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Use of isotopes to reduce neutron-induced radioactivity and augment thermal quality of the environment of an underground nuclear explosion

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USE OF ISOTOPES TO REDUCE NEUTRON-INDUCED RADIOACTIVITY

AND AUGMENT THERMAL QUALITY OF THE ENVIRONMENT OF AN

UNDERGROUND NUCLEAR EXPLOSION

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ABSTRACT

The use of isotopes to include radioactive waste pro-
ducts to reduce the neutron-induced activity of an under-
ground nuclear explosion and its application in the field
of geothermal power stimulation is discussed. A shield com-
posed of selected isotopes surrounding a fusion device will
capture excess neutrons producing isotopes with short half-
lives. Subsequent rapid decay will prolong the high
temperature in the vicinity of the explosion and decrease
the activity. Long-lived isotopes created by neutron cap-
ture in the host rock would be minimized. The investigation
is conducted for strontium-90, cesium-137, cerium-144,
technetium-99, rhodium-103 and -105. Of these, the first
three appear to be presently unacceptable due to their
physical properties and restrictions imposed during the
investigation. Rhodium-105 is unacceptable due to its
short half-life. The remaining two were determined to be
acceptable although comparatively expensive. Recommendations
are proposed to further study this concept for the disposal
of radioactive waste as well as power production.

ACKNOWLEDGMENTS

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I. INTRODUCTION

The search for an ideal electrical power source is
one of the primary concerns of contemporary man. His use
of inefficient and highly polluting methods of producing
electricity has led him into environmental problems he
has been unable to solve. Despite this, his requirement
for more electrical power to fill his needs increases
without apparent bounds. To balance this need against
the ability of the environment to withstand the impact
of the rapid depletion of the natural resources, a new
expedient should be sought to provide sufficient electrical
power, simultaneously dispose of waste products created
elsewhere, and preclude additional adverse effects on
the biosphere or resources. Briefly, the purpose of this
thesis is to introduce such a method. The concept of
the plan is to utilize the large neutron flux of an
underground thermonuclear explosion, the radioactive decay
of certain "waste" products from reactors, and the natural
geothermal gradient of the earth's mantle to provide a
heat source suitable for conversion to electrical power.

Neutron bombardment by an underground nuclear explo-
sion of certain isotopes (some created as waste matter from
reactors) would produce heavier isotopes which would then
decay by beta emission to stable isotopes. Properly
selected, these isotopes would possess a short half-life

and their decay would produce an increase in the heat of the
surrounding environment with a corresponding and rapid de-
crease in the radioactivity of the medium. An appropriate
amount of these isotopes surrounding the nuclear explosive
would reduce neutron capture by substances in the rock.
This would be regulated by enclosing the explosive first
with a moderating inner concentric shield, secondly the
reactant material, and lastly a reflecting outer concentric
shield, if required. Following a necessary delay period,
water would be introduced into the heated chimney zone and
the steam extracted to power generators. The quality of
the steam would be better than that of naturally occurring
hot springs and would possess relatively small amounts of
radioactivity. This paper examines selected isotopes for
use as a reactant for this purpose.

The presence of long-lived radioactive isotopes pro-
duced by neutron capture in the rock environment is one of
the important remaining restrictions to geothermal nuclear
power production. Among those produced is tritium. Nearly
all igneous rock, particularly pegmatite, contains traces
of lithium in minerals such as lepidolite ($K_2Li_3Al_4Si_7O_{21}$
$(OH,F)_3$), spodumene ($LiAl(SiO_3)_2$), petalite ($LiO \cdot Al_2O_3 \cdot 8SiO_2$),
and amblygonite ($Li(AlF)PO_4$).[1,2] The reaction:[3]

$$Li^6 + n \rightarrow T + He^4 + 4.7 \text{ MeV} \tag{1}$$
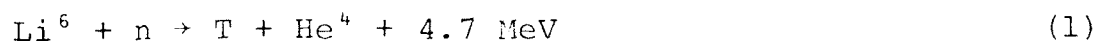
produces the tritium with its 12.26 year half-life. Oxides
and elements of a basalt which are affected by neutron

capture appear in Table I[3]. Only those nuclides with a longer half-life are shown. The composition of granitic rock would be similar. Changes in the percentage of minerals would affect only the amount of activity not the presence of any of the isotopes.

Thermonuclear or fusion devices require a fission trigger; a 99% fusion - 1% fission is considered normal in published data[3]. As this would confuse much of the data unnecessarily, it is assumed that a pure fusion device is used. The decay energy of fission products is about 20 MeV per fission or 29.2×10^{23} MeV per kiloton of explosive[3].

In order to assume a constant number of neutrons available from a fusion explosion, a yield of one megaton is assumed. Based on AEC guidelines[3] a radius of fuel is assumed at 30.48 cm (12 inches).

Further basic assumptions must also be made in order that reasonable investigation may be conducted of isotopes which might provide a suitable reactant in preventing neutron capture in the rock. Primary among these assumptions will be that concerning the neutron capture cross section. With one exception, Strontium - 90, the entire energy spectrum of capture cross sections is not available. For most of the nuclides to be considered, only the thermal neutron capture cross section is available. Miskel[4] assumed that all neutrons were eventually moderated to thermal speeds prior to capture in the rock surrounding the explosion. This allowed him to use the thermal capture cross

TABLE I. Composition of Basalt

Oxide	Weight (%)	Element	n-capture (%)	Isotope	Half-life	A_O (t=0)* (Curies)
SiO_2	48.8	Si	12.0	Si-31	2.62 hrs	7.1×10^7
TiO_2	2.19	Ti	14.8	-	-	-
Al_2O_3	13.98	Al	2.8	-	-	-
Fe_2O_3	3.59	Fe	43.4	Fe-55	2.7 yrs	7.6×10^5
FeO	9.78			Fe-59	45 days	3.9×10^4
MnO	0.17	Mn	2.5	Mn-56	2.58 hrs	8.5×10^8
MgO	6.70	Mg	9.7	-	-	-
CaO	9.38	Ca	6.7	Ca-45	165 days	4.3×10^4
Na_2O	2.59	Na	4.1	Na-25	15 hrs	1.9×10^8
K_2O	0.69	K	1.3	K-42	12.4 hrs	2.8×10^6
H_2O	1.80	H	2.8	-	-	-
P_2O_5	0.33	P	0.08	P-32	14.3 days	1.8×10^5

*Normalized for 1 MT.

sections in calculating the activation products in basalt. The assumption is logical in view of the fact that all measurable absorption cross sections are for neutron capture only, not fission or particle emission[5]. It will be assumed that the neutrons are slowed to thermal energies before capture in the various isotopes being considered.

