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Final Safety Evaluation of a Ten Watt Strontium-90 Fueled Generator For a Deep Sea Application - SNAP 7E

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Prepared by
H. N. Berkow
V. G. Kelly
Nuclear Safety
Martin Company
Nuclear Division
Baltimore 3, Maryland

Approved by:

W Hagis

W. Hagis
Nuclear Safety
Unit Supervisor

W L Kershaw

W. L. Kershaw
Project Engineer

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Summary

A safety evaluation was performed on the SNAP 7E generator system to determine if the safety design criteria with respect to mechanical, thermal and radiation integrity have been satisfied. Analyses were performed to assess the radiobiological effects in the event of release of the radioisotopic fuel. The shielding was evaluated to determine the safe working limits of personnel required to be near the generator. The entire safety evaluation is based upon a fuel loading of 31,000 curies of radiostrontium as measured by ORNL on March 27, 1962.

The results of the generator are as follows:

A. The radiostrontium fuel is encased in two independent pressure vessels, each of which will withstand the hydrostatic pressure occurring at the maximum ocean depth of 20,000 ft. (8,930 psi). The outer containment vessel was assumed to fail and the capsule cap, the critical member, was analyzed structurally. The strontium titanate fuel pellets with a minimum compressive stress of 25,600 psi were considered as support for the cap. The shear stress at the welded joint was found to be 37,450 psi while the allowable shear stress for Hastelloy C is 62,500 psi.

Hydrostatic pressure tests were performed on the capsule at pressures up to 10,000 psi without failures and only negligible deformation. Further tests will be performed on the complete unit after assembly and prior to shipment to its operating site.

An analysis of the generator dome showed that it would buckle at an external pressure of 40,800 psi, thus having a safety factor of 4.5. The maximum circumferential stress in the dome of 51,000 psi is significantly less than the compressive strength of mechanite, 150,000 psi.

B. Fuel containment will be maintained in the event of ground burial or transportation fire. If the generator were buried in a poor heating conducting soil at a temperature of 70°F, the shield container equilibrium temperature would be 197°F. This is only slightly higher than the normally operating temperature.

The generator will withstand the standard one hour fire, an AEC requirement on shipping containers of radioactive materials. The maximum temperature under this condition is 1700°F which is below the melting point (2100°F) of the outer shield container.

C. The generator shield container will withstand impact under conceivable collision accidents during transportation. An impact velocity of 220 mph is required to rupture the top plate covering the generator during transportation after fueling. The maximum velocity during truck transportation would occur if the carrier vehicle impacts with a vehicle of similar mass traveling 60 mph in the opposite direction. An accident postulated during rail transportation considered a train traveling at 100 mph impacting with an immovable object. In each case the maximum velocity of impact is below that required to rupture the container.

D. The chemical stability and insolubility of the fuel prevent it from becoming a radiobiological hazard even in the unlikely event of containment loss due to sea burial. A release analysis showed that if all of the fuel were released to the sea early in the generator life, the activity concentration in the water, 10 meters from the origin, would be 10^{-5} $\mu c/cm^3$. If one were immersed in this water, the resulting dose rate would be less than 1 microrem/hr.

E. With a capsule weld penetration depth of 0.090 in. and a Hastelloy C corrosion rate of 0.0001 in/yr., approximately 360 years of corrosive action and hydrostatic pressure (at 20,000 ft. depth) would be required to release the fuel from the exposed capsule. The remaining activity at that time would be 5 curies.

F. The radiostrontium is shielded sufficiently for shipment under ICC regulations. The maximum calculated dose rates are 110 mr/hr. and 10 mr/hr. at the surface of the generator shield and at one meter from the fuel capsule surface, respectively. Furthermore, a man can work around the generator for 15.6 eight hour days without exceeding his permissible quarterly whole body dose. He can work in direct contact, his hands on the side of the generator shield (with the rest of his body shielded) for 21 eight hour days without exceeding his permissible quarterly local dose.

I. INTRODUCTION

This report presents a safety evaluation of the SNAP 7E thermoelectric generator system for a U. S. Navy deep sea mission. The SNAP 7E generator produces electrical power from the energy released by the decay of 31,000 curies of radiostrontium fuel.

The safety evaluation is based upon analytical procedures and test results to determine if 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 the loss of a 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 were subjected to external pressures greater than those which will be encountered during operation (10,000 psi) without rupturing. Additional tests will be conducted on the generator and dome under similar external pressures. The results of the future tests will be included as an addendum to this report.

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II. SYSTEM APPLICATION AND CONFIGURATION

A. Application

This radiostrontium-fueled generator will be used to supply electrical power for a U. S. Navy deep sea application. For purposes of analysis, a value of 31,000 curies of Sr-90 was used as the source of the thermal energy. Lead telluride thermoelements, electrically in series, transform the thermal energy into useful electrical energy. A DC-DC Converter increases the voltage to charge a capacitor bank which discharges every 30 seconds.

The generator system will be transported to its operating site aboard ship by the U. S. Navy. The final destination of the system will be in the Atlantic Ocean and it will operate at an ocean depth between 12,500 and 18,000 feet.

B. Configuration

The strontium titanate fuel is compacted into pellets of approximately 1.4 inch diameter and 0.5 inch height. The pellets are encased in a cylindrical Hastelloy C fuel capsule (see Fig. II-1) having a minimum wall thickness of 0.179 inch. A total of 4 capsules contain all the radiostrontium fuel. The caps are welded on the fuel capsule in a helium atmosphere, permitting a non-oxidized weld to be effected. The capsules are inserted in a Hastelloy C heat accumulator having a diameter of 4.364 inches, a length of 5.540 inches and a minimum wall thickness of 0.150 inch. The heat accumulator is supported at its ends by slabs of Min-K insulation approximately 1.5 inches thick. Against flattened areas on its outer wall are 60 pair of lead telluride thermoelectric elements, each approximately 1 inch in length. The

