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WANL-TME-1345

May 1966

Westinghouse Astronuclear Laboratory

A SURVEY OF POTENTIAL SHIELD MATERIALS

(Title Unclassified)

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SPECIAL REVIEW FINAL DETERMINATION Class: <u>U</u>	Reviewed <u>KAW</u>	Class. <u>U</u>	Date <u>4-15-82</u>

Classification cancelled (or changed to)
by authority of
by H. F. C. TIC, date SEP 11 1973

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INFORMATION CATEGORY <u>C-R.D.</u>
<u>R. J. Ricks</u> <u>5-9-66</u>
AUTHORIZED CLASSIFIER DATE

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ACKNOWLEDGEMENT

The authors gratefully acknowledge the assistance of Mr. H. C. Woodsum who made many helpful suggestions in the preparation of the nuclear material comparisons code.

We also wish to acknowledge the contributions from Mr. R. N. Nassano, who performed the TAPAT system calculations, and Miss S. L. Zeigler for her patient data preparation and reduction.

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SUMMARY

Twenty-five shield materials for application in the NR-1 reactor have been evaluated from a shield weight, chemical, thermal, mechanical, and cost standpoint. Seven combinations of materials were found that appear to best satisfy these combined considerations for the NR-1 shield. Beryllium steel, a composite of $\text{TiH}_2\text{-B}_4\text{C-Al}$, and LiH-steel were selected as the best materials, each having its own advantages. It is recommended that the $\text{TiH}_2\text{-B}_4\text{C-Al}$ composite be developed for the NR-1 flight shield, with the beryllium-steel as a prime backup; and LiH as a secondary backup.

This report also describes the development of a fast, efficient, shield-weight optimization technique, containing a simplified physical description of the shield attenuation processes.

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BACKGROUND

The study of the shield effectiveness in an NR-1 reactor for a single reactor-propellant tank module or a triangular or in-line cluster of three modules, was begun in CY 1965 under project NE 1840 and is continued in CY 1966 under project NE 9200. The CY 1965 studies considered the effect of a shield above the reactor core support plate and vertical shields at the ID of the pressure vessel, extending from the bottom of the core to meet the top shield. The shield effectiveness was evaluated on the basis of propellant heating in single and clustered hydrogen propellant tanks. These studies resulted in the conceptual design of a flat, constant thickness shield of aluminum and beryllium oxide mounted above the reactor core with a vertical annulus extending down to the reflector at its perimeter. This design is reported in WANL-TME-1300⁽¹⁾. The effects of relocating the top shield below the core support will be considered in the NR-1 subsystems parametric study. The nuclear radiation heating in the tank directly above the reactor was reduced by 80 percent with the top shield⁽¹⁾ alone and 86 percent with the top shield and the vertical extension. The weight of the top shield was approximately 3000 pounds, while the weight of the extension was just over 1200 pounds.

In this study consideration is given to the evaluation of different shield materials and their optimum volume fractions to obtain a desired attenuation for a specified neutron and gamma radiation level. Any material that is ultimately chosen as a desirable shield material must meet not only nuclear requirements, but also practical requirements of physical and chemical properties and cost. Therefore, this investigation has given consideration to the nuclear, physical, chemical, and economic characteristics of the materials and their combinations or applications in an NR-1 reactor shield.

One or more of these factors was not evaluated in earlier surveys^(2, 3, and 4). Some of the important factors that must be included in the evaluation of materials for use in a practical shield design, and which are included in this study, are listed below:

- 1) Neutron and gamma ray attenuation properties;
- 2) Secondary gamma ray production;
- 3) Secondary gamma ray suppression;

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- 4) Slowing down of fast and intermediate energy neutrons;
- 5) Physical properties (mechanical, thermal, chemical, etc.);
- 6) Radiation effects;
- 7) Cost.

The nuclear evaluation of the shield materials considers, among other things: the attenuation, the volume fractions required of each material, and the shield weight. These quantities are evaluated with the aid of a code developed especially for this purpose. The physical properties include consideration of chemical, thermal, and mechanical properties, qualities of the material as well as their compatibility with the body of the shield and the reactor environment. The cost considerations of the shield include the raw material cost, development requirements, and fabrication.

Since at this time absolute shield criteria are not available and may well be completely dependent upon mission requirements, several shield radiation transmission levels were evaluated. These levels of transmission which included, 75, 80, 90, and 96 percent removal of the radiation incident on the shield, will affect the volume fractions and the optimum mixture of the shield materials. However, a review of the information presented herein will allow some judgment in determining the effects of various missions on changing shield requirements and on the shield design. Note that the flexible design of the conceptual shield presented in reference 1 should readily adapt to changes in either materials or volume fractions of the materials to accommodate a large variety of missions. Thus the purpose of this study is to determine the most desirable materials and combinations of materials to be incorporated in the design of a flight-type shield for an NR-1 reactor.

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I. BASIS OF EVALUATION

A. NUCLEAR EVALUATION

1. Preferred Qualities of Shielding Materials

A desirable lightweight reactor shield should include the characteristics of high gamma ray and fast neutron attenuation as well as rapid thermalization and resultant absorption of intermediate energy neutrons with little secondary gamma ray yield from neutron capture or inelastic scattering.

Hydrogen is very attractive for several of its characteristics - large values of gamma ray attenuation per unit weight (μ/ρ) for photon energies between 0.5 and 5.0 Mev, and high neutron slowing down power per unit weight (Σ_s/ρ) for intermediate energy neutrons (ev and Kev range). Many of the advantages of hydrogen are lost because of its low density high gamma ray build-up factor, and the high volume requirement for practical materials. When placed in a compound, its gamma ray shielding effectiveness is reduced since hydrogen electrons may be only a small fraction of those in the molecule. Although hydrogen is the most effective element for neutron slowing down, the overall effectiveness of a shield can often be improved by the addition of heavier atoms to help degrade high energy neutrons (>1 Mev) by the inelastic process where hydrogen is less effective and to attenuate low- and high-energy gamma rays by the photoelectric and pair production processes. Thus the shield material mixtures studied include low atomic weight atoms to aid in moderating the neutrons, high atomic weight atoms to attenuate the gamma rays, and boron to suppress the secondary gamma ray intensity. In an attempt to evaluate various materials for the desired qualities, three group representations of the gamma ray attenuation (μ/ρ), fast neutron removal (Σ_R/ρ), and neutron slowing down density Q (Σ_R, r) have been included in a Shield Materials Optimization Code described in Chapter I of the Appendix. Also included are the intensity and resultant attenuation of emergent secondary gamma rays from capture of the incident thermal and slowed neutrons. Boron is added to each material in small amounts to suppress the secondary gamma ray intensity. The gamma ray yield per capture in the non-boron or lithium

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materials ranges from about 2.2 Mev to over 7 Mev. Sufficient boron is added so that 90 percent of the neutrons will be absorbed in the boron and hence will give rise to only 0.45 Mev of gamma capture energy. (An exception in the case of Be and BeO is noted in the Appendix B. Figure 26 is a diagram of the input and output quantities considered in the shield materials optimization code.)

Factors considered in the selection of materials for nuclear evaluation were:

- 1) Gamma ray attenuation
- 2) Fast neutron removal
- 3) Intermediate neutron slowing down to thermal
- 4) Secondary gamma ray yield
- 5) Physical characteristics

The choice of materials is so wide that previous surveys^(2, 3, and 4) were used to narrow the choice to about 25 neutron materials to be combined with either steel or aluminum as supporting structure. While it was obvious that certain materials listed in table 19 would rate poorly in some of the above categories, it appeared that some of their advantages could outweigh their disadvantages. In any case, the attributes and liabilities of many material interact in such a way that their total effect may be difficult to discern without a detailed evaluation. Such an evaluation is the purpose of this study.

2. Assumptions

Assumptions made in the evaluation of potential shield materials involve:

- 1) Intensity of incident radiation on the shield;
- 2) Relative importance of the radiation components escaping the shield in terms of representing energy deposition in the propellant tank;
- 3) Adequacy of the two-group neutron treatment;
- 4) Adequacy of the three-group gamma ray treatment;
- 5) Absorption of 90 percent of the neutrons in boron* or lithium
- 6) Rocket stage weight change versus shield thickness.

* See exception noted in Chapter I B of the Appendix

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The neutron fluxes incident on the shield were taken from diffusion theory calculations^{*} presented in WANL-TME-1300⁽¹⁾. The gamma ray flux incident on the shield was determined by applying a distance correction factor to the unshielded point kernel calculations of the total radiation level at 10 feet above the core midplane⁽¹⁾.

