AD/A-001 210

RADIOISOTOPE SOURCES OF ELECTRIC POWER

G. M. Fradkin, et al

Army Foreign Science and Technology Center
Charlottesville, Virginia

20 September 1973
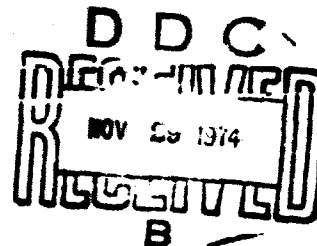
ABSTRACT: The basic principles of operation of radioisotope
current sources for various purposes are summarized. Structural
diagrams of contemporary radioisotope thermoelectric generators
are surveyed. Methods of obtaining radioisotope sources of
thermal energy are described and their physical and radiation
characteristics given. Problems of radiation safety during
work with radioisotope thermoelectric generators are reviewed.

This translation was accomplished from a xerox manuscript. The graphics
were not reproducible. An attempt to obtain the original graphics yielded
negative results. Thus, this document was published as is, in order to make
it available on a timely basis.

1

REPRODUCTION QUALITY NOTICE

This document is the best quality available. The copy furnished to DTIC contained pages that may have the following quality problems:

- Pages smaller or larger than normal.

- Pages with background color or light colored printing.

- Pages with small type or poor printing; and or

- Pages with continuous tone material or color photographs.

Due to various output media available these conditions may or may not cause poor legibility in the microfiche or hardcopy output you receive.

☐ If this block is checked, the copy furnished to DTIC contained pages with color printing, that when reproduced in Black and White, may change detail of the original copy.

Introduction

In recent years, side by side with the development of industry energetics, where the building of atomic electrical stations is of great importance, much attention has been given to the creation of low-powered self-contained sources of electrical energy based on utilization of radioactive isotope decay energy. Such sources have been developed and applied mostly for supplying the on board equipment of artificial earth satellites and space stations, oceanographic and navigation devices, meteorological stations, etc. [1 - 3].

The main forms of energy that may be transformed into electrical energy are: chemical energy, produced during the combustion of fuel or as a result of electrochemical processes in various kinds of batteries, and fuel cells; nuclear energy released during the splitting of heavy or fusion of light nuclei or during radioactive decay; the energy of solar radiation reaching the earth's surface or the space craft.

From the point of development of self-contained electrical energy sources, chemical sources of galvanic element or battery types have had wide application. (Traditional generators based on combustion of chemical fuel and mechanical methods of transforming thermal energy into electicity usually require systematic service; therefore, they cannot be used for the cases studied here.) In the last 15 - 20 years all possible combinations of galvanic pairs with different electrolytes have been investigated from the point of utilization as current sources. Only a few electrochemical systems, however, are of practical feasibility. Besides the classical lead and nickel batteries, silver batteries with a zinc or cadmium anode have become industrially important. There are also a few systems which have high short-term capacity. (In general, these are the ampule batteries.) The main characteristics of galvanic elements and storage batteries are given in Table 1 [2,3].

a) Система и тип электроэлементов	b) Э.д.с., В	c) Рабочее напряжение, В	d) Саморазряд в месяц, %	e) Минимальная рабочая температура, °C	f) Удельная энергоемкость		g) Отдача по емкости, %	h) Отдача по энергии, %	i) Срок службы (число циклов заряд—разряд или время)
					вт·ч/кг	вт·ч/дм³			
j) Свинцовые (кислотные)	2,1	1,9	21	—25	26	58	80—85	68—70	250—300
k) Железо-никелевые (ЖН) (щелочные)	1,4	1,2	18—25	—25	18	34	67	47	700
l) Кадмиево-никелевые, ламельные (КН)	1,35	1,2	11—18	—35	27	50	70	50	$(2—3)\cdot10^3$
m) Кадмиево-никелевые, безламельные (КНБ)	1,35	1,25	15—18	—35	38	62	75	62—65	$2\cdot10^3$
n) Кадмиево-никелевые, герметизированные (КНГ)	1,35	1,25	15	—40	27	80	85	72	10^3
o) Никель-цинковые (НЦ)	1,84	1,6—1,7	20	—30	60	110	87	75	100—150
p) Серебряно-кадмиевые (СК)	1,4	1,1	10	—30	70	130	75—80	70	500

Table 1

Key: a) System and type of electroelement
 b) Electromagnetic force, e.m.f.
 c) Operating voltage, volts
 d) Self-discharge per month, %
 e) Minimum operating temperature, °C
 f) Specific energy capacity
     watt hr/kg, watt hr/dm³
 g) Capacity efficiency, %
 h) Energy efficiency, %
 i) Service life (number of charge-discharge cycles or time)
 j) Lead (acid)
 k) Nickel-iron (alkaline) NI
 l) Nickel-cadmium, lamellar (NC)
 m) Nickel-cadmium, non-lamellar (NNC)
 n) Nickel-cadmium, sealed (NCS)
 o) Nickel-zinc (NZ)
 p) Silver-cadmium (SC)

(continued)

a) Система и тип электроэлементов	b) э.д.с. В	c) Рабочее напряжение V	d) Саморазряд в %	e) Минимальная рабочая температура, °C	f) Удельная энергоемкость		g) Отдача по емкости, %	h) Отдача по энергии, %	i) Срок службы (число циклов в заряд-разряд или время)
					вт·ч/кг	вт·ч/дм³			
g) Серебряно-цинковые (СЦ)	1,88	1,5	5—15	—40	120-150	220	85	75	150
h) Низкотемпературные топливные элементы Бэкона (H₂—O₂)	1,1	0,79	—	w) Рабочая температура 200°C	120—150	—	—	—	x) Несколько месяцев
s) Высокотемпературные топливные элементы (C—O₂); (H₂—O₂)	1,02	0,7	—	w) Рабочая температура 500°C	—	—	—	—	—
t) Окисно-ртутные элементы (ОРЭ)	1,35	1,3—1,1	—	—10	90—110	300—450	85—90		1,5 года y)
u) Марганцево-цинковые (МЦ) щелочные	1,5	1,3—1,1	—	—30	44	140	30—40		—
v) Медноокисные элементы (МОЭ)	0,96 0,88	0,7—0,5	—	—30	25—30	35—40	95—98	—	10—15 лет y)

z) Примечание. Все основные параметры приведены для температуры +20°C.

Table 1 (cont'd)

Key:
q) Silver-zinc (SZ)
r) Low temperature fuel elements, Bacon type (H₂-O₂)
s) High temperature fuel elements (C-O₂), (H₂-O₂)
t) Oxide-mercury elements (OME)
u) Manganese-zinc (MZ), alkaline
v) Copper-oxide elements (COE)
w) Operating temperature
x) A few months
y) Years
z) Note: All basic parameters are given for the temperature +20°C.

As follows from the table, the specific energy capacity of the best (silver-zinc) batteries is no more than 120-130 watt hr/kg under the most favorable operating conditions. The utilization of these sources in higher or negative temperature regions causes deterioration of their characteristics.

When making self-contained systems with longer service time, together with useful energy consumption, the self-discharge of chemical batteries becomes very substantial. It restricts the operating time and lowers the value of energy capacity.

One of the consumers of self-contained energy sources is the Hydro-meteorological Service. In the Soviet Union every year automatic radiometeorological installations are build in the Far North, in the deserts, in the taiga and in high-mountain regions where neither sources of electrical supply nor the usual meteorological stations existed and could not be set up because of severe climate or difficult access. The electrical supply of such stations is usually performed with nickel-cadmium batteries. The battery weight (for supplying the station for one year) is 1-2 tons, while the weight of the station itself does not exceed a few hundred kilograms.

If it is possible to charge the batteries with a wind turbine, then the weight of the supply source us slightly decreases. However, the normal operation of a wind turbine is possible only under certain wind conditions, making their exploitation difficult and not providing reliable non-interrupted station operation. It is also possible to charge the chemical batteries with the aid of thermoelectric generators operating on liquid or gaseous fuel, but such generators require systematic maintainance, not easy under self-contained conditions.

At the present time, the ground and floating weather stations have been developed for measuring a large number of parameters and transmitting them to regional and central meteorological centers. This requires a power increase of the supply source and, therefore, leads to an additional increase of weight and decrease of reliability.

The increase of specific energy capacity of self-contained electrical energy sources is the basic requirement for generators designed for supply of the on board equipment of space craft.

Solar elements together with buffer batteries are now widely used as electrical energy sources for supply of on board equipment for artificial earth satellites and space stations. At the present time 70-100 watts of electrical energy can be obtained from 1 m^2 of solar photoelement surface. As all listed sources of electrical energy, solar batteries have some shortcomings restricting their possible application. The surface of solar elements must be protected in the

radiation region of the earth from the shocks of micrometeorite dust, overheating and other harmful actions [1,9,15,16] .

Recently exclusive attention has been given to designing and creating electrochemical fuel elements having a specific energy capacity which exceeds by several times the capacity of chemical batteries. The most developed and potentially useful are apparently the hydrogen-oxygen elements. The electrical energy of such elements is obtained as a result of transformation of chemical energy into electrical energy. The element can operate at room temperature and atmospheric pressure, but the efficiency of the electrochemical process substantially increases with increasing temperature and pressure.

The process of creating fuel elements is very intense in the United States. In particular, the battery of fuel elements has been designed which provides electrical energy for on board equipment of the "Gemini" sattelite (the maximum power of the battery is 2 kwatts). The analogous battery for the "apollo" has been developed. This battery produces energy for the on board systems of the spacecraft and at the same time gives as a by-product drinking water for the astronauts. .

Despite the substantially higher specific energy capacity of the fuel elements, their application is restricted by devices with service time of about a few hundred hours. For longer operation, nuclear or solar systems are more advantageous.

The great necessity of self-contained electrical energy sources has arisen in connection with the program of surface and underwater navigation, development of oceanographic work, automation of the Meteorological Service, realization of long space flights and the creation of portable apparatuses for various uses. These sources must have longer operating times with specific energy capacity tens and hundreds of times higher than the existing sources. In a few cases, such sources can be created on the basis of utilization of the decay energy of radioactive isotopes. Radioactive isotopes are accumulated in large quantity in the waste solutions of the nuclear industry and heat producing elements of nuclear electrical stations and can also be obtained by neutron irradiation in nuclear reactors.

In the last few years, several scientific research institutes and technological companies in England, France, Japan, Canada and West Germany have been involved in the process of developing radio-isotope sources of electrical energy. Such great interest in this utilization of nuclear energy can be explained , first of all, by the substantial advantages of radioisotope thermoelectric generators over other self-contained electrical energy sources: high energy capacity (thousands of watt hr/kg), long service life (up to 10 years and more), and sufficiently high reliability.

The experience of development and operation of the radioisotope generators in different regions of the earth and also the growing possibilities of radioactive isotope production allows the hope that radioisotope energetics will be more widely used.

In this book the authors have sought to describe briefly the main problems that arise during the development and operation of radio-isotope thermoelectric generators and to answer many questions received from the builders and consumers of self-contained sources of thermal and electrical energy with long service life and high energy capacity.

General Information on Radioisotope Fuels

Radioisotope Fuels

Radioactive isotopes have the property of spontaneous decay. The following types of decay are usually distinguished: α-decay, β-decay, electron capture, spontaneous fission.

The decay results in continuous decrease in the atomic number of the initial radioactive isotope. If N = number of nuclei of the isotope at time t, then the number of decaying nuclei per unit time is proportional to N. Let us call the proportionality factor, determined by the probability of decay per unit time, λ (this value is usually called the decay constant). The law of radioactive decay can be written in the form

$$\frac{dN}{dt} = -\lambda N,$$

(2.1)

where dN = number of nuclei decaying in the time interval dt from time t to t + dt. We designate N_0 the number of atomic nuclei at time t = 0; from expression (2.1) we derive the basic equation

$$N_t = N_0 e^{-\lambda t}.$$

(2.2)

From the given equation one can see that the decay constant λ characterizes the rate of radioactive decay, or the rate of "burning" of the radioisotope fuel.

It is convenient to characterize the rate of the radioactive decay process by the time during which the number of nuclei decreases by half, $T_{1/2}$. This value is usually called the half-life of the radioisotope. The decay constant λ is related with the half-life according to

$$T_{1/2} = \frac{\ln 2}{\lambda} = \frac{0.693}{\lambda}.$$

(2.2')

The value of the half-life is constant for a given radioactive isotope and is practically independent of external conditions such as temperature and pressure, and also magnetic and electrical fields, chemical composition of the material, etc. The values of the half-lives may vary from fractions of a second to millions of years.

The value characterizing the number of radioactive decays per unit time is called the activity of the radioactive preparation (C = Nλ). The activity is measured in curies: 1 curie = the activity of a preparation having $3.7 \cdot 10^{10}$ decays per second.

Alpha-radioactive isotopes are usually associated with the heavy elements. In the process of their decay, α -particles (helium nuclei) are formed, with energies in the range of 5-9 Mev. (1 Mev = $1.6 \cdot 10^{-13}$ j = $1.6 \cdot 10^{-6}$ ergs.) The most important property of α -radioactive isotopes is the large amount of energy released in a single act of decay. In some α -emitters which are the originators of radioactive transformation chains (Ac^{227}, Th^{227}, U^{232}), the total energy released in one decay event of the mother isotope is 35-40 Mev, comparable to the energy of fission of heavy nuclei. It is also important that α -particles have very short paths (a range of millimeters) in the material of the preparation and that deceleration of the α -particle in matter results in practically no braking radiation.

During β -transformations, together with β -particles, neutrinos ν are usually emitted. Thus, the energy of the decay is distributed between β -particles, neutrinos and nuclear recoils. In some cases β -decay can be accompanied by γ -radiation.

During electron capture, there arises characteristic X-rays, electromagnetic radiation with energies in tens of kiloelectron volts.

During fission of heavy nuclei, neutrons and nuclear fragments are formed.

In the general case, the total energy of decay E$_{decay}$ see (2,3) is made up of the energy of the α -particles and the recoil nucleus (first term), the energy of the γ -quanta (third term), and the energy of the β -particles (second term). When there is spontaneous decay, it is necessary to include the energy of the fission decay products (fourth term):

$$E(\text{Mev/decay}) = \left(1 + \frac{m_\alpha}{m_{\alpha\lambda}}\right) \sum_i E_{\alpha i} n_{\alpha i} \cdots$$

$$\cdots \sum_i E_{\beta i} n_{\beta i} - \sum_i h\nu_i n_{\gamma i} + E_{\text{дел}} \qquad (2.3)$$

Here n$_\alpha$, n$_\beta$, n$_\gamma$ = the number of α -, β -particles and γ -quanta, respectively, in a decay act; m$_\alpha$ /m$_{\alpha\lambda}$ ratio of masses of the

-8-

alpha-particle and recoil nucleus; E_{α_i} = energy of the α-particles of the given group; $h\nu_i$ = energy of the γ-quanta of the given group; E_{β} = mean energy of the β-particles of the given group. The summation is carried out with respect to all groups of α-, β-particles and γ-quanta.

The majority of radioactive isotopes may be obtained from the wastes of atomic industry (fission products) or by means of irradiation in nuclear reactors (reactor isotopes).

At the present time more than 1000 radioactive isotopes are known; however, only a few of them may be used as fuel.

The energy of α- and β-particles is practically completely absorbed with the material of the radioactive fuel; γ-radiation is absorbed only partially.

Beta-radiation has a continuous energy spectrum; the energy of β-particles lies within the limits from zero to some limiting value, $E_{\lim}$, characteristic for the given isotope. Beta-particles have relatively short paths (several millimeters in all); however, their absorption in matter is accompanied by the appearance of braking radiation, the spectrum of which is continuous and extends from zero to some maximum electron energy. The mean energy of the beta spectrum E_{β} occurs in the range $0.25E_{\lim} \gtrsim E_{\beta} \lesssim 0.5E_{\lim}$.

After radioactive decay the nucleus of the daughter isotope is often in an excited state. The jump of the nucleus from this state to a lower energy level and to the normal state is accompanied by radiation of γ-quanta. Gamma-radiation of the radioactive isotope can be comprised of quanta of the same energy or of a group of quanta with discreet energy values. Gamma radiation has high penetration abilities which may be described by a linear absorption coefficient μ.

Fundamental requirements of Radioisotope Fuels

When making a radioisotope heat source, one gives preference to those chemical forms and physical states of the radioactive preparation which correspond to the minimum radiotoxicity with minimum (for the given isotope) specific heat evolution. The radioactive preparation should be solid, non-friable, practically insoluble in sea and fresh water, nonsublimating and non-reactive with air, water and the material of the ampoule. The preparation should have high radiation and thermal stability ensuring minimal leakage of the isotope during unforseen accidents damaging the radioisotope heat source. The lower limit of its melting and boiling points are regulated by the "Sanitation Regulations" (see Appendix) and are equal to 500° and 1500 C respectively (for generators with low temperature thermoelectric convertors).

As regards the radiation characteristics, the preparation should

contain a minimum amount of impurities of radioactive isotopes with hard γ-radiation and neutron radiation. The material of the chemical compound or of the carriers should be made from elements with low Z when making fuels based on β-active isotopes, and with high Z when making fuels based on α-active isotopes. The latter requirements necessitate the reduction of output of braking radiation in β-preparations and of neutron radiation in α-preparations.

Regarding the thermal characteristics, the preparation should possess sufficiently high thermal conduction and not contain a large quantity of impure radioactive isotopes with a half-life differing strongly from $T_{1/2}$ of the basic isotope. Low thermal conduction of the preparation leads to a substantial drop in the internal temperature of the preparation and the possibility of disturbance of its thermal stability. The presence of a significant quantity of short-lived isotope leads to a substantial drop in the initial capacity, and the presence of long-lived isotope — to a decrease in specific capacity.

When using fuels characterized by low specific capacity P_{sp} the size of the generator is excessively increased, its efficiency is lowered, and its weight increased. An acceptable value is $P_{sp} > 0.1$ watt/cm^3. The half-life should be large or at least equal to the service life of the generator. With short half-life and significant service life, it becomes impossible to regulate heat flow to the thermal convertor. As a rule, the half-life of the radioisotope should not be less than 100 days or longer than several hundred years. Isotopes with half-lives greater than 100 years have fairly low specific characteristics ($P_{sp} < 0.1$ watt/cm^3). The possibility of obtaining the fuel in sufficient quantity at relatively low cost is also important.

Basic Types of Fuel

Fission products. Among the fission products, the most suitable for radioisotope thermoelectric generators with long service life (1-30 years), one may include Sr^{90} — β-radiation of mean energy of the β-spectrum $E_\beta \sim 1.2$ Mev. When it is possible to use local protection (water environment, natural cover) and not impose rigid weight and size restrictions, Cs^{137} may be used. Present in its decay chain are γ-quanta of energy $E_\gamma = 0.660$ Mev; mean energy of β-spectrum is $E_\beta = 0.25$ Mev.

