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**FINAL SAFETY EVALUATION
OF A SIXTY KAT STROMTIUM 90
FUELED GENERATOR FOR A
U. S. COAST GUARD AUTOMATIC LIGHT STATION**

**SNAP 7B
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CONTENTS

CONTENTS	11
SUMMARY	111
I. Introduction	1
II. System Application, Installation and Configuration	3
A. Application and Installation	3
B. Configuration	4
C. Radiostrontium Fuel	5
III. Analytical Safety Evaluation	9
A. System Integrity	9
B. Fuel Solubility Analysis	15
C. Shielding Analysis	16
D. Release Analysis	21
IV. Quality Control	26
V. Conclusion	29
References	30
Appendix A - System Structural Calculations and Impact Analysis	32
Appendix B - Thermal Integrity Calculations	35
Appendix C - Discussion of Potential Accidents	36
Appendix D - Maintenance Requirements	39

SUMMARY

A safety evaluation was performed to determine the mechanical, thermal and radiation integrity of the thermoelectric generator. The evaluation included analyses to assess the radiological effects of an accidental fuel release, should the unit be dropped in a water environment. The shielding of the generator was evaluated to determine the safe working limits for personnel required to be near the generator. The detailed areas investigated and the results are as follows:

- A. The radiostrontium fuel is confined by two separate containers designed to withstand the external pressure occurring at 12,000 feet of water. Hydrostatic tests have been performed on simulated inner containers to substantiate the analytical study.
- B. The thermal integrity of the unit will be maintained in the event of soil burial or a standard one-hour fire as defined by the AEC.
- C. The generator unit can withstand forces imposed upon impact during transportation by truck carrier.
- D. The chemical stability and relative insolubility of the fuel compound prevents it from causing a radiobiological hazard even if containment is lost in a sea environment.
- E. Corrosive action of sea water combined with the maximum hydrostatic pressure will expose the fuel in approximately 510 years. By that time, the activity of the fuel will have decayed to less than one curie.
- F. The radiostrontium is shielded sufficiently for shipment under I.C.C. regulations. In addition, personnel can work on or around the unit's surface for a sufficient length of time to

make extensive repairs if necessary.

- G. Ionizing radiations from the fuel will have a negligible effect upon the structural integrity of the generator materials.
- H. Strict quality control procedures have been adhered to throughout manufacturing and assembly to insure conformance with all specifications.

Within the framework of this analysis, it is concluded that the safety criteria are met and there is reasonable assurance that this generator is safe for its intended use.

I. INTRODUCTION

This report presents a safety evaluation of the SNAP 7B thermoelectric generator system which is to be used as the power supply for an unattended automatic light station, a U. S. Coast Guard navigational aid. The generator produces electrical power from the energy released by the decay of the radiostrontium fuel.

The safety evaluation is based upon analytical procedures and test results which determine whether the safety criteria established for this system and its mission have been satisfied. These criteria were developed by evaluating the potential radiation hazards posed by the use of this radiostrontium fueled system. Potential hazards were defined as follows:

1. Internal radiation exposure from the ingestion and inhalation of the radiostrontium fuel as a consequence of loss of the fuel containment structure.
2. External radiation from surface contamination by the radioactive material.
3. External radiation exposure resulting from the inadequacy or loss of the biological shield.

Consequently, the safety criteria considered in the design and use of this generator system were:

1. Absolute fuel containment in the generator's operating environment and en-route to this environment under normal and accidental conditions.
2. Effective biological shielding and/or exposure limitations to the potentially hazardous radiations emitted by the fuel as defined by current Code of Federal Regulations and N. B. S. Handbook #69.

Absolute containment of the fuel is an important safety objective and is determined by means of a system integrity study. The effectiveness of the biological shield is determined by means of a shielding analysis including calculated dose rates and exposure limitations for the assembled generator system. Further, a dispersion analysis is included to determine the consequences of a fuel release to the operating environment.

A series of hydrostatic pressure tests were conducted to substantiate the findings of the structural analysis. Simulated fuel capsules withstood external pressures far greater than those which will be encountered during operation.

II. SYSTEM APPLICATION, INSTALLATION AND CONFIGURATION

A. Application and Installation

The SNAP-7B radiostrontium-fueled generator will supply electrical power for an unattended automatic light station, a U. S. Coast Guard Navigational aid. For purposes of analysis the generator is considered to be fueled with 245,000 curies of radiostrontium which produce 1546 thermal watts. Lead telluride thermoelements and a DC-DC converter transform the thermal energy into useful electrical power.

The first SNAP-7B system unit is scheduled to be installed at the U. S. Coast Guard Test Station at Curtis Bay, Maryland for functional evaluation. Figure 1 shows the location of the generator at the station. Plans for the performance evaluation at Curtis Bay include placing the 60-watt generator in a liquid-filled tank suitable for heat transfer purposes. The tank and other auxiliary equipment will be tied securely to a cement pad and the electrical output will be used for test purposes and evaluation of the system. Control of personnel in the vicinity of the SNAP-7B system, while it is located at Curtis Bay, will not pose any problems since this location is part of a federal military reservation.

The exact final location is not known at this time but it will be at one of the automatic light stations managed by the U.S. Coast Guard. The U. S. Coast Guard Aids to Navigation Manual, CG-222 describes automatic light stations by stating "These are flashing lights of relatively low candlepower, usually established in harbors, along channels, rivers and isolated locations. They are generally unattended and unwatched and operate automatically." Figure 2 shows a concept of a light station.