Of the various fusion reactions available for use in a thermonuclear device, the D - T reaction or:

$$D + T \rightarrow He^4 + n + 17.5 \text{ MeV} \tag{2}$$
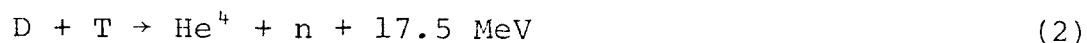
is assumed to be the preferred and most probable for consideration here. As the purpose is to reduce the amount of radioactivity in the rock medium, a D - D reaction or:

$$D + D \rightarrow T + p + 4.03 \text{ MeV} \tag{3}$$

would not be acceptable. As shown in Figure 1[6], however, energies to provide the D-D reaction must be in excess of 36 keV while the D-T reaction only requires energies of 4 keV, (1 keV being equal to 1.16×10^7 °K). It would therefore be logical to assume that only the D-T reaction need be considered to have an appreciable influence below 100 keV. At this energy, the probability of a D-T reaction occurring is 200 times greater than a D-D reaction.

Without knowing the exact configuration of a thermonuclear device, it is impossible to calculate any preferred direction of neutron release, if any. For this reason it will be assumed that the neutrons produced in reaction (2) are released isotropically. The medium in which they
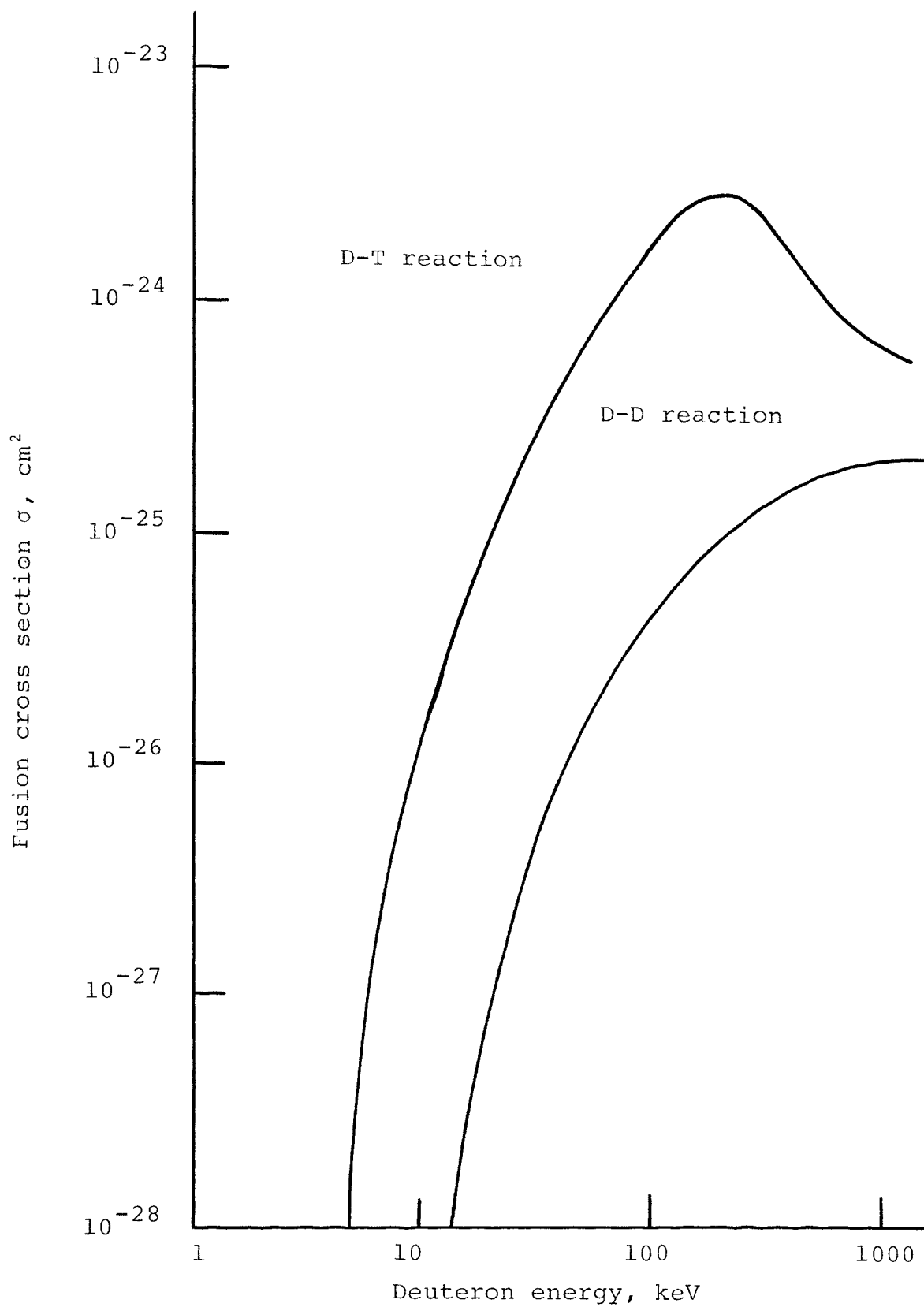
Figure 1. Fusion cross section versus deuteron energy.

will ultimately be released is a concentric shell surrounding the spherical device. The shell, by design, will moderate and capture all neutrons released. This would provide a spherical, infinite, homogeneous medium.

Any (n,γ) reaction within the spherical fuel can be disregarded due to the low capture cross section of deuterium (0.5 ± 0.1 mb at thermal and 29.4 ± 5.8 µb at 14.4 MeV) and tritium (less than 6.7 µb at thermal energy)[5,7]. The product He^4 of reaction (2) has an absorption cross section of essentially zero[5,7] and can therefore be disregarded. In view of this and the discussion later, it is assumed that the energies at which the released neutrons enter the moderating shield will range about 14 MeV.