For generators with shorter service life, Ce^{144} and Pm^{147} may be used. In the decay chain of Ce^{144} a series of β- and γ- transitions are observed (mean energy of decay $E_{\beta\gamma} = 1.4$ Mev). The presence in the spectrum of the daughter product Pr^{144} line $E_\gamma \sim 2.2$ Mev, necessitates the use of heavy shielding; however, in view of the high energy output, one may concentrate a significant part of the protection in the center of the installation and thereby decrease its weight. Pm^{147} compared to Ce^{144} has relatively low mean β-decay energy ($E_\beta = 0.062$ Mev) leading to a greatly reduced value of specific

energy output. At the same time, being a pure β-emitter, Pm^{147} has a comparatively low output of (braking) γ-radiation, permitting the use of shielding of relatively small thickness.

The basic specific characteristics of the fission products are given in Table 2.

a) Нуклид	b) Удельная активность, кюри/г	c) Удельное энерговыделение (мощность)			g) Период полураспада $T_{1/2}$
		d) вт/г	e) вт/см³	f) кюри/вт	
Sr⁹⁰	144	0,935	2,44	154	27,7 года h)
Cs¹³⁷	88,4	0,411	0,77	215	29,68 лет i)
Ce¹⁴⁴	3200	26,7	184,4	120	284,5 дня i)
Pm¹⁴⁷	918	0,338	—	2725	2,66 года h)

Table 2

Key: a) isotope
      b) specific activity
         (curie/g)
      c) specific power output
         (capacity)
      d) watt/g
      e) watt/cm³
      f) curie/watt
      g) half-life
      h) years
      i) days

Strontium-90 — a metal, having high radiotoxicity, may be used in the form of compounds which are inhert, stable and have acceptable physical properties (high heat conduction, high melting temperature). In Table 3 are given the most acceptable forms of the fuel and their characteristics (the metal is given for comparison).

The yield of Sr⁹⁰ as a fission product is ~ 5.8%; at the time of separation of Sr⁹⁰ , the preparation also contains Sr⁸⁹ (yield~4.8%, T$_{1/2}$ = 54 days), and stable Sr⁸⁸.

Cesium-137 — alkali metal, readily forms chemical compounds, the majority of which dissolve well in water. However, there is a series of compounds not having these undesirable qualities. The yield of Cs¹³⁷ as a fission product is 6.2%; separated with it is Cs¹³⁴ (T$_{1/2}$ ≈ 2.1 yr). Some characteristics of various cesium compounds are given in Table 4.

Table 3

a) Химическое соединение	c) Удельная мощность, вт/г	d) Температура плавления, С°	e) Плотность, г/см
b) Металл (чистый)	0,93	770	2,6
SrTiO₃	0,45	1910	5,1
SrO	0,78	2430	4,7
SrZrO₃	0,35	2700	5,4

Key: a) chemical compound
      b) metal (pure)
      c) specific capacity, watt/g
      d) melting temperature, $^{\circ}$C
      e) density, g/cm ^{3}

Table 4

a) Химическое соединение (топливо)	d) Удельная мощность, вт/г	e) Температура плавления, °С	f) Плотность, г/см³
b) Металл (чистый)	0,4	28,5	1,87
CsCl	0,3	646	3,97
CsF	0,21	864	3,58
CsO	0,24	600	3,77
c) Цезиевое полистекло (боросиликатное стекло с содержанием Cs 43 вес.%)	0,67	1100	3

Key: a) chemical compound (fuel)
      b) metal (pure)
      c) cesium polyglass (borosilicate glass with Cs content
         of 43 weight %)
      d) specific capacity, watt/g
      e) melting temperature,   C
      f) density, g/cm³

Cerium-144 — rare-earth metal, chemically highly active, pyro-
phoric, is an active reducing agent having hard γ -radiation. The
yield of Ce^{144} as fission product is equal to approximately 6%. The
corresponding characteristics of several cerium compounds are given
in Table 5.

Table 5

a) chemical compound	c) specific capacity, watt/g	d) melting temperature, °C	e) density, g/cm³
b) Металл (чистый)			6.7
CeO₂			7.13
CeCl₃			3.92
Ce₂O₃			6.8
CeF₃			5.8
CeB₆			—
CeN			—
CeS			5.9

Key: a) chemical compound
b) metal (pure)
c) specific capacity, watt/g
d) melting temperature, °C
e) density, g/cm³

Promethium-147 — a rare-earth element, β-emitter. Its yield as fission product is 2.7%. Table 6 gives its characteristics.

Table 6

a) chemical compound	c) specific capacity, watt/g	d) melting temperature, °C	e) density, g/cm³
b) Металл (чистый)	0.36	1357	1.1
Pm₂O₃	0.3?	1400	6.6
PmF	0.26	1650	--

Key: a) chemical compound
b) metal (pure)
c) specific capacity, watt/g
d) melting temperature, °C
e) density, g/cm³

Reactor isotopes. Among the large number of isotopes obtained by means of direct irradiation of stable isotopes in a nuclear reactor, the most interesting are Co⁶⁰ (for generators with service life up to 5 years), Tm¹⁷⁰ and Ir¹⁹² (for generators with service life to three or four months).

The basic characteristics of these isotopes without carrier are given in Table 7.

Table 7

Таблица 7

a) Изотоп	b) Удельная активность, кюри/г	c) Удельное энерговыделение (мощность)			g) Период полураспада T 1/2
		d) вт/г	e) вт/см³	f) кюри/вт	
Co⁶⁰	1140	17,5	152	65	5,21 года
Tm¹⁷⁰	9170	39,7	1335	151	55 дней
Po²¹⁰	4550	11,35	105,5	525	129 дней

Key: a) isotope
     b) specific activity
        (curie/g)
     c) specific power outout
        (capacity)
     d) watt/g
     e) watt/cm³
     f) curie/watt
     g) half-life
     h) years
     i) days

The specific eneroy outout achieved in oractice is several times less than the theoretically possible value qiven in the table.

Cobalt-60 — metal, obtained by irradiation of Co⁵⁹ in a reactor (Co_{27}^{59} + n → Co_{27}^{60} + γ). Its activation section is 30 barns. Co⁶⁰ gives hard γ-radiation (1.17, 1.33 Mev), mean soecific capacity is 1.6 watt/g (100 curies/g), density is equal to 8.7 g/cm³ corresponding to a volumetric specific capacity of 14 watt/cm³ .

Thulium-170 — a rare-earth element, obtained by irradiation of Tm¹⁶⁹ in a reactor; activation section is equal to aporoximately 130 barns; meltino temperature 1600°C.

Alpha-radioactive isotopes. When obtaining α -radioactive isotopes irradiated in a nuclear reactor, the taroets underco subsequent chemical treatment.

Obtaining α -radioactive isotopes in sufficient quantity presents a rather difficult oroblem. Therefore, when choosina one or ancther of the α-active isotopes for fuel, it is necessary to thorouqhly analyze the oossibility and economic exoediency of its oroduction in the necessary quantity.

Sufficient quantities of Po²¹⁰ , Pu²³⁸ , and Cm²⁴² can be obtained by irradiation of the startino materials in a reactor.

The specific characteristics of these isotopes are qiven in Table 8.

Table 8

a) Изотоп	b) Удельная активность, кюри/г	c) Удельное энерговыделение (мощность)			g) Период по- лураспада $T_{1/2}$
		d) вт/г	e) вт/см3	f) кюри/вт	
Po²¹⁰	4500	144,0	1340	31,4	138,5 дня i)
Pu²³⁸	17,45	0,58	9,25	30,3	86,4 года h)
Cm²⁴²	3320	122,5	1652	27,6	162,5 дня i)

Key: a) isotope
b) specific activity (curie/o)
c) specific power output (capacity)
d) watt/q
e) watt/cm ^{3}
f) curie/watt
o) half-life
h) years
i) days

Polonium-210 — obtained by irradiation of Bi^{209} in a reactor. It is a weak γ-emitter (10^{-6} γ-ouanta per decay). In Table 9 is oiven the characteristics of Po²¹⁰ and its compounds.

Table 9

a) Химическое соединение	c) Удельная мощность, вт/г	d) Температура плавления	e) Плотность, г/см3
b) Металл (чистый)	111,0	254	9,4
HgPo	72,0	—	11,1
PbPo	71,0	600	9,6

Key: a) chemical compound
b) metal (pure)
c) specivic capacity, watt/q
d) meltino temperature, C
e) density, o/cm ^{3}

Curium-242 — metal, obt-ined from Am^{241} recovered from fission products and then irradiated in a reactor (activation section of Am^{241} is aoproximately 750 oarns). The half-life of Cm^{242} is 0.447 year. The characteristics of Cm^{242} and several of its compounds are oiven in Table 10.

Table 10

a) Химическое соединение	c) Удельная мощность, вт/г	d) Температура плавления, °С	e) Плотность, г/см³
b) Металл (чистый)	121	950	13,5
Cm₂O₃	110	1700—2000	10,7
CmC	115	970	10
CmO₂	107	>1700	11,75
Cm—30Ni	85	—	9,05

Key: a) chemical compound
      b) metal (pure)
      c) specific capacity, watt/g
      d) melting temperature, °C
      e) density, g/cm³

Curium-244 may also be used as fuel. Its half-life $T_{1/2}$ = 18 years, specific capacity is 2.8 watt/g. Cm^{244} is obtained by irradiation of Am^{243}. It is a weak γ-emitter. It is at present obtainable in limited quantity.

Plutonium-238 — a metal, obtained by irradiation of Np^{237}. Its characteristics are given in Table 11.

Таблица 11

a) Химическое соединение	c) Удельная мощность, вт/г	d) Температура плавления, °С	e) Плотность, г/см³
b) Металл (чистый)			
Pu	0,58	640	16,5
Pu	0,54	1900	12,7
PuC	0,52	1650	13,6
PuO₂	0,49	2240	11,46
Pu₂O₃	0,50	2250	—

Key: a) chemical compound
      b) metal (pure)
      c) specific capacity, watt/g
      d) melting temperature, °C
      e) density, g/cm³

Actinium-227 ($T_{1/2} \approx 22$ yr; P_{spec} = 15 watt/g) is obtained by irradiation of Ra^{226} in a reactor. In its decay products are α-emitting isotopes. It is possible to use it in the form of Ac_2O_3 or AcF.

Thorium-228 ($T_{1/2} \approx 1.9$ yr., $P_{spec} = 170$ watt/g) may be obtained by irradiating Ra^{226}; it is also a daughter product in the decay of U^{232} and Ac^{228}.

Uranium-232 ($T_{1/2} \approx 74$ yr, $P_{spec} = 5$ watt/g) may be obtained by irradiating Th^{230} (ionium) or Pa^{231}. In its decay chain exist isotopes which are γ-emitters with hard radiation (Tl^{208}).

Estimation of General Resources of Radioactive Fuels

According to the estimates available at the present time, by the year 2000 there will be accumulated approximately 500 billion curies of radioactive wastes [4] which may be used for obtaining various fuels based on Sr^{90}, Cs^{137}, Ce^{144}, Pm^{147} and mixtures of fission products. Isotopes obtained by irradiation in a reactor (such as Pu^{238}, Po^{210}, Cm^{242}, Cm^{244}) may perhaps also be used. In Table 12 are presented the data of the AEC of the USA concerning the increase of production of eight of the above mentioned isotopes. In the upper line is shown the annual production for each of the isotopes (by year) in millions of curies, and in the lower — the corresponding thermal capacity in kilowatts.

Atomic Batteries

Development of radioisotope sources of electrical energy was begun in the 40's. The original sources had milli (micro) watt capacity and used various forms of interaction of radiation with matter. Transformation of the energy of radioactive decay of the mentioned sources is not related to a heat cycle. Such sources are usually called atomic batteries.

Among the atomic batteries known at the present time are primary, with direct collection of α - or β -particles emitted during radioactive decay (batteries with direct charge collection); secondary, where negative or positive charges are gathered which arise as a result of interaction of primary radiation with matter (batteries with p-n conversion, contact potential difference, with secondary electron emission); and tertiary, when electrical energy is obtained as a result of a double transformation (photoelectric batteries).

Such sources are being developed and at the present time they are used for supply of the grid current of various kinds of radiotechnical instruments, for charging dosimeters, for automatic charging of clocks, and for other uses, i.e. in those cases where high parameter stability with low power consumption is required.

Table 12

a) Исотоп	b) Единица измерения	g) Годовая производительность										i) Цена, долл/вт
		h) 1963 г.	h) 1964 г.	h) 1965 г.	h) 1966 г.	h) 1967 г.	h) 1968 г.	h) 1969 г.	h) 1970 г.	h) 1971 г.	h) 1972 г.	
Sr⁹⁰	c) Мкюри	3	5	5	5	10	10	10	10	10	10	77
	d) квт	19	32	32	32	63	63	63	63	63	63	
Cs¹³⁷	e) Мкюри	1	3,5	3,5	3,5	10	10	10	10	10	10	104
	d) квт	5	17	17	17	48	48	48	48	48	48	
Ce¹⁴⁴	c) Мкюри	3,5	3,5	3,5	3,5	100	100	100	100	100	100	5
	d) квт	25	25	25	25	700	700	700	700	700	700	
Pm¹⁴⁷	c) Мкюри	0,02	0,5	0,5	0,5	30	30	30	30	30	30	485
	d) квт	0,01	0,2	0,2	0,2	14	14	14	14	14	14	
Po²¹⁰	e) г	20	50	100	100	1000	1000	1000	1000	1000	1000	190
	d) квт	3	7	14	14	140	140	140	140	140	140	
Pu²³⁸	f) кг	3	3	6	9	11,5	15	17	20	23	25	1040
	d) квт	1,5	6	13,5	19	24	32	36	42	47	51	
Cm²⁴²	e) г	0	12	80	1000	1000	1000	1000	1000	1000	1000	165
	d) квт	0	1,5	9,5	120	120	120	120	120	120	120	
Cm²⁴⁴	f) кг	0	0,05	0,1	6	18	40	56	56		58	435
	d) квт	0	0,1	0,2	14	41	92	129	134	134	134	

Key: a) isotope   f) kilograms
      b) unit measured   g) production year
      c) millions of curies   h) year
      d) kilowatts   i) cost, dollars/watt
      e) grams

Batteries with Direct Charge Collection

If one takes two plates and on one of them (the emitter) deposites radioactive material, the radiated particles, as they accumulate on the plate opposite (the collector), charge it according to the sign of the charged particles.

In the case of a β-emitter, the plate is charged negatively: in the case of an α-emitter — positively. Beta-emitters are most frequently used.

Calculations show that for β-sources, the specific capacity, on the average, is several microwatts pere millicurie. Since the activity of the source does not exceed several curies, the output capacity is only several milliwatts. The output voltage of such sources is dependent on the energy of the β-particles as well as on the resistance of the insulation between the emitter and collector.

The first battery with direct collection of charge was made by Mosley in 1913. Using 20 microcuries of radium, he obtained a current of 10^{-11} amps under a voltage of 150,000 volts.

At the present time in some countries, industrial models of nuclear batteries are manufactured; various models of the battery based on Sr^{90} are marketed. battery D-50 (USA) contains a Sr^{90} source of 10 micro-curies activity and polystyrene insulator of $\sim$ 0.8mm thickness. The battery is placed in a lead container. The volume of the battery is approximately 16.5 cm^3, voltage 7000 v, current (short circuit) $4 \cdot 10^{-11}$ amp.

The actual parameters achieved by a battery with direct charge collection lie within the limits: voltage 1-100 kv; current 10^{-8} - 10^{-12} amp.

Batteries Based on Semiconductor Junctions

Batteries based on semiconductor junctions consist of a radiation source (β - or γ -emitter) and a semiconductor with a p-n junction. It is possible to use a γ -emitter, since γ -quanta, in the process of interacting with matter, knock out electrons from the crystal lattice of the semiconductor, forming numerous pairs of charge carriers — electrons (-) and holes (+). Thus, a kind of amplification of the primary charge of the β -particles occurs, reaching values on the order of 10^5 or the transformation of energy of the γ -quanta into energy of the electron-hole pair. Such sources may have relatively low voltage, but higher current than in atomic batteries with direct charge collection. The capacity of such batteries is limited by the radiation stability of the semiconductor junction.

Therefore, as an emitter, it is advantageous to select a soft radiation source (for example Pm^{147}). Experimental models have a capacity of 1 microwatt, voltage — a fraction of a volt, efficiency $\sim$ 1%.

Batteries Based on Contact Potential Difference

In every metal, there are so-called free electrons (electron gas), the concentration of which is approximately 10^{22} per cm^3. In order that the electrons can escape the metal, a necessary definite energy, called the work function P_{exit} , exists. The difference in the work functions for a given pair of metals is called the contact potential difference (this value is different for various metals). If between two unlike metals (electrodes) there is ionized gas, then (under the action of the contact potential difference), when the circuit connecting the electrodes is closed, current begins to flow. In such set ups may be used either self-ionizing radioactive gas, for example Kr^{85} or T, or a gas ionized by a special source. Since one β -particle may form 100 ion pairs, the current of such a battery is approximately 100 times higher than in a battery with direct charge collection. The e.m.f. is equal to the difference of the work functions of the electrodes (approximately several volts).

-19-

The strength of the current depends on the ion concentration which, in turn, is dependent on the activity of the emitter, the energy of the β-particles, the nature of the ionized gas, etc.

Photoelectric Batteries

In recent years a large number of materials have been discovered which luminesce intensively under the action of ionizing radiation. It is also known that under the action of light radiation on the so-called photoelements, electric current is formed. A combination of radioisotopic phosphorus and a photoelement allows one to make a photoelectric battery.

For obtaining a conversion of light energy into electrical energy for a photoelectric battery, a spectrum of the luminophor is selected in the range of maximum spectral sensitivity of the photoelements. The actual capacity of such an installation is on the order of tens of microwatts, efficiency 1-2%, voltage, several volts.

Batteries with Secondary Electron Emission

If a stream of, for example, β-particles is sent toward an electrode having a sufficient coefficient of secondary emission, then a stream of secondary charged particles is formed. By using this effect, one may make small-sized sources of electrical energy. Batteries with direct charge collection may be used to create accelerating voltages in subsequent cascades. The e.m.f. of single cascade batteries is approximately equal to the energy of the secondary electrons.