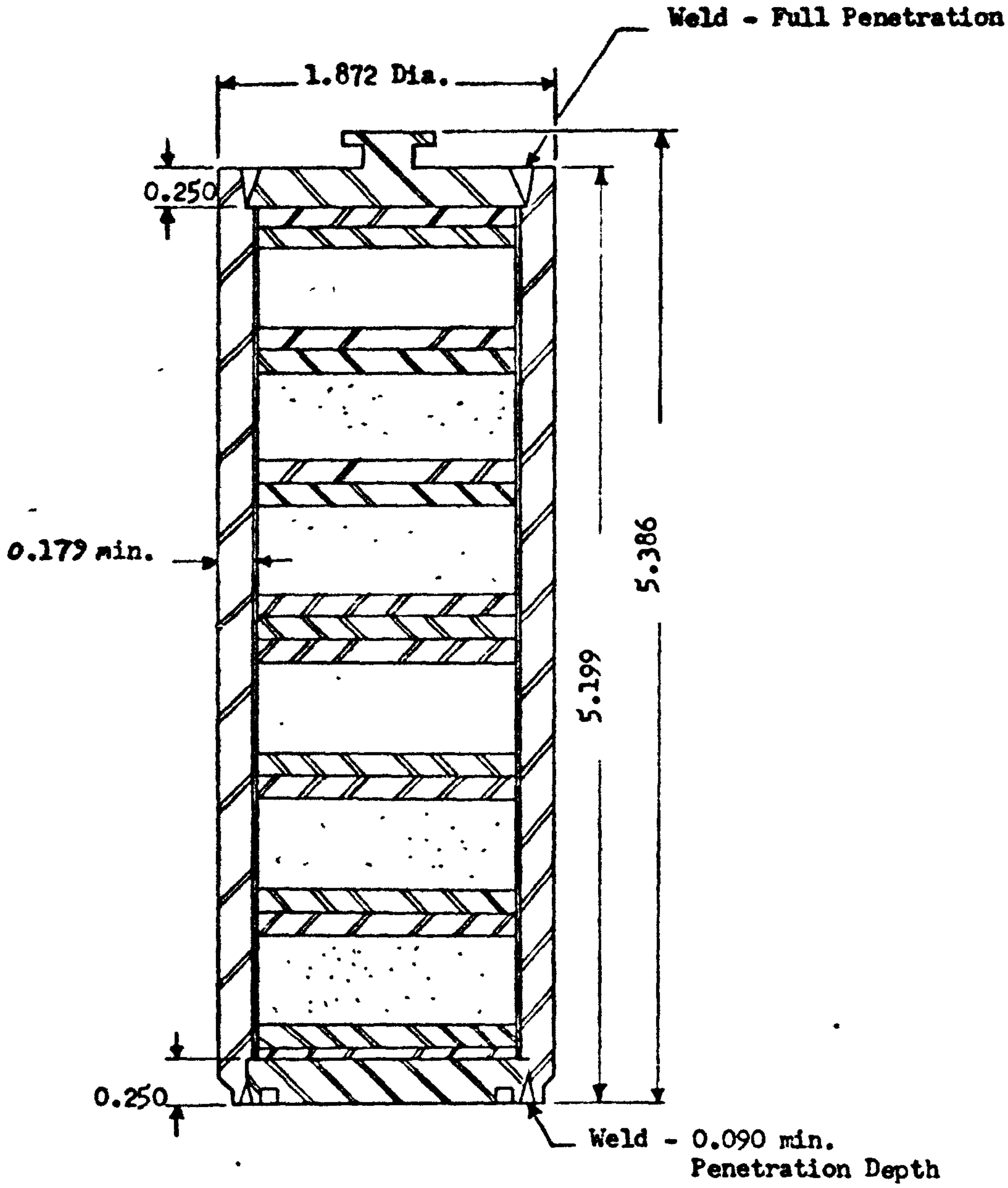


Figure II - 1

SNAP-7E Fuel Capsule

outer ends of the thermoelectric elements terminate in an aluminum heat sink which is a cylindrical ring 0.875 inches thick and 6.5 inches long. The generator assembly is evacuated and filled with an inert gas at atmospheric pressure. This procedure is designed to prolong the life of the thermoelectric elements.

The generator is inserted in a cylindrical shield container fabricated from meehanite grade GC. This material is a casting consisting of approximately 96% iron. A block of depleted uranium plated with nickel, 2.75 inches thick, is placed in the cavity over the generator to provide top shielding. During shipment of the fueled unit from ORNL to the Martin Company, the internal components are held in place by a removable bolted plate. Fins, located on the outer surface of the meehanite casting, aid in maintaining a surface temperature of 147°F. Mercury is added to the void between the shield container and heat sink to aid in the heat transfer characteristics of the generator and to provide additional biological shielding.

A dome is placed over the top portion of the generator to house Navy electronics equipment at final assembly. This dome is a cylinder of 17.375 inches diameter with a hemispherical top. The overall length is 45 inches and the wall thickness is 1.75 inches. The material is the same as that used for the shield container; meehanite. The generator configuration is shown in Figure II-2.

C. Radiostrontium Fuel

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

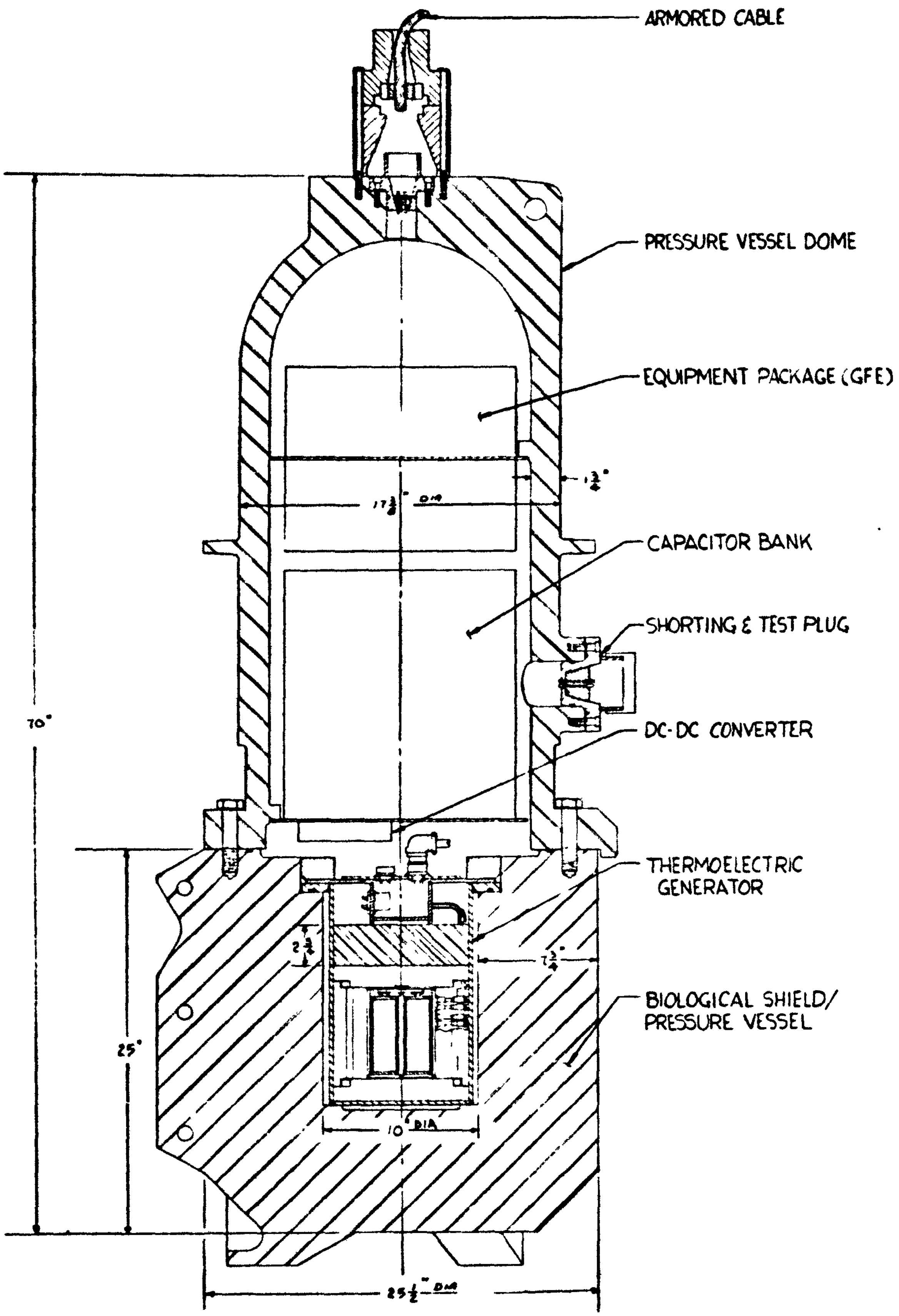


FIG. II- 2 SNAP-7E ELECTRIC GENERATION SYSTEM DETAILS

Strontium-90 is a fission product of uranium-235. It is recovered from reactor waste and shipped, as cakes of strontium carbonate from Hanford to ORNL, where it is processed into pellets of strontium titanate. Both Strontium-90 and Strontium-89 are present in the fuel at the beginning of the 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 II-3.