Certain numerical assumptions are also made through the extrapolation of the most reasonable known data for thermal energies. The capture cross sections of strontium-90, cesium-137, cerium-144, technetium-99, rhodium-103, -104, and -105 are known or approximated (Table II). The neutron rich isotopes of these nuclides have not had their cross sections determined. It is assumed that these values will vary according to whether the isotope is odd or even. As odd isotopes normally have a higher capture cross section than even ones, the odd isotopes will possess a cross section of one magnitude higher than the nearest measurable even one or equal to the nearest measured odd one. The even isotopes will possess a capture cross section of one

D-T
Fuel
Core

Beryllium
Moderator

Reactant
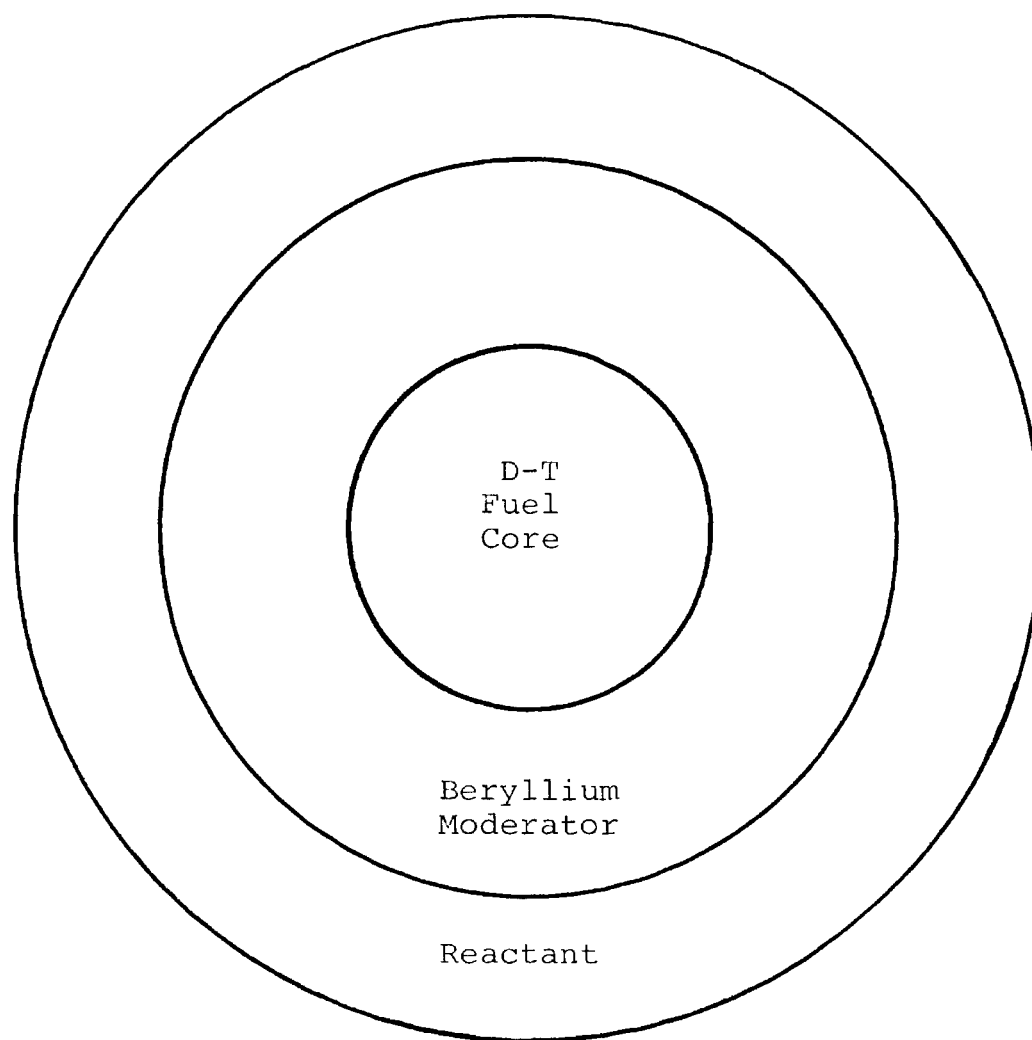
Figure 2. Device Schematic

TABLE II. Thermal Cross Section Data

Nuclide	σ_C (b)	σ_S (b)	D (cm)	L (cm)	ρ (g/cm^3)	Σ_a (cm^{-1})	Σ_s (cm^{-1})	Σ_t (cm^{-1})
Sr-90	1.0	10*	15.83	30.1	2.6	0.01742	0.1742	0.1916
-91	10		0.485	1.676		0.1623	0.1732	0.3446
-92	1.0		16.17	30.85		0.01704	0.1704	0.1874
Cs-137	110 mb*	20*	2.0	47.0	1.87	9.07 x 10^{-4}	0.1649	0.1658
-138	11 mb		2.04	150.0		9.07 x 10^{-5}	0.1636	0.1637
-139	110 mb		2.03	47.7		8.94 x 10^{-4}	0.1625	0.1634
-140	11 mb		2.07	152.8		8.87 x 10^{-5}	0.1613	0.1614
Ce-144	1.0*	9*	1.059	6.10	6.78	0.0284	0.255	0.2834
-145	10		0.296	1.025		0.281	0.253	0.534
-146	1.0		1.075	6.20		0.0279	0.251	0.2789
Tc-99	22*	5.8*	0.0353	0.1512	11.50	1.541	0.399	1.940
-100	2.2		0.0439	0.537		0.1525	0.395	0.5475
-101	22		0.0361	0.1545		1.510	0.391	1.901
-102	2.2		0.0448	0.546		0.1496	0.387	0.5366
Rh-103	144*	5*	1.04 x 10^{-3}	9.95 x 10^{-3}	12.41	1.05 x 10^{1}	0.364	10.82
-104	40*		1.14 x 10^{-2}	6.27 x 10^{-2}		2.88	0.360	3.24
-105	18000*		7.24 x 10^{-8}	7.52 x 10^{-6}		1.28 x 10^{3}	0.357	1281
-106	1800		7.26 x 10^{-6}	2.39 x 10^{-4}		1.27 x 10^{2}	0.353	127.3
-107	18000		7.35 x 10^{-8}	7.64 x 10^{-6}		1.26 x 10^{3}	0.350	1260
-108	1800		7.43 x 10^{-6}	2.44 x 10^{-4}		1.25 x 10^{2}	0.347	125.2
-109	18000		7.48 x 10^{-8}	7.78 x 10^{-6}		1.24 x 10^{3}	0.343	1236
-110	1800		7.53 x 10^{-6}	2.48 x 10^{-4}		1.23 x 10^{2}	0.340	122.8
Be-9	9.5*	7*	0.387	18.12	1.85	1.17 x 10^{-3}	0.8652	0.8664

NOTE: *-Measured Cross Section

magnitude less than the nearest measured odd isotope or equal to the nearest even isotope. Table II provides the assumed cross sections of the nuclides under consideration.

