.

- (1) Find the unshielded dose rate at a distance of one meter from one gram of Polonium-210. The flux-to-dose rate conversion factor is 1.6×10^{-3} mr/hr/photon/cm²/sec.
- (2) How much lead will be required to reduce the dose rate at one meter to 10 mr/hr? The absorption coefficient for 0.8-mev gammas in lead is 0.953 cm^{-1} .
- (3) Find the minimum dimensions a box must have such that the surface dose rate does not exceed 200 mr/hr. Maintain the 10 mr/hr of Part 2. Neglect absorption of the box.

Solution to Problem 1

$$-\frac{dP}{dt} = \lambda P$$

$$\frac{dP}{P} = -\lambda dt \quad \text{or} \quad \int_{P_0}^P \frac{dP}{P} = -\lambda \int_0^t dt$$

$$\ln \frac{P}{P_0} = -\lambda t \quad \text{or} \quad P = P_0 \exp(-\lambda t)$$

At the half life $t_{1/2}$, $P = P_0/2$

$$\frac{P_0}{2} = P_0 \exp(-\lambda t_{1/2}) \quad \text{or} \quad \lambda t_{1/2} = \ln 2$$

$$\lambda = \frac{\ln 2}{t_{1/2}} = \frac{0.693}{t_{1/2}} = \frac{0.693}{138} = 0.005021 \text{ days}^{-1}$$

$$P = P_0 \exp(-\lambda t) = P_0 \exp-(0.005021)(60) = 0.73948 P_0$$

Therefore, 26.052% of the original atoms present will decay in a period of 60 days.

Solution to Problem 2

At the end of 90 days, $\frac{25}{0.05} = 500$ thermal watts are required.

This means that $500/120 = 4.167$ grams of Curium-242 must be present at this point.

$$P = P_0 \exp(-\lambda t)$$

$$4.167 \exp(0.693)(90)/163 = P_0$$

$$P_0 = 4.167 \exp(0.38264) = (4.167)(1.46609)$$

$P_0 = 6.109$ grams must be present at loading. The heat generated from the excess 1.942 grams at the start of life must be dumped.

Solution to Problem 3

$$(1) \quad E = a + bI \text{ (open circuit voltage + voltage drop in volts)}$$

$$\text{At } E = 5.5 \text{ volts and } I = 0 \text{ amperes,}$$

$$a = 5.5.$$

$$\text{At } E = 0 \text{ volts and } I = 14 \text{ amperes,}$$

$$14b = -5.5 \text{ and}$$

$$b = -0.392857 \text{ (resistance in ohms).}$$

Therefore,

$$E = 5.5 - 0.392857I \text{ and}$$

$$\text{Power} = EI = 5.5I - 0.392857I^2.$$

For maximum power,

$$\frac{\partial (EI)}{\partial I} = 5.5 - 0.785714I = 0,$$

$$I = \frac{5.5}{0.785714} = 7.0000 \text{ amperes.}$$

$$E = 5.5 - (0.392857)(7.0) = 2.75 \text{ volts, and}$$

$$(EI)_{\max} = (2.75)(7.000) = 19.25 \text{ watts.}$$

- (2) The voltage is 2.75 volts, and
the current is 7.00 amperes.

$$\text{The load resistance, therefore, is } \frac{2.75}{7.00} = 0.393 \text{ ohm.}$$

- (3) The open-circuit voltage of the generator is 5.5 volts.

Solution to Problem 4

Rate of accumulation of Cm-242 = rate of production of Cm-242
minus rate of decay of Cm-242.

$$\frac{dc}{dt} = 0.1 - \lambda c$$

or

$$\frac{dc}{dt} + \lambda c = 0.1$$

where

c = accumulation of Cm-242

t = time

λ = decay constant of Cm-242.

Solve the complementary equation:

$$\frac{dc_c}{dt} + \lambda c_c = 0$$

$$\frac{dc_c}{c_c} = -\lambda dt, \text{ therefore } c_c = A_1 \exp(-\lambda t),$$

where A_1 is a constant of integration.