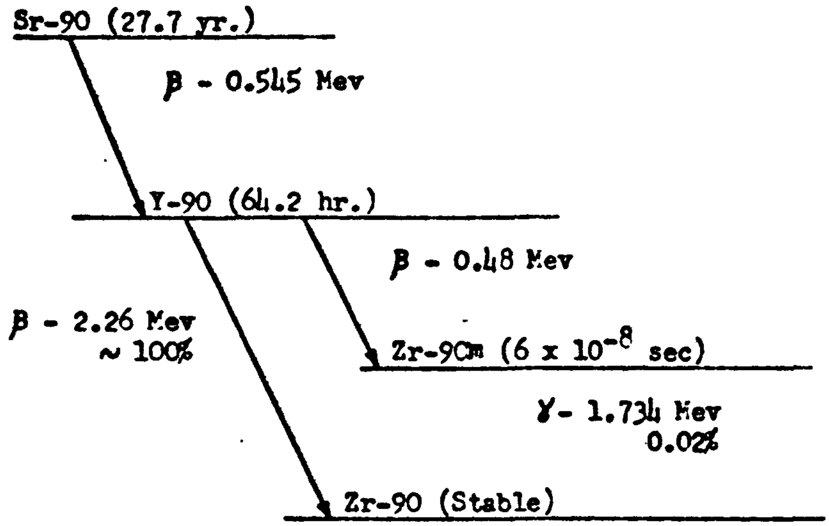
Strontium titanate is a chemically stable compound and is effectively insoluble in natural reagents. The titanate forms a eutectic with titania and forms solid solutions with the titanates of barium and calcium. The metatitanate form predominates, although an ortho-titanate can be prepared.

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.

Yttrium oxidizes in air and reacts with boiling water to form the hydroxide. It may also form the titanate ($Y_2Ti_2O_7$) and the zirconate ($Y_2Zr_2O_7$) (Ref. 1). However, the number of Y-90 atoms present in the fuel at any time is relatively small since its half life is only 64 hours.

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 if present in large quantities. The thermal output of such contaminants was 2 thermal watts at the time the fuel was placed in the capsules. Other contaminants are limited during processing as follows:

a. Decay Scheme of Strontium-90



b. Decay Scheme of Strontium-89

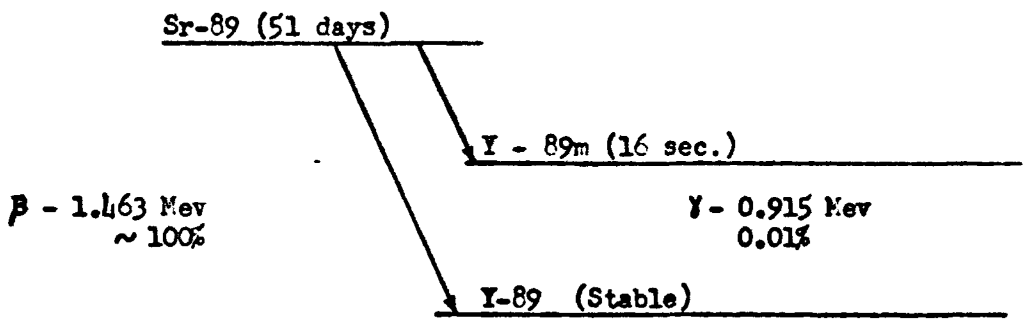


Figure II-3

Decay Scheme of Sr-89 and Sr-90

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a. beta emitters (e.g. cerium-144)	-	0.15 curies/gm Sr-90			
b. gamma emitters:					
0.2 to 1 mev	-	0.15	"	"	"
1 to 2 mev	-	0.015	"	"	"
2 mev or greater	-	0.0015	"	"	"

The principal proerties of the $SrTiO_3$ fuel form used in this generator are listed in Table II-1.

TABLE II-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

Actual density - 3.0 g/cc

Thermal conductivity - 0.0116 cal/sec-cm-°C

Specific Activity (theoretical) - 83 curies/g

Specific Activity (actual) - 36.4 curies/g

Specific Power (theoretical) - 0.456 watts/g

Specific Power (Actual) - 0.229 watts/g

Power Activity Constant - 6.3×10^{-3} watts/curie

Power Density (theoretical) - 2.36 watts/cc

Power Density (actual) - 0.687 watts/cc

Activity density (theoretical) - 430 curies/cc

Activity Density (actual) - 109 curies/cc

Note: Properties obtained from ORNL measurements on actual fuel capsules and reported to AEC on 3/27/62

III. Analytical Safety Evaluation

The 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 simplified. Analyses were performed to determine the generator integrity, fuel solubility rate, shielding effectiveness and the consequences of an accidental underwater activity release.

A. System Integrity

1. Hydrostatic Pressure

The radiostrontium is encased in two independent pressure vessels; the fuel capsules and a shield container. The shield container consists of the biological shield and the dome which houses electrical equipment. Each container has been designed to contain individually the fuel under maximum operating or accidental conditions. The maximum hydrostatic pressure to which the generator could be subjected would occur if it were accidently exposed in 20,000 feet of water (8,930 psi).

The dome was considered to be the critical member and an analysis was performed to determine the maximum external pressure that it can withstand. (See Appendix A) The buckling pressure or the pressure required to collapse the long cylinder was calculated to be 40,800 psi by the method defined in references 2 and 3 (See Appendix A). This is greater by a factor of 4.5 than the maximum pressure of 8,930 psi at a 20,000 foot depth.

The circumferential stress in the dome resulting from the hydrostatic pressure was found to be 57,500 psi by the method described in reference 4. The compressive strength of meehanite is 150,000 psi (ref. 5).

To establish further the inherent safety of the design, it was assumed that the outer containment failed and the fuel capsules were exposed to the hydrostatic pressure. The capsule cap welded to the cylindrical body was analyzed for failure. A previous analysis conducted on a similar capsule had indicated that the capsule cap would fail at an ocean depth of 8,600 feet (Ref. 6). This analysis was highly conservative in that it considered several assumptions for ease of calculations that resulted in a low rupture stress. One of the assumptions considered a high temperature drop across the wall, consequently a high thermal stress that was added to the stress from the external pressure. A detailed re-analysis of the capsule indicated a low temperature drop resulting in an almost negligible thermal stress. Another assumption did not consider the fuel as support for the capsule since no information was available on the compressive strength of strontium titanate.

The capsule cap was analyzed considering the additional support and neglecting thermal stresses. To substantiate the use of the fuel as a supporting force, great care was taken to control the gap between the fuel pellets and capsule cap to within a maximum of 0.032 inch. This tolerance then allows a clearance of 0.016 inch at the top and bottom of the fuel capsule.