Radioisotope Electrogenerators

Turboelectric Generators

Although radioisotope turboelectric generators have not as yet found practical application, they are of interest for making generators with capacities greater than 1 watt.

The principal scheme of radioisotope turboelectric generators is presented in Fig. 1. The generator consists of a radioisotope thermal block 1, where the operating medium is heated; a system 2 to transfer this medium to the turbine 3; refrigerator 4; and electro-generator (dynamo) 5. As operating medium, a liquid metal (Rankin cycle) or gas (Brayton cycle) may be used.
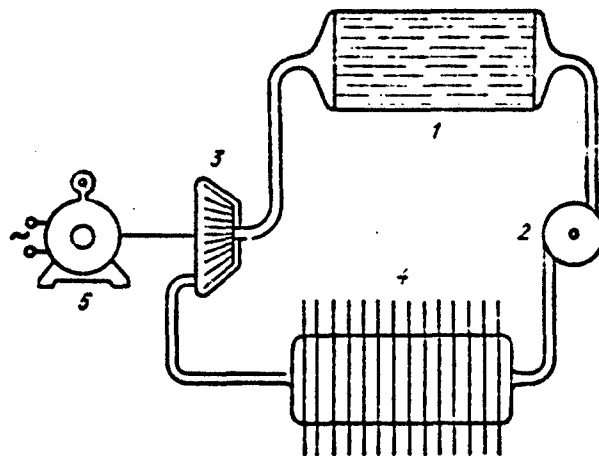
Fig. 1. Basic scheme of radioisotope turboelectric generator

The operating reliability of the turbine, generator, and pump plays a role of no small importance. However, even when achieving a relatively long service life, their reliability, because of the presence of rotary elements, will always be less than the reliability of a thermoelectric system. In view of the higher efficiency of turbo-electric generators in comparison to thermoelectric generators (at high capacity), turboelectric generators may in the long run be used, especially in those cases where both electrical and mechanical energy are required. Calculations show that the general efficiency is ~ 15%.

Thermoemissive Generators

There are two most usual forms of thermoemissive convertors (TEC): vacuum and plasma diode. As a source of thermal energy, an isotope with a large specific energy output is usually used ($Cm^{...}$, $Po^{...}$, etc.) in order to achieve high temperature. The scheme of the generator is presented in Figure 2. The heated cathode emitts electrons which cross the narrow interelectrode gap and strike the relatively cold anode. If the cathode and anode, having different work functions, are connected across the load, then, because of the potential difference which developes, an electric current will flow through it. With respect to the external circuit, the cathode will be the positive terminal of the thermoemissive

generator and the anode, the negative one. The thermal energy, arriving
on the cathode, is spend in surmounting the work function of the
electrons from the metal. In addition, there are energy losses due
to radiation, convection, and heat conduction. The energy lost by
the cathode is in general received by the anode. To protect the anode
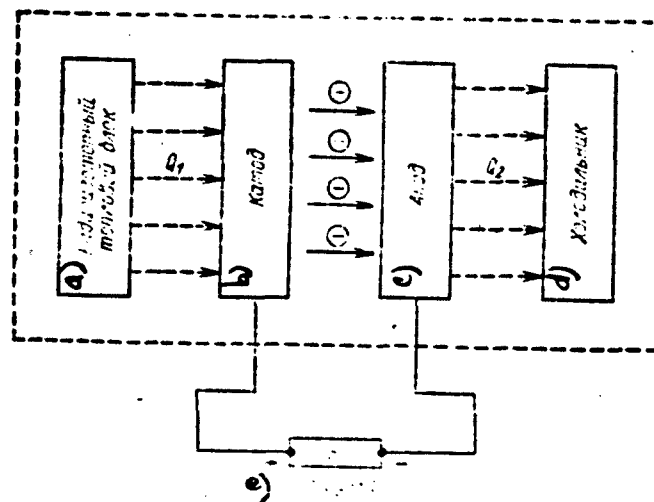from overheating, it is necessary to cool it.

Fig. 2. Basic scheme of radioisotope thermoemissive generator

Key: a) radioisotope thermal block
      b) cathode
      c) anode
      d) refrigerator
      e) load

   The most serious difficulty in making a TEC is the development
of a space charge in the anode-cathode gap, leading to restricted
current. In order to reduce the volumetric space charge, the goal
is to make the smallest possible gap between the cathode and anode
(d ~ 2·10^{-4} cm in a vacuum diode) or to use a pair of materials with
low ionization potential (e.g. caesium) in a plasma diode.

   The basic parameters of thermoemissive generators are the efficiency
and the specific capacity obtained through the convertor (watt/cm^2).
In the case of a gas-filled diode this value is a function of the

cathode (emitter) temperature T_C , the anode (collector) temperature T_A, vapor pressure p, the size of the interelectrode gap d, and the work functions of the anode and cathode.

Experiment has shown that the specific capacity for thermoemissive generators is equal to approximately 5-10 watts/cm^2, and an efficiency for a capacity of ~ 10 watts is approximately 10% (calculated efficiency ~ 30%).

It must be noted that the creation of thermoemissive generators entails serious technical difficulties combined with the selection of materials stable in relation to high temperature (T_C ~ 2000 K, T_A ~ 1500 K).

Thermoelectric Generators

In the thermoelectric method of conversion, the thermal energy evolved in the radioisotope block passes through a thermoconvertor to the refrigerator (Fig. 3). It is known that because of the development of a temperature drop along the thermoelements, there is a movement of positive charges (holes) toward the cold junction in the p-branch and of negative charges (electrons) in the n-branch [10-12] . As a result, a potential difference is created, proportional to the temperature gradient, and part of the heat flow is transformed into electrical energy.

In the general case, thermoelectric generators are heat sources on whose surface are installed thermoconvertor elements and structural connectors. The remaining surface is surrounded by insulation . The elements of the convertor are connected with the structural assembly of the generator, which dissipates heat to the surroundings. The output voltage in such sources is proportional to the temperature drop and the number of consecutively connected pairs of thermoelements. For the better semiconductor materials, the thermal e.m.f. α is considerably higher than for metals, and is equal to 1 millivolt/deg.

Thermoelectric materials of the p- and n-branches are characterized by three values: thermoelectromotive force α, specific electrical conductivity σ , and heat conductivity $\varkappa$.

The basic factor determining the efficiency of the convertor material is its quality factor z. For the case of a thermocouple consisting of a positive p- and negative n-branch, at optimal matching sections [10

$$ z = \frac{(\alpha_p - \alpha_n)^2}{\left[\left(\frac{\varkappa_n}{\sigma_n}\right)^{1/2} - \left(\frac{\varkappa_p}{\sigma_p}\right)^{1/2}\right]^2} . $$

-23-

In accordance with the general theory of thermoelectric convertors
[11, 12] the efficiency of the thermoelement is

$$\eta = \frac{T_{rop} - T_{xas}}{T_{rop}} \cdot \frac{\sqrt{1 + \frac{1}{2} z (T_{rop} + T_{xas})} - 1}{\sqrt{1 + \frac{1}{2} z (T_{rop} + T_{xas})} + \frac{T_{xas}}{T_{rop}}} ,$$

where T_{rop} and T_{xas} are the temperatures of the hot and cold junctions,
respectively. The first factor determines the efficiency of the ideal
heat engine (Carnot cycle), and the second, the actual thermoelectric
efficiency characterizing the reduction of the efficiency of the Carnot
cycle through irreversible losses.

Thermoconvertors from semiconductor materials of constant composition
with temperature difference in its junctions of $T \leq 300°$ have, on
the average, an efficiency of approximately 3-8%. Considerably higher
efficiency may be obtained if the optimal value of z at the given
temperature is achieved along the branches of the thermoelements.
The theoretically possible efficiency in the region from 250 to
1300 K is ~30%.

In accordance with the temperature dependence of z, all thermo-
electric materials are usually divided into low temperature (operating
in the interval from 200-600 K), medium temperature (600-1000 K),
and high temperature (1000-1200 K). Low temperature thermoelements
(for example, based on the triple junction Bi-Te-Se) have high service
life (greater than 50,000 hrs) and stability characteristics. Their
efficiency in the temperature interval 250 -600 K is approximately
equal to 6%. Medium temperature elements based on Pb-Te have some
worse characteristics: efficiency 4-5%, service life about 1 yr.
High temperature convertors based on Si-Ge have a 3% efficiency and a
service life greater than 1 year.

At the present time the technology has been developed to obtain
the necessary quantity of thermoelectric materials and to make models
of cascade batteries operating in the temperature range from 250 -
1300 K. The selection of one material or another (temperature range)
and making a specific structure of a thermoelectric convertor depends on
the operating conditions of the radioisotope thermoelectric generator,
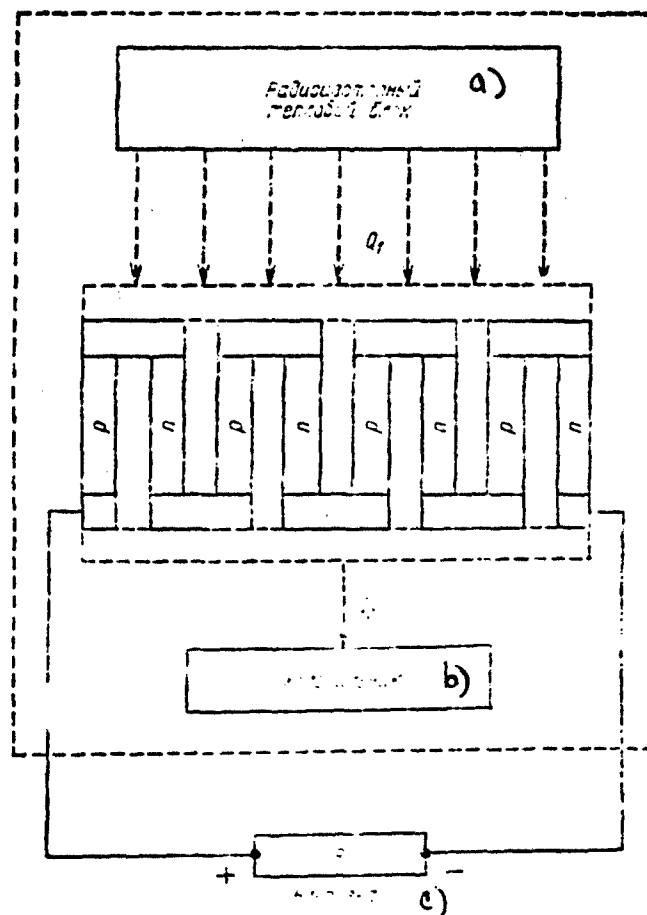the parameters of the heat source, and the necessary period of service.

-24-

Fig. 3. Basic scheme of radioisotope thermoelectric generator

Key: a) radioisotope thermal block
b) refrigerator
c) load

Existing experience of development and operation of radioisotope thermoelectric generators with capacities from one to several hundred watts has shown that the most (practically) acceptable conversion method at the present time is the thermoelectric one, which combines reliability, acceptable efficiency, and long service life. The accumulated service time of thermoelectric convertors is calculated at tens of thousands of hours. This condition is very important in the majority of cases, since the greatest advantages of radioisotope sources of electrical energy become apparent in their use in self-contained, long acting assemblies.

Radioisotope Thermal Blocks

Isotope Selection

The choice of isotope is to a large extent determined by the intended use of the generator. For operation in space, it is most expedient to use α-emitters having high energy output and not requiring heavy shielding. Among them the first are Po^{210}, Pu^{238}, Cm^{242} and Cm^{244}. For ground and underwater applications, γ- and β-emitters such as Sr^{90}, Cs^{137}, Ce^{144}, Pm^{147}, Co^{60} and others are widely used.

The energy output of the β-emitters is lower than that of α-emitters, but the former are readily available and cheaper.

The energy characteristics of radioactive isotopes may be calculated from the following expressions.

The specific activity C_{sp} of the pure radioactive isotope is found by the ratio

$$C = \frac{1.305 \cdot 10^8}{T_{1/2} A} \text{ (curie/g)}$$

where $T_{1/2}$ = half-life (in days); A = atomic weight.

If the isotope is used in a chemically combined state, then

$$C = \frac{1.305 \cdot 10^8 K}{T_{1/2} M} \text{ (curie/g)}$$

where K = number of atoms of the radioactive isotope in a molecule of the compound; M = molecular weight of the chemical compound.

The specific heating capacity of the radioactive isotope is calculated according to the formula

$$P_{\gamma \rho} \ (\text{watt/g}) = 7.5 \cdot 10^{5} \ \frac{KE_{tot}}{T_{\frac{1}{2}}M}$$

where E_{tot} = total energy of the act of decay, taking into account radiation leakage, Mev.

The specific heating capacity $P_{\gamma \rho}$ may be determined from a known value of C by the relationship:

$$P_{\rho} \ (\text{watt/g}) = 5.93 \cdot 10^{-3} \ C(\text{curie/g}) \ E_{\infty}(\text{Mev})$$

The volumetric heat output of the pure radioactive isotope is equal to

$$P_{\rho \gamma} \ (\text{watt/cm}^{3}) = P_{\rho} \ \rho$$

where ρ = density of isotope, g/cm^{3}

For chemical compounds

$$P_{\gamma \gamma} = P_{\rho} \rho \ \frac{AK}{M} \ (\text{watt/cm}^{3})$$

The activity per unit of heating capacity C may be calculated according to the formula:

$$C_{\gamma} = \frac{C}{P_{\gamma}} = \frac{1}{5.93 \cdot 10^{-3} E_{tot}} = \frac{169}{E_{tot}} \ (\text{curie/watt})$$

Structure of Ampoule

The requirements of the ampoule (Fig. 4) in which the radioactive preparation is to be placed are usually determined by the necessity that the integrity and savety of the ampoule be insured over a period of ten half-lives of the basic isotope and also under conditions of damage during operation and shipping. When using radioisotope blocks in generators for space uses, damage may also arise during launch and re-entry.

The ampoule material and sealing techniques should provide high corrosion resistance in relation to the surrounding air and ground water and to sea water. The ampule material should be compatible with the radioactive preparation and its decay products and also with those structural elements with which it is in direct contact.

In accordance with the "Sanitation Regulations" (see Appendix) the structure and manufacturing technology of low temperature units should insure its integrity and tightness with a high margin

-27-

of safety (by a factor of 10) in rela⁻ion to impact ($>$ 3000 g over
a period of 0.5 μ/sec) and oscillation (20-60 Hz with acceleration
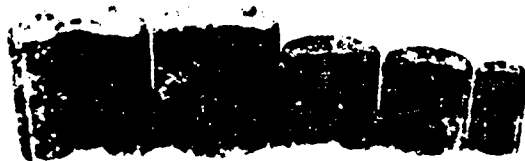$>$ 10 g in a period of 6 hrs)* load, internal and external pressure.

Fig. 4.   Radioisotope blocks based on Sr^{90}.

The ampoule material should be stable to heat (to $1100^\circ C$) and
radiation (loss of strength during the time of operation should not
exceed 10% of the original). Radiation stability should be insured
both in relation to primary radiation (α, β , γ , n) and to secondary
(braking and neutron) radiation coming from the radioactive preparation
or from the walls of the ampoule.

Radiation contamination should be minimal ($\leq$ 1 μ curie for β -
and 5×10^{-2} μ curie for α -radioactive isotopes) from all surfaces if
the block. The radiation output of the radioisotope block is re-
stricted to 300 rad/hr at a distance of 1 m. If the dose rate from
the radioisotope block exceeds the given value, this block should be
placed in protective shielding which satisfies all the requirements
with respect to shielding of the radioisotope block.

Determination of Thermal Output

Calculation of the thermal output in the isotope block involves
calculation of that portion of the energy released during radioactive
decay which is absorbed by the preparation itself and by the ampoule
walls. Since the paths of α-particles and fission fragments in
the radioisotope fuel consists of fractions of a millimeter, all the
kinetic energy is absorbed within the fuel itself and in the walls
of the isotope block; i.e. the energy absorbed within th radioisotope
block is equal to the total energy released, and the specific thermal
output may be taken, with sufficient accuracy, as constant with respect
to volume.

---

* Within the parentheses are shown the values specified in the
"Sanitation Regulations".

An analogous statement may also be made in relation to the thermal
output from β-particles, the energy of which is almost fully absorbed
within the fuel material and ampoule walls. Estimation of energy losses
with braking radiation shows that this effect can be disregarded, since
it does not exceed 1%.

The most difficult problem is the calculation of the thermal
output through absorption of γ-radiation which has rather signifi-
cant penetrating ability. For more complete collection of energy,
radioisotope blocks based on γ-emitters are usually enclosed in a
casing material with good absorption properties (wolfram, depleted
uranium). The combination of the radioisotope block with the pro-
tective (absorbing) casing is called the thermal block.

Table 13

a) Материал обшивки	e) плотность г/см³	f) Толщина поглотителя (см), обеспечивающая кратность ослабления K=100 и K=50 при различных энергиях излучения										
		0.5 Мэв g)		0.75 Мэв g)		1.0 Мэв g)		1.5 Мэв g)		2.0 Мэв g)		3.0 Мэв g)
		K=100	K=50	K=100	K=50	K=100	K=50	K=100	K=50	K=100	K=50	K=100	K=50
b) Уран	18.7	0.9	0.13	1.5	1.	3.7	3.2	2.1	1.4	6.3	5.4	7.0	6.0
c) Вольфрам	19.3	1.1	1.1	2.3	2.0	5.0	1.3	7.1	6.1	7.7	6.6	8.9	7.6
d) Свинец	11.34	1.6	1.1	3.0	2.5	7.0	6.0	7.6	8.2	11.3	9.6	12.2	10.6

Key: a) casing material   e) density, g/cm³
       b) uranium   f) absorber thickness (cm)
       c) wolfram   providing an attenuation factor
       d) lead   K = 100 and K = 50 for various
                                      energies of radiation
                                   g) Mev

In Table 13 are given the calculated values of casing thickness
of the thermal block from various materials (uranium, wolfram, lead)
providing absorption of 99% (K = 100) and 98% (K = 50) of the original
energy released in the form of γ-radiation. Here K = attenuation
factor, i.e. the value determining by how many times the radiation
is reduced in the presence of absorbers.

Calorimetry is a direct method which allows direct measurement of the thermal output in the radioisotope block. By this method it is possible to calculate the activity of the block for a given content of ingredients by using known values of the energy of one decay act. and measuring the thermal output.