Fig. 1 Locations of SNAP 7A and 7B Within Coast Guard Yard and Location of Yard in Curtis Bay Area

Figure 2
Minor Light Station

B. Generator Configuration and Assembly

The strontium titanate fuel is compacted into pellets of 1.48 inch diameter and 0.6 inch height. These pellets are encased in a cylindrical Hastelloy C fuel capsule having a nominal wall thickness of 0.181 inch. Each of the 14 fuel capsules contains 9 fuel pellets, making a total of 126 pellets. The top cap is welded on the fuel capsule in a helium atmosphere, permitting a non-oxidized weld to be effected and facilitating helium leak testing of the weld integrity. The capsules are inserted in a Hastelloy C heat accumulator having a diameter of 6.25 inches, a length of 18 inches and a minimum wall thickness of 0.075 inch. The heat accumulator is supported at its ends in slabs of Min-K insulation approximately 1.5 inches thick. Against flattened areas on its outer wall are 120 pairs of lead telluride thermoelectric elements, each approximately 1 inch in length. The outer ends of the thermoelectric elements terminate in an aluminum heat sink which is a cylindrical ring 0.875 inch thick and 18 inches long. The generator assembly is evacuated and then filled with an inert gas at atmospheric pressure. This procedure is designed to prolong the life of the thermoelectric elements.

The shield container is a double-walled Hastelloy C vessel with 0.25 inch thick inner and outer walls. The biological shield consists of 3.25 inches of depleted uranium between the walls. An additional uranium shield is placed above the generator, within the shielding vessel and is held in place by a 0.55 inch thick Hastelloy C plate welded to the top of the shielding vessel. Figure 3 shows detailed views of the generator unit.

Section A-A

FIGURE 3

Surrounding the generator unit is a finned aluminum container which is used to provide the additional heat rejection surface necessary to maintain the correct operating temperature of the generator.

C. Radiostrontium Fuel

Strontium was selected as the fuel for this generator application because of its long half-life, availability and ability to be processed into a highly insoluble compound.

Strontium-90 is a fission product of Uranium-235. It is recovered from reactor waste and shipped, in cake form as strontium carbonate from Hanford Atomic Processing Plant to ORNL, where it is processed into pellets of strontium titanate. Both strontium-89 and strontium-90 are present in the fuel at the beginning of generator life. However, the Sr-89 decays rapidly and is of little consequence in this safety evaluation. The thermal energy of the fuel is derived mainly from absorption of the betas produced in the decay chains shown in Figure 4.

Strontium titanate is a chemically stable compound and is effectively insoluble in natural reagents. The titanate forms an eutectic with titania and forms solid solutions with the titanates of barium and calcium. The metatitanate form predominates, although an orthotitanate can be prepared.

In the radioactive disintegration of Sr-90, the daughters Yttrium-90 and Zirconium-90 are formed. Since the valence of the cation increases, there is an oxygen deficiency. Consequently, yttrium and zirconium may exist in the fuel in an uncombined state. At the present time, there is no available data concerning the effect of fuel composition variation on solubility. A program of this nature has been proposed.

a. Decay Scheme of Strontium-90

b. Decay Scheme of Strontium-89

FIGURE 4
Decay Scheme of Sr^{89} and Sr^{90}

The raw feed materials will inevitably contain some Sr-89. Since the half life of this isotope is relatively short, it can affect the overall performance of the system. Therefore, the thermal output of such contaminants is limited to a maximum of 90 thermal watts at the time the generator is fueled. Other contaminants are limited during processing as follows:

a. beta emitters (eg. cerium-144) - 0.15 curies/gm Sr-90

b. gamma emitters

0.2 to 1 mev - 0.15 curies/gm Sr-90

1 to 2 mev - 0.015 curies/gm Sr-90

2 mev or greater - 0.0015 curies/gm Sr-90

The principal properties of the SrTiO_3 fuel form used in this generator are listed in Table 1. At the present time, very little is known of the mechanical properties of the compound at elevated temperatures or after long periods of decay and irradiation.

TABLE 1

Properties of Strontium Titanate

Molecular weight	- 183.5
Color	- white, glassy
Melting point	- 2080°C (3776°F)
Crystal structure	- cubic (persovskite)
Theoretical density	- 5.175 g/cc
Practical density	- 3.62 g/cc*
Thermal conductivity	- 0.0116 cal/sec-cm-°C
Specific heat	- 0.143 cal/g-°C
Coefficient of thermal expansion	- 1.12×10^{-5} in/in-°C
Specific Activity (theoretical)	- 83 curies/g
Specific Activity (practical)	- 31.4 curies/g*
Specific Power (theoretical)	- 0.456 watts/g
Specific Power (practical)	- 0.20 watts/g*
Power Activity Constant	- 6.3×10^{-3} watts/curie
Power density (theoretical)	- 2.36 watts/cc
Power density (practical)	- 0.724 watts/cc*
Activity density (theoretical)	- 430 curies/cc
Activity density (practical)	- 114 curies/cc*

*Based on 70% of theoretical density as a minimum value quoted by ORNL

III. ANALYTICAL SAFETY EVALUATION

An analytical treatment applied in determining the nuclear safety of a radionuclide-fueled generator involves many factors. However, by establishing safety criteria for design, handling and use, the task can be circumscribed. Analyses were performed to determine the system integrity, fuel solubility, shielding effectiveness and the consequences of an underwater activity release.

A. System Integrity

1. Hydrostatic Pressure Test

During transportation to its final destination, it is conceivable that the generator could be accidentally lost to a sea environment (See Appendix C). The maximum hydrostatic pressure that the generator could be exposed to is 5,350 psi or an equivalent depth of 12000 feet of sea water.