The high ductility of Hastelloy C (Ref. 7) indicates that the capsule cap will have to deflect more than 0.016 inch to rupture. Compressive tests on simulated pellets supplied by ORNL resulted in a minimum compressive stress of 25,6000 psi for the titanate pellets. Considering the forces acting on the capsule cap, failure would occur at the welded joint. A minimum of 0.090 inch has been specified for all capsule welded joints. Using this weld thickness, the shear stress was found to be 37,450 psi. The allowable shear stress for Hastelloy C is one half of the ultimate tensile strength or 62,500 psi.

To substantiate the above analysis and conclusions, additional tests were performed on simulated fuel capsules at the Naval Research Laboratory in Washington, D. C. These capsules were identical to actual fuel capsules except for the fuel. All fabrication techniques, including the remote cap weld, were performed as specified. Two capsules, one containing inert strontium titanate pellets and the other empty, were tested under pressure. At different intervals the test apparatus was shut down and the capsules removed for visual inspection and physical measurements.

The capsule containing the simulated fuel was first tested to 4000 psi and then removed from the test vessel. The measurements, read to the nearest 0.001 inch, indicated no permanent deformation on the caps or side walls. This agreed with the analysis in that the yield point of the capsule cap was above 4000 psi (in the order of 5000 psi). The capsule was returned to the pressure chamber and the pressure increased to 9000 psi to simulate the maximum design pressure at 20,000 feet of water. This test was conducted for a period of 23.5 hours at this pressure. During the hours at night it was not possible to control the pressure at a constant level and a pressure drop to 8,050 psi occurred over a 16 hour period. However, this was a gradual drop and for the last 5.5 hours of the test the pressure was maintained at a constant level of 9000 psi. The unit was removed, visibly inspected and measurements taken to determine the permanent deflection of the side walls and capsule cap. There was no change in the diameter of the side walls when measured to the nearest 0.001 inch. The measurement on the centerline of the capsule showed an overall deformation of 0.004 inch. This indicates that the yield point of the cap had been reached but the ultimate tensile strength of the material had not.

The empty capsule was then tested in a similar manner although different pressure levels were used. The test capsule was initially tested at 6000 psi to determine if the yield point of the cap had been surpassed. A difference in 0.001 inch in the centerline dimension indicated that the yield point of the capsule cap had been reached. The pressure was then increased to 10,000 psi and maintained at this level for a period of 20 hours. The results of this test were similar to those of the previous capsule, an overall deformation of 0.01 for the capsule length and no change in the diameter of the cylindrical walls.

In addition to the above tests, the capsule containing the simulated fuel was returned to the pressure chamber and the pressure raised to 10,500 psi which simulated the conditions at 23,800 feet of sea water. A permanent deflection of 0.010 inch was measured at the vertical centerline of the capsule.

The results of the capsule test program have verified the analysis and have shown that the capsule will not rupture when subject to pressures less than or equal to the design pressure of 8,930 psi. Calculations as shown in Appendix A indicate that the capsule will withstand the external pressure to a depth of 33,400 feet.

2. Impact

The most stringent impact conditions would occur if the generator unit were subjected to an accident during transportation from ORNL to the Martin Company after fueling. The unit was investigated to determine the members likely to rupture in the event of impact of the carrier with an immovable object. Two types of carriers were considered, truck and railroad car.

Two hypothetical accidents involving truck collisions can be postulated. The carrier moving at a constant speed of 40 mph might collide with an immovable object or with a vehicle of similar mass moving in the opposite direction with a velocity of 60 mph. The latter case would yield an equivalent impact force of 146 fps and represents the maximum force which can be encountered in a travel accident.

Conceivable rail accidents include collision with an immovable object, derailment and collision with another train traveling in the opposite direction with equal speed. An accident was postulated in which a train traveling at 100 mph collides with an immovable object.

In the analysis performed (See Appendix A) the tie down attachments are assumed to fail and the generator is exposed to the maximum impact velocity upon contain with the carrier's enclosure. The method for determining the uniform load from an impact velocity is described in reference 8. The weakest portion of the surface of the generator is the unsupported area of the top plate. To rupture this section a velocity of 323 fps is required to impose a uniform load of 9,230 psi on the unsupported area of the top plate. Although it is not presently planned to transport the generator by rail, it would require a uniform load of 1.9 times the pressure resulting from the maximum conceivable accident to rupture the external container.

An additional investigation indicated that for impact velocities greater than 83.5 fps the support holding the top uranium block may fail. However, the block will still be contained within the meehanite shield cavity by the top plate.

As a part of this safety evaluation, it is desirable to determine the potential hazards in the event of an accident that destroys the effectiveness of the biological shield. In the event of such an accident an exclusion area of 56 feet in diameter would have to be set up. The close rate at the edge of the exclusion area would be 100 mr/hr. Special equipment and personnel would then be required to handle the generator as the dose at 4 feet from the capsules would be 19.5 r/hr. Although there are no official maximum exposures set for rescue teams in extreme emergencies, a one time emergency dose of 25 rems is permissible (Ref. 9). This would allow one man working one hour and 17 minutes to place a shield around the unit for future disposal.

3. Hoisting Attachments

Provisions for lifting or installing the generator consist of three shackles attached to the fins on the outside surface of the generator. The maximum work load specified for each shackle and pin (a standard AN-116-20 fitting) is 6,380 pounds. This load represents one fifth of the breaking strength of the fitting. One fitting can hold five times the weight of the generator.

The shackles are attached to the fins by 0.875 inch diameter pins through a one inch diameter hole in the fin. The load required to shear through the one inch thick fin is 40,000 pounds or eight times the weight of the generator.

Therefore for failure to occur the generator must be subjected to a force of five times the weight of the generator. This is equivalent to a drop of 160 feet which is highly inconceivable in the present planning for lifting and transporting the generator. It should be noted that this analysis considers failure of two attachments. A force of 15 g's is required to rupture the three attachments to the generator.

4. Thermal Integrity

a) Fire

Although the mission of this generator precludes the occurance of fire during operation, the possibility of its being involved in a fire during transportation, even though remote, must be recognized. Two possible accidents that could occur are truck impact with a fuel carrier and fire aboard the transport ship. In either case, both the meehanite shield and the Hastelloy C fuel capsule would have to melt before the $SrTiO_3$ fuel could be released to the environment. Thus, the temperature of the 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, as described by the National Fire Protection Association (Ref. 10) and the American Society for Testing Materials (Ref. 11), was used as the basis for this analysis. The standard fire assumes a one hour exposure to a fire temperature that reaches a maximum of 1200°F. As shown in Table III-1, the melting point of the outer meehanite casting is 2100°F. Therefore, containment would be maintained under these conditions by the outer shield vessel.

If the outer surface of the generator is exposed to the above temperatures, the inner components will consequently rise above their normal operating temperatures.

At temperatures above 1200°F, the lead telluride thermoelectric elements will tend to decompose rapidly. If the lead telluride decomposes or melts (m.p. = 1690°F), the major path of heat conduction from the fuel capsules will be removed. Due to the thermal energy of the fire and fuel material, the capsule temperature will 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) Ground Burial

The thermal properties of the materials comprising the SNAP-7E generator are shown in Table III-1. The thermal integrity calculations are shown in Appendix B.

Although ground burial is unlikely, it could conceivably occur as the result of a truck, rail, or air transportation accident. The worst possible case from a heat transfer point of view would occur if the entire generator system, including shield, were buried 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 conservative results.