Solve for the particular solution:

$$c_p = B$$

$$\frac{dc_p}{dt} = 0$$

$$\lambda c_p = 0.1 = \lambda B \quad \text{or} \quad B = \frac{0.1}{\lambda}.$$

The total solution is then:

$$c = c_c + c_p$$

$$c = A_1 \exp(-\lambda t) + \frac{0.1}{\lambda}.$$

Boundary condition:

$$\text{at } t = 0, c = 0.$$

The constant of integration is then given by

$$0 = A_1 + \frac{0.1}{\lambda} \quad \text{or} \quad A_1 = -\frac{0.1}{\lambda}.$$

Therefore,

$$c = \frac{0.1}{\lambda} [1 - \exp(-\lambda t)].$$

The maximum quantity of Cm-242 which can ever be accumulated exists when $t \rightarrow \infty$.

$$\begin{aligned} c_{\max} &= \frac{0.1}{\lambda} \quad \text{and} \quad \lambda = \frac{0.693}{163} \\ c_{\max} &= \frac{(0.1)(163)}{0.693} = 23.52 \text{ grams} \\ \exp(-\lambda t) &= 1 - \frac{\lambda c}{0.1} = \frac{0.1 - \lambda c}{0.1} \\ t &= \frac{1}{\lambda} \ln \frac{0.1}{0.1 - \lambda c} = 235.21 \ln \frac{0.1}{0.1 - 0.025972} \\ &= 235.21 \ln 1.35084 = (235.21)(0.30069) \\ &= 70.73 \text{ days to produce 6.109 grams of Cm-242} \end{aligned}$$

Solution to Problem 5

$$p(t)v = n(t)RT(t)$$

where

$$p(t) = \text{pressure}$$

$$v = \text{void volume (constant at 1 cc)}$$

$$n(t) = \text{number of moles of helium present}$$

$$T(t) = \text{temperature}$$

$$T(t) = at + b$$

At the point $t = 0$,

$$b = 1500^\circ \text{ F}$$

At the point $t = 100$ days,

$$500^\circ \text{ F} = 100a + 1500$$

$$100a = -1000$$

$$a = -10$$

$$\therefore T(t) = 1500 - 10t$$

For every Cm-242 atom that decays, one alpha particle or helium atom is formed.

The number of active Curium-242 atoms present initially will be:

$$N_0 = \frac{2 \text{ gm (initial loading)} \times 6.03 \times 10^{23} \text{ (Avogadro's number)}}{242}$$

$$N_0 = 4.98 \times 10^{21} \text{ atoms of Cm-242 at } t = 0.$$

Therefore, the number of helium atoms present at any time t will be:

$$N_\alpha(t) = N_0 [1 - \exp(-\lambda t)]$$

and

$$n(t) = \frac{N_{\alpha}(t)}{6.03 \times 10^{23}}$$

$$n(t) = \frac{N_0 [1 - \exp(-\lambda t)]}{6.03 \times 10^{23}}$$

$$n(t) = \frac{2 [1 - \exp(-\lambda t)]}{242}$$

Therefore, the desired expression is

$$p(t) v = \frac{[1 - \exp(-\lambda t)] R}{121} (1500 - 10t)$$

and at the 100-day point,

$$p = \frac{(0.34623) (82.06) (1500 - 1000)}{(121) (1.8)}$$

where

82.06 = gas constant R in cc, atm, °K

1.8 = factor for conversion from °R to °K.

p = 65.0 atm.

Solution to Problem 6

$$dE = \alpha (T) dT$$

$$E_{oc} = \int_{T_1}^{T_2} \alpha (T) dT$$

A relation for $\alpha (T)$ may be obtained from the given data:

$$\alpha (T) = aT + b .$$

At the point 1,

$$300 \times 10^{-6} = a560 + b .$$

At the point 2,

$$600 \times 10^{-6} = a1560 + b .$$

Solving for a and b gives

$$a = 300 \times 10^{-9}$$

$$b = 132 \times 10^{-6} .$$

Therefore,

$$\alpha (T) = 300 \times 10^{-9} T + 132 \times 10^{-6}$$

$$E_{oc} = \int_{560}^{1560} (300 \times 10^{-9} T + 132 \times 10^{-6}) dT$$

$$= 1.5 \times 10^{-7} T^2 + 132 \times 10^{-6} T \Big|_{560}^{1560}$$

$$= 0.36504 + 0.20592 - 0.04704 \times 10^{-3} - 0.07392$$

$$= 0.49699 \text{ volt.}$$

Solution to Problem 7

Production rate of Q - decay rate of Q = rate of change of Q.