The details for the proposed concept will be developed in the discussion by calculating the radius and quantity of each proposed reactant material sufficient to capture the moderated neutrons. Decay of the neutron rich isotopes will be calculated in terms of heat available for transfer by conduction to the host rock. Finally, a conclusion will be made as to which nuclide would be suitable for this purpose.

II. REVIEW OF LITERATURE

The use of nuclear explosives to produce an acceptable environment from which geothermal energy can be extracted for conversion to electrical power was proposed as early as 1958 by Porzel[8] and Teller[9]. At that time, it was unknown whether or not the resulting chimney in underground explosions was accidental or normal. Porzel's concept provided for an empty or molten spherical cavity in granite or basalt. Subsequent shots indicated that retention of a cavity was abnormal[10]. Salmon and Gnome are the only two such cavities that have been produced. The previous in homogeneous salt and the latter in bedded salt. This incorrect assumption temporarily invalidated Porzel's theory until late that year when Johnson and Brown[11] and later Carlson[12] in 1959 and Kennedy[13] in 1964 proposed that the simple fracturing of rock in high geothermal areas by a 5 MT thermonuclear explosion would release sufficient heat energy to supply "... a 50,000 KW steam plant on a steady basis for approximately 10 years".[13] This concept has been given extensive consideration[14-19]. Rawson[20] has recently questioned this concept, however, which refers to the useful fracturing of the rock up to two or three cavity-radii. He holds that the effective fracturing will probably not exceed the cavity radius by significant amounts. This would affect the economics of the program, but not the theory behind it.

Rawson[17] proposed in 1970 that a multiple-shot field would liberate more available geothermal heat from the rock than a single large nuclear shot because more rock would be fractured. This field would be created from sequential explosions, thereby placing additional shear stress to fracture the rock. Ash[21] confirms this in terms of conventional explosives. The increased volume of rock per kiloton of explosive energy would make the program more economical.

The recovery of thermal energy from granitic geothermal deposits and conversion to electrical energy has been a reality for some time. At Larderello, Italy, a 40-HP generator was installed in 1905 using low pressure geothermal steam. Since then, the capacity of this plant has been increased to over 400 megawatts[22]. Iceland, Japan, U.S.S.R., and Imperial Valley, California, have successfully produced economical electrical power[23]. Such power conversion, however, is dependent on the availability of water in the hot intrusive rock. Problems concerning disposal of waste-water and presence of noxious gases continues to hamper these efforts[23].

The original use of explosive energy alone to produce electrical power has given way to the use of natural geothermal energy released from rock fractured by nuclear explosives. The latent energy in the rock is about three to four times the energy of the explosive.[3] Although a

higher quality of steam is produced in this method over that occurring naturally, the presence of neutron induced radio-activity complicates use of the steam for conversion to electrical power[3,16]. Rawson[17] suggests that flooding a chamber around the explosive device will shield the rock from neutron absorption and make available considerable amounts of additional gas to fracture more rock. The result of this method is the formation of considerable quantities of tritium. It would, however, be at a level about 10^{-6} lower than that of the usual production of tritium from lithium in rock.[17] This may be acceptable in natural gas stimulation, but questionably acceptable when steam is used to power electrical generating apparatus. Radiation deposits would accumulate on turbine blades and auxillary equipment, making operational safety a considerable problem.

Isotope production, from underground nuclear explosions, was proposed by Cowan[24] in 1959, Dorn[25], Karraker[26], and Heckman[27,28] in 1964 and by Teller[3] in 1968, although a classified report had been written by him much earlier[29]. Teller described theoretically an approach in that report toward maximizing exposure of a given target to the released neutrons of an explosion. Cohen[30] introduces new uses of the so-called "neutron bomb" which multiplys the number of neutrons released per kiloton. The additional neutrons provide an opportunity to increase the yield through eventual decay of bombarded isotopes. Detonation

of a thermonuclear device produces such an intense neutron flux that it "... can add 15 or more neutrons to a target nucleus ..." in less than a microsecond[26,29]. This specifically refers to the very heavy elements. No work has been published on the use of nuclear explosives to add neutrons to the nucleus of lighter isotopes, especially radioactive isotopes. Lack of probable application may be the reason.

Capture cross sections of radioactive isotopes are being calculated and measured at the present time at the Richland, Washington Hanford Engineering Development Laboratory[31]. Estimations of effects of underground nuclear detonations have recently been given considerable attention and study in this area is still being continued[32-35].

III. DISCUSSION

In order to minimize the effect of neutron capture in the host rock, it must follow that a maximum number of neutrons released by the detonation must be prevented from reaching the rock. To do this, a shield must be provided to absorb the neutrons. The use of radioactive and stable waste products was considered for this purpose. Primarily, the selection of a single isotope was based on the propensity of its more neutron-rich variations to decay with a short half-life to a stable isobar. Secondary considerations were a maximum energy of decay and reasonable thickness required; the latter being a factor in determining drill-hole diameter.

The discussion of the investigation of possible isotopes to be used for this purpose will be subdivided into five sections. These will consist of discussions of moderation of the neutrons prior to absorption, the ability of the isotope to absorb the required neutrons, decay processes to stable end-products, subsequent considerations of physical properties, and a comparison of results.