The ratio of the two-group neutron and gamma ray fluxes exiting the shield to their respective heating contributions in the central propellant tank was used as a normalizing factor for each of the radiation components. These heating contributions were calculated using the TIC-TOC-TOE codes^(1, 5).

The neutrons incident on and emergent from the shield are treated as two groups, fast and slow. The conservative assumption for this analysis is made that no captures occur while slowing down and hence the secondary gamma ray sources are displaced closer to the exit face of the shield by a distance proportional to the removal cross section and the age to thermal. One-group treatment, such as removal theory, does not take into account the epithermal neutrons that are captured and create 2 to 8 Mev of gamma radiation per capture in and beyond the shield. The two-group treatment goes a step further than the removal theory in considering the ability of various materials to thermalize and capture the removed neutrons in the shield. A more detailed multigroup analysis is made of the most promising material combinations using transport theory calculations as described in Chapter I B of the Appendix.

Gamma rays originating in parts of the reactor other than the shield are considered to have an effective photon energy of 2 Mev, as established by photon transport calculations. The secondary gamma rays are treated in two groups with average energies of 0.5 Mev and a hard gamma ray energy corresponding to the average energy of emission from capture in the shield material mixture. The 0.5 Mev attenuation coefficient is used for the gamma rays produced by neutron capture in the boron.

Boron is provided in sufficient quantity to absorb 90 percent of the neutrons in the boron of the shield and limit the secondary shield radiation to approximately 10 percent of the total gamma energy above the reactor. Where boron or lithium is not a basic part of the material compound, it is

* The agreement between diffusion and transport calculations of flux levels at the core inlet side of the shield was very good.

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assumed that adequate boron can be included in an acceptable form, such as in extrusions, pressed inserts, cladding, mixtures, or boron bearing washers or spacers.

While 90 percent suppression has been assumed for the majority of this study (see the exception in the case of Be and BeO (Appendix B) the boron concentration will be determined from the shields' secondary gamma ray contribution to the total radiation level above the reactor and from practical considerations concerning how much boron can be physically added. The determination of optimum boron concentration is considered beyond the scope of this study.

Weighting, or units normalizing, coefficients, K_i , are included in the optimization program formulation for each radiation component leaving the shield. These factors can be determined from propellant heating calculations in the central propellant tank by taking the ratio of unshielded neutron and gamma ray heat Q_i in the tank to the respective incident flux, ϕ_i from Reference 1.

$$K_i = \frac{Q_i}{\phi_i}$$

While the shield weight is of major significance in the choice of materials, its volume should not be neglected. The weight penalty associated with a large shield envelope is not well defined. However, some of the effects can be expressed in terms of increased reactor vessel size and weight, increased chemical-to-nuclear rocket interstage length and weight, and decrease in mass of chemical or nuclear stage propellant caused by decreased maximum height.

The present study included allowance for only the increase in reactor vessel weight with increased height, 24 lb/in., and 1/2.7 times the interstage sleeve weight increment of 37.6 lb/in. The fraction, 1/2.7, for the effective interstage weights, determined by NASA's MSFC for a lunar mission, is applied because this mass is carried only for the boost stage of the mission. Hence, the equations for the penalized weight in terms of a reference shield

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thickness are given as

$$W_p = \pi R^2 (t_1 + t_2) + (W_p \left(\frac{H}{SF} - H_o \right))$$

where

W_p = penalized weight, grams

R = shield radius, cm

t_1, t_2 = mass thickness of materials 1 and 2, g/cm²

WP = weight penalty per cm (or inch) for increased height above reference H_o , g/cm

H_o = reference shield thickness (9 inches or 22.86 cm)

H = linear thickness of the shield $\frac{t_1}{\rho_1} + \frac{t_2}{\rho_2}$, cm

SF = solid fraction of material (1 - coolant volume fraction)

ρ_1, ρ_2 = are material densities for materials 1 and 2, g/cm³

Adequate consideration of the effect of decrease in maximum available propellant height is beyond the scope of this study. Note that 1 foot of propellant in the nuclear stage contains about 3,700 pounds of propellant, which could change the weight transferred from earth orbit to a lunar trajectory by several thousand pounds. Other assumptions and details pertinent to the analytical model are described in Appendix A.

B. PHYSICAL AND CHEMICAL CONSIDERATIONS

The chemical properties of a shield material, related to the environmental fluids, reactor materials, and temperatures which they reach, determine whether or not a given material is chemically compatible. A chemically incompatible material must be sealed in a leak-tight capsule. Chemical decomposition may also set temperature limits for the material. In

this study, it was required that shield materials be used in such a way that no gross chemical decomposition would occur and that there be no exposure of incompatible materials.

Physical phase changes such as melting or boiling also exercise an important limiting influence on shield materials. The maximum temperature rise of a shield element is determined by the element size, the volumetric heat generation rate due to absorption of radiation, and the thermal conductivity of the material. Shield elements made of continuous, solid materials are subject to thermal stress created by the above internal temperature difference interacting with mechanical and thermal properties such as modulus of elasticity, thermal expansion coefficient, and Poisson's ratio. In this study, it was required that shield elements be sized so as to prevent melting, chemical reactions, sublimation, or thermal stress exceeding the modulus of rupture.

All of these chemical and physical effects are temperature dependent. In each case the shield element must be sized so that its internal temperature difference, when combined with environmental temperature and convection film temperature difference, will result in a maximum temperature that is below the temperature limit set by chemical or physical changes. In addition, the element must be sized to maintain the thermal stress below the limiting stress set by the material.

To screen the materials under consideration as to their relative desirability from the standpoint of their physical and chemical characteristics, it was necessary to set up several relatively simple thermal models. In all of these models, a cylindrical shield element was assumed as a representative shape. The maximum radius the cylindrical shield element can have and still satisfy the thermal criterion for each of three operating conditions was calculated. These three criteria are:

- 1) Maximum allowable center temperature of the shield material must not be exceeded under steady-state, full-power conditions;
- 2) Maximum allowable center temperature of the shield material must not be exceeded under the condition of cooldown following reactor shutdown (ambient gas cooling with hydrogen and nitrogen was used to evaluate this condition since it represents the condition that is most extreme for the shield during cooldown);

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- 3) Limiting thermal stress must not be exceeded in any solid, self-supporting, shield element under condition of steady-state full power operation.

There is a possibility that during the latter stages of cooldown, after the cooling fluid has been shut off, that the ambient temperatures in the vicinity of the shield may rise to somewhere between 250 and 350 degrees F. It is necessary, of course, for the shield material to have limiting temperatures in excess of these values.

Clearly, a material that requires that the shield be made of a large number of very small elements is not desirable since it will cause the shield to be more costly and probably less structurally sound.

A digital computer code, LIMRAD, was written and used to evaluate the maximum shield element radius which meets all chemical and physical limitations.

While the models used in setting over-temperature are reasonably accurate, the thermal stress model is quite conservative. It is assumed that the materials have a linear stress-strain relation up to the point of rupture. Actual materials have a non-linear characteristic which results in values for the strain at failure that are usually several times the failure strain predicted by the linear model. In view of this, the oversimplified thermal stress model should not be taken too seriously. In most cases the limiting radius calculated by this model can be exceeded by a reasonable amount and still maintain adequate margin below rupture.

The shield element must be designed to withstand the vibratory and mechanical shock loads as well as the steady loads imposed by the environment. Structural integrity of shield elements can be obtained in either of two ways. First, if the mechanical properties of the shield material are adequate, it can be used as a self-supporting shield element. Second, if the mechanical properties of the shield element are not suitable, structural integrity can be imparted by encapsulating the material. In this case, the strength and rigidity of the capsule substitutes for mechanical properties of the shield material itself. The loads imposed by vibration, mechanical shock, and pressure differences are low and should be no great problem for materials with even modest strength and ductility.

Mechanical properties of all candidate shield materials were inspected to determine whether or not they were suitable for use as a reliable self-supporting element.

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An unfortunate characteristic of most of the otherwise acceptable solid shield element materials is that they have characteristically low ductility. This is true not only for room temperature, but in addition, most materials have rather sharply reduced ductility at cryogenic temperatures. The actual ductility or percent of elongation at cryogenic temperatures is known only for beryllium, aluminum, and steel out of all the shield materials considered. Room temperature ductilities are known for several other materials, but the reduction in ductility from room temperature to the cryogenic operating temperature is not known for these materials. One tenth percent elongation at failure is taken as the minimum desirable for the operating condition. Considering the known variation of elongation in Be, it appears that, conservatively, minimum elongation of about 1.5 percent at room temperature is desired.