Table III - 1

Thermal Properties of Major Generator Components

Component	Material	Melting Point	Specific Heat	Weight	Heat Equivalent	Average Working Temperature	Heat to Reach 1700°F
		(°F)	(Btu/lb-°F)	(Lbs.)	Btu/°F	(°F)	(Btu's)
Shield Casting	Meehanite	2100	0.119	3100	368	90	592,000
Top Shield Block	Uranium	2067	0.028	54	1.5	90	2,400
Heat Sink	Aluminum	1220	0.260	10	2.6	150	5,100 (1)
Fuel Block	Hastelloy C	2350	0.092	14.8	1.36	1000	950
Fuel Capsules	Hastelloy C	2350	0.092	7.64	0.71	1090	430
Misc.				39.86			
Total (2)				3226			600,880

(1) Includes the heat of fission of the component (1070 Btu's)

(2) This does not include dome as the generator is shipped after fueling without dome to Martin Company

From the equation for concentric spherical shells surrounding a heat source (Ref. 12), the steady state temperature differential between the shield surface and the undisturbed soil is found to be 125°F. This calculation assumes that the generator is buried in an infinite soil medium whose thermal conductivity was conservatively chosen as that of Healy clay at 40°F, which has a low thermal conductivity when compared with other common soils (Ref. 12). Thus, the outer vessel wall of the generator, when buried in soil, will reach a steady state temperature of 165°F.

The equation for the steady state temperature of concentric cylinders (Ref. 12) was used to determine the temperature drop across the meehanite shield. This temperature differential was calculated to be 2°F. Thus, the shield container inside surface temperature would be 167°F.

The above analysis is based on an equilibrium soil temperature of 40°F. Even if this temperature were as high as 70°F, the shield container I. D. temperature would be only 197°F. This is only slightly higher than the normal operating temperature of the generator and consequently there is no danger of meltdown due to soil burial.

5. 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 0.170 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. 13). The tensile and yield strengths of the material increase while the ductility decreases. This will have a negligible effect upon the structural integrity of the capsule as analyzed in Section III-A.

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. 14).

B. Fuel Solubility Analysis

The fuel form with the radiostrontium combined as the arthotitanate, $SrTiO_3$, 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. 15). 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 III-2.

TABLE III - 2 (Ref. 15)

Summary of Strontium Titanate Solubility

A. In Water

Run No.	Method of Analysis	Environmental Solution	Exposure Temp. (°C)	Exposure Time (Hr.)	Volume of Solution (cc)	Titanate Sample Weight (gm)	Titanate State	Solubility Strontium (ppmSr)
1)	Flame Photometry	Sea Water	100	238	250	1.10	Pulverized	1.4
2)			100	336	250	4.98	"	30
3)			100	168	250	1.26	"	0.02
4)			20	2208	100	8.195	Pellet	16.
5)			20	2208	100	7.936	"	13
6)	Tracer	Sea Water	65	2448	3786	2.608	Pellet	0.113
7)			65	2448	3786	1.241	"	0.042
8)			65	2448	3786	0.730	"	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	"	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	"	3

TABLE III - 2 (Cont.) (Ref. 15)

B. In Reagents

Concentration	Exposure Time (Hr.)	Solubility, PPM					Method of Analysis
		HNO_3	HCl**	H_2SO_4	NH_4OH	NaOH	
0.1N	336	2	2	4	1	3	Flame photometry
6 N	336	50	625	0	0	245	Flame photometry
0.1N	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

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 cm^2, 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/cm^2-day, or twice the average rate, was used for calculations in this safety evaluation.

At the present time, there is no available data concerning the effect of fuel composition variation on solubility. A program to investigate nature of this effect has been proposed.

C. Shielding Analysis

The generator is permanently shielded by a block of depleted uranium on the top surface, and a meehanite casting on the bottom and peripheral surfaces to provide biological shielding against the beta, gamma and bremstrahlung radiation generated by the decay of the radiostrontium fuel. Additional shielding is provided by the liquid mercury located in the cavity between the generator and rechanite shielding.

An analysis based upon the method reported in reference 19 was performed to determine the dose rates on the surface of the generator and at a distance of 1 meter from the fuel capsules. The fuel loading was 31,000 curies of Sr-90. The results of the dose rate calculations are shown in Table III-3. The dose rates are below the maximum permissible levels for transportation as defined by ICC ragulations. The ICC regulations permit a maximum dose of 200 mr/hr on the surface of the shipping unit and 10 mr/hr at a distance of 1 meter from the centerline of the cask (Ref. 16).

TABLE III - 3

Calculated Dose Rates from Assembled Generator, mr/hr.

(Based upon 31,000 curies of radiostrontium fuel)

Location	Shielding	Surface	One Meter from Outside Surface of Fuel Capsules
Top	2.75 in. depleted uranium	96*	10*
Side	8 in. meehanite + 0.375 in. mercury	110	10
Bottom	9.13 in. meehanite	105	10

*These top values include the shielding effect of the top dome. When shipping the generator without the dome an additional steel plate, ½ inch thick, will reduce the surface and one meter dose rates to 60 and <10 mr/hr, respectively

Actual measurements will be taken after the generator is fueled to assure that the unit meets the ICC regulations for shipment. The analysis is conservative so that it is likely that the actual measurement taken at ORNL will be lower than those in Table III-3.

The shielding analysis considers the three types of radiation emitted by the radiostrontium fuel; beta, gamma and 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 III-4. However, beta emitters may be effective x-ray sources because of the bremsstrahlung generated as the electrons decelerate in the atomic field (Ref. 18). 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. 19). The bremsstrahlung from the 0.545 mev beta of Sr-90 were neglected since they are 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. However, this emission is negligible since the Sr-89 rapidly decays to stable yttrium.

Based upon the calculated dose rates in Table III-3, a man can work around the generator (at one meter distance) for 15.6 days, assuming an 8 hour working day, without exceeding his permissible quarterly whole

TABLE III - 4

Beta Energies and Ranges (Ref. 17)

Beta Energy (Mev)	Material	Density(g/cc)	Range (mg/cm^2)	Maximum Penetration Distance (cm)	Maximum Penetration Distance (in.)
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
	Hastelloy C	8.9	1000	0.118	0.046

body dose of $1\frac{1}{4}$r (Ref. 20). He can work in direct contact, his hands on the sides of the generator (with the rest of his body shielded) for a period of 21 eight hour days without exceeding his permissible quarterly local dose of $18\ 3/4$ r.

D. Release Analysis

The structural integrity analyses have shown that an activity release to the atmosphere is not possible. In addition, a simple calculation based on the corrosion rate of Hastelloy C shows that if the fuel capsule were to be exposed to its sea water environment, it would take approximately 360 years (13 Sr-90 half lives) for corrosive action of the sea water and an external pressure of 9000 psi to cause release of the fuel. This analysis is based upon a corrosion rate of 0.0001 in/yr for Hastelloy C (Ref. 15) and a weld thickness of 0.090 in. By that time, the remaining activity would be a maximum of 5 curies.