A. Moderation

A concentric shell is provided which surrounds the deuterium-tritium core of the explosive device. This shell acts as a moderator for the neutron flux released during the initial phase of the explosion. As the nature of the

moderator is not primarily the intent of this discussion,
a material was arbitrarily selected to provide sufficent
data to investigate the remaining portions of the subject.
Beryllium, which is too expensive for use in normal shield-
ing or moderating, was selected because it offered a smal-
ler thickness capable of slowing the 14 MeV neutrons to
thermal energies.

The central core of the device precludes the treatment
of the moderator as a medium surrounding an ideal point
neutron source. The source, based on AEC recommendations
for a 1 MT explosion, would possess a radius of 30.48 cm.
This source, however, would be isotropic and could be
considered as a point source when compared to an infinite
surrounding medium. Furthermore, none of the released
neutrons escape without encountering the moderator. In
calculating the required thickness of the moderating
beryllium shell, the D-T core was considered to be a point
source. The probability that a neutron emitted from the
source acquires Fermi Age τ between r and r + dr is:

$$p(r) = \frac{4\pi r^2 \exp(-r^2/4\tau)}{(4\pi r)^{3/2}} \tag{4}$$

and the second moment of p(r) gives the average radius
squared.

$$\bar{r}^2 = \int_0^\infty r^2 p(r) dr = 6\tau \tag{5}$$

The average thickness for thermalization (t_m) is defined as being equal to $\bar{r}$ and is therefore $\sqrt{6\tau}$.

The Fermi Age required from 14 MeV to 2 MeV is given by:

$$\tau(E) = \int_{E}^{E_O} \frac{D(E)}{\xi\Sigma_s(E)} \frac{dE}{E} \qquad (6)$$

The energy dependent values of D and Σ_s can be considered a constant for values of E from 2 MeV to 14 MeV. In this case, values of $D = 0.387$, $\Sigma_s = 0.865 cm^{-1}$ and $\xi = 0.207$ were used giving a value of $\tau = 3.72 \ cm^2$. The value of τ from 2 MeV to 0.025 ev is given as 102 cm^2, giving a total quantity of $\tau_T = 105.72 \ cm^2$ to thermal.[36] Based upon the $\sqrt{6\tau}$ factor, the thickness of beryllium required to thermalize the 14 MeV neutrons would be 25.2 cm. This value represents the maximum required thickness as its volume from a radius of 30.48 cm to 55.68 cm would be larger than that of a sphere of 25.2 cm radius by 13.8%. The total radius to include fuel and moderator would be 55.68 cm.

The preceeding is based on the assumption that the most energetic neutrons will be at 14 MeV. These neutrons must, however, be transported through a portion of the D-T fuel mixture and, as the reaction approaches completion, an increasing amount of $_2He^4$. Absorption and loss of energy by scattering within the fuel must also be considered.

In a balanced D-T reaction for a 1 MT explosion, the total of the released neutrons would equal the total

number of deuterium and tritium nuclei persent and in equal
quantities, or 0.745 x 10^{27} atoms each. The resulting
fusion product, helium-4, would exist in quantities from
zero to 1.49 x 10^{27} nuclei as the reaction progresses
towards completion.

As the absorption and scattering cross sections for
deuterium and tritium are neglible, the worst possible
condition for either case would be at the end of the re-
action where there would be, theoretically, 100% helium
present in the fuel core. Calculating τ for a radius of
30.48 cm of helium gives: $\tau = 155 \; cm^2$. Further, using
$\tau = \int_{E}^{E_O} (D/\xi\Sigma_S) dE/E$ as before, yields a value of 13.9 MeV
for E or a loss of energy of 0.1 MeV from scattering.
This would not contribute significantly. The loss of
neutrons due to absorption can be estimated by measuring
the mean free path of a neutron in helium, which is
$\lambda s = 46.6$ cm for scatter, $\lambda a = 74.4$ x 10^4 cm for absorption
and $\lambda t = 47.6$ cm for both. For a fuel core of 60.96 cm
diameter, one interaction by scatter may occur (loss of
0.1 MeV) but it is highly improbable that any absorption
interaction would take place. The number of neutrons
reaching the moderator shield would be 1.49 x 10^{27} for a
1 MT explosion.

B. Absorption

To effectively reduce the long-lived radiation within
the immediate proximity of the explosion, all the released

neutrons must be absorbed. For computational purposes, it is assumed that all of the 14 MeV neutrons have been thermalized by the beryllium shield. This leads to the investigation of each of the six isotopes for absorption ability.

Values of D and L were calculated from:

$$D = \frac{\Sigma s}{3\Sigma t^2} \quad \text{and} \quad L = \sqrt{\frac{D}{\Sigma a}} \tag{7,8}$$

Various sources were used to calculate the macroscopic cross sections (Table II)[5,7,36]. Once more considering the fuel as a point source in an infinite medium and the second moment of $p(r)$ to be equal to $6L^2$, the average thickness was determined[36] as $L\sqrt{6}$. Table III provides these values under t_r and the volumes of the absorbing shell as V_r.

Not only must all the neutrons be absorbed in (n,γ) reactions, but also the radioactive isotopes must all be reacted upon to produce neutron-rich isotopes which will rapidly decay. To allow any of the radioactive isotopes to remain with long half-lives would be to defeat the purpose altogether. With 1.49×10^{27} neutrons each being absorbed by a nucleus, a maximum number of 1.49×10^{27} nuclei of reactant is allowable for the reaction. Values for volumes of each nuclide are indicated in Table III as well as the corresponding maximum allowable thickness of the shell. The last column indicates the excess of the

TABLE III. Absorption Data

Nuclide	Average Absorber thickness (cm)		Average Volume of Reactant (cm^3)		Excess Volume (cm^3)
	t_r	t_{max}	V_r	V_{max}	
Sr-90	73.8	10.75	83.97×10^5	8.57×10^4	83.11×10^5
Cs-137	115.2	16.00	30.70×10^6	18.09×10^4	30.52×10^6
Ce-144	14.96	8.3	7.51×10^5	5.25×10^4	69.85×10^4
Tc-99	0.371	5.4	1.45×10^4	2.13×10^4	0
Rh-103	0.0243	5.31	0.05×10^4	2.05×10^4	0
Rh-105	18.42×10^{-6}	5.35	0	2.09×10^4	0

t_r - Thickness of reactant required to absorb 1.49×10^{27} neutrons

t_{max} - Thickness of reactant based on 1.49×10^{27} atoms

V_r - Volume of shell with thickness t_r

V_{max} - Volume of shell with thickness t_{max}

NOTE: Thickness of moderator shell t_m is 25.2 cm and radius of fuel core is 30.48 cm.

nuclide which would not be reacted upon if the absorption thickness (t_r) is used. Sr-90, Cs-137, and Ce-144 all require too great a thickness to preclude an excess of unreacted nuclei remaining. Tc-99, Rh-103, and -105 provide a greater allowable thickness. Rh-103 is not radioactive, however, and is included as a comparison.