The ability of ambient materials to withstand the environmental radiation without serious degradation of their physical and chemical characteristics is an important factor in the selection of materials. The expected radiation environment is as follows:

- 1) Neutrons - total integrated flux after 1 hour of full-power operation is estimated to be 10^{19} nvt. Approximately 80 percent of these neutrons will be epithermal or higher energy.
- 2) Gamma Dosage - the total environmental exposure will be the equivalent of 10^{10} rads-carbon.

The radiation damage resistance of materials has been evaluated against these criteria.

C. ECONOMIC CONSIDERATIONS

For evaluation, the total cost of flight shields for NERVA reactors can be thought of in two categories, the cost of development of the first one or two shields, and the hardware cost of the shields.

1. Development Cost

Detailed knowledge of material characteristics in relation to the reactor environment is lacking in varying degrees on all materials not already used in reactor tests. Radiation damage resistance has been experimentally determined for only a few of the materials. The thermal conductivities of most of the materials are not known under cryogenic conditions, particularly

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as influenced by radiation damage. In some cases, thermal conductivities of unirradiated materials at room temperature are lacking. The specific heats of some of the materials have not been determined at low temperatures. Mechanical properties for low temperatures are in some cases unavailable.

The more unknown factors there are regarding the characteristics and properties of a material in the service environment, the more materials research, tests, and experiments must be made to evaluate these properties correctly. This means greater development cost, longer elapsed time until evaluation is complete, and greater risk that the material will not, in the end, measure up satisfactorily to the environmental requirements.

The greater the complexity of the shield design, the more effort will be needed in engineering analysis and design for complete evaluation. It is also likely that a more complex design will require more manufacturing development effort. As an example of this, encapsulated lithium hydride required a considerable amount of manufacturing development and engineering analysis to arrive at what was considered to be a satisfactory result for the KIWI-B-5 reactor shield.

2. Hardware Cost

One of the major determinants of the cost of a finished shield is the cost of raw materials. Some of the materials under consideration are quite expensive, for example, beryllium base materials and metal hydrides in massive forms. The cost of pieces of these materials ready for use in a shield may run from \$35 to \$70 a pound. The cost of B_4C - Al or TiH_2 - B_4C -Al inserts are estimated to be about \$12 per pound.

The cost of fabricating raw materials into a shield is heavily dependent upon the size of the shield elements and characteristics of the materials and methods used. If large elements can be used, fewer elements are required to make a complete shield. The maximum permissible shield element size is determined mainly by the combination of thermal characteristics with the physical and chemical temperature limitations of the materials discussed in Section B above. The fabrication cost is also greatly dependent on the method of construction of the element. For example, materials that can be used as self-supporting shield elements and can be fabricated by extrusion, would be much less expensive to fabricate than materials that must rely on a

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structural enclosure for their physical strength and protection from environmental fluids. Materials that must be contained in leak tight capsules because of chemical incompatibility are even more expensive.

Unusual manufacturing requirements can greatly increase the fabrication cost for a material. An example of this is the requirement for dry box assembly of hygroscopic shield elements. Similarly, unusual toxicity or stringent leak tightness requirements would greatly increase the fabrication cost of the shield. Adaptability to established quality manufacturing methods such as forging, extrusion, machining, etc, is a quality to be desired in materials for a flight shield.

The total cost of the flight shield to the program is dependent on the total number of shields used. It is possible to incur heavy development costs, and by virtue of a low fabrication cost per shield multiplied by substantial number of shields used, secure a significant overall cost saving for the program.

3. Overall Cost Evaluation

Accurate evaluation of developmental hardware costs for flight shields is difficult to obtain at this stage of the project. Establishing even the relative order of desirability of the various materials is difficult except in a rather crude fashion. With this in mind candidate materials were grouped into categories of high, medium, or low cost for development and for hardware cost.

A shield requiring a large amount of developmental effort could cost \$500,000 in developmental effort other than hardware cost. At the other end of the scale, the shield made from the most familiar materials already tested in the reactor could possibly be developed for a small fraction of that amount. The expected range of hardware cost for a single flight shield for the NR-1 reactor is between an upper limit of \$750,000 and a possible lower limit of about \$150,000.

Eight of the more promising combinations of materials are subjected to a more detailed examination of hardware cost. Rough estimates of the cost of shields made of each of these materials are given in Chapter II.

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II. RESULTS OF EVALUATION

A. EVALUATION OF SHIELD WEIGHTS

The results of the nuclear evaluation of over 25 neutron moderators, with aluminum or steel structure, are presented in tables 1 through 4, and figures 1 through 23. Water was included in the study as a familiar reference for comparison.

The hydride materials show superior qualities because of their ability to moderate neutrons to thermal energy within a small displacement distance from the fast neutron source group. This rapid slowing down, or low "age, " causes the secondary gamma rays to be produced near the core side of the shield and reduces the intensity of neutrons escaping the shield. While the non-hydrides may have more efficient gamma ray mass attenuation coefficients, they tend to leak more secondary gamma rays and epithermal neutrons.

The effects of the boron secondary gamma ray suppression and the shield gamma ray contribution to the total gamma level can be obtained from a typical case presented in table 5 and in figure 4. If the boron were not included, the secondary source would be about three times greater and displaced further into the shield. The total gamma radiation exiting the shield would be increased by more than one third. This effect is very important for large attenuations in non-hydrogeneous shields.

The materials generally have different relative positions, when ordered by increasing total weight, for different neutron-to-gamma energy ratios. The order of the materials of interest for possible use in NR-1 was essentially unchanged for different attenuations (table 5 and figure 24) over the range of 0.25 to 0.04 studied here. The materials selected for the more detailed investigation were chosen after consideration of their physical, chemical, and radiation properties (see Chapter II F). These materials are:

- 1) Lithium hydride
- 2) Boron carbide
- 3) Hydrogen
- 4) Titanium hydride

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TABLE 1

OPTIMUM MIXTURES OF ALUMINUM AND NEUTRON MODERATOR
FOR A SHIELD ATTENUATION FACTOR OF 0.25

Material	Moderator (gm/cm ²)	Al (gm/cm ²)	Total Penalized Thickness (gm/cm ²)	Volume Fraction of Moderator
LiNH ₂	22.82	0	23.14	1.0
Nylon 66	22.98	0	23.55	1.0
Li(BH) ₄	19.99	0	25.57	1.0
BeH ₂	20.94	0	23.81	1.0
H ₂ O	22.66	0	23.97	1.0
LiH	21.71	0	24.55	1.0
LiOH	25.56	0	25.44	1.0
TiH ₂	28.48	0	25.54	1.0
Be(OH) ₂	26.83	0	25.57	1.0
MgH ₂	25.66	0	25.67	1.0
TaB ₂	30.43	0	27.10	1.0
Ti-Nb-H ₂	30.75	0	27.44	1.0
Al(OH) ₃	29.29	0	27.45	1.0
CaH ₂	28.56	0	28.12	1.0
Fe(OH) ₃	31.56	0	28.88	1.0
YH ₂	32.68	0	29.70	1.0
BeO	32.27	0	30.0	1.0
Boron	30.45	0	30.02	1.0
AlB ₂	32.52	0	30.11	1.0
Ca(OH) ₂	29.88	1.814	30.25	0.95
ZrH ₂	35.8	0	32.26	1.0
WB ₂	32.12	8.089	36.39	0.45
H ₂	1.27	32.49	37.28	0.6
Be	39.81	0	40.81	1.0
PbO	47.49	0	43.81	1.0
Pb	64.11	3.814	64.58	0.8

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TABLE 2

OPTIMUM MIXTURES OF ALUMINUM AND NEUTRON MODERATOR
FOR A SHIELD ATTENUATION FACTOR OF 0.2

Material	Moderator (gm/cm ²)	Aluminum (gm/cm ²)	Total Penalized Thickness (gm/cm ²)	Volume Fraction of Moderator
LiHN ₂	26.99	0	28.35	1.0
Nylon 66	26.89	0	28.47	1.0
Li(BH) ₄	23.26	0	28.90	1.0
BeH ₂	24.66	0	29.00	1.0
H ₂ O	26.57	0	29.03	1.0
LiH	22.59	4.12	30.35	0.95
Be(OH) ₂	31.14	0	30.56	1.0
MgH ₂	29.76	0	30.64	1.0
TiH ₂	33.23	0	30.70	1.0
LiOH	30.06	0	30.88	1.0
Al(OH) ₃	33.84	0	32.56	1.0
Ti-Nb-H ₂	35.61	0	32.64	1.0
CaH ₂	33.13	0	33.49	1.0
Fe(OH) ₃	36.48	0	34.24	1.0
YH ₂	37.94	0	35.36	1.0
Ca(OH) ₂	36.24	0	35.41	1.0
B ₄ C	35.42	0	35.81	1.0
AlB ₂	37.79	0	35.88	1.0
BeO	37.37	0	36.36	1.0
ZrH ₂	41.70	0	38.48	1.0
	32.8	13.05	42.63	0.35
LH ₂	1.467	37.40	43.74	0.6
WB ₂	34.24	13.1	44.13	0.35
Be	46.33	0	48.38	1.0
PbO	57.85	0	54.56	1.0
Pb	70.69	11.21	79.56	0.6