The radiostrontium is encased in two independent containers, the fuel capsule and the shield container. Each will individually contain the fuel in the event of an accidental burial at sea. In analyzing the outer container, the top plate was found to be the critical member. The area on which the hydrostatic pressure acts is the unsupported 4 X 4 inch square at the plate center. The bending stress was 86,500 psi which is below the ultimate tensile strength of 125,000 psi for Hastelloy C plate (Ref. 1).

The evaluation then considered that the outer shield container failed and the individual fuel capsules were exposed to the hydrostatic pressure of 5,350 psi. The shearing and bending stresses in the capsule

cap were investigated and the shearing stress in the area of the weld was taken to be the critical stress. A minimum of 0.090 inch is specified for the weld penetration. However, on the capsules tested, a 0.083 inch weld penetration was obtained. This value was used in the analysis to correlate the test data with the calculations. It should be noted that the weld penetration was as low as 0.064 in one of the test capsules and still maintained its integrity during the test program. Such a capsule, however, would have been rejected if loaded with radioactive strontium titanate. The shearing stress based upon the 0.083 inch weld penetration and a depth of 12,000 feet is 24,400 psi. The allowable shear stress for Hastelloy C material is 62,500 psi.

The results of the analysis indicate that both containers will withstand the external hydrostatic pressure if accidental burial at sea occurs. The mechanical properties of the materials are reported in reference 1. The calculations for the above analysis are shown in Appendix A.

Several tests were performed on simulated fuel capsules and indicate that the above analysis on the fuel capsule is conservative. Two capsules, one containing inert strontium titanate pellets and the other empty, were tested under various pressures. The capsule containing the pellets was first subjected to 4,000 psi and then removed for inspection and calibration. Measurements taken showed no permanent deformation in the cap or side walls. The capsule was returned to the pressure apparatus and the pressure raised to 9,000 psi for 23.5 hours. Measurements taken after removal showed a permanent deflection of 0.004 inch at the vertical centerline of the capsule but no deflection on the diameter of the capsule shell.

The empty capsule was tested at external pressures of 6,000 and 10,000 psi and then checked for permanent deflections. The results of the tests showed no deflection on the diameter but a maximum deflection of 0.010 inch after 20 hours under a hydrostatic pressure of 10,000 psi. Although the tests were conducted in relatively short periods of time, the results over an extended period of time would be the same. Data on Hastelloy C indicate that rupture stress does not vary as a function of time unless the material is subjected to high temperatures (Ref. 2). The temperature of the capsule submerged in water should not exceed 200°F. The pressure tests indicate that the fuel capsule is capable of withstanding greater pressures than the maximum which the generator could accidentally encounter.

2. Impact

The most stringent impact condition would occur if the generator were subjected to an accident during transportation by truck. A hypothetical accident considered was one in which the carrier truck traveling at 40 mph impacted with another truck traveling in the opposite direction at a speed of 60 mph (see Appendix C).

The impact analysis assumes that the mounting provisions fail and the generator impacts on a rigid medium at a velocity of 146 fps. The pressure on the surface of impact is 3340 psi as calculated from the method shown in reference 3. The pressure would move through the shield material and decrease in intensity. However, the shield material is assumed to be under a uniform external pressure. The tensile yield stress of uranium is 27,000 psi (Ref. 1), which is well above 3340 psi (see Appendix A for calculation).

3. Thermal Integrity

The thermal properties of the materials comprising the SNAP-7B generator are shown in Table 2. All thermal integrity calculations are shown in Appendix B.

a. Ground Burial

Although ground burial is unlikely, it could conceivably occur as the result of a truck, rail or air transportation accident. The worst possible case would occur if the entire generator system (including shield and outer container) were buried in the soil intact.

To determine the temperature of the outer vessel wall in the event of soil burial, a spherical model of equivalent surface area was considered. This approach yields a conservative result.

From the equation for concentric spherical shells surrounding a heat source (Ref. 4), the steady state temperature differential between the shield surface and soil is found to be 1040°F . The calculations assumed that the generator is buried in an infinite soil medium. The thermal conductivity of the medium was conservatively selected as that of a common soil, Healy Clay which has a low thermal conductivity when compared with other common soils (Ref. 3). There are other soils that have lower thermal conductivities and consequently would result in higher equilibrium temperatures. However, these soils are of a relatively rare dry sand type and are found only in some interior areas of the country. Due to the nature of this generator's mission, it is highly improbable that this type of soil would ever be encountered. Thus, the outer vessel wall of the generator, when buried in soil, will reach a steady state temperature of $1040 + 40 = 1080^{\circ}\text{F}$.

TABLE 2
THERMAL PROPERTIES OF THE GENERATOR COMPONENTS

Component	Material	Melting Point °F	Specific Heat Btu'a/lb-°F	Weight lbs.	Heat Equivalent Btu'a/°F	Average Working Temperature °F	Btu's to Reach 1700°F Btu's
Shield	Uranium	2070	0.028	3570	100	100	160,000
Shield Container	Hastelloy C	2350	0.092	315	29	90	46,700
Inner Shield Conductor	Solder 50% Pb, 50% Sn	425	0.289	42	12.1	100	19,600*
Outer Shield Conductor	Solder 50% Pb, 50% Sn	425	0.289	74	21.4	100	34,500*
Heat Sink	Aluminum	1220	0.260	45	11.6	150	25,750*
Heat Accumulator	Hastelloy C	2350	0.092	75	6.87	1000	4820
Misellllaneous				142			
Totals				4263			291,370

*Includes heat of fusion

The temperature drop across the shield cylindrical wall was then considered. The temperature drop across the Hastelloy C inner and outer container was considered to be small when compared to the thick uranium shield. The temperature drop across the shield was found to be 15°F. Thus, the temperature at the inner surface of the shield would be 1095°F, well below the melting point of the materials.