As with the value for the thickness of the moderator, the calculated thickness of the various absorbing nuclides is a maximum based on the source as a point and a homogeneous medium immediately surrounding it.

C. Decay

The chain of decay for each nuclide following neutron capture must be such that its total energy of decay is deposited within a short period, or at least faster than is normally experienced within the surrounding rock. This would preclude forming any isotope which has a long half-life isotope in its decay chain. Of the neutron-rich isotopes of the six selected nuclides, only Rh-107 is extensively long in that its daughter Pd-107 decays to Ag-107 with a 7×10^6 year half-life. To be of value the total half-life should not exceed about 30 hours. In Table IV the various decay chains are represented. Only the values of β^- decay were used with the exception of the ones noted. Some isotopes have only been measured for total energy of decay with β^- decay the primary source of

TABLE IV.   β⁻ Decay Chain to Stable Nuclides

Nuclide	Half-life	Dau.	Decay Energy (MeV)	Half-life	Grand-dau.	Decay Energy (MeV)	Remarks
Sr-90	27.7 yr	Y-90	0.546	64 hr	Zr-90	2.27	
-91	9.67 hr	Y-91	2.67	58.8 days	Zr-91	1.145	
-92	2.71 hr	Y-92	1.5	3.53 hr	Zr-92	3.63	
Cs-137	30.0 yr	Ba-137	1.176	-	-	-	
-138	32.2 min	Ba-138	3.40	-	-	-	
-139	9.5 min	Ba-139	4.0*	82.9 min	La-139	2.3	
-140	66 sec	Ba-140	6.1*	12.8 days	La-140	1.02	Decays to Ce-140
Ce-144	284 days	Pr-144	0.316	17.3 min	Nd-144	2.99	Decays to Sm-144
-145	3.0 min	Pr-145	1.7	5.98 hr	Nd-145	1.8	
-146	14 min	Pr-146	0.73	24.2 min	Nd-146	2.1	
Tc-99	2.12 x 10⁵ yr	Ru-99	0.292	-	-	-	
-100	15.8 sec	Ru-100	3.38	-	-	-	
-101	14.0 min	Ru-101	1.32	-	-	-	
-102	4.5 min	Ru-102	4.4	-	-	-	
Rh-103		-------Stable-------					
-104	43 sec	Pd-104	2.44	-	-	-	
-105	35.88 hr	Pd-105	0.568	-	-	-	
-106	30 sec	Pd-106	3.54	-	-	-	
-107	22.4 min	Pd-107	1.2	7 x 10⁶ yr	Ag-107	0.035	
-108	17 sec	Pd-108	4.5	-	-	-	
-109	30 sec	Pd-109	2.5*	13.46 hr	Ag-109	1.03	
-110	5 sec	Pd-110	5.5	-	-	-	

*Primarily β⁻ decay energy plus some minor secondary γ energy.
NOTE: Underlined nuclides are stable isotopes.

the energy. Decay constants were calculated based upon[36] $\lambda = \ln 2/t\ 1/2$ and appear in Table V. The activity expressed in Curies was calculated from $A = \lambda N$. This can be compared with the values measured for radiative capture in basalt[3] (Table VII). Both Tables VI and VII include the amount of initial activity and that after 60 days. This includes the activity of the parent nuclide and any daughter or grandaughter nuclide present initially or after 60 days.

D. Deposited Energies

The energy deposited in the form of heat is a bonus contribution to the decay process.[37] Fully 90% of the released energy of the explosion, (3.57×10^{12} Btu for a 1 MT explosion) is deposited in the rock[27]. In a geothermal field where the latent heat of fusion is 0.318 Btu, per gram and mean volume of rock fractured by a 1 MT explosion is 9.85×10^{12} cm^3 with an <u>in situ</u> density of 2.66 gm/cm^3, the amount of available heat is 8.29×10^{12} Btu.[16] This is only two times the amount of heat deposited by the nuclear explosion. Such an amount of heat cannot be ignored.

Table VIII shows the amount of heat deposited in basaltic rock by a 1 MT explosion by the decay process. The heat deposited within 60 days (q) and total heat deposited (Q) both include the parent as well as daughter nuclides. The total deposited heat Q_t equals 5.68×10^8 Btu or about

TABLE V. Decay Constants of Decay Chains

Nuclide	Decay Constant	Dau.	Decay Constant	Grand-dau.	Decay Constant	Grand-dau.
Sr-90	7.94×10^{-9}	Y-90	3.01×10^{-5}	Zr-90	–	
-91	1.99×10^{-5}	Y-91	1.36×10^{-7}	Zr-91	–	
-92	7.10×10^{-5}	Y-92	5.45×10^{-5}	Zr-92	–	
Cs-137	7.33×10^{-10}	Ba-137	–	–	–	
-138	3.59×10^{-4}	Ba-138	–	–	–	
-139	1.22×10^{-3}	Ba-139	1.39×10^{-4}	La-139	–	
-140	1.05×10^{-2}	Ba-140	6.27×10^{-7}	La-140	4.79×10^{-6}	Ce-140
Ce-144	2.83×10^{-7}	Pr-144	6.69×10^{-4}	Nd-144	----to Sm-144----→	
-145	3.85×10^{-2}	Pr-145	3.22×10^{-5}	Nd-145	–	
-146	8.26×10^{-4}	Pr-146	4.82×10^{-4}	Nd-146	–	
Tc-99	1.04×10^{-13}	Ru-99	–	–	–	
-100	4.39×10^{-3}	Ru-100	–	–	–	
-101	8.26×10^{-4}	Ru-101	–	–	–	
-102	2.57×10^{-3}	Ru-102	–	–	–	
Rh-103	–	–	–	–	–	
-104	1.61×10^{-2}	Pd-104	–	–	–	
-105	5.37×10^{-6}	Pd-105	–	–	–	
-106	2.31×10^{-2}	Pd-106	–	–	–	
-107	5.40×10^{-4}	Pd-107	3.40×10^{-12}	Ag-107	–	
-108	4.07×10^{-2}	Pd-108	–	–	–	
-109	2.31×10^{-2}	Pd-109	1.43×10^{-5}	Ag-109	–	
-110	1.39×10^{-1}	Pd-110	–	–	–	

NOTE: Underlined isotopes are stable.