The design of calorimeters for measurement of thermal output in a block for α - and for β -emitters may be essentially the same, since the penetrating ability of their radiation is not high (Fig. 5). Calorimeters for measuring the thermal output of radioisotope blocks based on γ -emitters are usually made with thick walls for nearly com- plete absorption of the γ -radiation, since calculation of the unabsorbed part of the radiation is difficult, especially for complex geometry and heterogeneity of the absorbing material.
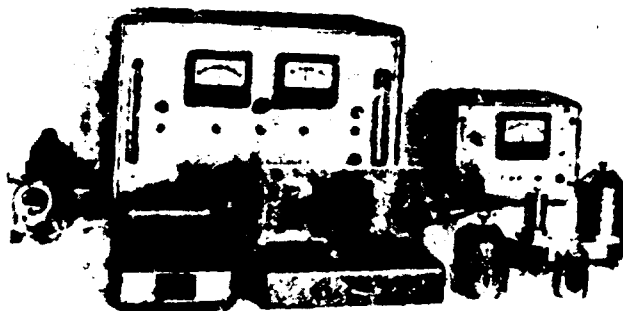
Fig. 5 Calorimetry set up

Thermoelectric Convertors

Basic Requirements

In order to insure reliable and efficient operation of the con- vertor, the thermoelement should meet the following basic requirements:
1. High quality factor $z = \alpha^2 \sigma / \lambda$, i.e. high electrical conduc- tivity σ and thermo-e.m.f. coefficient α for the lowest possible thermal conductivity λ .
2. Sufficiently high mechanical strength, stability toward thermal shock, and radiation and chemical interactions.
3. The possibility of using simple manufacturing technology for the branches and its coupling to the block by switching which is reliable for the given temperature level and has minimal D.C. resistance and also

-30-

BEST AVAILABLE COPY

metallurgical and chemical compatability with the material of the
thermoelements.

Since the temperature behavior of α , σ , $\varkappa$ depends essentially
on the material chosen, the maximum value of z for various materials
lies within various temperature ranges. Fig. 6 shows the temperature
dependence of the quality factor z for several semiconducting materials
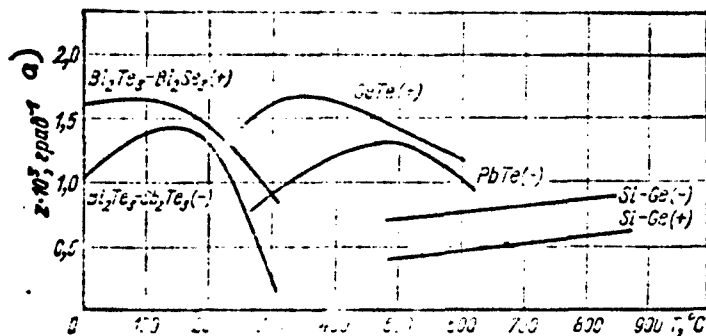of the p- and n-types.

Fig. 6.   Temperature dependence of quality factor of some thermo-
          electric materials   p(+) —positive branch; n(-) —
          negative branch

          Key: a) deg^{-1}

The level of operating temperature influences not only the choice
of thermoelectric material, but also the design of the generator, which,
in turn, is also determined by its intended use.

Low Temperature Convertors

In the temperature interval (200 -600°K) the following materials
are most efficient and technologically feasible to manufacture:

$$Bi_2Te_3, \quad Sb_2Te_3, \quad Bi_2Se_3, \quad SbZn, \quad Bi_2Se_3 - Bi_2Te_3,$$
$$Bi_2Te_3 - Sb_2Te_3.$$

Solid solutions of $Bi_2Ti_3 - Sb_2Te_3$ (p-type) and $Bi_2Te_3 - Bi_2Se_3$
(n-type) have been found most practical for manufacture of thermoelements.

It is apparent that low temperature limits are determined by
the possibility of heat dissipation from the cold junction. The
operating temperature T_{hot} of the thermopile is not high, essentially
lowering the negative influences of such factors as oxidation, diffusion,
vaporization of the alloying and basic materials.

Experience of application in domestic radioisotope thermoelectric
generators of convertors based on ternery alloys of Bi-Te in the tem-
perature interval from 200 -600°K has shown that they have a long
service life (no less than 3-5 years), stable characteristics during
that time, and an efficiency where T_{hot} - T_{cold} ≥ 250° up to 6%.
When the temperature interval is raised, the efficiency is slightly
increased (to ∼7%). Low operating temperatures of such convertors
make it easier to creat directed heat flow, i.e. allows one to make
generators with small heat losses (10-20%) with acceptable dimensions
of heat insulation, leading to lower weight and size characteristics.

The temperature dependence of the parameters σ, τ, and $\varkappa$ for
ternery alloys based on Bi-Te are given in Fig. 7. Table 14 gives
the parametric values, averaged with respect to temperature, in the
interval from 20 to 270°C.
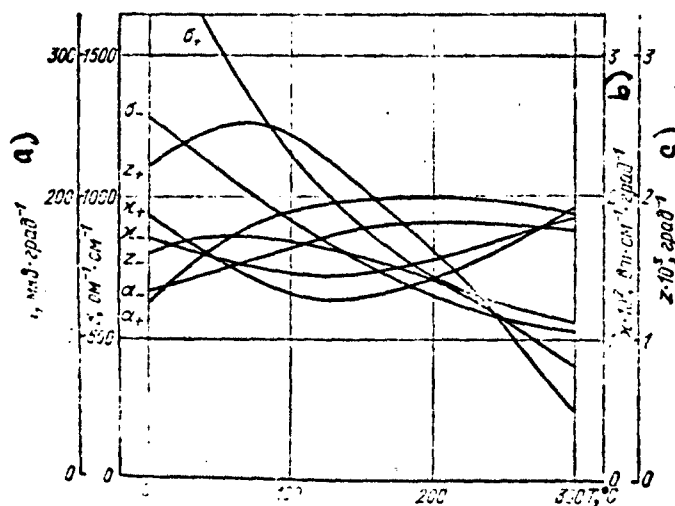
Fig. 7. Temperature dependence of σ, τ, and $\varkappa$ for Bi_2Te_3-
        Sb_2Te_3 (p-branch) and Bi_2Te_3 - Bi_2Se_3 (n-branch).

      Key: a) microvolt·deg^{-1}
            b) watt cm^{-1} deg^{-1}
            c) deg^{-1}

-32-

Table 14

Таблица 14

$\bar{x}_p = \bar{x}_n$	$\bar{c}_p = \bar{c}_n$	$\bar{x}_p = \bar{x}_n$
180 мкв/град a)	740 ом $^{-1}$·см $^{-1}$ b)	$1.6 \cdot 10^{-2}$ вт/(см·град) c)

Key: a) microvolt/deg
       b) ohm $^{-1}$ cm $^{-1}$
       c) watt/(cm·deg)

Medium Temperature Convertors

In the temperature interval from 600 -1000°K the basic thermo-electric materials are currently Pb-Te, Ge-Te, Pb-Se, Ag-SbTe$_2$ and solid solutions based on them. Elements manufactured on a Pb-Te basis have found practical application in thermoelectric generators. However, alloys of Pb-Te are subject to plastic deformation, especially at high temperatures, and have insufficient mechanical strength. Its service life is less than that of the low temperature convertors. If sublimation properties are not decisive in the selection of low temperature materials, then during transition to medium temperature materials this question will become highly significant. Therefore, in the majority of instances, thermoelements of these materials must be specially coated. The efficiency of medium temperature materials based on Pb-Te in the stated, higher temperature interval is ~ 4-5%; service life — on the order of a year. In Fig. 8 the basic characteristics of medium temperature thermoelectric materials are shown.

High Temperature Convertors

The operating condition of high temperature elements (T$_{out}$ ⊆ 1300 K) is incomparably more difficult than that of medium and low temperatures. Questions of sublimation, oxidation and diffusion here play a highly important role. Problems of structural mounting of the thermopile, commutators and electrical insulation are significantly complicated. The efficiency of the convertor (from calculations at 100°C) of the high temperature materials currently known is approximately 1.5 times less than the efficiency of medium temperature convertors and two times less than that of low temperature ones. Of the high temperature materials which have been investigated up to the present time, silicon-germanium (Si-Ge) alloys are most widely used. They have sufficiently high mechanical properties. The most efficient temperature region lies

-33-

in the interval 900 -1300^{0}K ($\sim$3% efficiency); in the interval 600 -1300^{0}K, the efficiency reaches 5-6%. Fig. 9 oives the basic characteristics of hioh temperature materials.
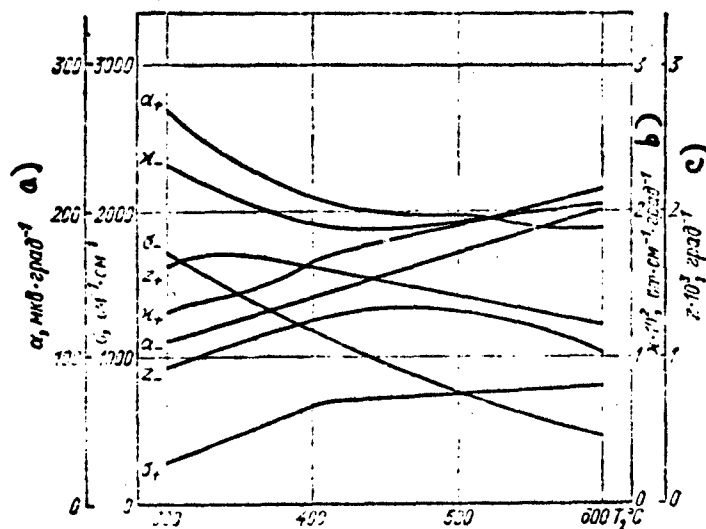

Fio. 8. Temoerature deoendence of α , σ , $\varkappa$
for Ge-Te (p-oranch) and Pb-Te
(n-branch).

Key: a) microvolt deo^{-1}
      b) watt cm^{-1} deo^{-1}
      c) deo^{-1}
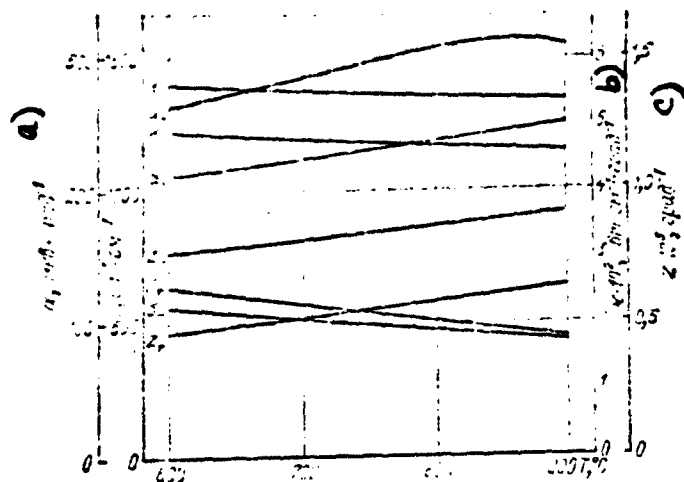
Fig. 9. Temperature dependence of $\sim$, τ, $\varkappa$ for n- and p-branches based on Si-Ge.

Key: a) microvolt deg⁻¹
     b) watt cm⁻¹ deg⁻¹
     c) deg⁻¹ —

Key: a) microvolt deg^{-1}
     b) watt cm^{-1} deg^{-1}
     c) deg^{-1} —

Calculation of Parameters of
Radioisotope Thermoelectric Generators

Selection of Heat Circuit

From the standpoint of the heat circuit, the radioisotope thermo-
electric generator is a system with an internal source of thermal out-
put and a thermal field in which is determined the basic processes of
heat transfer through heat conduction or radiation within the system,
and convective (radiant) or contact heat exchange with the surroundings.

The fundamental problem in the selection of a heat circuit is
to provide maximum heat flow to the thermoconvertor and the creation in
it of a temperature difference corresponding to the optimal temperature
range. In this, special consideration is given to making equal heat
flow through the elements of the thermoelectric convertor and to de-
creasing the resistance along its route.

Development of the heat circuit consists of analysis of the
distribution of the heat flows and temperature fields inside the

generator stemming from the operation of the convertor, conditions of
hea. exchange with the surroundings, thermophysical properties of the
materials, geometry and relative positions of the generator elements ,
its capacity and intended use.

Radioactive isotopes may be divided according to power density
into isotopes with lower power density (less than 1 watt/cm^3) (in
this group are Cs^{137}, Sr^{90}), those with medium density (1-10 watt/cm^3)
(Ce^{144}, Pu^{238}, Co^{60}), and high density (greater than 10 watts/cm^3)
(Cm^{244}, Po^{210}).

Isotopes with low and medium power density are advantageously
used in generators with heat circuits based on low and medium temperature
thermoelectric convertors, since for considerable increase in temperature
it is necessary to significantly increase the thickness of the heat
insulation, leading to worsening of the weight and size indices of
the generator.

The selection of the heat circuit of the generator is also dependent
on the radiation characteristics of the fuel and allowable radiation
level during assembly and operation of the generator. For isotopes
with intense γ- and neutron radiation, considerable biological protection
is required to reduce the radiation intensity to an admissible level.
The distribution of the biological shielding (which constitutes the
greater part of the generator's weight) in relation to the radioisotope
block has a determining influence on the heat circuit of the generator
and its overall weight and size characteristics. Of the operating
factors, the greatest influence on the generator's heat circuit is
shown by the value of dynamic loads which arise during shipping, assembly
and operation of the generator, and also the conditions of heat exchange
with the surroundings.

Under significant dynamic load and limited supporting power of
the thermoelectric convertor, structural connections are envisaged
in the generator in order to sustain the mechanical load and insure
the integrity of the thermoelements. The structural connections are
sources of heat loss; these losses may considerable exceed or be com-
mensurate with losses through the thermal insulation. Therefore, one
of the fundamental construction problems is the maximal shortening
of these connections. The best method is the direct fastening of the
radioisotope block to the thermoconvertor. From the standpoint of
the heat exchange with the surroundings, there are two variations of
the heat circuit: with contact and with convective (radiant) heat
exchange. In the first case the problem amounts to making good heat
contact between the generator and the surroundings (the surrounding
ground, water); in the second it is necessary to make an extended
surface of the heat exhaust (to the air, to space).

Heat Calculations

Correct calculation of the heat circuit of the thermoelectric generator is quite a complex problem, since it is practically impossible to consider all the factors: specific geometrical units, structural connections, temperature changes, heat conduction, actual values of thermal contact resistances, and a whole series of other factors influencing the selection of the limiting conditions [13, 14].

The aim of the thermophysical calculations is the determination of the temperature field and distribution of heat flows within the elements and units of the generator.

In the general case the heat transfer processes are represented as differential equations of heat conduction which establish the connection between time and spatial changes in temperature:

$$\frac{\partial}{\partial x}\left(\lambda\frac{\partial t}{\partial x}\right) + \frac{\partial}{\partial y}\left(\lambda\frac{\partial t}{\partial y}\right) + \frac{\partial}{\partial z}\left(\lambda\frac{\partial t}{\partial z}\right) + Q = c\rho\frac{\partial t}{\partial \tau}.$$

where λ = coefficient of thermal conduction , t = temperature, ρ = density, c = specific heat capacity, τ = time, Q = density of heat sources.

In the general case the values of λ, ρ, c and Q are functions of the coordinates, time, and temperature. During engineering estimations this equation can be considerably simplified. For example, if the thermal conduction coefficient is taken as constant, then the general equation thermal conduction takes the following form

$$\frac{\partial^2 t}{\partial x^2} + \frac{\partial^2 t}{\partial y^2} + \frac{\partial^2 t}{\partial z^2} + \frac{Q}{\lambda} = \frac{c}{\lambda}\cdot\frac{\partial t}{\partial \tau}.$$

For fixed conditions, i.e., when t is a function of the coordinates only and does not depend on time

$$\frac{Q}{\lambda} + \frac{\partial^2 t}{\partial x^2} + \frac{\partial^2 t}{\partial y^2} + \frac{\partial^2 t}{\partial z^2} = 0.$$

Finally, for the fixed problem, the calculation of the units in the absence of sources and flows of heat

$$\frac{\partial^2 t}{\partial x^2} + \frac{\partial^2 t}{\partial y^2} + \frac{\partial^2 t}{\partial z^2} = 0.$$

-37-

In essence, thermophysical calculations amount to solution of one of the differential equations given above. The basic difficulty, as discussed above, is the sufficiently correct determination of the limiting and initial conditions. In this process one obtains temperature fields, the knowledge of which allows one to find the distribution of heat flows within the separate elements of the generator design and the value of the thermal efficiency of the generator as a whole.

The heat circuit of the generator may also be presented in the form of a dual circuit (Fig. 10).
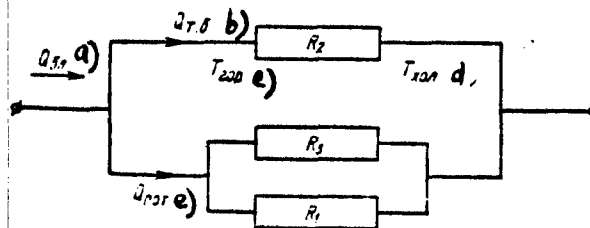
Fig. 10.   Dual circuit of radioisotope thermoelectric generator.

Key: a) radioisotope block
      b) thermal block
      c) $T_{гор}$
      d) $T_{хол}$
      e) Q

The symbols $Q_{б1}$ = complete heat output of the radioisotope block; $Q_{т.б}$ = heat flow arriving at the convertor; $Q_{пот}$ = heat losses; R_1, R_2, R_3 = thermal resistances of heat insulation, thermopile, and structural elements, respectively. Calculation is carried out under the given assumptions. It is assumed that the temperature on the surface of the thermal block and housing is constant. In accordance with the dual circuit, the thermal efficiency of the radioisotope thermoelectric generator may be determined from the following ratios

$$\eta = \frac{Q_{т.б}}{Q_{б1}} = \frac{Q_{т.б}}{Q_{т.б} + Q_{пот}}, \quad Q_{шт} = \frac{Q_{т.б} R_2 (R_1 + P_3)}{R_1 R_3},$$

-38-

from which

$$\tau_1 = \frac{R_1 R_1}{R_1 R_3 + R_2 R_1 + R_1 R_2}.$$

Calculation of Thermoelectric Convertor

On the basis of the adopted values of T_{hot} and T_{cold} according to the known properties of the chosen thermoelectric materials and given electrical output parameters, the calculation of the thermoconvertor is carried out. As a basis for the calculation, one may use the method of "average parameters" proposed by A.F. Ioffe [10, 11]. The efficiency of the convertor is determined according to the formula given on p.26. The heat flow $Q_{n.t.}$ to the convertor, necessary for obtaining the given electrical capacity Q_{el} is determined by the equation

$$Q_{n.t.}^{т. б} = \frac{Q_{эл}}{\tau_1}.$$

Proceeding from the given output voltage U, the e.m.f. E is determined

$$E = U \frac{M + 1}{M},$$

where

$$M = \sqrt{1 + z\bar{T}}, \qquad \bar{T} = \frac{T^{(hot)} + T^{cold}}{2}.$$

The necessary number of thermoelement pairs n is found by the formula

$$n = \frac{E}{(\bar{z}_p + \bar{z}_n)\,\Delta T}.$$

where $\Delta T = T_{hot} - T_{cold}$, the temperature difference of the hot and cold junctions.