Elevated temperature properties of lead telluride and Min-K are not well known. At temperatures above 1200°F, PbTe decomposition is rapid. If the lead telluride decomposes or melts (m.p. = 1690°F), the major path of heat conduction from the fuel capsules will be removed. This will cause the fuel capsule temperature to rise until the Min-K decomposes or the capsule container melts. It is probable that the Min-K will decompose prior to melting the Hastelloy C fuel container. If this were not to occur, it is unreasonable to assume that the strontium titanate fuel could melt (m.p. = 3776°F). Even if one could postulate melting of the titanate fuel, if the shield was intact, the fuel would not escape from the generator since it would solidify upon approaching the generator surface.

b. Fire

Although the mission of this generator precludes the occurrence of fire during operation, the possibility of its being involved in a fire during transportation, however remote, must be recognized. Two possible accidents that could occur are truck impact with a fuel carrier and fire aboard a transport ship. In either case, both the shield and the Hastelloy C fuel capsules would have to melt before the SrTiO_3 fuel could be released to the environment. Thus, the temperature of the entire generator system would have to reach 2350°F, the melting point of Hastelloy C, before a fuel release could occur.

A standard one-hour fire has been described by the National Fire Protection Association (Ref. 5) and the American Society for Testing Materials (Ref. 6). This standard fire assumes a one hour exposure to a fire temperature of 1700°F. Since the generator system must reach 2350°F for meltdown to occur, it is obvious that the standard one-hour fire would not result in loss of containment.

4. Radiation Damage to Containment Materials

An investigation was conducted to determine the effects of the radiations emitted by the fuel upon the containment materials. The sources of potential radiation damage are the 2.26 Mev betas, x-rays produced by the bremsstrahlung, and the gammas of Yttrium-90, a daughter of Strontium-90. The maximum range of the betas in the Hastelloy-C fuel capsule will be 0.045 inch. Since the minimum wall thickness of the capsule is 0.181 inch, the betas cannot penetrate more than one-fourth of the wall thickness. The radiation effect to the inner surface of the capsule would be similar to the effects produced by cold working the material (Ref. 1). The tensile and yield strengths of the material increase while the ductility decreases.

The gamma and x-radiation will be attenuated by both the capsule and shielding materials. The amount of damage, if any, caused by these radiations during the life of the generator will be negligible (Ref. 7).

B. Fuel Solubility Analysis

The fuel form with the radiostrontium combined as the ortho-titanate, SrTiO_3 , was chosen principally because of the relative insolubility of this compound. A series of tests were conducted to

determine the solubility of this fuel form in sea, tap, and deionized water. The simulated fuel pellets were prepared from stable strontium. Solubility measurements were made by flame photometry and tracer methods. Flame photometry has limited sensitivity at the low concentrations involved and was further handicapped during the sea water measurements due to the fact that natural sea water contains about 8 ppm of strontium (Ref. 8). For the tracer tests the simulated fuel pellets contained Sr-90 in small quantities. This method offers greater precision than flame photometry but the tracer concentration was low and the counts obtained were not much above background level. This was especially apparent in the sea water tests where the background count is increased by the natural radioactivity of Potassium-40. The results of the various tests are summarized in Table 3.

The most significant solubility measurements were those made by tracer methods on fuel pellets held in sea water at 60°C for 100 days. Three pellets, having surface areas of 4.56, 2.17, and 1.97 cm², showed solubilities of 115, 30, and 15 parts per billion Sr-90 respectively. The average rate of solution of the titanate compound was found to be about 1.6×10^{-6} gm/cm² - day (Ref. 8). However, because of the uncertainty of some of the tests results and the fact that the initial rate of solution may be somewhat higher than the average, a solubility rate of 3.2×10^{-6} gm/cm² - day, or twice the average rate, was used for calculations in this safety evaluation.

C. Shielding Analysis

Biological shielding against radiation from the radiostrontium fuel is provided by 3.25 inches of depleted uranium on the sides and top of the generator.

TABLE 3 (Ref. 8)
SUMMARY OF STRONTIUM TITANATE SOLUBILITY

A. In Water

<u>Run No.</u>	<u>Method of Analysis</u>	<u>Environmental Solution</u>	<u>Exposure Temp. (°C)</u>	<u>Exposure time (hr)</u>	<u>Volume of Solution (cc)</u>	<u>Titanate Sample Weight (gm)</u>	<u>Titanate State</u>	<u>Solubility Strontium (ppmSr)</u>
1)	Flame photometry	Sea Water	100	238	250	1.10	Pulverized	1.4
2)			100	336	250	4.98	Pulverized	30.
3)			100	168	250	1.26	Pulverized	0.02
4)			20	2208	100	8.195	Pellet	16.
5)			20	2208	100	7.936	Pellet	13.
6)	Tracer	Sea Water	65	2448	3786	2.608	Pellet	0.113
7)			65	2448	3786	1.241	Pellet	0.042
8)			65	2448	3786	0.730	Pellet	0.021
9)	Flame photometry	Tap Water	100	168	250	1.99	Pellet	0.02
10)			100	168	250	0.63	Pulverized	0.02
11)			20	2208	100	8.079	Pellet	2.
12)			20	2208	100	8.058	Pellet	0.02
13)	Tracer	Tap Water	65	2400	3786	Not reported	Pellet	0
14)	Flame photometry	De-ionized water	100	168	250	2.09	Pellet	0.02
15)			100	168	250	1.40	Pulverized	0.02
16)			20	2208	100	8.087	Pellet	6.
17)			20	2208	100	7.913	Pellet	3.