$$\text{Production rate of Q} = -\frac{dP}{dt} = \lambda_1 P.$$

$$\text{Decay rate of Q} = -\frac{dQ}{dt} = \lambda_2 Q.$$

$$\text{Rate of change of Q} = \frac{dQ}{dt}.$$

Therefore,

$$\frac{dQ}{dt} = \lambda_1 P - \lambda_2 Q.$$

However,

$$-\frac{dP}{dt} = \lambda_1 P \quad \text{and} \quad P = P_0 \exp(-\lambda_1 t)$$

$$\frac{dQ}{dt} = \lambda_1 P_0 \exp(-\lambda_1 t) - \lambda_2 Q$$

$$\exp(\lambda_2 t) \frac{dQ}{dt} + \exp(\lambda_2 t) \lambda_2 Q = \lambda_1 P_0 \exp[(\lambda_2 - \lambda_1) t]$$

$$\frac{d}{dt} [Q \exp(\lambda_2 t)] = \lambda_1 P_0 \exp[(\lambda_2 - \lambda_1) t]$$

$$Q \exp(\lambda_2 t) = \frac{\lambda_1 P_0}{(\lambda_2 - \lambda_1)} \exp[(\lambda_2 - \lambda_1) t] + C.$$

Boundary condition: $Q = 0$ at $t = 0$.

$$C = \frac{\lambda_1 P_0}{\lambda_2 - \lambda_1}, \quad \text{and therefore,}$$

$$Q = \frac{\lambda_1 P_0}{(\lambda_2 - \lambda_1)} [\exp(-\lambda_1 t) - \exp(-\lambda_2 t)]$$

which is an expression for the power from Q at any time t.

The total power produced then is:

$$P + Q = \frac{\lambda_1 P_0}{(\lambda_2 - \lambda_1)} [\exp(-\lambda_1 t) - \exp(-\lambda_2 t)] + P_0 \exp(-\lambda_1 t)$$

Solution to Problem 8

(1) Energy per disintegration (Mev):

$$\begin{array}{lll}
 \beta & 0.287 \times 0.312 & = 0.0895 \\
 \gamma_1 & 1.17 \times 1.0 \times 0.9 & = 1.0530 \\
 \gamma_2 & 1.33 \times 1.0 \times 0.9 & = 1.1970 \\
 \gamma_3 & 2.16 \times 1.2 \times 10^{-5} \times 0.9 & = \frac{0.00002}{2.3395}
 \end{array}$$

For 1 curie of Co-60,

$$3.7 \times 10^{10} \times 2.34 \times 1.6 \times 10^{-13} = 0.0138 \text{ watt/curie}$$

$$\frac{\text{disintegrations per sec}}{\text{curie}} \times \frac{\text{Mev}}{\text{disintegration}} \times \frac{\text{watt-sec}}{\text{Mev}}$$

$$30 \text{ curies/gram} = 0.414 \text{ watt/gm}$$

$$\frac{100 \text{ thermal watts}}{0.414 \text{ watt/gm}} = 241.5 \text{ gm of cobalt at 30 curies / gram}$$

$$30 \times 241.5 = 7.245 \times 10^3 \text{ curies.}$$

(2) Dose rate at one meter (without shield)

$$= \frac{S}{4\pi r^2} \times \text{conversion factor (where } S = \text{disintegrations per second)}$$

E	S (E)	Conversion Factor	S x Conversion Factor
1.17	$7.245 \times 10^3 \times 3.7 \times 10^{10} \times 1.0$	2.18×10^{-3}	5.84×10^{11}
1.33	$7.245 \times 10^3 \times 3.7 \times 10^{10} \times 1.0$	2.42×10^{-3}	6.49×10^{11}
2.16	$7.245 \times 10^3 \times 3.7 \times 10^{10} \times 1.2 \times 10^{-5}$	3.42×10^{-3}	0.00011×10^{11}

$$12.33 \times 10^{11}$$

$$\frac{1.233 \times 10^{12}}{4 \pi \times (100)^2} = 9.81 \times 10^6 \text{ mr/hr with no absorption.}$$

With 90% absorption, 10% of gammas escape; therefore,

$$\text{dose rate} = 0.1 \times 9.81 \times 10^6 = 9.81 \times 10^5 \text{ mr/hr.}$$

$$(3) \text{ Dose rate} = \frac{BS \exp(-\mu t)}{4 \pi r^2} \times \text{conversion factor}$$