When obtaining a fractional value, the number of thermoelement pairs should be rounded off, then the voltage U fed into the external circuit is recalculated according to the formula

-39-

$$U = \frac{n(\bar{a}_p + \bar{a}_n)\Delta TM}{M+1}.$$

The value of the current flowing in the thermopile — load circuit is given by

$$I = \frac{P_{el}}{U}$$

The optimal resistance of the load R_{ℓ} is calculated by the formula

$$R_{\ell} = \frac{U}{I}$$

The internal resistance r_{int} of the converter, not counting the resistance of the commutators, is determined by the condition of maximum efficiency: $R_{\ell} = r_{int} M$, from which $r_{int} = R_{\ell}/M$.

The ratio of the length of the thermoelement 1 to its cross section S is determined from the relationship

$$\frac{1}{S} = \frac{r_{int}\bar{a}}{2n}$$

where $\bar{a} = a_p = a_n$; $S = S_p = S_n$, $1 = 1_p = 1_n$.

Proceeding from the structural value of the thermoelement cross-section S, we can find the value of 1 and conversely. Sometimes one has to change the value of the output voltage because of the impossibility of making structurally reliable elements corresponding to the cross sections and lengths.

Extreme conditions of operation of thermoelectric convertors are short circuit conditions ($R_{\ell} \rightarrow 0$) and no load ($R_{\ell} \rightarrow \infty$). Calculation shows that in a short circuit condition the temperature of the hot junction is decreased (for low temperature elements by $\sim$ 10%), and in no load condition, is increased (by $\sim$ 20%). A rise in temperature of the hot junction in some cases may lead to breakdown of the convertor and also lower battery reliability. Under any circumstances the no load condition is not desirable.

Description of Radioisotope Generator Designs

Design Features of Generators

Up to the present time in the Soviet Union and abroad, radio-isotope thermoelectric generators of various designs have been developed which depend on the intended operating conditions and the radioactive isotopes used.

The main design differences of the generators are in the connection of the thermal or radioisotope block, the design of the thermoelement block, the thermal decoupling on the route of the main heat flow from the thermal block toward the thermoelements and from the thermoelements toward the heat exhaust system, the types of thermal insulation, etc.

Usually in foreign radioisotope thermoelectric generators, the thermoelectric convertor is designed as a number of separate thermo-elements placed on the surface of the thermal block. The tolerances and thermal expansions are compensated for by spring-controlled heat passage along the cold side. One of the advantages of this scheme is the possibility of obtaining more evenly distributed temperature along the surface of the thermal block. The assembling of the separate thermoelements, however, is highly complicated, and in emergencies their replacement is difficult. Moreover, the thermoelements along the side of the cold junctions have to be hermetically sealed. Since the number of thermoelements in some designs reaches the hundreds of pieces, the presence of a great number of seals substantially lowers the reliability of the whole generator.

From the point of assembly and operation, the block grouping of thermoelements placed on one or several surfaces of the thermal (radioisotope) block is the more expedient. This scheme has insignif-icantly lower thermal efficiency (because of the existance of a temperature drop along the height of the thermal block) compared with the previous one, and it has recently become increasingly applied in foreign and domestic designs of radioisotope thermoelectric generators.

Description of Domestic Generator Designs

In the "Beta-1" generator, the radioactive isotope Ce was used as fuel. In order to maintain constant generator power over the period of a year, a thermal regulation system is provided. Constant heat flow to the thermopile is obtained through regulation of the heat exhaust by radiation, which is carried out with the aid of movable shielding insulation.
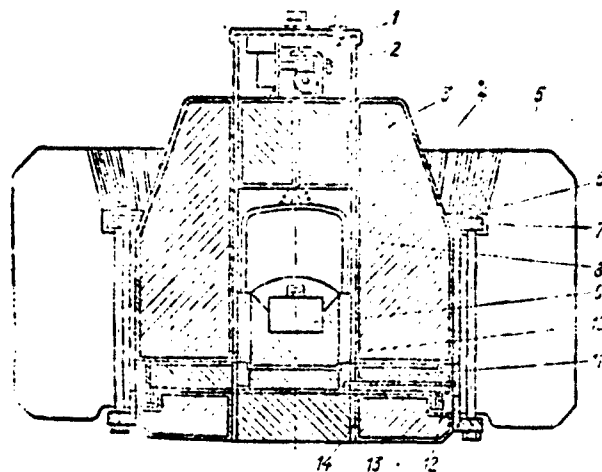
Fig. 11. Scheme of radioisotope generator
"Beta-1" based on Ce^{144}.

The design of the "Beta-1" generator (Fig. 11) consists of the
following elements: 1 — end shield; 2 — regulating mechanism; 3 —
lead plug; 4 — biological shielding; 5 — heat exhaust system; 6 —
coupling stud; 7 — rib flange; 8 — thermal insulation; 9 — isotope
block; 10 — thermal block; 11 — thermoelement block; 12 — electrical
terminal; 13 — base; 14 — copper plug.

Under operating conditions, the thermal block has contact with
the thermoelement block under the action of its own weight. During
shipping, the thermal block is fixed with the aid of two locking lugs
which grip the lateral surface of the thermal block (raising the
thermal insulation). The thermoelement block is fixed into a chassis
which allows its change (ov radial displacement when the thermal block
is raised). The thermal insulation is made in the form of hood screens;
the spaces between them are filled with inhert gas.

The advantanes of this scheme are the simplicity of design, absence
of structural elements for fastening the thermal block and, therefore,
the absence of heat losses through them, and the partial unloading
of the thermoelement block during shipping.

The shortcomings of this scheme are the absence of reliable contact of the plates of the hot junctions of the thermoelement block with the thermal block, and of the plates of the cold junctions with the generator base, the necessity of dependable securing of the thermal block during shipping, and the possibility of displacement of the thermal block relative to the thermoelement block axis, which could later result in substantial increase in the thermal resistance between the thermal block and the thermoelement block. If the size and weight of the thermal block is increased the construction of locking elements becomes significantly complicated. These elements are also heat conductors. The great weight of such a scheme is also a disadvantage.

Tests and operating experiments of the generator have demonstrated that the "Beta-1" generator had a total efficiency of about 4% at a surrounding temperature of +20 C.

The heat exhaust system, in the form of a detachable radiator, provided heat exhaust in the conditions of natural convection of the surrounding air (the temperature drop between the radiator and the air was 5-7°C at a total heat flow of ~ 135 watts and cooling surface of about 8 m²).

Figure 12 shows the design scheme of the "Beta-2" generator (Sr[*] as fuel) with electric power of 5-7 watts.

The fastening of the thermal block in the "Beta-2" generator was performed with the aid of structural ties and spring suspension. Such a scheme has additional heat losses because of heat leakage through the carrying studs. The design of the carrying studs and springs was based on the conditions of providing the necessary strength during dynamic loading and of the absence of disturbance of heat contact of the thermal block with the thermoelement block during shipping as well as under operating conditions. To decrease the thermal contact resistance between the thermal block and the thermoelement block, the lead gaskets of 1 mm thickness were inserted, causing a decrease in the temperature difference between the contacting surfaces to 2-3°C, compared with 7-8 C in the absence of lead gaskets. The presence of spring-suspension in the "Beta-2" generator provides the necessary connection of the thermobattery with the thermal block, but does not eliminate the mechanical stresses of the elements of the thermoelectric convertor. The thermal insulation of the "Beta-2" generator is provided by screens, placed along the equitemperature surfaces, preventing heat leakage, especially in the transition zone from the heated thermal block to the cool generator end-shield.

The design of the "Beta-2" generator requires special thoroughness of assembly, high accuracy of manufacture of the elements of the spring suspension, strict warming-up sequence of the thermoelement block, and maintainance of the necessary contact of the thermopile and the surface of the thermal block.
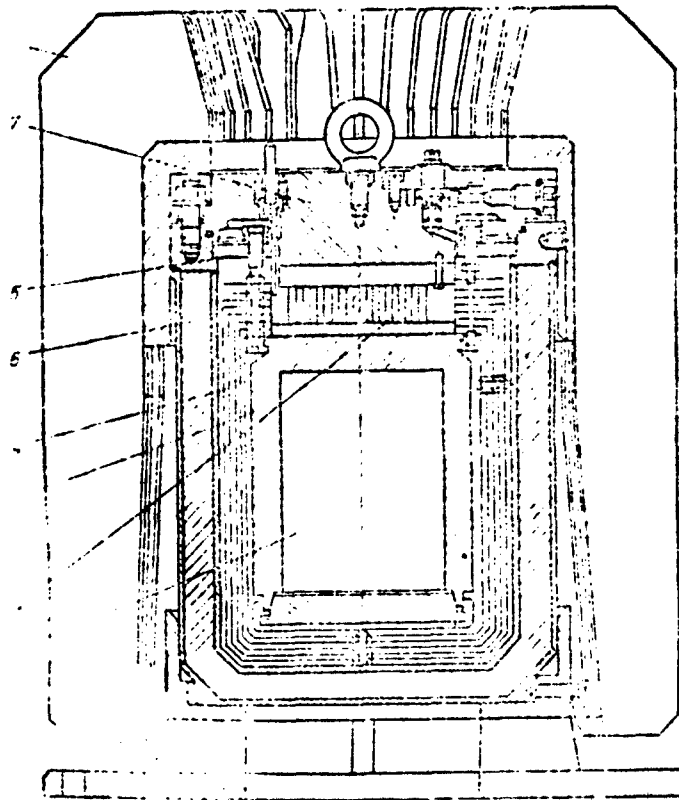
-43-

Fig. 12.   Scheme of radioisotope generator "Beta-2" based on Sr^{90}.

    1 — radioisotope block
    2 — thermoelectric convertor (thermopile)
    3 — thermal block
    4 — thermal insulation
    5 — thermal block suspension
    6 — housing
    7 — end shield
    8 — radiator

    Similar to the "Beta-2" generator is the "Beta-S" generator
which is shown in Fig. 13.   In 1968 domestic industry began the
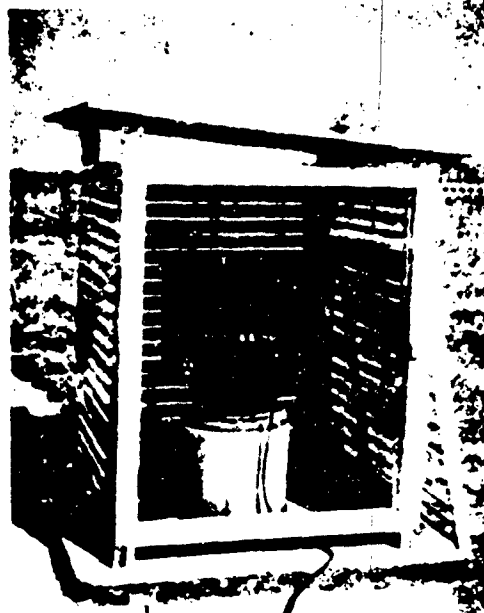serial production of such generators.

Fig. 13 "Beta-S" radioisotope generator in operating condition

The design scheme of the "Beta-2" generator was further improved
in the radioisotope generator "Beta-3", the cross-section of which is
shown in Fig. 14. This generator is intended for operation in the
conditions of the Far North and Antarctic.

The thermal block is fastened by three studs, which carry the
weight of the block and the spring stress in order to provide thermal
contact with the thermoelement block. In this scheme the spring
suspension and carrying ties must provide the connection of the thermal
block and the thermoelement block (considering the thermal expansion
and compensating for tolerances) only in operating conditions.
During shipping of the generator, the thermal block is disconnected
from the thermoelement block with the aid of a special rod, which
is simultaneously used as a heat dissipator. In this case the thermo-
element block is completely unloaded from the stress created by the
thermal block during shipping of the generator.

-45-
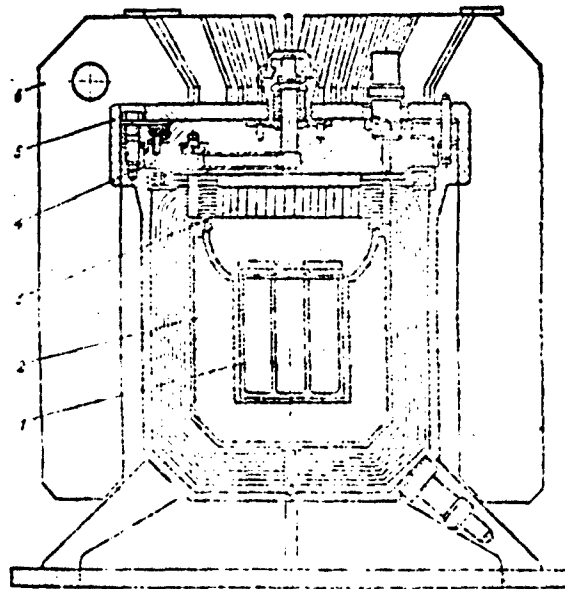
Fio. 14. Scheme of "Beta-3" radioisotope nenerator based on Sr$^{''}$.

    1 — radioisotope block
    2 — thermal block
    3 — thermopile
    4 — end-shield
    5 — housing
    6 — radiator (heat dissipation system)

    The shortcomings of this scheme are the disturbance of thermal
contact between the thermoelement block and the cover of the thermal
block durino the oeriod of transition from the oneratino state to
shinoino conditions, and also the nossibility of misalignment of the
thermal block, resultino in the impairment of contact of the thermo-
element block in the oneratino state, and, finally, the reduction of
the power output of the nenera*or.

    The distinctive feature of the "Beta-3" nenerator is the nrotection
by the deoleted uranium olaced inside the thermal insulation,
allowino the total weioht of the device to be decreased to 250 kc,
while the "Beta-S" nenerator (tooether with snioping container) has
a weiaht of 500 ko, havino oractically the same electrical nower.
The reductinr of the nene-ator weinht nermitts its installation in
inaccessible reoions without soecial loadino mechanisms.

Radioisotope generators of the D-1, D-2b, and N-1 types have schemes which basically differ from those studied above. In these generators, rigid fastening of the thermal block is used, and thermal expansions and tolerances are compensated for with the aid of a heat passage placed either between the thermal block and the thermo-element block or between the thermoelement block and the generator housing. In this case, the thermoelement block is almost completely unloaded from the dynamic action of the thermal block weight, increasing the reliability of the generator with small reduction of thermal efficiency.

Thus, in generators with Cs^{137}, D-1 (Fig. 15), the thermal block (200 kg weight) is fastened with the aid of three stainless steel X18H10T posts of 4x30 mm cross section and 70 mm height, welded to the casing of the thermal block and to the support ring. The heat passage between the thermoelement block and the thermal block is performed with the aid of copper plates, fixed to the corresponding covers and released by four cylindrical springs. Such a scheme of heat passage results in increased heat losses through high thermal resistance (the temper-ature drop in the heat passage is 40-50°C for a heat flow density of about 1 watt/cm^2) and increase of height of the hot zone.

In various modifications of the N-1 generator (Fig. 16) the following versions of heat passage are developed: non-contacting, wire-coupled, and hydraulic.

The non-contacting heat passage was made as a system of extended surfaces of the thermal block and cover of the hot junctions of the thermoelement block in such a way that the thermal expansions and tolerances were compensated for by gaps between the surfaces. The heat transfer, in that case, is carried out generally by radiant and convection heat exchange. At a heat flow density from the thermal block surface of about 5 watt/cm^2 and gas clearance (xenon) of about 0.5 mm, the calculated temperature difference of the thermal block and cover of the thermoelement hot junctions is ~ 50°C. Due to significant thermal resistance, such a scheme can be expedient for generators in which the radioactive isotopes with high energy output are used. Because of the small size of the thermal block, the greater part of its surface can be used to accomodate the thermoelements, and the increase of the heat losses becomes insignificant.

The heat passage (wire-coupled) is in the form of two plates connected with each other by a great number of wires soldered to them. If the couplings with diameter 1.2 mm (the single-wire thickness is 0.14 mm) are placed along a square with a 3 mm side (at a height of 10-12 mm), then the compensating ability of such a heat passage is 2-3 mm. The temperature drop is 10°C at a heat flow density of 2.5 watt/cm^2.
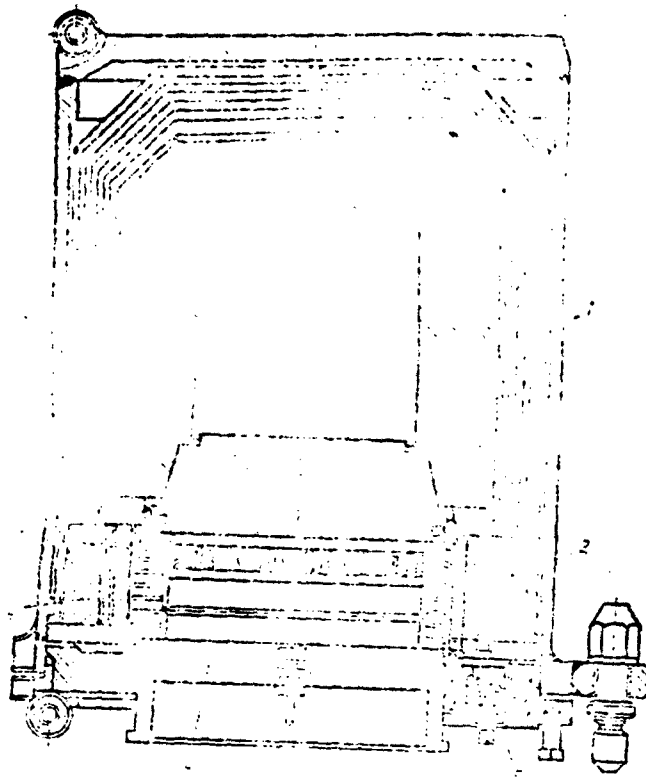
Fig. 15. Scheme of D-1 radioisotope generator based on Cs[137].