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TABLE 3

**OPTIMUM MIXTURES OF STAINLESS STEEL AND NEUTRON MODERATOR
FOR A SHIELD ATTENUATION FACTOR OF 0.2**

Material	Moderator (gm/cm ²)	Stainless Steel (gm/cm ²)	Total Penalized Thickness (gm/cm ²)	Volume Fraction of Moderator
LiNH ₂	26.99	0	28.35	1.0
Nylon 66	26.89	0	28.47	1.0
Li(BH) ₄	23.67	0	28.90	1.0
BeH ₂	24.66	0	29.00	1.0
H ₂ O	26.57	0	29.03	1.0
LiH	18.17	9.81	29.84	0.95
Be(OH) ₂	31.14	0	30.56	1.0
MgH ₂	29.76	0	30.64	1.0
LiOH	30.06	0	30.88	1.0
Al(OH) ₂	33.84	0	32.56	1.0
Ti-Nb-H ₂	35.61	0	32.64	1.0
TiH ₂	35.34	0	32.69	1.0
CaH ₂	33.13	0	33.49	1.0
Fe(OH) ₃	36.48	0	34.24	1.0
YH ₂	37.94	0	35.36	1.0
Ca(OH) ₂	36.24	0	35.41	1.0
Boron	35.21	0	35.56	1.0
AlB ₂	37.79	0	35.88	1.0
TaB ₂	37.87	0	35.97	1.0
H ₂	1.26	35.75	35.22	0.8
ZrH ₂	41.7	0	38.38	1.0
BeO	42.79	0	41.55	1.0
Be	41.41	0	42.67	1.0
WB ₂	34.24	13.1	44.13	0.35
PBO	57.85	0	54.56	1.0
PB	70.69	11.21	79.56	0.6

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TABLE 4
SEVERAL ATTENUATIONS THROUGH OPTIMUM MIXTURES OF SELECTED MATERIALS

Material	0.04				0.1				0.2				0.25			
	Moderator (gm/cm ²)	Aluminum (g/cm ²)	Total (g/cm ²)	Volume Fraction (Moderator)	Moderator (gm/cm ²)	Aluminum (gm/cm ²)	Total (gm/cm ²)	Volume Fraction (Moderator)	Moderator (gm/cm ²)	Aluminum (gm/cm ²)	Total (gm/cm ²)	Volume Fraction (Moderator)	Moderator (gm/cm ²)	Aluminum (gm/cm ²)	Total (gm/cm ²)	Volume Fraction (Moderator)
Be	100.37	0	111.16	1.0	67.76	0	73.2	1.0	46.33	0	48.38	1.0	39.81	0	40.81	1.0
BeO	74.15	0	75.99	1.0	53.19	0	52.97	1.0	37.37	0	36.36	1.0	32.27	0	30.00	1.0
B ₄ C	71.57	0	77.89	1.0	50.96	0	53.89	1.0	35.42	0	35.81	1.0	30.45	0	30.02	1.0
H ₂	2.85	72.83	90.31	0.6	2.06	52.65	63.79	0.6	1.46	37.40	43.74	0.6	1.27	32.49	37.28	0.6
H ₂ O	55.69	0	66.82	1.0	38.94	0	35.09	1.0	26.57	0	29.03	1.0	22.66	0	23.97	1.0
LiH	25.62	38.02	72.17	0.7	26.11	15.95	48.34	0.85	22.59	4.12	30.35	0.95	21.71	0	24.55	1.0
Nylon 66	56.29	0	65.66	1.0	39.34	0	44.16	1.0	26.89	0	28.47	1.0	22.98	0	23.55	1.0
TiH ₂ -B ₄ C-Al	67.52	0	68.01	1.0	47.98	0	46.75	1.0	33.23	0	30.70	1.0	28.48	0	25.54	1.0
		Steel				Steel				Steel				Steel		
Be	40.60	44.14	87.51	0.8	29.24	31.78	61.50	0.8	41.41	0	42.67	1.0	35.49	0	35.80	1.0
BeO	57.70	26.88	85.81	0.85	60.81	0	61.35	1.0	42.79	0	41.55	1.0	37.14	0	35.35	1.0
Baron	71.14	0	77.40	1.0	50.65	0	53.54	1.0	35.21	0	35.56	1.0	30.25	0	29.78	1.0
H ₂	33.63	67.26	81.86	0.86	2.405	48.10	57.00	0.85	1.26	35.75	38.22	0.8	1.088	30.82	32.20	1.0
H ₂ O	55.69	0	66.82	1.0	39.94	0	45.09	1.0	26.57	0	29.03	1.0	22.66	0	23.97	1.0
LiH	22.98	41.60	69.46	1.0	20.41	23.26	46.89	0.9	18.17	9.81	29.84	0.95	15.36	8.291	24.38	0.95
Nylon 66	56.29	0	65.55	1.0	39.34	0	44.16	1.0	26.89	0	28.47	1.0	22.98	0	23.55	1.0
TiH ₂	70.50	0	70.61	1.0	50.42	0	48.95	1.0	35.34	0	32.69	1.0	30.50	0	27.46	1.0

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Figure 1. Shield Weights for ZrH₂, Ca(OH)₂, Be(OH)₂, BeH₂-Steel

Figure 2. Shield Weights for Fe(OH)₃, CaH₂, LiOH, Li(BH)₄-Steel

Figure 3. Shield Weights for YH₂, Al(OH)₃, MgH₂, LiNH₂-Steel

Figure 4. Shield Weights for PbO, WB₂, TaB₂, and Pb

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Figure 5. Gamma Ray Flux Above Reactor Unshielded and with 100 Percent BeO Shield

Figure 6. Required Shield Thickness for Mixture of Aluminum and Water

Figure 7. Required Shield Thickness for Mixture of Aluminum and Lithium Hydride

Figure 8. Required Shield Thickness for Mixture of Aluminum and Nylon 66

Figure 9. Required Shield Thickness for Mixture of Aluminum and Titanium Hydride

Figure 10. Required Shield Thickness for Mixture of Aluminum and Titanium Hydride-Boron Carbide

Figure 11. Required Shield Thickness for Mixture of Aluminum and Liquid Hydrogen

Figure 12. Required Shield Thickness for Mixture of Aluminum and Boron Carbide

Figure 13. Required Shield Thickness for Mixture of Aluminum and Beryllium Oxide

Figure 14. Required Shield Thickness for Mixture of Aluminum and Beryllium

Figure 15. Required Shield Thickness for Mixture of Steel and Water

Figure 16. Required Shield Thickness for Mixture of Steel and Lithium Hydride

Figure 17. Required Shield Thickness for Mixture of Steel and Nylon-66

Figure 18. Required Shield Thickness for Mixture of Steel and Titanium Hydride

Figure 19. Required Shield Thickness for Mixture of Steel and Liquid Hydrogen

Figure 20. Required Shield Thickness for Mixture of Steel and Boron Carbide

Figure 21. Required Shield Thickness for Mixture of Steel and Beryllium Oxide

Figure 22. Required Shield Thickness for Mixture of Steel and Beryllium

Figure 23. Required Shield Thickness for Mixture of Steel or Aluminum and Beryllium Oxide with no Hydrogen Coolant

Figure 24. Attenuation through Optimum Weight Shields

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TABLE 5
SHIELD REGION MATERIAL
(n, γ) CONTRIBUTIONS*

100 Percent BeO Shield with Steel Hardware			
Material	Average Mev per Absorption	Region Absorptions/ Neutron - cm ²	Total Gamma Energy Release (Mev/cm ² - Neutron)
Cr	9.21	1.028 (- 4)	9.47 (- 4)
Ni	8.32	6.906 (- 5)	5.75 (- 4)
Fe	7.79	3.613 (- 4)	28.14 (- 4)
Be	6.82	8.847 (- 5)	6.03 (- 4)
B	0.45	5.979 (- 3)	26.90 (- 4)
H	2.23	1.215 (- 4)	2.71 (- 4)
Total		6.722 (- 3)	78.99 (- 4)
$\frac{\text{Boron Absorptions}}{\text{Total Absorptions}} = 0.89$			
$\frac{\text{Boron } \gamma \text{ Energy Release}}{\text{Total Energy Release}} = 0.34$			
<p>Fractional total energy release due to boron capture gammas ≈ 0.075</p>			
<p>* From transport calculations for one source neutron</p>			

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- 5) Beryllium oxide
- 6) Beryllium
- 7) Water

While water is not considered practical for NR-1 application, it is included for comparison.