TABLE VI.   Activity of Reactant Nuclides

Nuclide	A_O (t=0) (Curies)	A (t=60 days) (Curies)	Remarks
Sr-90	3.20×10^8	3.20×10^8	Cont. decay to Zr-90
-91	8.03×10^{11}	4.01×10^{11}	Incl. dau. decay
-92	2.86×10^{12}	6.55	" " "
Cs-137	2.95×10^7	2.30×10^7	Stable
-138	1.44×10^{13}	negligible	"
-139	4.86×10^{13}	"	Incl. dau. decay
-140	4.23×10^{14}	3.06×10^{14}	" " "
Ce-144	1.14×10^{10}	1.14×10^{10}	Continues to Sm-144
-145	1.55×10^{15}	8.49×10^7	Incl. dau. decay
-146	3.33×10^{13}	negligible	" " "
Tc-99	1.17×10^4	1.17×10^4	Stable
-100	1.77×10^{14}	negligible	"
-101	3.33×10^{13}	"	"
-102	1.04×10^{13}	"	"
Rh-103		------Not Radioactive------	
-104	6.48×10^{14}	negligible	Stable
-105	2.16×10^{11}	1.52×10^{10}	"
-106	9.32×10^{14}	negligible	"
-107	2.17×10^{13}	2.17×10^{13}	Incl. dau. decay
-108	1.64×10^{15}	negligible	Stable
-109	9.32×10^{14}	5.71×10^{11}	Incl. dau. decay
-110	5.58×10^{15}	negligible	Stable

TABLE VII.   Activity of Nuclides in Basalt

Nuclide	A_O (t=0) (Curies)	A (t=60 days) (Curies)	Decay Mode	λ (sec^{-1})
Si-31	7.1×10^7	2.17×10^{-9}	β^- to P-31	7.34×10^{-5}
Fe-55	7.6×10^5	7.53×10^4	ϵ to Mn-55	8.14×10^{-9}
Fe-59	3.9×10^4	3.45×10^4	β^- to Co-59	1.78×10^{-7}
Mn-56	8.5×10^8	1.30×10^8	β^- to Fe-56	7.47×10^{-5}
Ca-45	4.3×10^4	4.13×10^4	β^- to Sc-45	4.86×10^{-8}
Na-24	1.9×10^8	negligible	β^- to Mg-24	1.28×10^{-4}
K-42	2.8×10^6	8.90×10^2	β^- to Ca-42	1.55×10^{-5}
P-32	1.8×10^5	1.33×10^5	β^- to S-32	5.60×10^{-7}
Total	11.2×10^8	1.30×10^8	–	–

TABLE VIII.   Heat of Decay Transferred to Rock by Basaltic Nuclides

Nuclide	n_o	n_{60}	n_d	E	q	Q
Si-31	3.58×10^{22}	1.09×10^6	3.58×10^{22}	1.47	8.00×10^6	8.00×10^6
Fe-55	3.46×10^{24}	3.42×10^{23}	3.12×10^{24}	0.006	2.85×10^5	3.16×10^6
Fe-59	8.10×10^{21}	7.17×10^{21}	9.30×10^{22}	0.27	3.82×10^4	3.28×10^5
Mn-59	4.21×10^{23}	6.45×10^{22}	3.56×10^{23}	2.8	1.52×10^8	1.79×10^8
Ca-45	3.27×10^{22}	3.15×10^{22}	1.20×10^{21}	0.255	4.66×10^4	1.27×10^6
Na-24	5.48×10^{22}	0	5.48×10^{22}	1.4	1.17×10^7	1.17×10^7
K-42	6.68×10^{21}	2.13×10^{18}	6.68×10^{21}	3.55	3.61×10^6	3.61×10^6
P-32	1.19×10^{22}	8.80×10^{21}	3.10×10^{21}	1.7	8.02×10^5	3.08×10^6

n_o - Number of parent nuclei

n_{60} - Number of parent nuclei after 60 days

n_d - Number of parent nuclei decayed after 60 days

E - Energy of decay in MeV/disintegration

q - Heat deposited in 60 days = $1.52 \times 10^{-16} n_d E$ Btu

Q - Total heat of decay = $1.52 \times 10^{-16} n_o E$ Btu

0.6% of that achieved by any of the selected nuclides (Table IX). Of the selected nuclides in Table IX, only technetium deposits all of its decay thermal energy within 60 days. Rhodium (with the exception of the odd isotopes) decays completely and cerium and cesium approach complete deposition of their heat in the same time period.

E. Physical Properties

The crushed and heated zone surrounding the point of detonation is to be injected with water to provide the required steam for power generating operations. Subjecting any of the isotopes of strontium, cesium or cerium to water results in decomposition of the nuclide. Deposition of radioactive particles carried up by the steam would introduce a measurable safety hazard. Use of technetium or rhodium, which are insoluble in water, would greatly enhance filtration and purification of the steam prior to introduction into the turbine complex. Although groundwater leakage from the vicinity of the chimney would be minor, use of nuclides which are insoluble in water would minimize the danger of radioactive particles escaping into the biosphere.

Of all the selected isotopes, only cesium melts at a temperature (28.6°C) below the ambient temperature at great depths in a geothermal environment. This might hamper its use for this purpose.