TABLE 3 - (Cont) - (Ref. 8)

B. In Reagents*

Concentration	Exposure Time (hr)	Solubility ppm					Method of Analysis
		HNO ₃	HCl**	H ₂ SO ₄	NH ₄ OH	NaOH	
0.1N	336	2	2	4	1	3	Flame photometry
6 N	336	50	625	0	0	245	Flame photometry
0.1 N	750	2	2	5	2	4	Flame photometry
6 N	750	85	660	0	5	150	Flame photometry

* Exposure Temperature - 49°C

** Gastric Juice - approximately 0.1 N HCl, P_H - 1.5 - 2.0

An analysis was performed to determine the dose rates on the generator surface and at a distance of 1 meter from the centerline. The results of the dose rate calculations are shown in Table 4 and are below the maximum permissible levels for transportation as defined by ICC regulations (Ref. 9) which permit a maximum dose of 200 mr/hr on the surface and 10 mr/hr at a distance of 1 meter from the centerline.

The shielding analysis considers the three types of radiation emitted by the radiostrontium fuel; beta, gamma, and X-rays (resulting from bremsstrahlung). The beta particles are absorbed within the fuel capsule. The principal beta energies and their ranges in the fuel and cladding materials are shown in Table 5. However, beta emitters may be effective x-ray sources because of the bremsstrahlung generated as the electrons decelerate in the atomic field (Ref. 11). The magnitude of this radiation energy is dependent upon the initial beta energy, the absorber, and the resultant x-ray spectrum.

An average bremsstrahlung energy of 236 kev was calculated for the 2.26 mev beta from Y-90 (Ref. 12). The bremsstrahlung from the 0.545 mev beta of Sr-90 was neglected since it is in the low kilovolt range and will be attenuated within the first few mils of shielding.

A 1.734 mev gamma is emitted from the daughter, Zr-90m, but it is infrequent (0.02% of Y-90 disintegrations). The impurity, Sr-89 produces a 0.915 mev gamma in 0.02% of the disintegrations. Due to the infrequency of the gamma radiation, it was not included in shielding analysis.

Based upon the calculated dose rates in Table 4, a man can work around the generator (at one meter distance) for 71 days, assuming an 8 hour working day, without exceeding his permissible quarterly whole body dose of 1-1/4r (Ref. 13). He can work in direct contact, his hands on,

TABLE 4
CALCULATED DOSE RATES FROM GENERATOR ASSEMBLY

<u>Location</u>	<u>Shielding</u>	<u>Dose Rate, mr/hr</u>	
		<u>Surface</u>	<u>One Meter from Centerline</u>
Top	3.25 in. of depleted uranium	11	2.2
Side	3.25 in. of depleted uranium	39	2.2

TABLE 5
BETA ENERGIES AND RANGES (Ref. 10)

<u>Beta Energy (mev)</u>	<u>Material</u>	<u>Density (g/cc)</u>	<u>Range₂ (mg/cm²)</u>	<u>Maximum Penetration Distance</u>	
				<u>(cm)</u>	<u>(in)</u>
0.545(Sr-90)	SrTiO ₃	4.8	175	0.035	0.013
	Hastelloy C	8.9	175	0.019	0.007
2.26(Y-90)	SrTiO ₃	4.8	1000	0.219	0.086
	Hastalloy C	8.9	1000	0.118	0.046

on the generator (with the rest of his body shielded), for a period of 60 eight hour days without exceeding his permissible quarterly local dose of 18-3/4r.

D. Release Analysis

The structural integrity analyses have shown that an activity release to the atmosphere is not credible. In addition, calculations indicate that if the fuel capsules were to be exposed to a sea water environment of 12,000 feet, it would take approximately 500 years (18 8r-90 half-lives) before a release of the fuel would occur. This analysis was based upon a corrosion rate of 0.0001 in/year for Hastelloy C and a cap weld penetration of 0.083 inch (see Section III-A-1). The calculations are shown in Appendix A for the minimum wall thickness required to withstand the 5350 psi hydrostatic pressure.

If, however, an improbable accident did occur, causing rupture of the fuel capsule and release of the pelletized fuel to the sea, the rate of solution would be of greater significance than the absolute solubility or total activity. The fuel pellets have a total surface area of about 5100 cm². Using the solution rate derived from the solubility tests, a fuel erosion rate of 1.63×10^{-2} g/day is obtained. The specific activity of the fuel at the time of loading is 31.4 curies/gm. The rate of activity release would then be 5.12×10^{-1} curies/day or 5.93×10^{-6} curies/sec.

Diffusion in the ocean is limited almost completely to horizontal layers. This is particularly true below the thermocline (500 meter depth) where temperature gradients are small but regular and there is

little seasonal variation (Ref. 14). One investigator released fission products at a considerable depth in the ocean and was able to trace the diffusion over an area of 100 km^2 . The vertical thickness of the diffusion layer was approximately one meter (Ref. 15). The rate of horizontal diffusion varies from 0.3 to 1.0 cm/sec (Ref. 14, 16, 17).

If a small source continuously releases fission products in the ocean at the rate of Q curies/sec, an equilibrium state will be reached. At a distance, r , the activity will be passing through an area of $2\pi r h \text{ cm}^2$ at the same rate as it is being released.