Since the majority of the nuclear evaluation of the several materials was performed with a recently developed and unproven code, it was desirable to evaluate the accuracy of this code with more accurate and detailed methods. The significant factors to be confirmed concerning the shield materials comparisons code were:

- 1) Optimum mixture of materials
- 2) Order by weight of materials for a given attenuation
- 3) Correct overall attenuation
- 4) The relative weight difference between materials

To determine whether the shield materials optimization code predicted the correct optimum mixture of shield materials, the TAPAT one-dimensional transport code system (Section A-1.2) was used to determine the radiation transmitted through three mixtures of beryllium oxide-aluminum and three mixtures of lithium hydride-steel. These two materials were selected to represent the hydride and non-hydride materials. The transport calculations confirmed that the Shield Optimization Code had correctly predicted the optimum mixtures of materials.

To determine whether the shielding materials comparisons code predicted the correct order of materials, seven materials were run using the TAPAT Transport Code System. Multigroup neutron flux distributions, gamma sources, and resulting primary and secondary gamma ray fluxes were determined. The results of the transport calculations showed the materials to be in the same relative order as predicted by the shield materials comparison code. The mass thickness for a given attenuation as predicted by the transport calculations were found to be up to 15 percent lower than those predicted by the material comparison code. This difference, however, increased for the non-hydride materials as the volume fraction of the neutron moderator was decreased. The difference is believed to be due to the simple treatment

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of the hydrogen coolant and lack of epithermal captures in the materials comparison model and the single, high-energy, neutron group used in the transport calculation. The transport model contained multigroup neutron and gamma treatment of the hydrogen coolant and plena and took into account boron absorptions above thermal energies. The relative difference in mass thicknesses calculated by the transport code and materials code for a given material was small for the hydride materials as shown in figures 6 and 16.

The shield thicknesses calculated by the materials comparison code diverged and became much larger than the transport code values for the non-hydride materials and for low volume fractions of neutron materials (see figure 13). (This difference is again probably due to the simple treatment of the hydrogen coolant.)

A more complete description of the methods is presented in the Appendix. The shield mass thicknesses presented in figures 1 through 23 show the total shield weight per unit area versus neutron moderator volume fraction. The total weights are penalized for increase in linear thickness above the reference case (9 inches) to account for the change in reactor vessel and chemical-to-nuclear stage connection. The 9-inch dimension was chosen from the conceptual design study as reported in WANL-TME-1300. Each shield includes 20 percent volume fraction of coolant at an average density of 0.00405 g/cm^3 . Several of the graphs have corresponding thicknesses calculated by the TAPAT Transport code (Chapter 1 B of the Appendix) for a design attenuation factor of 0.1 as indicated by "x's". Table 6 presents the attenuation through several shield configurations as calculated by the transport code.

The attenuations calculated here were for infinite plane slab geometry, whereas, the values of attenuation in earlier tank heating studies and the conceptual shield design were the effective attenuation obtained by integrating over the solid angle subtended by the propellant tank. The shield weights calculated here are approximately 10 percent less than those in reference 1.

Most of the optimum mixtures resulted in shields of 100 percent neutron material, which generally are not mechanically and structurally sound. Weights were therefore compiled for off-optimum or practical compositions and are presented in table 7 and figure 25.

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TABLE 6

ATTENUATION THROUGH SEVERAL SHIELD ARRANGEMENTS
FROM S_n TRANSPORT CALCULATIONS

Shield	Shield Thickness (gm/cm ²)		Gamma Ratio	x 0.726	Fast Neutron Ratio	x 0.182	Slow Neutron Ratio	x 0.0909	Total Exit Radiation	Mass Thickness for 0.1 Attenuation
	Penalized	Actual								
No Shield	0.0		1.0	0.726	1.0	0.182	1.0	0.0909	1.0	----
0.304 BeO,* 0.496 Al	54.2	54.2	0.111	0.08072	0.089	0.01618	0.147	0.01336	0.1103	56.6
0.496 BeO,* 0.304 Al	54.2	54.2	0.113	0.0823	0.058	0.01054	0.109	0.00990	0.103	54.7
0.8 BeO	54.2	54.2	0.115	0.0836	0.028	0.00509	0.068	0.00618	0.0949	52.9
0.8 BeO**	54.2	54.2	0.093	0.0676	0.027	0.0049	0.038	0.00345	0.0759	48.3
0.56 LiH, 0.24 Steel	54.2	54.2	0.064	0.0464	0.060	0.0109	0.019	0.00172	0.0591	44.2
0.33 LiH, 0.47 Steel	51.9	54.2	0.075	0.0545	0.192	0.0349	0.060	0.00545	0.0949	50.6
0.8 LiH	49.9	48.8	0.111	0.0807	0.0013	0.000236	0.0052	0.000472	0.0814	46.4
0.72 TiH ₂ , 0.08 Al	54.2	54.7	0.067	0.0486	0.045	0.00818	0.0059	0.000536	0.0573	43.6
0.296 TiH ₂ , 0.504 Al	54.2		0.082	0.0595	0.085	0.01545	0.02	0.00181	0.0768	48.5
0.08 TiH ₂ , 0.72 Al	54.2	54.2	0.103	0.075	0.149	0.0271	0.108	0.00981	0.1119	56.0
0.801 H ₂ , 0.199 Al	57.7	45.2	0.096	0.0709	0.0091	0.0016	0.0016	0.000145	0.0727	50.5
0.8 Be	54.2	52.4	0.134	0.0973	0.0642	0.0127	0.01	0.00172	0.1103	54.2
0.8 Al	54.2	54.2	0.081	0.059	0.207	0.0376	0.27	0.0245	0.121	59.1
0.8 H ₂ O	51.0	49.0	0.082	0.0596	0.0014	0.00025	0.0026	0.000236	0.0601	41.8

* Material fractions are by volume, additional 0.014 for structure and 0.186 for coolant.
 ** Contains 20 times as much boron as previous case for BeO.

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TABLE 7
OPTIMUM VERSUS PRACTICAL SHIELD WEIGHTS

Alloy	Optimum		Practical		Percent Increase Versus Practical LiH
	Moderator Volume Fraction	Mass Thickness	Moderator Volume Fraction	Mass Thickness	
LiH - Steel	0.90	47.0	0.70	50.0	0
TiH ₂ - Steel	1.0	49.3	0.70	53.7	7
TiH ₂ - B ₄ C - Al	1.0	47.0	0.30	53.9	8
BeO - Al	1.0	53.5	0.85	56.0	12
H ₂ - Steel	0.85	57.5	0.85	57.5	15
B ₄ C - Al	1.0	54.0	0.30	61.1	22
B ₄ C - Steel	1.0	54.0	0.65	58.4	17
Be - Steel	0.80	62.0	0.80	62.0	24

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Table 7 also shows the percent weight increase when compared to the lowest weight practical shield, LiH-steel.

Of the materials considered, $\text{TiH}_2\text{-B}_4\text{C-Al}$ appears to be the most promising. While it is shown to be 8 percent heavier than a practical LiH-Steel shield, as a result of the material comparison code calculation, it must be recognized that the chosen upper limit for the volume fraction of neutron shield material, 30 percent, may be conservative. Development of the $\text{TiH-B}_4\text{C-Al}$ may result in a significant increase in allowable volume fraction which would be closer to the optimum. Also note that the attenuation through the $\text{TiH}_2\text{-B}_4\text{C-Al}$ should be greater than that presented in figure 11 and table 7 since no credit is taken for neutron capture in boron above the 0.025 ev thermal energy. The estimated cost difference between the $\text{TiH}_2\text{-B}_4\text{C-Al}$, TiH_2 -steel, or LiH shield, \$168,000 versus \$440,000 or \$660,000, seems to make undertaking a $\text{TiH}_2\text{-B}_4\text{C-Al}$ development program desirable.

B. EVALUATION OF THERMAL PROPERTIES

The effect of physical-chemical characteristics is manifested strongly in shield element size. As set forth in Chapter 1 B there is a maximum size for shield elements of each material, which will conform to the thermal stress limits (if applicable) and temperature limitations of the material. These maximum sizes are set forth in table 8 as cylindrical shield element, limiting diameters. These are not necessarily the diameters that would actually be used. They simply indicate whether there is a thermal limitation that would restrict the shield elements to undesirably small sizes.