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 821 cm^2. Using the solution rate derived from the solubility tests, a fuel erosion rate of 2.63×10^{-3} gm/day is obtained. The specific activity of the fuel at the time of loading was 36.4 curies/gm. The rate of activity release would than be 9.57×10^{-2} curies/day or 1.12×10^{-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. 21). 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. 22). The rate of horizontal diffusion varies from 0.3 to 1.0 cm/sec (Ref. 21, 23, 24).

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

$$D = Q/2\pi rh \qquad \text{eq. (1)}$$

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:

$$X = D/K = Q/2\pi Krh \qquad \text{eq. (2)}$$

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 196 thermal watts* or 47.5 cal/sec. This quantity of energy is capable of raising the temperature of a column of water 4.4 cm in diameter and 312 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 = 31.2 cm and K = 0.5 cm/sec (Ref. 24). Then eq. (2) gives an activity concentration in the water, 10 meters from the origin, of 1.1 x 10^{-5} $\mu c/cm^3$. This analysis considers only diffusion, not advection.

* Measured by ORNL on 3/20/62

The maximum recommended concentration in water for Sr-90 is 10^{-6} $\mu c/cm^3$. But man does not normally drink ocean water. The dose rate, if one were to bathe in the water would be less than 1 μ rem/hr. Fish and other marine organisms which may enter the diet do not take up strontium from the water. The concentration factor for the fish body compared with the ocean water is 10 (Refs. 21, 25, 26). 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.

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 III-5 (Ref. 27).

TABLE III - 5

Maximum Permissible Sr-90 and Y-90 Concentrations in Air and Water

(Ref. 27)

Radionuclide and Type of Decay	Form	Organ of Reference	Maximum Permissible Concentration (μc/cc)			
			40-hour week		168-hour week	
			Water	Air	Water	Air
$_{38}Sr^{90}$	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}
$_{39}Y^{90}$	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}

* GI (LLI) - Gastrointestinal Tract (Lower Large Intestine)

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. 28). The Sr-90 bone concentration is commonly expressed in Strontium Units (1 SU=1 μμc 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. 29).

IV. QUALITY CONTROL

To insure the integrity of the generator unit rigid quality control procedures have been maintained during and after fabrication and assembly of the various components. The general procedures follow the Quality Control System Requirements of Military Specification MIL-Q-9858 and are as follows:

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 certification program. Where ASME welding is specified, the welder is ASME certified. Wherever practical, sample welds are made and analyzed to verify repeatibility.

3. Materials are ordered as specified in drawings. Alternate materials are specified, when feasible, to reduce costs, and improve delivery schedules. All substitutes 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. Visusl and dimensional inspection as well as required chemical and physical tests are performed as specified. These include dye penetrant tests and radiograms. Special consideration is given to welded assemblies requiring absolute containment. All seal welds to be performed by the Martin Company are lead tested with inert gas and must meet the following requirement: "There shall be no leakage when

tested with equipment having a sensitivity of 10^{-7} cubic feet of inert gas per hour at 4 mm Hg and 70°F."

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

Specific Quality Control procedures performed for the capsule and shield container were:

1. The certification report of the material used in the construction of the fuel capsule assembly received from the Haynes Stellite Company was reviewed by Quality Control on December 21, 1961.
2. Detailed visual and dimensional checks of the parts to Engineering drawings was accomplished.
3. Welding of the capsule cap was done by a certified welder (see part 2 above). A full weld penetration was specified for the top cap weld.
4. A dye-penetrant check was performed and acceptance is based upon specifications MIL - I - 6866 and MIL - I - 6868.
5. The final capsule weld was performed at ORNL by placing the capsule in a remote welder, the welder purged seven times with helium, then approximately two curies of Kr-85 introduced and the chamber pressurized to 30 psi with helium. The welds were made in two passes at 60-70 amps and 15 inches per minute. Each capsule was leak tested by the Kr-85 method with the following results:

Capsule No.	CPM Above Background
1	197
2	69
3	76
4	20

The capsules were then ultrasonically cleaned. Minimum thickness was 0.090.

6. The certification report for the meehanite casting furnished by Queens City Foundry, Charlotte, N. C., was reviewed by Quality Control on 1/15/62.

7. Radiographic inspection was made at the source and forwarded to the Martin Company Engineering for approval. Acceptance specifications stated that surface defects, such as sand holes, burn-in, etc. should not exceed 3 inches in any direction and not be deeper than 0.5 inch.

8. Detailed visual and dimensional check of material to engineering drawings was performed.

V. CONCLUSION

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 mission.

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Appendix A - Structural Integrity Calculations

I. Hydrostatic pressure at 20,000 ft of water:

1. P = depth x specific weight

$$= 20{,}000 \text{ feet} \times \frac{64.3 \text{ lb/ft}^3}{144 \text{ in}^2/\text{ft}^2}$$

$$= 8930 \text{ psi}$$

II. Shield Container - External Pressure

1. Buckling pressure

$$P_c = \frac{E}{4(1-v^2)} \left(\frac{t}{Ri}\right)^3$$

P_c = Pressure, psi

E = Young's Modules = 17×10^6 psi

v = Poisson's ration = 0.3

t = Metal thickness = 1.75 inches

R_i = Inner radius = 8.5 inches

$$P_c = \frac{17 \times 10^6}{4(1-0.09)} \times \left(\frac{1.75}{8.5}\right)^3$$

P_c = 40,800 psi

2. Circumferential Stress

$$S_c = -\frac{2\,R^2{}_o}{R_o{}^2 - R_i{}^2} \times P$$

S_c = Stress, psi

R_o = Outer Radius = 10.25 inches

P = Hydrostatic pressure = 8930 psi

$$S_c = -\frac{2 \times 105 \times 8930}{105 - 72.4}$$

S_c = 57,500 psi (compression)

III. Capsule Container - External Pressure

1. Circumferential stress

$$S_c = - \frac{2R_o^2}{R_o^2 - R_i^2} \times P$$

R_o = Outer radius = 0.936 inch

R_i = Inner radius = 0.755 inch.

$$S_c = - \frac{2 \times 0.877 \times 8930}{0.877 - 0.570}$$

S_c = 51,000

2. Shear force on capsule cap weld

$$S_s = \frac{Pr}{2t_w}$$

r = Cap radius = 0.755 inch

t_w = Weld thickness = 0.090 inch

$$S_s = \frac{8930 \times 0.755}{2 \times 0.09}$$

S_s = 37,450 psi

3. Maximum external pressure

$$P = \frac{2S_s \times tw}{r}$$

S_{ult} = Shearing stress of Hastelloy C equals ultimate tensile stress divided by 2 = 62,500 psi

$$P = \frac{2 \times 62{,}500 \times 0.09}{0.755}$$

P = 14,900 psi

D = 33,400 ft. depth of water

4. Minimum cap thickness

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

S_s = Ultimate shearing stress of Hastelloy C

= Ultimate tensile strength /2

= 125,000/2 = 62,500 psi

$$t = \frac{8930 \times 0.755}{2 \times 62{,}500}$$

t = 0.054 inch

5. Time required for release of fuel from corrosion and hydrostatic pressure.

$$T = (t_w - t)/C_r$$

T = Time in years

t_w = Original thickness of weld = 0.090 inch

t = Weld thickness at time of failure = 0.054 inch

C_r = Corrosion rate of Hastelloy C = 0.0001 in/yr

$$T = (0.090 - 0.054)/(0.0001)$$

T = 360 years

IV. Uranium Shield Container - Impact

1. Maximum pressure required to rupture support

$S_{ult} = k \frac{Pr^2}{t^2}$ or $P = \frac{S_{ult}\ t^2}{kr^2}$

Where $k = f\left(\frac{r}{rc}\right)$

S_{ult} = Ultimate tensile stress 125,000 psi

P = Pressure in psi

t = Plate thickness = 0.35 inch

r = Radius of plate = 4.25 inch

r = Radius of cutout = 1.875 inch

$k = f\left(\frac{4.25}{1.875}\right) = f\ (2.26) = 0.483$ (Ref. 30)