Then:

$$D = Q/2\pi r h \quad \text{eq. (1)}$$

where: D = diffusivity coefficient, curies/ cm^2 - sec.

h = thickness of diffusion layer, cm.

The concentration at any point, distant r , from the origin is then:

$$X = D/K = Q/2\pi K r h \quad \text{eq. (2)}$$

where: X = activity concentration, curies/ cm^3

K = diffusion velocity, cm/sec.

The quantity, h , depends upon the heat released at the origin. The generator produces about 1546 thermal watts or 370 cal/sec. This quantity of energy is capable of raising the temperature of a column of water 5 cm in diameter and 190 cm high at the rate of 0.1°C per second. The thermal column produced would be limited by the heat available and by the temperature gradient in the ocean.

Let $h = 190 \text{ cm}$ and $K = 0.5 \text{ cm/sec}$ (Ref. 17). Then eq. (2) gives an activity concentration in the water, 10 meters from the origin, of $9.9 \times 10^{-6} \mu\text{c}/\text{cm}^3$. This analysis considers only diffusion, not advection.

The maximum recommended concentration in water for Sr-90 is $10^{-6} \mu\text{c}/\text{cm}^3$. But man does not normally drink ocean water. The dose rate, if one were to bathe in the water, would be less than $1/\text{rem}/\text{hr}$. Fish and other marine organisms which may enter the diet, do take up strontium from the water. The concentration factor for the fish body compared with the ocean water is 10 (Refs. 14, 18, 19). However, the fish comes into equilibrium with the element, not the isotope. The ocean abundance of stable strontium is relatively high, so that the isotope dilution factor is small and a fish in equilibrium with water of this concentration would contain an insignificant amount of radioactivity. (Isotope dilution factor is calculated by dividing the atomic concentration of the particular isotope by the atomic concentration of the element as a whole).

This analysis considers the generator early in its life. For a later time, the value of Q should be modified by the exponential decay rate. Conversely, the fuel may lose its ceramic form and the solution rate may increase. However, an increase of several orders of magnitude would be required to produce a significant hazard.

In considering the radiobiological hazards of a Sr-90 release, a clear distinction must be made between the soluble and insoluble forms. For all practical purposes, the properties of insoluble Sr-90 apply to the titanate, while the soluble properties apply to the minute quantities of Sr-90 which are dissolved from the fuel under various conditions.

The critical organs for insoluble strontium are the lungs and lower intestine. The lungs might be exposed through inhalation of Sr-90, which need not be considered for this application. Exposure of the

intestine could be brought about by ingestion of the fuel. The maximum permissible exposures are given in Table 6 (Ref. 20).

Since strontium and calcium are chemically similar, the biological paths of both are similar. Soluble radiostrontium, when ingested, deposits in the bone by exchanging with the calcium and then serves as an internal source of radiation to the bone and blood forming mechanism. The average human body contains about 1 kilogram of calcium and 0.7 gram of strontium (Ref. 21). The Sr-90 bone concentration is commonly expressed in Strontium Units ($1 \text{ SU} = 1 \mu\text{c Sr-90/gm Ca}$). At present, the average concentration of Sr-90 in the bones of children is about 1.5 SU. This is equivalent to a bone marrow exposure of about 3 mr/yr. The maximum permissible body burden for the general population is 67 SU, resulting in a bone marrow exposure rate of 134 mr/yr (Ref. 21).

TABLE 6
MAXIMUM PERMISSIBLE Sr-90 AND Y-90 CONCENTRATIONS IN AIR AND WATER
(Ref. 20)

Radionuclide and Type of Decay	Form	Organ of Reference	Maximum Permissible Concentration ($\mu\text{c/cc}$)			
			40-hour week		168 hour week	
			Water	Air	Water	Air
^{90}Sr	Soluble	Bone	4×10^{-6}	3×10^{-10}	10^{-6}	10^{-10}
		Total Body	10^{-5}	9×10^{-10}	4×10^{-6}	3×10^{-10}
		G I (LLI)*	10^{-3}	3×10^{-7}	5×10^{-4}	10^{-7}
	Insoluble	Lung	--	5×10^{-9}	--	2×10^{-9}
		G I (LLI)	10^{-3}	2×10^{-7}	4×10^{-4}	6×10^{-8}
^{90}Y	Soluble	G I (LLI)	6×10^{-4}	10^{-7}	2×10^{-4}	4×10^{-8}
		Bone	10	5×10^{-7}	4	2×10^{-7}
		Total Body	80	3×10^{-6}	30	10^{-6}
	Insoluble	G I (LLI)	6×10^{-4}	10^{-7}	2×10^{-4}	3×10^{-8}
		Lung	--	3×10^{-7}	--	10^{-7}

*G I (LLI) = Gastrointestine (Lower Large Intestine)

IV. QUALITY CONTROL

To insure the integrity of the generator unit, rigid quality control procedures will be maintained during and after assembly of the various components. In general, the procedures follow the Quality Control System Requirements of Military Specification MIL-Q-9858.