Of the materials considered only Be, $\text{TiH}_2\text{-B}_4\text{C-Al}$, BeO, B_4C , YH_x , LiH, AlB_2 , ZrH_x , CaH_2 , and TiH_x appear to be in the practical range, assumed to be approximately 0.25 inch in diameter. (See Chapter 2 D.) The others must be used in such small sizes that an inordinate number of elements would be required for a shield.

There is a large group of materials that are in the form of powders or weak brittle crystals. Unless it is known that these materials can be pressed into dense, pellets, it is assumed they will be packed into tubular capsules and densified to 80 percent of theoretical density by some

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TABLE 8

LIMITING SHIELD ELEMENT DIAMETER FOR VARIOUS MATERIALS

Material	Diameter (Inches)	Material	Diameter (Inches)
Be	2.1	CaOH	0.19
TiH ₂ -Al	1.8	MgH	0.19
BeO	0.97	WB ₂	0.19
ZrH _x	0.67	LiOH	0.15
TiH _x	0.60	Al(OH) ₃	0.15
YH	0.55	Be(OH) ₂	0.13
LiH	0.53	Nylon	0.13
B ₄ C	0.42	Fe(OH) ₃	0.13
AlB ₂	0.41	LiNH ₂	0.12
CaH ₂	0.35	PbO	0.08
Li(BH) ₄	0.23	BeH	*
TaB ₂	0.22		

method such as swaging. These materials are limited in size mostly by maximum center temperature at full power. The thermal stress criterion is not applicable; and only one, beryllium hydride, was limited by the cooldown condition.

Where it is known that powder or crystalline materials can be pressed into dense pellets, the thermal stress limitation is ignored since the pellets can be allowed to crack inside their capsule. However, the thermal conductivity of the cracked material was reduced to one-fourth of the value for the solid material.

The thermal stress limit is valid only for those materials that are to be used as self-supporting structural elements, and it is considered to be a very conservative limit because of

* This material could not meet the ambient gas cooldown requirement regardless of size.

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the simple model used (Chapter 1 B).

C. EVALUATION OF STRUCTURAL PROPERTIES

Of all the materials considered, only beryllium, beryllium oxide, $\text{TiH-B}_4\text{C-Al}$, aluminum, and the steel used in hydrogen-steel shields can be considered to be reliable, self-supporting shield elements. Some pertinent properties of these materials are indicated in table 9. All of the other materials must be enclosed in a structural support or encapsulated.

TABLE 9

STRUCTURAL PROPERTIES OF SELECTED MATERIALS AT ROOM TEMPERATURE

Material	Modulus of Rupture (psi)	Modulus of Elasticity (psi)	Elongation (%)
Be*	40,000	40×10^6	1.5
Be	35,000	45×10^6	0.2
BeO	16,000	50×10^6	0.06
$\text{TiH}_2\text{-B}_4\text{C-Al}$	30,000	10×10^6	2.9
Borated Steel**	100,000	30×10^6	18
Aluminum 1100-HO	$13,000^+$	10×10^6	38

* Unborated
** 1 percent boron, by weight, in austenitic stainless steel
+ Ultimate tensile strength

D. EVALUATION OF COSTS

A rough evaluation of the development cost and the hardware cost per shield has been made based on the factors outlined in Chapter I C. The results of this evaluation appear in table 10. The relative ranking of each of the cost factors is set forth along with an overall ranking of the development and hardware costs. From this table, the materials that are desirable for other reasons have been singled out for more detailed treatment. Estimates of the cost of a shield made of each of these materials are given in table 11.

The raw materials cost in the shield was calculated. The element fabrication cost was computed to be the estimated fabrication cost per element times the number of elements used, considering the limiting size from thermal and chemical factors given in table 8 (Note: Beryllium, BeO, and the Aluminum based composite materials were arbitrarily limited to 0.75 inch diameter. Use of the limiting element size would result in lower costs for both of these classes of materials.)

Contingency factors were allowed consistent with the degree of uncertainty associated with each shielding material. The overall cost shown in the right-hand column of table 11 is thought to be a fair ranking of the various materials and adequate to cover the actual costs.

The table shows that there is a very considerable spread in cost among the top candidate materials. The difference between the least expensive and the most expensive is \$538,000 per shield and \$2,700,000 for five shields used over the life of the program. This sum is a large multiple of the cost of developing low-cost shielding materials.

E. EVALUATION OF RADIATION DAMAGE RESISTANCE

The shield materials considered have been divided into three categories as listed in table 12. Materials whose radiation damage resistance is not known to be satisfactory are considered in general to be poor choices for shielding materials. This view is based on the uncertainty of success, development expense, and elapsed time required to evaluate radiation damage resistance and develop a satisfactory material.

TABLE 10

RELATIVE COST RANKING OF SHIELD MATERIALS

Material	Development Cost					Hardware Cost			
	Materials Research and Evaluation		Design and Manufacturing Development		Raw Materials Cost (\$) Low < 10/lb High ≥ 40/lb	Construction	Fabrication		Overall Cost Dev/Hardware
	Physical Properties	Radiation Damage	Design Complications	Manufacturing Development			No. of Elements	Unusual Mfg Requirements	
Al B ₂	High	High	High	High	Low	High	Medium	Low	High/Medium
Al OH ₃	High	High	High	High	Low	High	High	Low	High/Medium
B ₄ C	Medium	Medium	Low	Medium	Low	Low	Low	Low	Medium/Low
Be	Low	Low	Low	Low	High	Low	Low	Low	Low/Medium
Be H	High	High	High	High	High	High	High	High	High/High
Be O	Low	Low	Low	Low	High	Low	Low	Low	Low/Medium
Be OH	High	High	High	High	High	High	High	High	High/High
Ca H	Medium	Medium	High	High	Low?	High	Medium	Medium	Med/High
Ca OH	High	High	High	High	Low	High	High	Low	High/High
Fe OH ₃	High	High	High	High	Low	High	High	Low	High/High
Hydrogen-Steel	Low	Low	High	High	Low	Low	Low	Low	Low/Low
Li BH ₄	High	High	High	High	?	High	High	High	High/High
Li H	Medium	High	High	Medium	Low	High	Medium	High	High/High
Li NH ₂	High	High	High	High	?	High	High	High	High/High
Li OH	High	High	High	High	Low	High	High	Low	High/High
Mg H	High	High	High	High	Low	High	High	Medium	High/High
Nylon	Medium	High	High	High	Low	High	High	Low	High/High
Pb O	High	High	High	High	Low	High	High	Low	High/High
Ta B ₂	High	High	High	Medium	High	Medium	High	Low	High/High
Ti H	Low	Low	Medium	Medium	High	Medium	Medium	Low	Medium/Medium
TiH-B ₄ C-Al	High	Medium	Low	Medium	Low	Low	Low	Low	Medium/Low
W B ₂	High	High	High?	Medium	Low	Medium	Medium	Low	High/Medium
Y H	Medium	Low	Medium	Medium	High	Medium	Low	Low	Medium/Medium
Zr H ₂	Low	Low	Medium	Medium	High	Medium	Medium	Low	Medium/Medium

TABLE 11
ESTIMATED COST OF SELECTED SHIELDS

Materials	Raw Materials (\$1000)	Element Fabrication Cost (\$1000)	Structure & Assembly (\$1000)	Total (\$1000)	Contingencies Percent of Total	Grand Total (\$1000)
Hydrogen-Steel*	9	--	66	75	100	150
TiH ₂ -B ₄ C-Al	15	35	40	90	100	180
B ₄ C-Al	17	38	40	95	100	190
Be-Steel	94	16	40	150	50	225
BeO-Al	102	16	40	158	50	237
TiH ₂ -Steel	169	115	40	324	35	437
ZrH ₂ -Steel	170	115	40	325	35	439
LiH-Steel	15	518	40	573	20	688

* Hydrogen-steel shield is not constructed of discrete elements.

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TABLE 12
RADIATION DAMAGE RESISTANCE OF SHIELD MATERIALS

Material	Satisfactory	Unknown	Unsatisfactory
Be	X		
BeO	X		
B ₄ C	X		
LiH	X		
TiH ₂	X		
TiH ₂ -B ₄ C-Al*	X		
YH ₂	X		
ZrH ₂	X		
AlB ₂		X	
BeH ₂		X	
Be(OH) ₂		X	
CaH ₂ **		X	
Ca(OH) ₂		X	
Fe(OH) ₃		X	
Li(BH) ₄		X	
LiNH ₂		X	
LiOH		X	
MgH ₂		X	
PbO		X	
TaB ₂		X	
WB ₂		X	
Al(OH) ₃			X
Nylon			X

* Radiation damage of all components tested separately, and in mechanical mixture of B₄C-Al is satisfactory. It was not tested in a mechanical mixture of all three.