E	S (E)	
1.17	$0.1 \times 7.245 \times 10^3 \times 3.7 \times 10^{10} \times 1.0$	2.681×10^{13}
1.33	$0.1 \times 7.245 \times 10^3 \times 3.7 \times 10^{10} \times 1.0$	2.681×10^{13}
2.16	$0.1 \times 7.245 \times 10^3 \times 3.7 \times 10^{10} \times 1.2 \times 10^{-5}$	3.217×10^8

Buildup factors

$$E = 1.17; t = 20 \text{ cm}; \mu = 0.69; \mu t = 13.8.$$

$$B = 2.715 \times \exp \left[- (-0.0421) \times 13.8 \right] - 1.715 \exp \left[- 13.8 \times 0.1406 \right]$$

$$= 2.715 \exp 0.7190 - 1.715 \exp (-1.9403)$$

$$B_{1.17} = 5.33$$

$$E = 1.33; t = 20 \text{ in}; \mu = 0.63; \mu t = 12.6.$$

$$B = 2.758 \exp \left[- (-0.0451) \times 12.6 \right] - 1.758 \exp \left[- 0.1402 \times 12.6 \right]$$

$$= 2.758 \exp(0.5683) - 1.758 \exp(-1.7665)$$

$$B_{1.33} = 4.57$$

$$E = 2.16; \mu = 0.51; t = 20; \mu t = 10.2.$$

$$\begin{aligned} B &= 2.603 \exp[-(-0.0421 \times 10.2)] - 1.603 \exp[10.2 \times (-0.1374)] \\ &= 2.603 \exp(0.4294) - 1.603 \exp(-1.4015) \\ &= 3.6 \end{aligned}$$

$$\text{Dose rate} = \frac{ES e^{-\mu t}}{4\pi r^2} \times \text{conversion factor}$$

E	E	S	$\exp(-\mu t)$	Conversion Factor	$ES \exp(-\mu t) \times$ Conversion Factor
1.17	5.33	2.681×10^{13}	9.8×10^{-7}	2.18×10^{-3}	3.053×10^5
1.33	4.57	2.681×10^{13}	3.3×10^{-6}	2.42×10^{-3}	9.785×10^5
2.16	3.60	3.217×10^8	3.65×10^{-5}	3.42×10^{-3}	0.001×10^5 12.839×10^5

$$\text{Dose rate} = \frac{1.2839 \times 10^6}{4\pi (100)^2} = 10.2 \text{ mr/hr at 1 meter with 7-7/8-inch lead shield.}$$

Solution to Problem 9For Thorium-228

$$T_{1/2} = 1.9 \text{ years}$$

$$\lambda_t = \frac{0.693}{1.9 \times 365} = 0.0010 \text{ days}^{-1}$$

$$N_t \text{ per gm} = \frac{6.023 \times 10^{23}}{228} = 2.64 \times 10^{21}$$

For Radium-224

$$T_{1/2} = 3.6 \text{ days}$$

$$\lambda_R = \frac{0.693}{3.6} = 0.1952 \text{ days}^{-1}$$

$$N_R = \frac{6.023 \times 10^{23}}{224} = 2.69 \times 10^{21}$$

Decay of thorium

$$N_T = N_T(0) \exp(-\lambda_T t)$$

where

$$N_T(0) = \text{Number of atoms present at } t = 0$$

Buildup of Radium-224

$$\frac{d(N_R)}{dt} = \lambda_T N_T - \lambda_R N_R$$

$$= \lambda_T N_T(0) \exp(-\lambda_T t) - \lambda_R N_R$$

This equation has the following solution

$$N_R = N_R(0) \exp(-\lambda_R t) + \frac{\lambda_T N_T(0)}{\lambda_R - \lambda_T} \left[\exp(-\lambda_T t) - \exp(-\lambda_R t) \right]$$

$$\frac{\lambda_T N_T(0)}{\lambda_R - \lambda_T} = \frac{0.0010 \times 2.64 \times 10^{21}}{0.19254 - 0.001} = 1.378 \times 10^{19}$$