1 — shielding thermal insulation
2 — thermoinsulated insert
3 — heat passage
4 — thermal block suspension (mounting supports)
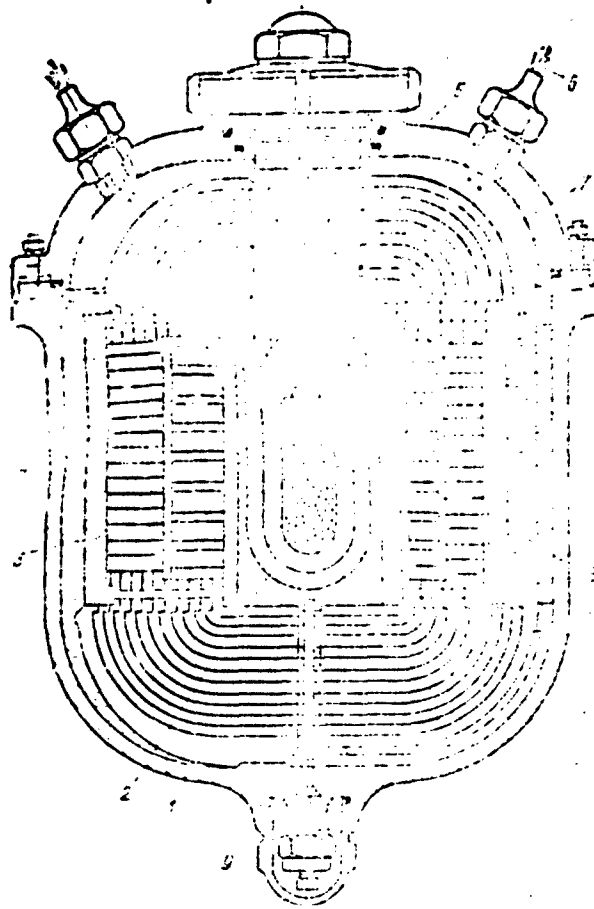5 — valve
6 — thermal block
7 — radioisotope block
8 — thermopile

Fig. 16. Scheme of radioisotope generator N-1 based on Cm^{242}:

1 — housing
2 — radioactive preparation
3 — ampoule
4 — thermoelement block (thermopile)
5 — siphon
6 — heat passage
7 — shielding thermal insulation
8 — electrical terminal
9 — valve
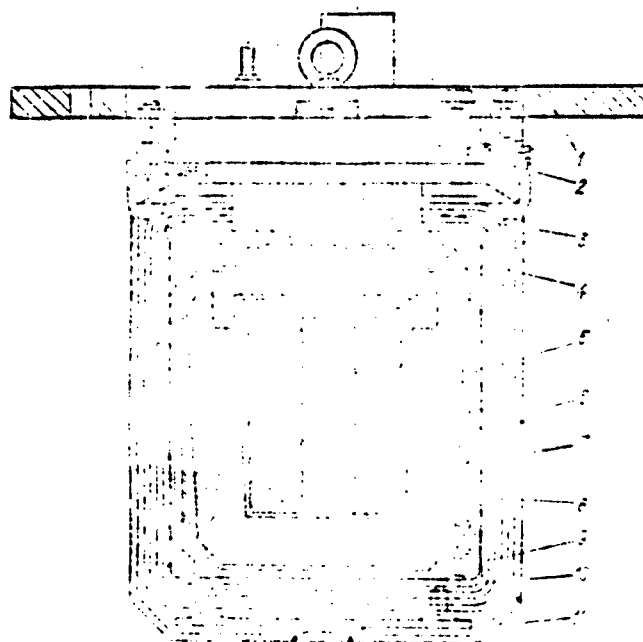
Fig. 17. Scheme of radioisotope generator "Beta-A" based on $Sr^?$:

1 — mounting disk
2 — end-shield
3 — thermopile
4 — thermal insulation
5 — heat passage cylinder

At the axial load the thermal passage has the tendency toward longitudinal displacement, which complicates the assembling technology of the thermoelement block. The heat passages with wire couplings are more likely to be used for generators which do not require the possibility of changing the block in case of emergency, because both plates of the heat passage are rigidly coupled with the casing and the thermoelement block. In order to replace the thermoelement block, the generator must be completely dismantled and the radioisotope block extracted.

The hydraulic heat passage is in the form of a cavity between the casing with a specially shaped siphon and the inner surface of the generator housing. This cavity is filled with a liquid with a high coefficient of thermal conductivity (mercury, eutectic alloys based on indium, gallium or cadmium). Contact with the thermoelement block is provided by maintaining the corresponding pressure inside this cavity. Calculations show that the temperature drop in the hydraulic heat passage is 3-5°C at a heat flow density of 0.5-1 watt/cm². The hydraulic thermal passage permits the regulation of the stress on the thermoelement block. If the base of the siphon adjacent to the thermoelement block is sufficiently thin (on the order of 0.1-0.2 mm), then there is the possibility of pitch of some of the thermoelements relative to the siphon without loss of heat contact. The hydraulic heat passage, however, is difficult to manufacture and requires the maintainance of appropriate inner pressure either through a gas cushion or with the aid of springs.

Heat passage in the form of a "heat pump" is of definite interest. It is made and tested experimentally on the O-28 model generator with an electrical immitator of the radioisotope block. The heat passage is made in the form of a cylinder which enveloped the outside surface of the thermal block, providing the unloading of the thermoelement block from radial and axial stresses. The thermal block is fastened rigidly, and thermal expansions and tolerances are compensated for by clearances between the cylinder and thermal block. The weight of the cylinder from materials with high heat conductivity is 3-5 kg and it has insignificant dynamic influence on the thermoelement block . The thermoelement block is placed between the generator end shield and the cylinder, and the correctness of placement can be verified on the electric immitator before placement of the radioisotope block in the generator.

The temperature drop between the "heat pump" and the thermal block is 20-50°C at a heat flow density of 0.2-0.5 watt/cm² and gas clearance of about 0.5-0.8 mm. As a result of shielding of the surface of the thermal block by the heat pump, the losses through insulation increase

insignificantly, and additional heat leakage through the butt-end of
the thermal block can be compensated for by increasing the insulation
thickness at these places.

The heat pump was used in the "Beta-A" generator. In order to
reduce the temperature drop between the thermal block and the cylinder
cover of the hot junctions of the thermoelement block, spring rings
were used (7 in Fig. 17). Such a scheme showed sufficient reliability
(can sustain dynamic loadings up to 8 G's) with thermal efficiency
about 90%.

In conclusion, we give a description of the "Kobra" (Cobra) type
generator (Fig. 18), which has Co^{60} fuel. This generator is made in
the form of separate blocks: biological shielding, frame, transformer
with thermal screens, and thermal block. Assembly of the generator
is performed on the working site. The radioisotope block is shipped
in a special shipping container which has a re-charging device for
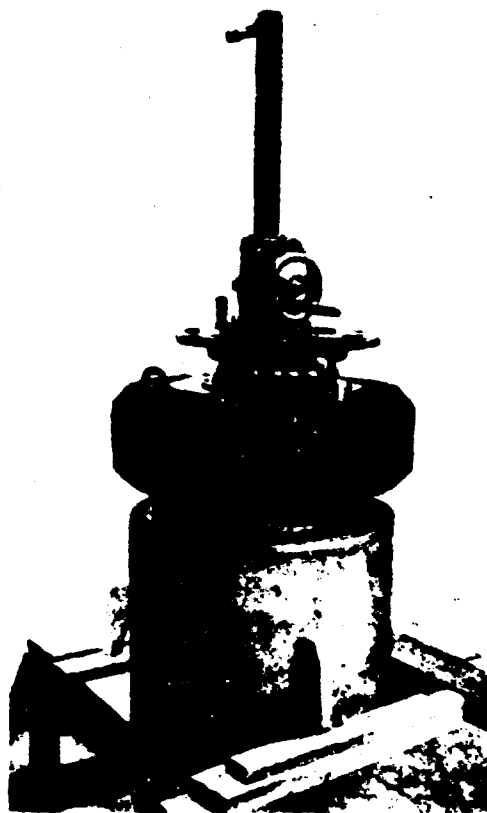placement of the radioisotope block into the generator.

Fig. 18. Radioisotope generator "Kobra" based on Co^{60}.

Foreign Generators

In the U.S. development of radioisotope sources of electricity began in 1947. Before the SNAP program was approved (1952-56), only low-power sources (about a few milliwatts) of the atomic battery type were manufactured.

The SNAP program provides for the manufacture of radioisotope sources of electricity with power from one to 500 watts. In these sources, usually, the thermoelectric method of heat transformation, produced through decay of radioisotopes is used. There have also been attempts to use the thermoemissive and mechanical methods of conversion.

Most of the radioisotope thermoelectric generators in the U.S. already developed or in design operate on radioactive preparations based on Sr^{90}. Usually the generators are intended for operation as a supply source for automatic meteorological stations, hydroacoustic and navigation devices.

The first radioisotope generator RSG in the U.S. was made by the Martin-Marietta company in 1961 and was put into operation together with an automatic weather station. Strontium-90 was used as fuel in the generator, in the form of strontium titanate; the activity is 17500 curies. The thermal capacity of the generator is 117 watt, electrical capacity — 4.2 watt. The strontium titanate is placed in a fuel capsule made from the alloy hastelloy . The heat flow from this capsule is released by 60 pairs of thermoelements, based on lead telluride, which are placed radially at the holes of the ring belt of the thermal insulation of Min-K. The thermal block and convertor are assembled in a steel frame, which is then placed in a lead biological shielding casing of 119 mm thickness. In order to improve the thermal contact the clearance between the steel block and the biological shielding is filled with mercury. Protection is provided up a dose rate of 3 mrad/hr at a distance of one meter from the center of the generator.

At the working site, the generator was put in a steel tube with 240 cm length and 75 cm diameter. The generator is situated in the low section of the pipe; above it — the nickel-cadmium batteries, voltage transformer (from 3.5-4 to 25 v) and the measuring part of the weather station. The heat exhaust of the generator is drained off through the lead shielding to the ground.

After the development program of the SNAP-7 series of generators was adapted, the following generators were designed in the U.S.: SNAP-7A, SNAP-7B, SNAP-7C, SNAP-7D, SNAP-7E, and SNAP-7F.

In these generators, the radioisotope fuel in the form of strontium-90 titanate (in fuel capsules) is placed in the holes of the thermal block made from the alloy hastelloy on the planer sides of which

thermoelements (based on lead telluride) are situated. The thermal
block surface is covered with aluminum oxide as electrical insulation.

The thermal block with the thermoelements and thermal insulation
Min-K are placed inside the housing, made from hastelloy ,
which is placed in the biological shielding. Thermal contact with
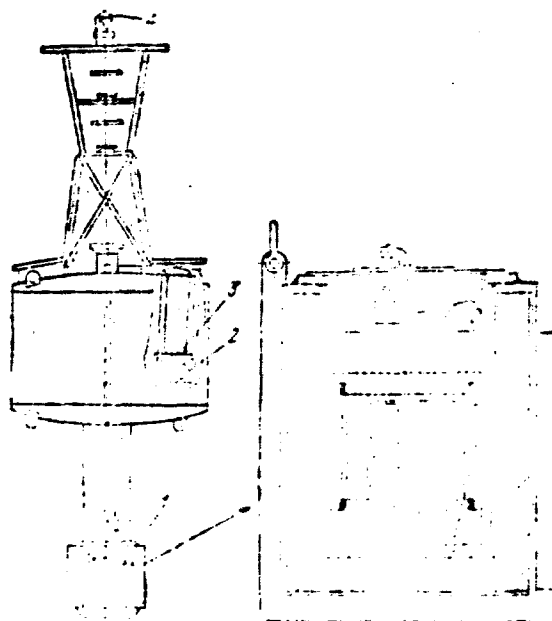the biological shielding is through a mercury layer.

Fig. 19.   Arrangement of radioisotope generator SNAP-7A
            in a floating bouy:

            1 — SNAP-7A generator
            2 — batteries
            3 — voltage transformer
            4 — electric lamp

Figure 19 represents the scheme of the SNAP-7A generator designed
to provide electrical energy for a bouy. This buoy was put into action
at the end of 1961 in Chesapeake Bay. The electrical capacity of the
generator is 10 watts; the activity of the fuel is 40,000 curies. The
biological shielding is made from depleted uranium, providing a radiation
level up to 10 mrad/hr at a distance of one meter from the generator
surface.

-54-

The radioisotope generator SNAP-7C with thermal capacity of 256 watts (activity ~ 40,000 curies) and electrical capacity of 10 watts was developed in 1962 and in the same year was put in place for supplying the automatic weather station in the Antarctic. The generator is situated on wooden planking to protect the station from submerging in the snow.

The generator SNAP-7E is the prototype of generators SNAP-7A and SNAP-7C. It was developed and put into operation in 1962 as a part of a deep-water station. The generator provides pulsed consumption of capacity of 1 kwatt in a 10 m.sec period every 30 seconds. The activity of the fuel is 31,000 curies Sr^{90}.

The scheme of the SNAP-7B and SNAP-7D generators is shown in Fig. 20.

In 1964 the Coast Guard of the U.S. Navy put into action a light house with the SNAP-7B generator. The electrical power of the generator is 60 watts, activity of the fuel is 225,000 curies Sr^{90}. The generator is put in a special container with extended cooling surface. The interior cavity of the container is filled with a mixture of water and ethylene glycol. In the upper part of the container are placed the storage battery block and voltage transformer.

The SNAP-7D generator was set up in a floating weather station in the Gulf of Mexico. The design scheme of the generator is analogous to that of SNAP-7C. The generator SNAP-7D is situated in the lower part of the floating station, which is filled with oil, providing heat dissipation from the generator to the housing and to the surroundings.

The SNAP-7P with capacity of 60 watts, activity 225,000 curies Sr^{90} was put into service with a navigation light house. The service time of the generator is 10 years.

In Table 15 the basic parameters of the SNAP-7 type generators are given. The main feature of the SNAP-7 generators is that the thermoelectric convertor is made as a complex of separate thermo-elements placed on the thermal block surface.

Temperature expansions and tolerances are compensated for by a flexible heat passage along the cool side, provided by springs.

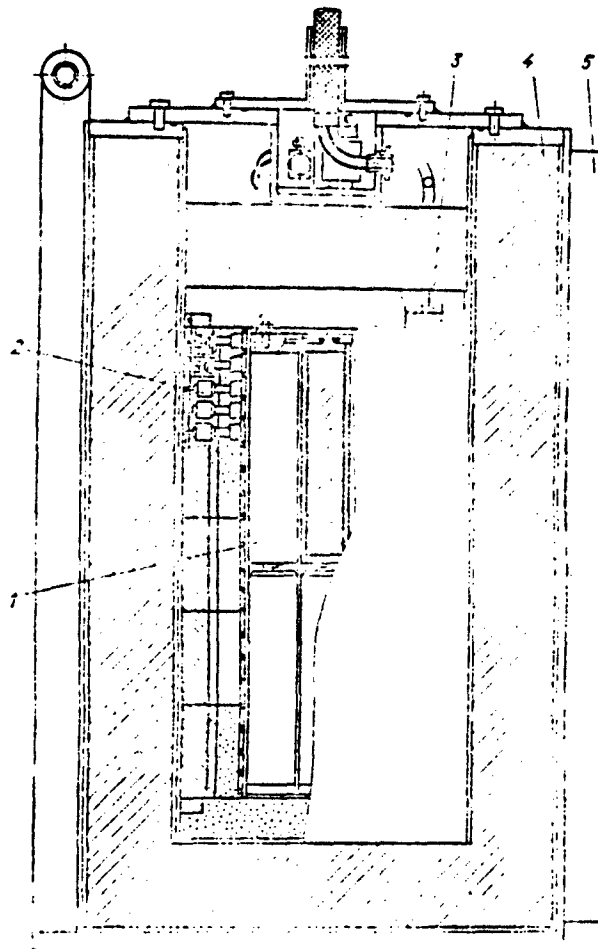Similar generators have been developed in England and in other countries.

-55-

Fig. 20. Scheme of CHAP-7B (7D) generator:

　　　1 -- radioisotope block
　　　2 -- thermoelements
　　　3 -- thermal insulation
　　　4 -- biologic 1 shielding
　　　5 -- radiator

Table 15

Таблица 15

a) Характеристика	WSG	SNAP-7A SNAP-9A	SNAP-7B SNAP-7D	SNAP-7E	SNAP-7F
b) Активность, *кюри*	17500	40000	225000	310000	225000
c) Тепловая мощность, *вт*	117	276	1410	395	1410
d) Максимальная электрическая мощность, *вт*	4,2	11,6	68	6,5	60
e) Коэффициент полезного действия, %	3,4	4,5	4,7	3,3	4,15
f) Напряжение, *в*	3,5—4,0	5,0	12,0	4,5	—
g) Материал термоэлементов	m) Теллуриды свинца				
h) Количество пар термоэлементов	60	60	60	60	60
i) Температура горячих спаев, °К	726	783	763	683	—
j) Температура холодных спаев, °К	326	459	333	290	—
k) Вес, *кГ*	765	870	2080	1020	2100
l) Проектный срок службы, годы	2	10	10	5	10

Key: a) characteristic
   b) activity, curies
   c) heat capacity, watts
   d) maximum electrical capacity, watts
   e) efficiency, %
   f) voltage, volts
   g) thermoelement material
   h) number of thermoelement pairs
   i) temperature of hot junctions, °K
   j) temperature of cold junctions, °K
   k) weight, kg
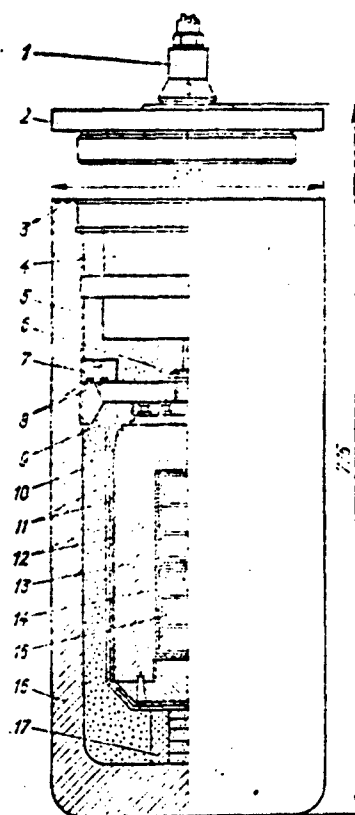   l) designed life time, years
   m) lead telluride

Fig. 21. Scheme of radioisotope generator based on Cs[137]:

```
1  —  electric terminal
2  —  end-shield
3  —  sealing gaskets
4  —  cavity for electrical energy accumulation system
5  —  thermal insulation
6  —  electrical terminal
7  —  push-ring
8  —  seal
9  —  thermoelement block
10 —  thermal insulation (fill)
11 —  shields
12 —  thermal insulation
13 —  thermal block
14 —  radioisotope block
15 —  radioactive preparation
16 —  housing
17 —  thermal block mounting
```

The desian scheme of radioisotope generator based on Cs^{137} is shown in Fig. 21. The thermoelement block is made in the form of sequentially connected thermoelements placed between the plates. In the generator, the thermal block is rigidly fastened. Temperature expansion and tolerances are compensated for through individual connection of the thermoelements.