** Tests up to 7×10^{17} nvt gave satisfactory results.

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F. OVERALL EVALUATION OF MATERIALS

The overall evaluation of materials was based on integrity, efficiency, design suitability, and cost.

Proven ability of a material to withstand the radiation environment without excessive damage or deterioration of properties is a very important integrity factor. Where materials are known to be unsatisfactory from this standpoint they were dropped from further consideration. Complete lack of knowledge of the radiation damage resistance of a material was considered to be a major deterrent to its use in the NERVA shield. A great deal of time and money must be spent both evaluating the characteristics of the material and, if required, finding additives or processing techniques to increase radiation damage resistance.

Structural strength and ductility are important integrity factors because they provide the design margins needed for high reliability. Low melting or decomposition temperatures are very strong negative integrity factors. Without generous temperature margins, uncertainties in properties such as thermal conductivity, specific heat, etc. an approach too close to the limiting temperature and container pressure may result.

The efficiency or attenuation-to-weight ratio for a shielding material is a most important factor in evaluation. The weight of the shield subtracts directly from the payload that can be carried by the engine. The shield weight for the materials considered is dependent on the incident and exit neutrons and gamma ray fluxes and the desired radiation attenuation ratio.

A number of small but important factors are grouped under the heading of design suitability. One of the items that comes under this heading is flexibility. Do we have the ability when using a certain material to vary the attenuation of the shield locally without changing the overall dimensions? If so, the material possesses the desirable characteristic of flexibility, and we can add attenuation at the places in the shield where it is needed most while saving weight in other places. Another item included under the heading of design suitability is the desirability of the mechanical configuration dictated by the properties of the material. Some materials must be divided into very small elements to prevent overheating. If we make the

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shield elements in the form of cylindrical rods, these rods would have entirely too high a ratio of length to diameter. They would be too flimsy for self-support and would require intermediate supports along the length of the rod. Such characteristics make poor mechanical design, are unduly complicated, and lack the high degree of reliability required.

The cost of development and hardware fabrication is a factor that must not be overlooked in the comparison of shield materials. Some materials, though otherwise desirable, would require excessively large expenditures of time and money for their development. In addition, many of these would be quite expensive to build as practical shields, which is particularly important because the hardware cost of the shield must be borne each time a shield is used. The development cost, though it may be high, is a non-recurring.

An overall evaluation of the 25 materials considered has been made on the basis of the points discussed above and is summarized in table 13. This table shows the relative rating assigned to each material for each of the factors considered. An overall rating was also assigned to each material on a number scale from 0 to 4, 4 representing the most desirable and 0 the least desirable material. This table was used for screening out relatively undesirable materials.

The top seven materials were taken from table 13 for more detailed consideration in evaluation. The characteristics of these materials are given in table 14.

Three combinations of materials shown in table 14 appear to have the most desirable characteristics for NERVA flight shields. These combinations are beryllium-steel, titanium hydride - boron carbide - aluminum composite, and lithium hydride-steel. The choice among these three combinations depends upon the relative importance of factors such as cost and weight. The beryllium steel shield is one that can be designed and built in a relatively short time and for low development cost. The risk of arriving at a shield with insufficient integrity and reliability is lower with this material than with any other. On the other hand, a top shadow shield designed for an attenuation ratio of 0.1 and made of beryllium-steel will weigh approximately 770 pounds more than one made of lithium hydride, the lightest of the three.

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TABLE 13
OVERALL EVALUATION OF MATERIALS

Material	Efficiency Shielding	Radiation Stability	Design Suitability	Cost	Overall Figure of Merit ⁺
Al B ₂	Fair	Unknown	Fair	High	1
Al (OH) ₃	Fair	Unsatisfactory	Poor	High	0
B ₄ C	Fair	Satisfactory	Good	High	3
Be	Fair	Good	Fair	Medium	3
Be H	Good	Unknown	Unsuitable	High	0
Be O	Fair	Good	Fair	Medium	3
Be(OH) ₂	Fair	Unknown	Poor	High	1
Ca H ₂	Fair	Unknown	Fair	High	1
Ca(OH) ₂	Fair	Unknown	Poor	High	1
Fe(OH) ₃	Fair	Unknown	Poor	High	1
H ₂ -Steel	Fair	Good	Fair	Low	3
Li(BH) ₄	Good	Unknown	Poor	High	1
Li H	Good	Satisfactory	Fair	High	4
Li NH ₂	Good	Unknown	Poor	High	1
Li OH	Good	Unknown	Poor	High	1
Mg H ₂	Fair	Unknown	Poor	High	1
Nylon	Good	Unsatisfactory	Poor	High	0
Pb O	Poor	Unknown	Poor	High	0
Ta B ₂	Poor	Unknown	Poor	High	0
Ti H _x	Fair	Good	Good	Medium	3
TiH ₂ -B ₄ C-Al	Fair	Good	Good	Low	4
W B ₂	Poor	Unknown	Poor	High	0
Y H _x	Fair	Good	Fair	Medium	2
Zr H _x	Fair	Good	Good	Medium	2

⁺ Figure of merit is based on combination of various factors from poorest (0) to best (4)

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TABLE 14
CHARACTERISTICS OF SELECTED SHIELD MATERIALS

Beryllium-Steel

Form - Slugs of unclad beryllium alternated with wafers of borated stainless steel.
Steel support framework.

Shield Weight * - 3,940 lbs.

Confidence Level - Characteristics of both materials are well known and adequate for the environment. Beryllium has been thoroughly tested in the actual reactor environment.

Design Suitability - Flexibility is fair. The shield can be readily shaped or increased for gamma ray attenuation but not for neutron attenuation. Mechanical design is straightforward and has few unknowns. Large elements can be used.

Cost - Less development required than for any other material. Hardware cost per shield - \$225,000.

Beryllium Oxide-Aluminum

Form - Rods of unclad borated beryllium oxide supported at their ends by an aluminum structural framework.

Shield Weight - 3,580 lbs.

Confidence Level - Most of the characteristics of both materials are well known and adequate for the environment. Some data are lacking on beryllium oxide at cryogenic temperatures.

Design Suitability - Flexibility is rather poor. Mechanical design is relatively straightforward and has few unknowns. Large elements can be used.

Cost - Development costs lower than for most other shield materials. Hardware cost per shield - \$240,000.

* For shield attenuation ratio of 0.1.

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TABLE 14 (CONTINUED)

Boron Carbide - Aluminum

Form - Extruded mixture of boron carbide and aluminum similar to "Boral"

Shield Weight - 3,730 lbs.

Confidence Level - Characteristics of the individual components are well known and adequate for the environment. "Boral" has been tested thoroughly in reactor environmental conditions near room temperature.

Design Suitability - Flexibility is poor. The density of the mixture cannot be varied much by varying the constituents. Mechanical design is straightforward and has few unknowns. Large shield elements can be used.

Cost - A medium amount of development is expected to be required with this material. Hardware cost per shield - \$190,000.

Hydrogen - Steel

Form - Honeycomb or stacked tubes of thin-walled steel with void space filled by high-density, low-temperature hydrogen.

Shield Weight - 3,710 lbs.

Confidence Level - The characteristics of both materials are well known and adequate for the environment. Both have been tested in the reactor environment.

Design Suitability - Flexibility is poor. High attenuation shields would have a large height penalty and poor neutron attenuation at its perimeter. The wall thickness of the tube or honeycomb structure can be varied to suit central attenuation requirements. Mechanical design will probably be the most complicated of all the shield concepts and would also require special pressure vessel penetration.

Cost - Development cost will be relatively modest. Hardware cost per shield - \$150,000.

Lithium Hydride - Steel

Form - Pressed pellets of lithium hydride encapsulated in steel tubes. Steel support framework.

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TABLE 14 (CONTINUED)

Shield Weight - 3, 180 pounds.

Confidence Level - Radiation damage resistance of lithium hydride is believed to be adequate but is highly dependent on the exact materials composition and processing techniques used. Methods of determining radiation damage resistance of a particular batch of material without actual irradiation of samples are not known. There are questions regarding the effective thermal properties of the irradiated material. Lithium hydride has not been tested in our reactor environment.

Design Suitability - Flexibility is moderately good. Wafers of steel can be inserted between pellets inside the tube to vary attenuation characteristics to suit requirements. The shield must be divided into relatively small elements to avoid overheating. Mechanical design will be more complicated than for unclad elements.

Cost - High development expenditure required for satisfactory results. Hardware cost per shield - \$690,000.

Titanium Hydride - Steel

Form - Hydrided rods of titanium encased in steel tubes. Steel support framework.