Time Days	$e^{-\lambda_T t}$	$e^{-\lambda_R t}$	$e^{-\lambda_T t} - e^{-\lambda_R t}$	N_R	gm Ra-224 = $\frac{N_R}{2.69 \times 10^{21}}$	Dose Rates mr/hr	
						Th-228	Ra-224
1	0.9990	0.8249	0.1741	0.24×10^{19}	8.93×10^{-4}	1.159×10^{-3}	1.38×10^5
10	0.9900	0.1459	0.8441	1.163×10^{19}	4.33×10^{-3}	1.148×10^{-3}	6.71×10^5
100	0.9048	0	0.9048	1.247×10^{19}	4.64×10^{-3}	1.049×10^{-3}	7.19×10^5

Total dose rates at one meter (mr/hr)

1 day	1.39×10^5
10 days	6.72×10^5
100 days	7.20×10^5

Solution to Problem 10

$$(1) \text{ Fission half life} = 7.2 \times 10^6 \text{ years} = 2.270 \times 10^{14} \text{ sec}$$

$$\lambda_{\text{Fission}} = \frac{\ln 2}{2.27 \times 10^{14}} = 3.05 \times 10^{-15} \text{ sec}^{-1}$$

$$N = \frac{6.023 \times 10^{23}}{242} = 2.49 \times 10^{21} \text{ atoms/gram}$$

$$\lambda N = 3.05 \times 10^{-15} \times 2.49 \times 10^{21} = 7.59 \times 10^6 \text{ fissions/sec/gram}$$

$$2.65 \times 7.59 \times 10^6 = 2.01 \times 10^7 \text{ neutrons/sec/gram}$$

$$(2) = \frac{S \exp(-\mu t)}{4 r^2}$$

With no absorption and at a distance of 1 meter,

$$\phi = 2.01 \times 10^7 \times 7.958 \times 10^{-6} = 160 \text{ neutrons/cm}^2/\text{sec}$$

$$160 \exp(-\Sigma_r t) = 100$$

$$\exp(-\Sigma_r t) = 0.6250$$

$$-\Sigma_r t \log e = \log(0.6250) = -0.20412$$

$$t = \frac{0.20412}{0.111 \times 0.43429} = 4.23 \text{ cm} = 1.66 \text{ inches}$$

Solution to Problem 11

$$\begin{aligned}
 (1) \quad T_{1/2} &= 138 \text{ days} = 1.1923 \times 10^7 \text{ sec} \\
 \lambda &= \frac{\ln 2}{T_{1/2}} = 5.813 \times 10^{-8} \text{ sec}^{-1} \\
 N &= \text{atoms per gram} = \frac{6.023 \times 10^{23}}{210} = 2.868 \times 10^{21} \\
 \text{Disintegration rate} &= \lambda N = 5.813 \times 10^{-8} \times 2.868 \times 10^{21} \\
 &= 1.667 \times 10^{14} \text{ dis/sec-gm} \\
 \frac{\text{gammas}}{\text{sec-gm}} &= \frac{1.667 \times 10^{14}}{10^5} = 1.667 \times 10^9 \\
 \text{Dose rate} &= \frac{S}{4\pi r^2} \times \text{conversion factor} = \\
 &= \frac{1.6 \times 10^{-3} \times 1.667 \times 10^9}{4 (100)^2} \\
 &= 21.2 \text{ mr/hr} \\
 (2) \quad \text{Dose rate} &= \frac{S}{4\pi r^2} \exp(-\mu t) \times \text{conversion factor} \\
 21.2 \exp(-\mu t) &= 10 \\
 \exp(-0.953t) &= \frac{10}{21.2} = 0.4717 \\
 -0.953 t \log e &= \log 0.4717 \\
 -0.953 t &= \frac{-0.32633}{0.43429} \\
 t &= \frac{0.32633}{0.953 \times 0.43429} = 0.788 \text{ cm} = 0.310 \text{ inch}
 \end{aligned}$$

$$\begin{aligned}
 (3) \text{ Dose rate at } R_2 &= \text{dose rate at } R_1 \times \frac{R_1^2}{R_2^2} \\
 200 &= 10 \times \frac{10^4}{R_2^2} \\
 R_2^2 &= \frac{10 \times 10^4}{2 \times 10^2} = 500 \\
 R &= 22.36 \text{ cm} = 8.80 \text{ inches}
 \end{aligned}$$

APPENDIX B

TRANSIT IV-A POWER SUPPLY

The successful launching and operation of a Transit satellite in June 1961 heralded the first application of nuclear power in America's space program. After more than five years of development by Martin Company engineers under AEC auspices, a SNAP (systems for Nuclear Auxiliary Power) generator was put to work to supply power for two transmitters aboard the Navy's Transit IV-A navigation satellite.