Fields of Application of
Isotope Thermoelectric Generators

Technical and Economic Preconditions

One of the conditions of widespread radioisotope thermoelectric generator utilization is the low cost (compared to other sources) of 1 watt·hr of energy produced.

Table 16 gives (according to foreign press data) the comparative costs of one watt·hr of electrical energy from utilization of various isotopes in generators with equal capacity (efficiency of generator assumed equal to 5%).

Table 16

Таблица 16

a) Изотоп	Sr⁹⁰	Cs¹³⁷	Ce¹⁴⁴	Pm¹⁴⁷	C.⁶⁰	Pu²³⁸	Po²¹⁰	Cm²⁴²	Cm²⁴⁴
b) Срок службы генератора, годы	10	10	1	2,6	5	0,5	10	0,5	10
c) Стоимость, долл/(вт·ч)	0,023	0,031	0,002	0,86	0,031	2,12	0,26	1,64	0,12

Key: a) isotope
     b) service life of generator, years
     c) cost, dollars/watt·hr

Generators for Ground Service

At the present time battery supply sources or storage batteries with charging units are, for the most part, used as the general supply sources for communications and testing instruments for ground, surface, and underwater apparatus. Due to the demands of the times, however, they must be replaced by more reliable, higher capacity supply sources such as radioisotope thermoelectric generators. The more this can be widely used in various spheres of the national economy. Thus, their application in automatic research and weather stations in nearly inaccessable and remote, uninhabited regions allows one to reduce their maintainance costs and to obtain data from regions earlier inaccessable.

In the Soviet Union, the U.S. and other countries, radioisotope generator models for various purposes have been developed. In the Soviet Union in recent years, several types of generators are in operation. One of the first made was the "Beta-1" generator based on Ce (Fig. 22).

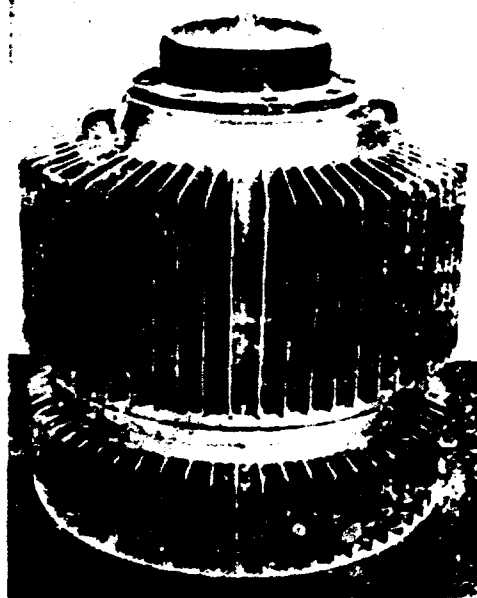Fig. 22. Radioisotope generator "Beta-1".

The generator has average electrical capacity of $\sim$ 5.35 watts, its e.m.f. is 6.9 v, the voltage at $R_{load} \sim$ 2.7 ohm is equal to 3.8 v. The total efficiency is equal to 3.7-4%, γ-background at a distance of 1 m from the shipping container does not exceed 1 mrad/hr. The generator operated successfully during eight months as a part of an automatic radiometeorological station.

The "Beta" series generators are generally designed for supplying various automatic devices which are placed in remote and nearly inaccessable regions. In the generators of the "Beta-2", "Beta-3", and "Beta-5" types and in some others, strontium-90 titanate is used as fuel. The "Beta-2" and "Beta-5" types of generators are designed for operation in the middle regions and in temperate climate (Fig. 23 and 13). The "Beta-3" generator (Fig. 24), is specially made to operate in the Far North regions and in the Antarctic. The other generators of that series ("Beta-A") are designed to operate in the high mountain regions. In "Beta-A" generators, the exhausted heat is used for thermal regulation of the instrument compartment (Fig. 25). The generators of the "Beta series have electrical capacity of 5-25 watts, e.m.f. of 8-30 v, efficiency of 4-6%. The majority of generators operate successfully in almost inaccessible regions. In the U.S. a model of automatic research station with radioisotope generator has also been designed for the Antarctic. The station is to be used for investigation of various effects which take place in the ionosphere, geomagnetism, earth movement, and other geophysical phenomena, and also for gathering and transmitting the meteorological data.

In addition, various types of generators have been developed for automatic radiometerological stations, retransmission points, etc.

In the U.S., the SNAP-7 series of generators has been developed. The series has five generators with capacity from 6 to 60 watts. The output voltage is 4.5-12 v, efficiency is 3.3-4.7%, weight is 1-2 tons. The shielding material is uranium, cast iron. The first generator of this type as a part of an automatic weather station was tested in Iceland, where it successfully operated over a period of two years. As an energy storage system, nickel-cadmium batteries were used.
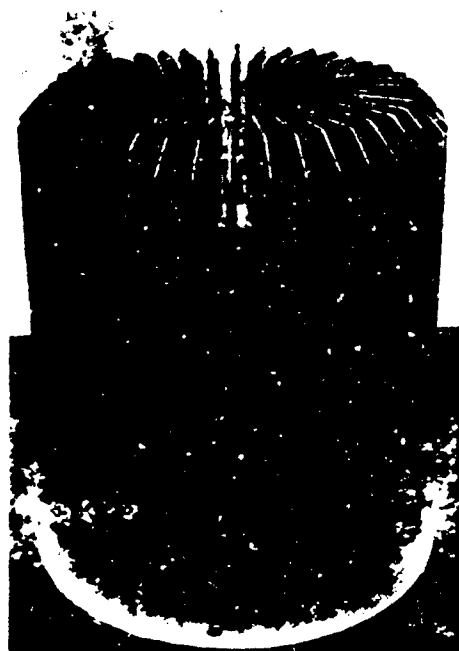
Fig. 43.   Radiographs cut rotor w t 4-2

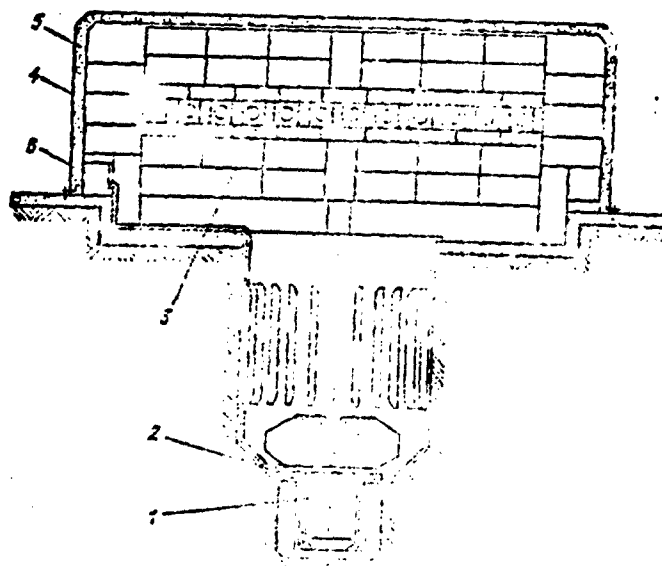Fig. 44.   Radiographs separator 4-1=

Fig. 25. Arrangement of radioisotope generator Beta-A in the
          thermostat box of a cosmic radiation registration system.

          1 — radioisotope generator
          2 — housing of heat dissipation system
          3 — cosmic radiation detectors
          4 — housing of thermostat box
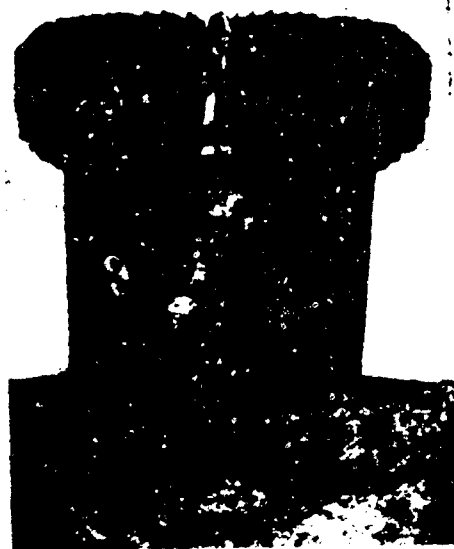   5 — plastic foam
          6 — voltage transformer
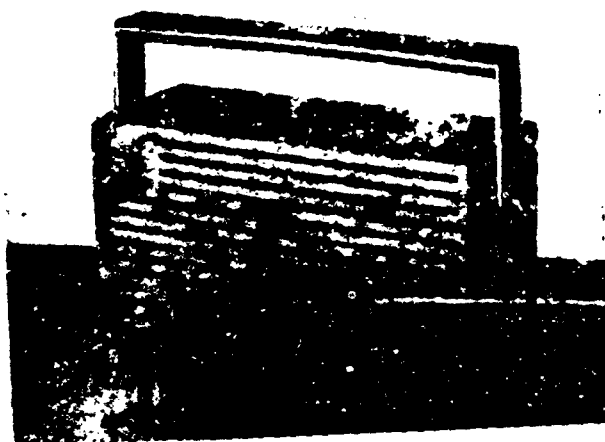
Fin. 25. Radioisotope generator of "O" series

Fig. 27. Radioisotope generator Niho-5

Generators for Surface and Underwater Purposes

Radioisotope thermoelectric generators may find wide use in buoys and underwater devices for various purposes. The "D" type generators (Fig. 26) are designed to operate in a water environment. They consist of the generator and shipping container. In the "D" type generators, a preparation based on Cs137 is used as fuel in the form of caesium-lead silicate glass. The thermopile is unloaded; the generator housing is designed for increased pressure, allowing the generator to operate at the required depths. The generators have the following principal characteristics: electrical power output — 10-25 watts; e.m.f. — 12-24 v; efficiency — 3-5%. Radioisotope generators of such type also operate in the U.S. Thus, the radioisotope generator of the SNAP-7 type has been used since January, 1964 as a supply source in a floating oceanographic weather station. In addition, generators of such type are used in navigation light houses placed in distant or inaccessible regions. Several hundreds of such generators are required. To make the navigation period longer, the U.S. Coast Guard has suggested that radioisotope supply sources be used in the sound signal beacons which are placed on the bottom of the sea or ocean to mark the navigable channel in zones where use of surface signal buoys is restricted by floating ice-floes. Radioisotope generators can also be used in navigation devices placed on the sea floor to designate underwater rocks and other underwater dangers for submarines. Underwater radioisotope generators can be used for supplying geophysical and other testing instruments whose purpose is the registration of earthquakes, underground nuclear explosions, etc.

The U.S. Coast Guard plans to place seismometers on the ocean floor to register underground nuclear explosions and earthquakes. Some of these devices will have cable communication with the coastal stations; the others will be self-contained. Radioisotope generators may be used in both cases: in the first — for supplying the cable amplifiers; in the second — as the sole supply source of the whole device. Similar systems can also be used for warnings concerning tidal waves formed by underwater quakes.

In foreign publications, experts have analyzed the possibility of using radioisotope supply sources in sound navigation transmitters which provide position-location of surface and underwater vessels with high accuracy. It is assumed that the sound generators will be needed with power supply sources of 0.01-1 watt, depending on the strength, time and repeat velocity of the sound signals. The operating time for these devices is from 5 to 20 years.

It is assumed that a few sound generators will be established in areas of definite interest. Following an inquiry from a ship, these generators must transmit encoded impulses by which the ship's location may be determined with sufficient accuracy. The variants for utilization of various acoustic devices have been analyzed to designate the

location of submarines. Such devices could transmit the data through the cable communication system to the coastal stations, the supply of which can be provided by utilization of radioisotope generators in a special complex of testing instruments. The power of the supply source must be about 100 watts.

Isotope current sources can also be used during prospecting and exploitation of mineral resources on the bottom of the sea and ocean. Several companies in the U.S. are very interested in this problem.

Radioisotope generators can be used to supply seismic apparatus which is used for exploration of oil and also for supplying the special apparatus used for mapping the sea floor. During exploitation of sea wells and exploratory drilling, the generators can be used for supplying the telemetering apparatus, the oil pipeline regulation system, etc. They can be used for supply of the equipment of underwater stations for various purposes, in particular for photo- and telemetering devices of communications systems, etc.

Generators for Space Uses

A wide program of work on exploration of outer space is connected with the necessity of the availibility of long term reliable energy supply sources, especially sources for supply of life-support systems on the space craft,

In the U.S. radioisotope energy supply sources for space craft life support systems are being developed by several firms. Radioisotope sources may be used to perform many important functions of the life support system, particularly for burning and processing of waste, heating food, conversion of liquid oxygen to gas, and also for removal of impurities and regulation of the air medium in the manned space craft.

In the Manned Spacecraft Laboratories, 30-day experiments are carried out on a system in which drinking water is recovered from human liquid wastes (wash water, urine, condensate). As fuel a preparation based on Pu^{238} is used, with heating capacity of 280 watts.

For supply of on-board electronic regulation and control, scientific and telemetric instruments and communications facilities, electrical energy sources having long term service and high reliability are also required. In some circumstances, the parameters of solar and chemical batteries (weight, reliability, operating life, etc.) cannot meet the requirements demanded of the energy sources for specific types of manned spacecraft. In such cases, it is advantageous to use radioisotope generators. Such a generator was used in the U.S. in satellites of the Transit-4 type, in the the Soviet Union — or satellites of the "Kosmos" type. On satellites, highly varios purposes may also be served by such generators of thermal and electrical energy, the power

-66-

of which, depending on the purpose and service life, varies from fractions
of a watt to kilowatts.

In contrast to terrestrial cere ators, which are based on fission
products and for which the critical value is usually cost, the critical
parameter for gene ators for use in space is weight. For space uses,
therefore, it is preferable to use preparations based on α -emitters
which have high specific power and which do not require the use of
heavy shielding.

Miniature Generators

As a rule, in miniature generators it is advantageous to use α -
radioactive isotopes or, as a last resort, soft β-emitters.

In the USSR and the U.S., operative miniature generators have
been developed, one of whose possible applications is in medicine.
Developed in the USSR, miniature generators of the IG-67 type (Fig. 27)
have capacity of $\sim$ 1 watt, service period of 10 years, fuel — Pu^{238},
generator weight — $\sim$0.5 kg.

A radioisotope generator has been dev loped in the U.s. for supply
of heart stimulators. The characteristics of the generator are as
follows: electrical capacity $\sim$ 162 watt, service life $\sim$ 10 years,
fuel — Pu^{238}, generator weight' 100 g, volume 70 cm , dose rate
of irradiation — not more than 5 mrad/hr.

Work is being done on artificial (mechanical) hearts with radio-
isotope sources of energy supply.

In conclusion, one might mention that in the past 10 years a
whole series of radioisotope thermoelectric generators have been
made which have long service life with high reliability. The specific
power capacity (watt·hr/kg) of such generators exceeds by 10 times
that of the best model sel -contained chemical source of electrical
energy in wide use at the present time.

Questions Concerning Radiation Safety During
Development and Operation of Isotope Thermoelectric Generators

Types of Radiation Effects

Radioactive substances possess potenti l danger nd can have
harmful effects on both humans and equipment.

Concerning the action on humans, one may distinguish two types
of irradiation: external nd internal. In the case of external

radiation, the danger is from neutron- , γ- and braking radiation, and also, in some cases, β-radiation. Internal irradiation is caused by radioactive material getting inside the organism.

External Irradiation. For the same density of radiation flow, the greatest danger is from fast neutrons. This is because under the the action of this sort of particle with tissue of the human organism containing significant amounts of hydrogen, primarily recoil protons are formed. Protons create powerful ionization along their paths since they are heavy, single-charged particles. This ionization by many times exceeds the ionization caused by electrons formed as a result of the interaction of braking or γ-radiation with the surroundings. Potential danger is also present from thermal neutrons, since during their interaction with living tissue, nuclear reactions occur with emission of γ-quanta and protons. The higher the energy of the neutrons, the greater is the biological danger they present, i.e., with increase of neutron energy, there is a decrease in the magnitude of the maximum permissible neutron flux. For γ-, braking- and β-radiation, such dependence is practically absent up to an energy of 2-3 Mev.

For persons constantly occupied at work with source of ionizing radiation there has been established a limit of 0.1 rem per week, however, in separate occurrances a single dose may have a value up to 3 rem per any 13 consecutive weeks (quarter) on condition that the annual dose will not exceed 5 rem . The stated levels are such that there is no measurable change in the state of health of the person.

Internal Irradiation. Radioactive materials inside the human organism present the greatest danger since they interact directly with the living tissue. The exposure time sharply increases, since in the majority of cases, the radioactive materials enter into chemical reactions with various elements of the biological tissue and are released from it rather slowly. In addition, certain radioactive materials have the tendency to concentrate in specific organs, which increases still more their biological danger. It is important to bear in mind that when radioactive material falls inside the organism, the usually measures of protection by distance, time, shielding materials are inapplicable and removal of the radioactive material is sometimes practically impossible. The maximum danger is present in the contamination of the surrounding air, food, and water. Of these the greatest danger is from contamination of the air. This is explained by the fact that the volume of air inhaled by a person per 24 hours exceeds by many times the volume of water or food ingested in the same period. In addition to contamination of air, water, and food, one must also keep in mind contamination of various objects, articles, clothing, etc. Such contamination represents potential danger from its possible introduction into the organism.

-68-

Principal Safety Measures

To insure safe operation with radioactive materials, the following precautions are necessary: maximum possible (reasonable) reduction of the dose of external irradiation and limit of intake of radioactive material into the body. Radioactive materials occur in covered and uncovered forms. The greatest biological danger, undoubtedly, is presented by uncovered preparations. Therefore, one must try to use radioactive materials in a covered state, insuring maximum protection from intake of radioactive material from the immediate surroundings. For this it is possible to transform the radioactive material to a suitable state of aggregation and then encase it in a special envelope, etc. In the covered state, radioactive material presents danger as a source of external irradiation. From this standpoint pure α-emitters are essentially not dangerous, since they are easily absorbed by relatively thin layers of material and require practically no special protection.