The specific procedures performed by the Martin Marietta, Nuclear Quality Control Department are:

1. Periodically inspect and calibrate all gages and precision measuring instruments used in manufacturing.
2. Maintain 100% surveillance of the Certificate of Qualification Card of welders to insure their validity. Welding is performed by personnel certified under an Air Force controlled Martin Marietta certification program. Where ASME welding is specified, the welder is ASME certified. Wherever practical, sample welds are made and analyzed to verify repeatability.
3. Materials are ordered as specified in the drawings. Alternate materials are specified when feasible, to reduce costs and improve delivery schedules. All substitutions must be approved by Engineering. All incoming materials, parts, or units are checked for dimensions, physical conformity, workmanship, and shipping damage. In addition, all raw materials are analyzed chemically to verify composition.
4. Visual and dimensional inspection as well as required chemical and physical tests are performed as specified. Dye penetrant tests are performed per Martin Marietta Process MIL-I-6866 and MIL-I-6868. Radiograms are performed wherever possible. Special consideration is given to welded assemblies requiring

absolute containment. All seal welds are helium leak tested and must meet the following requirement: "There shall be no leakage when tested with equipment having a sensitivity of 1×10^{-7} cubic feet of helium per hour at an absolute pressure of 4 mm of mercury and 70°F ambient temperature."

5. Quality Inspection Log Control is maintained to document in-plant inspection steps during manufacturing, assembly, and tests.

Specific quality control procedures performed on the capsules and shield by Martin Marietta, Nuclear Quality Control are:

1. The uranium shield material was visually inspected for surface imperfections and measured for correct dimensions. It was not possible to radiograph the shield after assembly due to its large mass. The material specifications for the depleted uranium, which was government furnished equipment, stated that the minimum attenuation measured at any point shall equal to 95% of the calculated attenuation of 100% pure uranium. An additional check was made by the Martin Marietta Corporation to determine the density by accurately weighing the individual blocks. The results of this check indicated the density of the blocks to be a minimum of 97.5% of the theoretical density. After final assembly, the generator will be monitored to verify material density and shielding analysis.
2. All welding of the biological shield container is performed by personnel certified under ASME Boiler Code (see part 2 above).
3. Acceptance of the welds is based upon liquid dye penetrant checks performed to specifications MIL-I-6866 and MIL-I-6868.

4. Voids in the solder bonds between the uranium shield and Hastelloy containers are checked by ASTM-E-113-55T. Voids in the bonds do not exceed 5 square inches and shall not be closer than 5 inches on the same surface between perimeters.
 - .. The total void does not exceed 10 per cent of the area.
5. Vendor's material certifications are reviewed by Martin Quality Control prior to fabrication of the capsule details. The materials certification report as prepared by the Haynes Stellite Company states the following about the capsule material:
 - a. Hastelloy alloy C satisfies all requirements of specification AMS-5750.
 - b. The results of the chemical analysis of the heat containing this material are:

Cr	N	Fe	Si	Co	Mn	V	Mo
15.4	3.75	5.24	0.55	1.73	0.39	0.22	15.18
 - c. A tensile test of the heat indicated an ultimate strength of 113,000 psi, yield strength at 0.2% offset of 54,100 psi and 55% elongation of one inch.
6. The top capsule weld will be performed by the Martin personnel certified under ASME Boiler Code (see part 2 above).
7. Acceptance of the weld is based upon a helium leak test as defined above.
8. Top or closure weld will be made remotely after fueling and the acceptance specification is by a helium leak test as defined above.
9. A full weld penetration is specified for the manually formed bottom weld and a weld penetration of 0.090 inch is specified for the remotely formed top weld.

V. CONCLUSION

Most of this safety analysis has been based upon objective data and information. Assumptions were necessary in some instances where experimental data were lacking and where possible were derived from objective information. In a few limited cases subjective assumptions, based upon reasonable estimates, were used. However, when all factors and results are considered in their interrelationships and in relation to the overall safety of the unit, it can be concluded that within the framework of this analysis, the safety criteria have been met and there is reasonable assurance that this generator is safe for its intended mission.

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APPENDIX A
SYSTEM STRUCTURAL CALCULATIONS AND IMPACT ANALYSIS

I. Hydrostatic Pressure

$$P = \frac{D \times C}{144}$$

where: P = Hydrostatic pressure at depth D = psi

D = Depth of water = 12,000 ft.

C_w = Specific weight of sea water = 64.3 lbs/ft³

$$P = \frac{12,000 \times 64.3}{144}$$

$$P = 5,350 \text{ psi}$$

II. Shield Container - Hydrostatic Pressure

1. Maximum Stress at Top Cover Plate

$$M = \phi P a^2$$

where: M = Maximum moment on square plate = in-lb/in

P = Uniform pressure on plate = 5,350 psi

a = Minimum width of plate = 4 in.

ϕ = Plate constant as a function of the plate length and width
= 0.0513

$$M = 0.0513 \times 5,350 \times 16$$

$$M = 4,400 \text{ in-lbs.}$$

$$I = \frac{bh^3}{12}$$

I = Moment of inertia of elemental section = in⁴

b = Width of elemental section = 1 in.

h = Plate thickness = 0.550 in.

$$I = \frac{1 \times 0.167}{12}$$

$$I = 0.014 \text{ in}^4$$

$$S = \frac{Mc}{I}$$

S = Stress on plate = psi

c = One-half the plate thickness = 0.275 in.

$$S = \frac{4,400 \times 0.275}{0.014}$$

S = 86,500 psi

III. Fuel Capsule - Hydrostatic Pressure

1. Maximum Shear Stress on Capsule Cap

$$S = \frac{Pr}{2t}$$

S = Shear stress = psi

P = Uniform load on cap = 5,350 psi

r = Radius of Cap = 0.755 in

t = Weld thickness = 0.083 in

$$S = \frac{5,350 \times 0.755}{2 \times 0.083}$$

S = 24,400 psi

IV. Shield Container - Impact

$$P = \frac{\rho v^2}{g144}$$

P = Pressure on impact surface = psi

ρ = Bulk density of unit = 725 lbs/cu. ft.