The Transit IV-A satellite, Fig. B-1, a development of the Applied Physics Laboratory of The Johns Hopkins University, is designed to produce constant frequency transmission for a minimum of five years to ships at sea. Through doppler shift techniques, accurate navigational information is made available to such vessels. One of the major problems associated with this satellite was the development of a power source capable of meeting the five-year life requirement. Early Transit satellites used solar cell battery power supplies. The battery system, however, was required to go through many recharge cycles over the five-year period and therefore presented a major reliability problem.

It was a natural development that the Transit program and the SNAP program should merge. Initial discussions were held approximately one year ago when it was decided that the first system should be a low powered prototype in order to demonstrate operational capability.

Extensive work has been performed on the testing and modification of the SNAP-3 system for operational use. This was a logical starting point for the prototype Transit generator. One of the major modifications required was the use of a long-life fuel in place of Polonium-210. Several radioisotopes were considered for the Transit system. The criteria for selecting a core material are as follows:

- (1) It must possess sufficient power density to maintain the hot junction at the required temperature.
- (2) It must have a sufficiently long half life for the intended space mission.
- (3) The nuclear radiation level should not be hazardous to those working with it using reasonable precautions.

Fig. B-1. The Transit IV-A Satellite

In considering these three criteria, it became apparent that the non-fissionable radioisotope Plutonium-238 was an ideal core material for the Transit satellites. The Plutonium-238 is an alpha-emitter with a half life of 89.6 years. The thermal power output of the Plutonium-238 used on the Transit IV-A satellite was 52 watts. This source of heat was quite sufficient to maintain the hot junction at the desired temperature.

The half life of 89.6 years for this material was quite compatible with a design objective operating life of five years. As a result of radioactive decay, the power output of the Transit IV-A generator will steadily decrease with time. For the five-year design objective operating life of the Transit satellites, there will be a power reduction of approximately 3.7% due to this decay in core energy. By providing 10% more heat source capacity than is required at the time of launch, a 10-year life for the nuclear-powered thermoelectric generator could be realized.

The third item which makes the Plutonium-238 an ideal core material is that, since it is principally an alpha-emitter, no heavyweight shielding is required to protect personnel. At a one-foot distance from the center of the generator, the total dose rate was measured to be only 4 mrem/hr. These low radiation levels made it possible to handle the device during testing and prelaunch operations with only minimum care. These levels are actually sufficiently low so that personnel handling the device were not required to wear film badges or any other type of radiation detector. Thus, a new fuel core was designed and built which permitted the use of Plutonium-238. Fuel was provided and the core was loaded by Mound Laboratory.

Other generator modifications included those required to optimize (1) the system for the Transit configuration, (2) heat balance for the Transit orbit, and (3) efficiency of the thermoelectric system for the new fuel. The details of these modifications cannot be discussed for security reasons. The generator system which finally evolved from these modifications contains 1600 curies of Plutonium-238. The total system is 4-3/4 inches in diameter and 5-1/2 inches in height and weighs approximately 4 pounds. The generator power output is 2.7 watts. The design life of the system is 5 years with a possibility of exceeding this life, provided that the electronics continue to operate.

In order to derive the greatest significance in the use of a nuclear-powered thermoelectric generator in the Transit IV-A satellite, it was decided to power several distinct electronic devices in the satellite. One of these devices was the satellite's ultra-stable crystal oscillator which is the heart of the electronic system that provides the capability for doppler tracking of the Transit satellite. In addition, it was decided

to power a buffer-divider-multiplier unit, a 54-mc multiplier, a 54-mc transmitter and a 324-mc transmitter. The operating voltages of these instruments are considerably higher than the output voltage of the generator itself. Consequently, a dc-dc converter was designed and manufactured. Output taps of 21, 11.5, 0 and -4 volts were provided. The tolerance permitted in voltage was $\pm 5\%$ with a maximum of 1% ripple. The converter provided met these specifications, and is operating at an efficiency of greater than 70%.