$P = \frac{125,000 \times 0.123}{0.483 \times 18.1}$

P = 1760 psi

2. Maximum velocity required to rupture support

$P = \frac{pV^2}{144g}$ or $V^2 = \frac{144g\ P}{p}$

V = Velocity at impact in fps

p = Bulk density of uranium block = 1170 lbs/ft^3

g = Gravitational constant = 32.2 ft/sec/sec

$V^2 = \frac{144 \times 32.2 \times 1760}{1170}$

$V^2 = 6950$

V = 83.5 fps

VI. Impact on Unsupported Top Plate

1. Maximum load to rupture plate

$$S_{ult} = \frac{3\ Pa^2}{4\ t^2} \quad \text{or } P = \frac{S_{ult}\ 4t^2}{3a^2}$$

S_{ult} = Ultimate tensile stress = 125,000 psi

P = Uniform load to rupture plate in psi

a = Radius of unsupported plate = 4.25 inch

t = Thickness of plate = 1 inch

$$P = \frac{125,000 \times 4 \times 1}{3 \times 18.1}$$

P = 9230 psi

2. Velocity required for maximum force

$$P = \frac{pV^2}{144g} \quad \text{or} \quad V^2 = \frac{144g\ P}{p}$$

V = Velocity in fps

g = Gravitational constant = 32.2 ft/sec/sec

p = Bulk density of material= 412 lbs/ft^3

$$V^2 = \frac{144 \times 32.2 \times 9230}{412}$$

V^2 = 104,000

V = 323 fps.

VII. Load Required to Shear fin during Lifting Operation

$$S_s = \frac{P}{A} \quad \text{or} \quad P = S_sA$$

P = Load required to shear material = lbs.

S_s = Shearing stress of meehanite = 40,000 psi

A = Area of fin = 1 in^2

P = 40,000 lbs.

APPENDIX B

Thermal Integrity Calculations

1. Equivalent Spherical Model:

   $SA = 4 \pi r^2$

   Where: SA = total exposed surface area = 3130 in^2

   r = Equivalent radius, in.

   Substituting, r = 15.7 in.

2. Steady State Temperature Differential Between Shield and Soil

   $$\Delta T = \frac{q}{4 K}\left(\frac{1}{r_1} - \frac{1}{r_2}\right)$$

   Where: ΔT = temperature differential, °F

   q = heat flow rate = 6.78 x 10^2 btu/hr

   K = thermal conductivity of soil = 0.33 btu/hr-ft.-°F

   r_1 = radius of heat source = 15.7 in. = 1.308 ft.

   r_2 = outer radius =

   $$\Delta T = \frac{6.78 \times 10^2}{4(3.14)(0.33)(1.308)} = 125°F$$

3. Steady State Temperature Differential Across Meehanite Shield

   $$\Delta T = \frac{q}{2 KL} \cdot \ln\frac{r_2}{r_1}$$

   Where: ΔT = temperature differential, °F

   q = heat flow rate = 6.78 x 10^2 btu/hr

   r_1 = shield container inner radius = 5 in = .416 ft.

   r_2 = shield container outer radius = 12.75 in = 1.063 ft.

   L = core length = 26.375 = 2.198 ft.

   $$\Delta T = \frac{6.78 \times 10^2}{2(3.14)(25)(2.198)} \ln\frac{1.063}{.416} = 2°F$$

Addendum "A"

A change was made in the assembly of the generator flange to the mechanite casting as shown on drawing 398-3021200. The flange assembly, which previously was to be welded, is being attached to the mechanite casting by eight 1/4 inch screws. The retaining ring (part no. 398-3021201-9) is then held in place by eight 1/4 inch bolts which connect the ring and cover to the flange. This arrangement is shown in figure A-1.

The maximum pressure that can be exerted by the uranium block before rupture of the cover plate occurs, is 1760 psi as shown in the impact analysis in Appendix A. This pressure occurs of an area of 2.94 square inches for a total force of 5,170 pounds. Thus, each bolt must be capable of withstanding a force of 646 pounds. A 1/4 inch bolt having a stress area of 0.027 square inch and ultimate tensile strength of 58,000 psi[1] will withstand a load of 1560 pounds.

The revision to the assembly procedure will not change the structural integrity of the top cover assembly.

[1] Marks, Lionel S.; Mechanical Engineering Handbook, McGraw-Hill, Inc., Sixth Printing, May, 1956.

-3	TI-170310‡ OR TI-170330	EYE BOLT	
1	AN970-4	WASHER	
8	AN960C416	WASHER	
8	MS20074-04-26	BOLT	1/4-20
8	AN4C7A	BOLT	1/4 8
-			
-9	PART NUMBER	DESCRIPTION	STOCK SIZE

LIST OF MATERIAL

‡ 4. EYE BOLT (TI-170310) ALREADY PROCURED IF REQUIRED.

B 3. APPLY PRIMER COAT AND BOND SURFACES A AND B PER MARTIN PROC. P60026 (BONDING WITH SILICONE ADHESIVES).

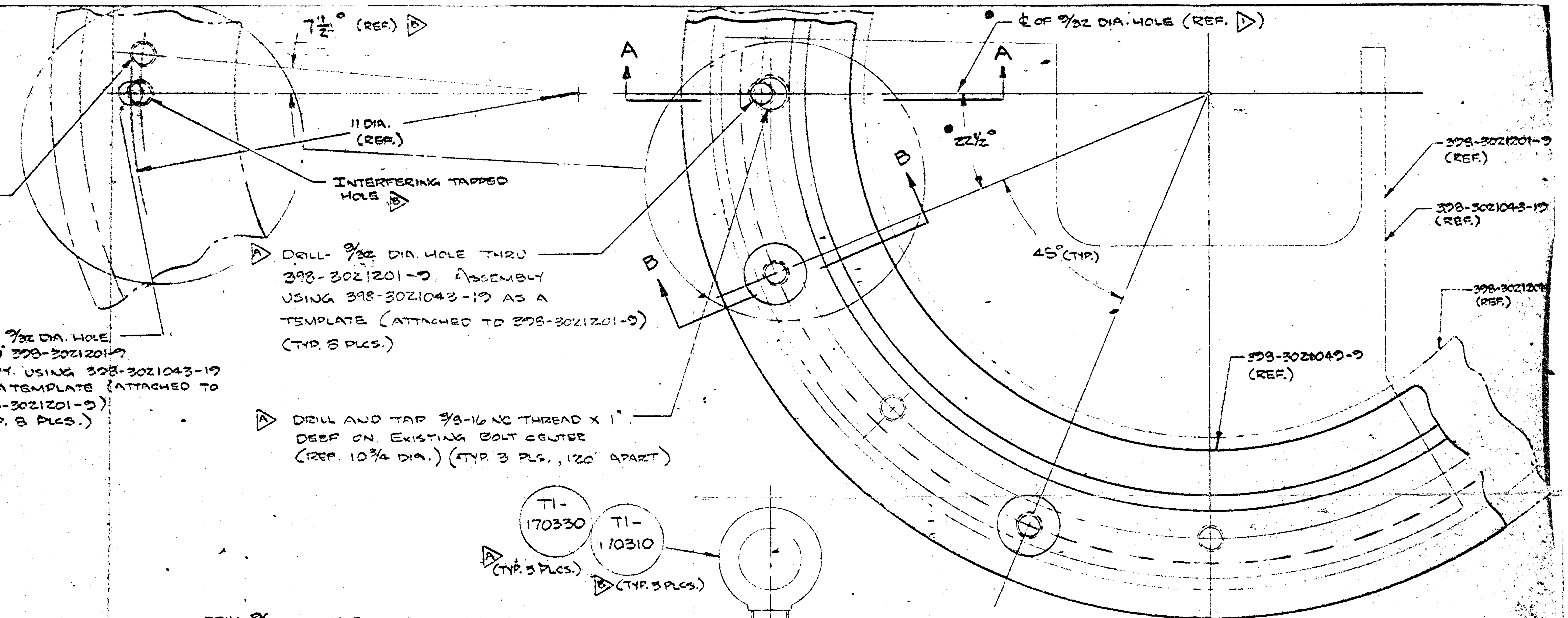
* 2. 9/32 DIA. HOLES MUST MATCH EXISTING TAPPED HOLES IN FLANGE OF CONTAINER 398-3021049-9.