TABLE IX. Heat of Decay Transferred to Rock by Reactants

Nuclide	n_{60}		$n_d \times 10^{27}$		E (MeV/dis)		$q \times 10^{11}$	$Q \times 10^{11}$
	Parent	Daughter	Par.	Dau.	Par.	Dau.	Btu	Btu
Sr-91	4.70×10^{22}	0	1.49	0.74	2.67	1.55	7.80	9.55
-92	1.49×10^{12}	1.20×10^{5}	1.49	1.49	1.50	3.63	11.60	11.62
Cs-138	0	–	1.49	–	3.40	–	7.70	7.70
-139	0	0	1.49	1.49	4.00	2.30	14.30	14.28
-140	0	4.88×10^{20}	1.49	1.49	6.10	1.02	16.13	16.13
Ce-145	0	2.64×10^{12}	1.49	1.49	1.70	1.80	7.93	7.93
-146	0	0	1.49	1.49	0.73	2.10	6.41	6.41
Tc-100	0	–	1.49	–	3.38	–	7.66	7.66
-101	0	–	1.49	–	1.32	–	2.99	2.99
-102	0	–	1.49	–	4.40	–	9.97	9.97
Rh-104	0	–	1.49	–	2.44	–	5.53	5.53
-105	1.05×10^{26}	–	1.39	–	0.57	–	1.12	1.23
-106	0	–	1.49	–	3.54	–	8.02	8.02
-107	0	1.49×10^{27}	1.49	0	1.20	0.035	2.72	2.80
-108	0	–	1.49	–	4.50	–	10.20	10.20
-109	0	3.99×10^{16}	1.49	1.49	2.50	1.03	7.99	7.99
-110	0	–	1.49	–	5.50	–	12.46	12.46

n_o – Number of radioactive atoms at t=0 (1.49×10^{27})
n_{60} – Number of radioactive atoms at t=60 days
n_d – Number of radioactive atoms decayed at t=60 days; $n_d = n_o - n_{60}$
$q = 1.52 \times 10^{-16} n_d E$
$Q = 1.52 \times 10^{-16} n_o E$

Technetium and rhodium, at present, are processed at a
cost far in excess of that considered economical. The
quantity of technetium necessary would cost in excess of
$20 million whereas rhodium (in a variety of isotopes)
would cost about $280,000 for the quantity required[1].
Rhodium-103 is a stable nuclide and produced either from
decay of ruthenium-103 (a fission product with a half-life
of 39.5 days) or refined from naturally occurring ore.
Other isotopes of the element, especially Rh-105, are
fission products capable of being separated and utilized.

F. Comparison of Results

In comparing the results of the calculations, five
aspects must be examined. The primary consideration is to
minimize the neutron capture in the host rock. To accomplish
this, a maximum number of neutrons must be captured by the
reactant. All the nuclides investigated are acceptable for
this purpose. For strontium-90 and cesium-137, however,
the diameter of the device would exceed 100 inches. The cost
of drilling such a large hole to the depth of 10,000 feet
would be in excess of $10 million. Use of cerium would de-
crease the diameter to about 54 inches for a cost of $8.6
million, (a cased hole would be $11.7 million). Use of
technetium or rhodium would necessitate a minimum diameter
of 45 inches to a maximum of 50 inches. This would cost
from $3 million to $4.5 million for an uncased hole or

$6.0 million to $9.5 million for a cased hole[16].

The second consideration would be existence of parent nuclide residue in a long half-life state. As seen in Table III a residue of 83.11×10^5 cm³ of Sr-90 remains following the explosion, 305.2×10^5 cm³ of Cs-137 residue remain and 6.985×10^5 cm³ of Ce-144 remain. This trend is reversed for both Tc-99 and rhodium. Both nuclides require less than the allowable volume (based on one neutron per nucleus reaction for a 1 MT fusion explosion) to capture the released neutrons.

The third consideration is the presence of any long-lived isotopes after a reasonable length of time. All nuclides, with the exception of technetium, include a possibility of some long-lived isotopes remaining after a period of 60 days. The odd isotopes of rhodium indicate some activity (Table IV) after 60 days. The exact amount is unknown and can be surmised only when based upon an arbitrary selection of probable capture cross sections. As the odd isotopes habitually possess higher capture cross sections, it is logical to assume that if multiple capture occurs, it would leave a preponderance of even isotopes of the element. These exhibit negligible activity after 60 days.

The fourth consideration deals with the quantity of heat deposited by radiative decay. Cesium would deposit the most heat based on the decay energies of Ce-145 and -146. The remainder of the nuclides decay with a varying amount

of energy dependent on the particular isotope and the per-
centage of that isotope actually produced. Exact quantities
can be determined only through experimentation or computer
simulation. A comparison of Tables VIII and IX, however,
indicates a considerable increase in the average amount of
energy deposited as heat by the selected nuclides than in
the nuclides of basaltic rock.

The last aspect for consideration is the physical pro-
perties and cost of the nuclide. Of those selected, only
technetium and rhodium and their products are insoluble in
water. These both can only be produced, however, at a cost
in excess of that reasonable for the purpose. Of the two,
rhodium would be the least expensive and virtually any of
its isotopes above Rh-103 may be used, were it not for short
half-lives. Only Rh-103, which is stable, could be
realistically utilized.

IV. CONCLUSIONS

Based upon the five considerations from the comparison
of results, only rhodium-103 appears to hold promise for
use in minimizing the radiative capture effect in the host
rock of an underground nuclear explosion. Strontium,
cerium and cesium should be eliminated at the present time
due to the thickness of shield required, amount of parent
nuclide residue remaining, activity present after reasonable
periods of time, and their solubility in water. These
problems may be overcome, however, with the use of an outer
cadmium reflector to lessen the amount of reactant requir-
ed, acceptance of a longer decay period, and installation
of a dual-loop heat exchanger to eliminate contamination
in the turbine. Technetium-99 and rhodium-103 are accept-
able from an application viewpoint. Until such time as
technetium is made available at reasonable costs, its use
is uneconomical.

Further study is recommended not only in reduction of
activity, but also in the disposal of nuclear waste through
exposure to the large neutron flux of a thermonuclear ex-
plosion. This concept can have even greater applications
if the neutron flux is first exposed to an isotope with a
large (n,2n) cross section. This would increase the econ-
omical aspect of waste disposal and power generation.

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VITA

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