The generator-converter system was first subjected to flight qualification tests. These qualification tests included vibration and thermal vacuum simulating launch and orbital conditions. The system was then mounted on the satellite for flight acceptance tests. The system satisfactorily passed all of these requirements.

Safety Analysis and Test

One of the most critical aspects in the use of radioactive material in space is the assurance that the operational safety of the device is conclusively demonstrated. The first step in providing this assurance is the analysis of the environments to which the generator might be subjected under any possible missile abort conditions. Thus, the missile trajectory was analyzed in great detail to establish those points along the trajectory which conceivably could subject the generator to maximum environmental conditions. The conclusions drawn from this analysis are as follows:

- (1) During launch and several seconds thereafter the primary environment to be considered is that of fire and/or explosion. The intensity of this environment was calculated using the methods of Space Technology Laboratory, Inc., that were verified by experimental measurements in actual missile fires. Fire temperatures were determined to be of the order of 5000° F for millisecond periods; these temperatures dropped rapidly to temperatures of the order of 1600° to 2000° F with burning duration of approximately 15 minutes. Maximum shock overpressures resulting from launch pad explosions of the Thor Able Star vehicle were calculated to be 643 psi.
- (2) As the missile proceeds along the trajectory the primary environmental condition to be considered becomes possible impact of the fueled generator on land or sea. It was determined that the terminal velocity of the generator was 214 ft/sec and the terminal velocity of the bare fuel core was 250 ft/sec.

- (3) Re-entry heating becomes a significant parameter as the vehicle approaches orbital velocity. Failure to inject the satellite into orbit could result in a re-entry heating condition sufficient to burn off the outer shell of the generator and result in bare core impact.

In order to demonstrate that the generator could withstand these environmental conditions, a series of safety tests were performed. One-third scale explosion tests and full-scale fire tests were conducted by Aberdeen Proving Ground. These tests accurately simulated the anticipated environmental conditions of fire and explosion. Multiple generator and core samples were exposed. In no instance was there any indication of fuel core damage. Impact studies were performed at 360 ft/sec (100 ft/sec greater than anticipated) against granite for both the generator and the bare fuel core. Some plastic deformation was observed; however, the cores were pressure-tested after impact and found to be structurally sound. Re-entry heating was simulated using a plasma arc. It was established that orbital re-entry would result in complete burnup of the fuel core at altitudes in excess of 100,000 feet. Under these conditions the isotope would be widely dispersed over the surface of the earth and ground concentration would be insignificant. Suborbital conditions produce a number of heating curves, some of which result in partial burnup. The partial burnup area, however, was determined to be in one of the most remote regions of the Indian Ocean. No population would be affected by the activity released.

As an extra measure of conservatism, meteorological studies were performed for the Cape Canaveral area. Despite the fact that the probability of fuel release is extremely small, if not zero, a study was performed which assumed that the Plutonium-238 fuel was released in a missile fire; the fallout field from the resulting cloud was then calculated. From these data, it was established that no member of the public domain would be exposed to ingestion or inhalation doses in excess of the maximum permissible body burden.

Additional tests were performed on internal core pressure resulting from helium gas buildup (alpha decay), corrosion in salt water, and burial in typical soil materials. All these tests gave positive results of satisfactory fuel containment.

The results of this program were reviewed by the Atomic Energy Commission, the Joint Committee on Atomic Energy, and the Air Force Special Weapons Center. These organizations concurred in the conclusion that the generator was operationally safe.

Having satisfactorily proved the operational capability and safety of the system, the process of obtaining flight approval was initiated. This matter was taken up by the National Space Council and on Friday, June 23, final flight approval was obtained. This approval is extremely significant in that a precedent has been set concerning nuclear space systems, providing all safety requirements can be met. The decision paves the way for larger, more advanced space-nuclear devices, and opens the door to reliable, compact space power.

Conclusions

On June 28, 1961 at 11:22 P. M. Eastern Standard Time, the Thor Able Star successfully launched the Transit satellite into a 600-statute mile circular orbit. The launch was the most successful of any experienced to date in the Transit program. No unusual circumstances occurred, and the generator is operating satisfactorily at this time. It can be said with confidence, however, that if for some reason the missile had not achieved its design performance, no person--here or abroad--would have been endangered by the launch of this device. The generator used in this shot was small, but the significance of its approval and success will have far-reaching effects on our future satellite systems.

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