Safety while working with β-emitters may be insured by special, relatively small, screening (ampoule casing, etc.). However, in the presence of significant quantities of radioactive materials considerable danger may be present from braking radiation arising from the inter-action of β-radiation with matter. Braking radiation is similar to γ-radiation in the character of its interaction with matter. Consequently, the protective measures it requires are the same as those for γ-radiation.

Biological effects are principally based on the cumulative radiation dose, which is dependent on the radiation intensity (dose rate) and exposure time. It is apparent that, if it is not possible to shorten the exposure time, it is necessary to decrease the dose rate. One may decrease it either by increasing the object-source distance (usually the dose rate decreases with the square of the distance), or by use of special attenuating shielding (lead, tungsten, etc.). As a rule, in determination of protective conditions, one takes into account all three factors: exposure time, object-source distance, attenuation of radiation by the material of the protective shielding.

For protection from neutron radiation, neutron absorbing material (paraffin, water, etc.) is used.

All covered sources must be kept in a safety device. Touching them by hand is prohibited; during manipulations with them, one must use various adaptors, manipulators, protective shielding, etc.

Radiation Monitoring

Radiation monitoring was organized for monitoring the exposure level to which persons working with sources of ionizing radiation are

-69-

subject. The task of individual radiation monitoring is to measure
the dose received by each worker.

Individual radiation monitoring can be achieved by the use of
three methods: photomethod, and ionization and luminescence methods.
In practice, the first two are most often used. The IRM kit (Individual
Radiation Monitoring) is used for determination of the cumulative
dose of exposure (for a certain cut-off time). With this method it
is not immediately possible to monitor the dose received on the site,
since it requires special photographic processing.

The ionization method exists in two forms: the first, for
measurement of the dose, a condenser ionization chamber (kit KID)
is used; in the second, a read-out ionization chamber with direct
reading of the dose by the scale of an electroscope connected with
the chamber (DK). At the present time sufficiently complete sets of
specialized instruments for measurement of the dose rate, surface
contamination and aerosols have been developed.

Radiation Protection

Practical work with sources of ionizing radiation will show
that observance of suitable rules for handling radioactive materials
and sources of nuclear radiation may provide complete safety for the
operating personnel. Restricted permissible exposure levels for all
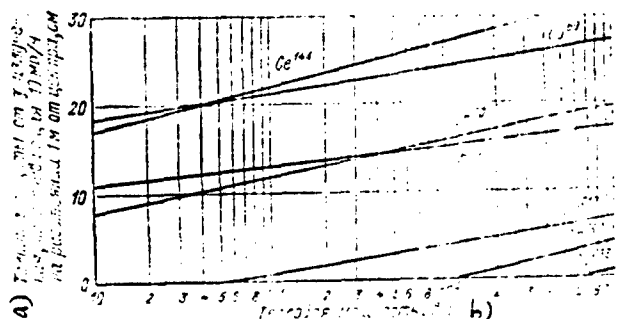forms of ionizing radiation have been established.

Fig. 28.   Thickness of lead shielding v.thermal power of
           radioisotope blocks based on Ce^{144}, Co^{60}, Sr^{90},
           Cs^{137}, Po^{210}, Cm^{242}, Pu .

    Key: a) thickness of shielding against γ -radiation,
               providing 10 mrad/hr at a distance of one meter
               from the center, cm.
           b) thermal power, watts

-70-.

In radioisotope generators based on β-radioactive isotopes, the greatest attention must be given to protection from γ- and braking radiation.

The best protection from γ-radiation is afforded by materials with high atomic number Z (lead, tungsten, uranium).

Fig. 28 shows the graphic dependence of the thickness of the lead shielding on the thermal power of the radioisotope block for generators based on various isotopes. According to the given graph, it is possible to estimate the required protection for the restricted permissible dose rate of γ-radiation of 10 mrad/hr at a distance of one meter from the radioisotope block. If an increase of dose rate is permissible, it is possible to use the same graph, decreasing the value of the thermal power by as many times as the permissible dose rate is increased, and conversely.

Sanitation Regulations for Installation and Operation
Of Isotope Current Sources Based on β-active Isotopes
For Ground Purposes (SR No. 619-66)

Introduction

The requirements of the present regulations are compulsory during manufacture, storage, shipping, and operation of current sources based on β-active isotopes for automatic meteorological stations. The requirements grew out of the "Sanitation Regulations for Work with Radioactive Materials and Sources of Ionizing Radiation" No. 333-60.

The responsibility for execution of the present regulations lays with the administration of the institutions and enterprises of manufacture, shipping, and operation of the isotope current source (ICS).

ICS's are intended for supply of automatic ground meteorological stations and other installations established in remote and nearly inaccessable regions.

The application of an ICS for supply of stations is determined by its operating reliability in comparison with other self-contained supply sources, continuity of work without monitoring and replacement of the device itself or its components, and also by economic advantages.

From the standpoint of providing radiation safety, the ICS is a covered radiation source of high activity (20-200 kcuries) with a high degree of tightness, mechanical, thermal and corrosion stability.

The ICS differs from covered sources applied in radioisotope devices in that its highest operating efficiency is reached when the maximum radiation absorption inside the source itself is provided. This allows the enclosure of the radioactive preparation in a capsule (primary packing)* with a wall thickness by far exceeding the penetrating ability of the primary (working) radiation.

In order to provide radiation safety in relation to external, penetrating radiation, the isotope block is placed in a shielding block (working container) and during shipping, additionally in a shipping container.

Potential radiation danger during shipping, installation and operation of the ICS can include the following factors:
braking radiation and γ-radiation from impurities;
surface radioactive contamination of the ICS and shipping containers.

---

*In technical descriptions, it is acceptable to call an encapsulated radioactive preparation an 'isotope block'.

-72-

The fundamental sanitation-technical characteristics of the ICS are:
    rated (actual) activity contained in the isotope block;
    isotope used;
    chemical and radiochemical purity of the isotope;
    mechanical, thermal, and corrosion stability of the isotope
block and of the ICS as a whole;
    level of radioisotope contamination of the unit and external
surfaces of the ICS;
    radiation levels from the ICS.

I. Requirements for the Isotope Block

1. When making the heat source for the ICS, preference should
be given to an isotope with maximum heat output with low penetrating
ability of the principal radiation, with low output of accompanying
and secondary penetrating radiation, and with minimal possibility
of radiotoxicity. Criteria for selection (when various technical-
economic indexes are equal) should be the minimal value of the product
of activity of the isotope block and the reciprocal of the maximum
permissible concentration of the isotope in the air.

2. Radioactive preparation should be a solid, non-friable, in-
soluble in water, non-sublimating, and not entering into reactions
with air and the material of the capsule, with high radiation and
temperature stability, providing minimal leakage of the isotope
during rupture of the isotope block. The melting temperature should
not be less than $500^{\circ}C$, and the boiling temperature not less than
$1500^{\circ}C$.

3. The design of the isotope block should guarantee its inte-
grity and tightness under normal operation and storage of the ICS
over a period of ten half-lives of the isotope and during emergency
situations which may arise during shipping by all forms of transportation.

4. The material of the isotope block and the technology of its
sealing should provide integrity and tightness in an atmosphere of air
and ground water over a period of ten half-lives of the isotope under
any climatic condition.

5. The design and manufacturing technology of the isotope block
should insure its integrity and tightness with a safety factor of
ten in the case of possible increase in internal pressure. Pressure
should be estimated for a temperature of $1100^{\circ}C$.

6. The isotope block should maintain its integrity and tightness
under the following stresses:
    vibration with frequency of 25-60 Hz under an acceleration
of $\gtrsim$ 10 G's over a period of 6 hrs.
    single impact with acceleration $\lesssim$ 3000 G's and action time
of 0.5 msec.
    temperature of $1100^{\circ}C$ over a period of 2 hours.

7. The capsule material of the isotope block should be stable toward radiation. Losses of strength of the isotope block from radiation damage should not exceed 10% from the original with absorption of a dose corresponding to an operating dose for an exposure period equal to ten half-lives.

The capsule material of the isotope block should be chemically stable in relation to the preparation and the radioactive decay products.

8. Surface contamination of the isotope block should not exceed $5 \cdot 10^{-5} \mu$curies/cm^2 for β-active isotopes and $1 \cdot 10^{-7} \mu$curies/cm^2 for α-active isotopes with a cumulative activity no greater than 1μcurie for β- and $5 \cdot 10^{-2} \mu$curie for α-active isotopes.

9. The dose rate from the isotope block (without protection unit) should not exceed 300 rad/hr at a distance of one meter. If the dose rate exceeds that stated above, the protection unit should maintain, without disruption of its integrity, the same mechanical and temperature stresses as the isotope block.

II. Requirements of the Protection Unit (Working Container)

10. The protection unit should insure reduction of the radiation level from the isotope block to the following values:
        on the surface of the protection unit — no more than 30 rad/hr,
        on the exterior surface of the heat exhaust — 10 rad/hr.
        at a distance of one meter from the protection unit — 1 rad/hr.

11. The material and design of the protection unit should insure its integrity and tightness under normal operating conditions of the ICS and most probable accidents.

12. The protection unit must withstand the following mechanical and temperature stresses:
        vibrations of 20-66 Hz and acceleration $\cong 10$ G's in a period of 6 hrs,
        single impact with acceleration not less than 100 G's during an action time of 5 msec.
        temperature of 1110°C for a period of 2 hrs.

13. The design of the protection unit should provide for possible removal of the heat exhaust system and ICS packing in the shipping container, connection of the cable, short-circuit of the thermopile terminal, replacement of the thermal convertor, and also the reverse operations with the aid of manual remote-control devices providing removal of the operator from the ICS by a distance no less than 0.5-1 m during an execution time for each of the stated operations no more than 3 min.

-74-

14.   The total weight of the ICS in the protection unit, together with the heat exhaust system, should not exceed 250 kn.

15.   The design of the protection unit should insure the possibility of carrying the ICS with the aid of rods of length no less than 2 m. The protection unit should have a device for rapid securing and various other shipping aids.

16.   The protection unit accessible for contact should not have a temperature over 80°C.

17.   The surface contamination of the protection unit (working container) should not exceed $1 \cdot 10^{-4}$ μcuries/cm^2 for β - and $1 \cdot 10^{-5}$ μcuries/cm^3 for γ-active isotopes.
       The surface of the working container accessible to contact (devices for fastening, installation and shipping of the ICS and the external surface of the heat exhaust system) should not have surface contamination exceeding $3 \cdot 10^{-4}$ curies/cm^4 for β - and $1 \cdot 10^{-5}$ curies/cm^2 for γ-emitting isotopes.

18.   The design of the protection unit should exclude the possibility of its disassembly by ordinary mechanics tools.

19.   On the surface of the protection unit, there should be a sign "Radiation Danger" indelibly marked in red.

III.   Requirements for Shipping Container

20.   The design of the shipping container should allow the possibility of quick extraction of the protection unit (working container) with the aid of manual remote-control devices.

21.   The shipping container should be stamped in accordance with "Regulations for the Transportation of Radioactive materials" No. 349-61.

22.   The shipping container should have devices for its securing in the transportation facilities and for use by hoisting facilities.

23.   The dose rate on the surface of the shipping container should not exceed 200 mrad/hr at a distance of one meter from it — 10 mrad/hr, i.e. the level corresponding to the III shipping category of crating radioactive materials.

24.   Contamination of the outer surface of the shipping container should not exceed 200 γ-particles or 5000 β-particles per 150 cm^2 in one minute ($3 \cdot 10^{-5}$ for β - and $1 \cdot 10^{-7}$ curies for γ-active isotopes).

25.   The exterior surface accessible to contact of the shipping container should not have a temperature above 80°C.

26. The material and design of the shipping container should insure its integrity under the following stresses:

single impact with acceleration no less than 10 G's and duration of 50 msec,
temperature of 500 C for a period of 2 hrs.

IV. Storage and Shipping of Radioisotope Current Sources

27. The ICS should be stored and shipped in its shipping container in accordance with the "Regulations for Transportation of Radioactive Materials" No. 349-1.

28. During shipping and storage of ICS's, the thermopile terminal should be short-circuited. It is not permissible to ship or store the ICS without the heat exhaust system (finned surface).

29. Shipping of the ICS without a shipping container is permitted only on that portion of the route not accessible for the transport of the shipping container and during its execution by specially trained workers.

30. Delivery of the ICS to the storage site during shipping in the working container should be done after preparation of the site (preparation of the bunker or embankment, arrangement of barriers and signs, assembly of all units of the weather station, etc.).

31. During shipping of the ICS in the working container, it is necessary to use all available means of transportation which speed up the shipping or increase the distance between the ICS and personnel (helicopters, cross-country vehicles, pack animals, etc.)

V. Operation of Isotope Current Sources

32. All work connected with transport of the ICS, site preparation, its installation, thermopile r placement, inspection of the ICS and return (recovery) should be carried out by specially trained workers.

33. Work of assembly and inspection of the weather station units not on the site of the ICS installation, may be carried out by weather service personnel. During this time they should be instructed concerning and acquainted with current regulations.

34. Installation of weather stations with ICS's are permitted only by authorization of the Chief of the Sanitary-Epidemiological Control of the Ministry of Public Health of the USSR and notification of the organs of the militia.

35. As a rule, the ICS should be used in the shipping container.

36. During operation of an ICS in its shipping container in areas accessible to people, the ICS installation site should be fenced with barbed with 1.5 m high for a distance at which the dose rate does not exceed 0.01 mrad/hr.

37. To reduce the size of the fenced off area, it is recommended that the ICS be buried in the ground or surrounded by an embankment. In this case, the fenced off area should be such that the dose rate at a height of 1.8 m above the ground should be 0.01 mrad/hr.

38. When the site is located in inaccessible area, it may be unfenced.

39. Operation of the ICS in the working container may be carried out only in remote sites.

40. When utilized in its working container, the ICS should be buried in the ground, or surrounded by an embankment such that the dose rate 3 m from the ICS does not exceed 10 mrad/hr (at a height of 1.8 m above the ground). The site should be fenced off by barbed wire. The minimum distance between the fence and the installation site should be 3 m.

41. On the fence and also on the ICS installation site, there should be radiation danger signs picturing the skull and crossbones and the warnings "Keep Out!", "Hazard to Life", easily seen at a distance of 10 m.

42. All units of the automatic weather stations, when using the ICS in its working container, should be set up at a distance no less than 3 m from the ICS.

43. The specialized organizations which carry out the installation of the ICS must be strictly accountable for it. The weather service operating automated stations with an ICS, is required to notify the specialized organization in the case of absence of broadcast from the station for a period of a week and immediately take joint measures for finding the cause of the cessation of station operation.

The specialized organization should ask the weather service about the operation of the weather station with an ICS at least once a month.

The presence of the ICS at the installation site and the integrity of the isotope block should be verified for weather stations located in accessible regions no longer than 10 days from cessation of broadcast from the station and, in accessible regions, in the shortest possible period. The integrity of the ICS may be determined by the level of external radiation and absence of radioactive contamination on its surface.

-77-

44.   When disturbance of integrity (or disappearance of the ICS) is discovered, the organ of the militia, the Sanitation Service, and the specialized organization should be promptly informed and measures taken to erase the damage and locate the ICS.

45.   When discontinuing a weather station when it is no longer needed or owing to failure of the ICS which cannot be repaired on site, the ICS should be delivered   to its place of manufacture for repair or to the nearest point of radioactive waste disposal. Such work should be performed only by specially trained personnel. Shipping of the ICS for these purposes should be carried out in the shipping container. When using the ICS in the working container, it should be placed in the shipping container at the nearest place available for delivery of the shipping container.

46.   Before changing the thermopile or before repair of a non-functioning ICS on site, it is necessary to convince oneself of its integrity.

Literature

1. Chang, Sh., *Preobrazovanive enernii* (Energy transformations), Atomizdat, Moscow, 1965.

2. Orlov, V.A., *Malooabaritnyye istochniki toka* (Small-sized current sources), Voyennoye izd-bo MO SSSR, 1965.

3. Baootskiy, V.S., V.N. Flerov, *Noveyshive oostizheniya v oblasti khimicheskikh istochnikov toka* (Latest achievements in the area of chemical sources of current), Gosenernoizdat, Moscow-Leninorad, 1963.

4. *Otkhooy atomnoy oromyshlennosti* (Wastes of atomic industry), Gosatom-izdat, Moscow, 1963.

5. Aolintsev, K.K. et al, *Prikladnava dozimetriva* (Aonlied Oosimetry), Atomizdat, Moscow, 1962.

6. Gusev, N.G., *O predel'no donustimykh urovnyakh ioniziruvushchikh izlucheniy* (Concerning the toleraole levels of ionizino radiation), Meonis, Moscow, 1961.

7. Ardashnikov, S.N., et al, *Zashchita ot radioaktivnykh izlucheniy* (Protection from radiation), GNTizdat literaturv oo chernoy i tsvetnoy metallurnii, Moscow, 1961.

8. Gorshkov, G.V., *Gamma-izlucheniye radioactivnykh tel.* (Gamma-radiation of radioactive materials), Izd-vo LGU, Leninorad, 1966.

9. Fradkin, G.M., V.M. Kodvukov, A. Rocvinskiy, *"Izotoonvve istochriki elektricheskoy enernii"* (Isotope sources of electrical enernv), Trudy VNiIP vyp. 1, Atomizat, Moscow, 1967.

10. Ioffe, A.F. *Enercheticheskive osrovv termoelektricneskikn oatarev iz oolurovodnikov* (Enerov orinciples of thermoelectric semi-corouctor oatteries), Izd-vo AN SSSR, Moscow-Leninorad, 1950.

11. Ioffe, A.F., *Poluprovodnikovyve termoelementy* (Semiconductor thermoelements), Izd-vo AN SSSR, Moscow-Leninorad, 1960.

12. Bursntevn, A.I., *Fizicheskive osrovv rascheta poluprovodnikovvkh termoelektricheskikh ustrovstv* (Phvsical orincioles of the calculation of semiconductor thermoelectric devices), Fizmatniz, Moscow, 1962.

13. Kutateladze, S.S., *Osrovv teorii teplonomena* (Principles of the theory of heat exchnooe), Mashoiz, Moscow-Leninorad, 1962.

14. Shneyder, P. Inzhenernyye problemy teploprovodnosti (Engineering problems of thermal conductivity), Izd-vo inostr. lit., Moscow, 1958.

15. Fradkin, G.M., V.M. Kodyukov, "Radioizotopnyye termoelektricheskiye generatory" (Radioisotope thermoelectric generators), Atomnaya energiya 26, p. 169, 1969.

16. Corlis, U., D. Harvy, Istochniki energii na radioaktivnykh izotopakh (Energy sources based on radioactive isotopes), "Mir", Moscow, 1967.

Contents

Best Available Copy