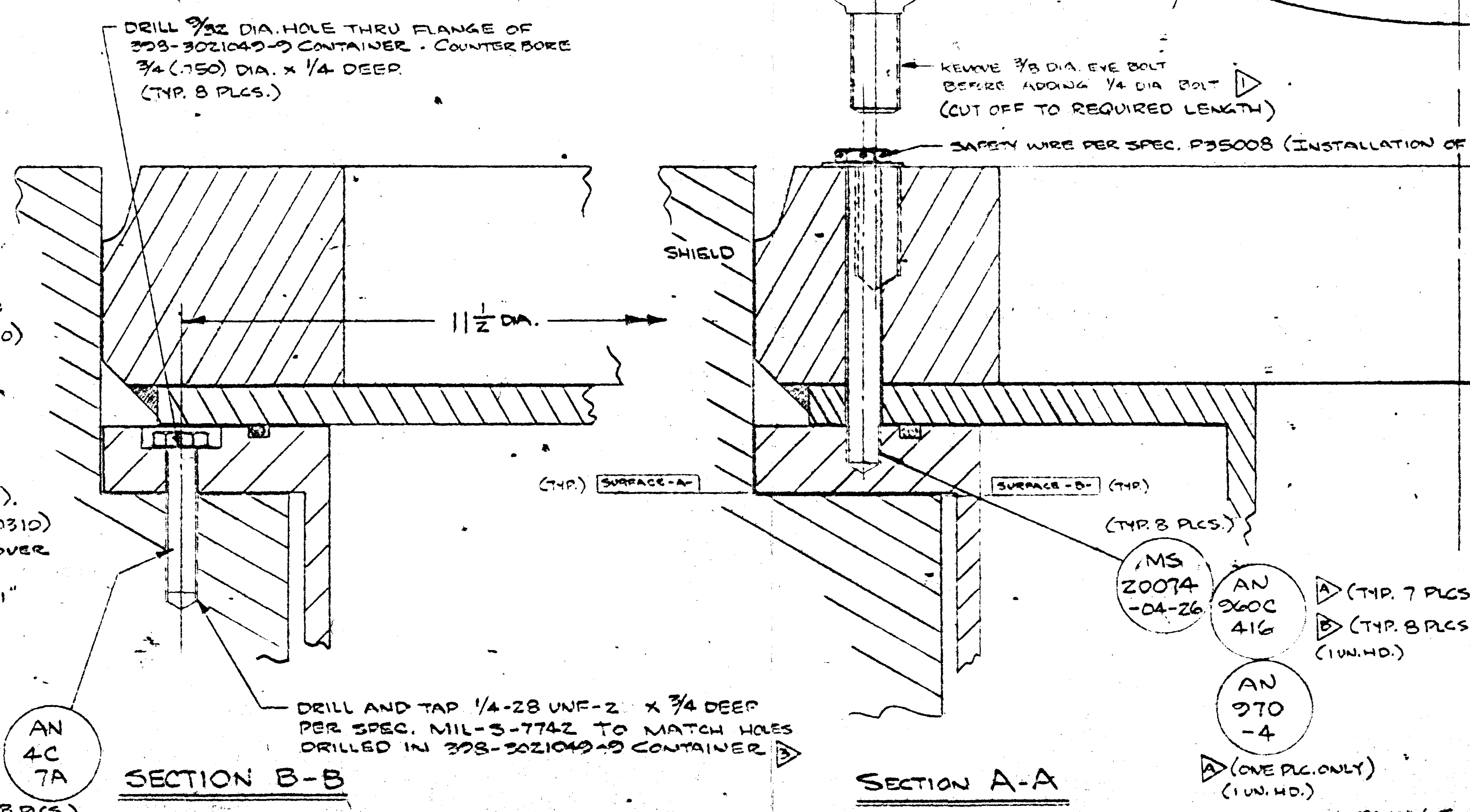
1 1. MANUFACTURING SHALL PERFORM ONE (1) OF THE FOLLOWING:

A A. IF ANY OF THE THREE (3) 1/4-20 UNC-2B EYE BOLTS (TI-170310) ARE POSITIONED DIRECTLY ABOVE ANY OF THE HOLES LOC. IN COVER ASS'Y. 398-3021043-19, INCREASE THE EXISTING EYE BOLT TO 3/8 DIA. (TI-170330) AND DRILL AND TAP 3/8-16 NC x 1" DEEP AT THE EXISTING LOCATIONS. DRILL EXISTING 9/32 DIA. HOLES* (ALREADY LOCATED IN ASS'Y. 398-3021043-19) THRU GENERATOR COVER ASS'Y. (TYP. 8 PLCS.).

B B. IF ANY OF THE THREE (3) 1/4-20 UNC-2B EYE BOLTS (TI-170310) ARE POSITIONED DIRECTLY ABOVE ANY OF THE HOLES LOC. IN COVER ASS'Y. 398-3021043-19; ROTATE APPROX. 7 1/2° FROM THE INTERFERING HOLE AND DRILL AND TAP 1/4-20 UNC-2B x 1" DEEP (ONLY TAP NEW HOLES WHERE REQUIRED). DRILL EXISTING 9/32 DIA. HOLES* (ALREADY LOC. IN ASS'Y. 398-3021043-19) THRU GENERATOR COVER ASS'Y. (TYP. 8 PLCS.)

GENERAL NOTES :





DRAWN	RMT.	6-5-62	THIS DRAWING FORMS PART OF DCN -D
CHECK	RSS	6-5-62	
STRESS			
ENGNR.	RSS	6-5-62	
GROUP ENGNR.	W. Schultz	6-6-62	
APPR'D	[illegible]		GENERATOR ASSEMBLY AND SHIPPING ATTACHMENTS
			398-3021209
			SUPPLEMENT NO. 1