V = Velocity at impact = 146 fps.

g = Gravitational constant = 32.2 ft/sec-sec

$$P = \frac{725 \times 21,300}{32.2 \times 144}$$

P = 3340 psi

VI. Time of Capsule Cap Failure Due to Corrosive Action and External Pressure

1. Thickness of Cap at Which Failure Will Occur Due to Hydrostatic Pressure

$$t = \frac{Pr}{2S}$$

t = Cap thickness = in.

P = Uniform load on cap = 5,350 psi

r = Cap radius = 0.755 in.

S = Ultimate shear strength of Hastelloy C = 62,500 psi

$$t = \frac{5,350 \times 0.755}{2 \times 62,500}$$

$$t = 0.032 \text{ in.}$$

2. Time of Failure

$$T = \frac{t_1 - t_2}{K}$$

T = Time = years

K = Corrosion Rate = 0.0001 in/yr

t₁ = Original weld penetration = 0.083 in.

t₂ = Weld thickness at time of failure = 0.032 in.

$$T = \frac{0.083 - 0.032}{0.0001}$$

$$T = 510 \text{ years}$$

APPENDIX B

THERMAL INTEGRITY CALCULATIONS

I. Soil Burial

1. Equivalent Spherical Model

$$A_s = 4 \pi r_s^2 = A_c$$

where: A_s = Spherical model surface area = in^2

r_s = Spherical model radius = in.

A_c = Actual container surface area = 2700 in^2

r_s = 14.6 in.

2. Steady State Temperature Differential Between Container and Soil

$$\Delta T = \frac{\dot{Q}}{4\pi K} \left(\frac{1}{r_s} - \frac{1}{r_D} \right)$$

$\dot{Q}$ = Heat flow rate = $5.29 \times 10^3 \text{ Btu/hr}$

K = Thermal conductivity of soil = $0.33 \text{ Btu/hr-ft.-}^\circ\text{F}$

r_s = Radius of spherical model = 1.225 ft.

r_D = Outer radius of soil = ∞

$$\Delta T = \frac{5.29 \times 10^3}{4 \times 3.14 \times 0.33} \left(\frac{1}{1.225} - 0 \right)$$

$$\Delta T = 1040^\circ\text{F}$$

3. Steady State Temperature Differential Across Uranium Shield

$$\Delta T = \frac{\dot{Q}}{2\pi KL} \ln \frac{r_2}{r_1}$$

ΔT = Temperature differential = $^\circ\text{F}$

$\dot{Q}$ = Heat flow rate = $5.29 \times 10^3 \text{ Btu/hr}$

K = Thermal conductivity of uranium = $19 \text{ Btu/hr-ft.-}^\circ\text{F}$

r_1 = Shield container inner radius = 0.468 ft.

r_2 = Shield container outer radius = 0.781 ft.

L = Length of heat source = 1.51 ft.

$$\Delta T = \frac{5.29 \times 10^3}{2 \times 3.14 \times 19 \times 1.51} \ln \frac{0.781}{0.468}$$

$$\Delta T = 15^\circ\text{F}$$

APPENDIX C

DISCUSSION OF POTENTIAL ACCIDENTS

The potential accidents to which this generator is vulnerable can be classified as occurring either during transportation or operation.

1. Transportation Accidents

The frequency and magnitude of transport accidents involving the generator can be significantly diminished by the proper selection of shipping routes, optimization of shipping procedures, proper location of the generator on the carrier, regulation of shipping time and utilization of an escort. The four modes of transportation considered are truck, rail, water and air.

a. Truck

Truck accidents could involve collision, explosion and fire. The frequency of truck accidents has been estimated as 1/230 per 1000 mile shipment (Ref. 23). The frequency of such accidents involving explosions is 1/115,000 per 1000 mile shipment, and about 1/10,000 per 1000 mile shipment involve fire.

Two hypothetical accidents involving truck collisions can be postulated to yield maximum impact forces. The carrier moving at a constant speed of 40 m.p.h. (68.6 fps), might collide with an immovable object or with a vehicle of similar mass moving in the opposite direction with a velocity of 60 m.p.h. The latter case would yield an equivalent impact velocity of 146 fps and represents the maximum force which can be encountered in a truck accident. Even if the second vehicle contained explosives, it is not credible to achieve intimate contact of the generator with the explosive.

For accidents complicated by fire, it is necessary to consider only those where the second vehicle contains a large quantity of combustible material, e.g., a gasoline truck. The fuel tank capacity of the generator carrier alone is not capable of inflicting significant thermal damage upon the generator.

b. Marine

The statistics for marine accidents on a per mile shipment basis are not available. However, several cases may be examined. The most credible mishap involves dropping the generator from the cargo loading equipment into the hold of the ship. This would probably yield impact velocities of less than 100 fps. Other accidents might include collision with other vessels or objects, grounding, storms, or fires. The ultimate fate of the generator as the result of a marine accident would be underwater burial. The maximum depth of water that the generator could be exposed to during a marine accident is assumed to be that of the Gulf of Mexico or 12,500 feet.

2. Operational Accidents

Once the generator is placed in the weather station and has reached steady state conditions, there are few significant events which could materially affect its integrity. Some incidents which can be postulated are:

- a. Complete burial of the generator due to earth motion (land slides, earthquakes, etc.)
- b. Destruction of the light station with resulting submersion of the generator in shallow water.
- c. Deliberate tampering by individuals. The intruder would have to be equipped with special tools in order to gain entrance to the generator and fuel capsules and it is unlikely that he would be

able to destroy the multiple containment.

The analysis has shown that none of the three above incidents would result in hazardous conditions.