Shield Weight - 3, 430 pounds.

Confidence Level - Characteristics of both materials are well known at room temperature and above. Data are lacking on cryogenic characteristics. Titanium hydride has not been tested in our reactor environment.

Design Suitability - Flexibility is fair. Mechanical and thermal design are more complex than for unclad elements. The titanium hydride shield will be divided into elements of moderate size.

Cost - The development cost is expected to be moderate. Hardware cost per shield - \$440,000.

Titanium Hydride - Boron Carbide - Aluminum Composites

Form - Cylindrical or hexagonal extruded rods of unclad composite material supported at each end by an aluminum framework.

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TABLE 14 (CONTINUED)

Shield Weight - 3,440 pounds.

Confidence Level - Characteristics of the individual constituents of the composite are well known and adequate for the environment. Experimental information on the composite material characteristics does not exist.

Design Suitability - Flexibility is good. The volume fraction of the heavier titanium hydride used in the mixture can be varied to suit attenuation requirements. Mechanical design is straightforward and has few unknowns. Large shield elements can be used without overheating.

Cost - A medium amount of development is expected to be required. Hardware cost per shield - \$180,000.

* For shield attenuation ratio of 0.1

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The titanium hydride - boron carbide - aluminum composite material is unquestionably the least expensive of the three shields from the standpoint of the fabricated cost of each shield. Development cost will be intermediate between the beryllium and the lithium hydride shield. A shield made of this composite material is lighter than the beryllium shield but weighs approximately 270 pounds more than a lithium hydride shield for an attenuation ratio of 0.1.

A lithium hydride-steel shield is without doubt the lightest of the three. On the other hand, both the cost of development and the fabricated cost of each shield are expected to be considerably higher than for either of the other materials. In addition, a longer time will probably be required for development, and there is somewhat more uncertainty regarding the outcome.

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III. CONCLUSIONS

The most promising sets of the 25 materials considered in this study are lithium hydride-steel; titanium hydride-steel; titanium hydride-boron carbide-aluminum, beryllium oxide-aluminum, beryllium-steel, boron carbide-aluminum and hydrogen-steel (table 14).

It is recommended that the development and design of the $\text{TiH}_2\text{-B}_4\text{C-Al}$ composite shield be pursued. The shield design should be such as to permit substitution of beryllium-steel as a backup material.

Of the materials considered, beryllium-steel appears to be the only desirable material combination that could be built with essentially no materials development.

LiH appears to have poor growth potential for higher power densities or relocation of the shield nearer the core. It is also expected to be the highest cost shield. Therefore, it is recommended that LiH not be developed for the NR-1 flight shield, but be considered as a backup because of its low weight potential.

The least costly of the above seven shields are estimated to be boron carbide-aluminum; titanium hydride-boron carbide-aluminum and hydrogen-steel. The lowest weight shields are: lithium hydride-steel, titanium hydride-steel, and titanium hydride-boron carbide-aluminum.

The estimated cost for a finished flight prototype shield with an attenuation ratio of 0.1 ranged from a low of \$150,000 for hydrogen-steel to \$690,000 for lithium hydride-steel (table 11). The calculated weight for a practical design shield with an attenuation ratio of 0.1 ranged from 50 g/cm^2 for lithium hydride-steel to 62 g/cm^2 for beryllium-steel (table 7).

The shield development and design can proceed without final definition of the allowable radiation levels, since the material recommended retains its relative advantages over the transmission range of 0.25 to 0.04 considered in this study (table 4, figure 25).

While it may not be possible in many cases to build an optimum (minimum weight) shield because of materials and structural limitations, a practical off-optimum incurs a weight penalty of 0 to 14 percent for materials that appear useful for NR-1 application (table 7).

Materials, other than lithium compounds, that do not contain sufficient boron for suppression of secondary gamma ray production will suffer severe weight penalties.

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A shield material evaluation code, SOC-DS, has been developed for optimization of two materials. The code is simple and economical to use (see Appendix) and has been found to be relatively accurate in comparison to more exact (and more time consuming) transport calculations.

Figure 25. Attenuation Through Several Shields of Practical Compositions

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APPENDIX

EVALUATION METHODS AND COMPILATION OF DATA

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APPENDIX A

I. NUCLEAR METHODS

1.1 SHIELD OPTIMIZATION CODE - DIRECT SEARCH

The shield optimization routine, Shield Optimization Code - Direct Search (SOC-DS), used in the evaluation of shield material combinations is based on a gross model of a two-material homogeneous shield. The optimization variables selected in the shield model are total shield thickness and the volume fractions of the two constituents. The hydrogen coolant in the shield is represented explicitly as the third constituent, but the hydrogen volume fraction is a fixed parameter for each optimization problem. The output of the program is printed and/or automatically plotted.

1.1.1 Analytical Model

The attenuation of neutrons and gamma rays in the homogeneous shield is treated by simple analytical expressions for the radiation components of incident neutrons (fast and thermal), incident gamma rays, and radiative capture gamma rays born in the shield materials (figure 26). The attenuation of the primary components (incident neutrons and gamma rays) is solved from simple exponentials using removal theory cross sections for neutrons and a single representative gamma ray absorption coefficient for gamma rays. The radiation sources originating in the shield are gamma rays from (1) neutron absorption following neutron slowing down, and (2) absorption of thermal neutrons incident upon the shield.

Gamma ray energy release from neutron absorption in the shield is represented by two groups of gamma rays. Neutron capture in the shield material is assumed to yield a single, high-energy photon (table 19) group with a corresponding representative absorption coefficient. The second group of gamma rays is the low energy emission associated with parasitic or suppressor capture in boron. Hence, the overall shield attenuation model consists of a two-group neutron attenuation model with a three-group gamma-ray attenuation model.

One-group age theory is used to obtain neutron absorption rates from neutron slowing down of removal theory fluxes in the shield. A simplified Fermi age model is obtained from the assumptions that the neutron scattering cross section, $\sigma_s(u)$ average logarithmic energy decrement,

Figure 26. Considerations for Shield Materials Comparison Code

$\xi(u)$, and average cosine of scattering angle, $\bar{\mu}(u)$, can be represented as constant values over a broad neutron lethargy range.

The simplified age model is derived from the Fermi age equation:

$$\tau = \int_{u_1}^{u_2} \frac{du}{3.0 \bar{\Sigma}_s^2(u) \bar{\xi}(u) [1 - \bar{\mu}_0(u)]} \quad (1)$$

If one assumes that neutron absorption occurs only at the upper bound of the lethargy range (u_2) and that $\bar{\Sigma}_s$, $\bar{\mu}_0$ and $\bar{\xi}$ are invariant with lethargy, the age equation (age from removal theory neutron energies to thermal neutron energies) reduces to the approximation:

$$\tau = \frac{18.0^+}{3.0 \bar{\Sigma}_s^2 \bar{\xi} (1 - \bar{\mu}_0)} \quad (2)$$

For the two-material shield model with hydrogen included explicitly, the equations used to calculate $\bar{\Sigma}_s$, $\bar{\mu}_0$, and $\bar{\xi}$ for the homogeneous shield are as follows:

$$\bar{\Sigma}_s = \left[\Sigma_{sn} V_n + \Sigma_{sg} V_g \right] \cdot [1 - V_h] + \Sigma_{sh} V_h \quad (3)$$

$$\bar{\mu}_0 = \frac{[\mu_n \Sigma_{sn} V_n + \mu_g \Sigma_{sg} V_g] \cdot [1 - V_h] + \mu_h \Sigma_{sh} V_h}{\bar{\Sigma}_s} \quad (4)$$

$$\bar{\xi} = \frac{[\xi_n \Sigma_{sn} V_n + \xi_g \Sigma_{sg} V_g] [1 - V_h] + \xi_h \Sigma_{sh} V_h}{\bar{\Sigma}_s} \quad (5)$$

⁺ An implicit conservative assumption is made that neutrons initially at 8 - 10 MEV are scattered to thermal without absorption.

Where subscripts n, g, and h refer to neutron shielding material, gamma ray shielding material, hydrogen coolant, respectively and

$\sum_s$ = macroscopic neutron scattering cross section for each constituent

V = volume fraction of each constituent in the shield

NOTE: The two-material shield optimization routine is written so that V_h is a preselected value and does not enter into the optimization. Hence $V_n + V_g = 1.0$ is used within the optimum solution and final optimum volume fractions are reported as such.

ξ = logarithmic energy decrement, which is approximated by $6.0 / (2.0 + 3A)$

μ_o = cosine of the scattering angle, which is approximated by the equation $\mu_o = 2 / (3A)$

A = atomic weight of the moderating, or lightest, atom in the molecule of each material. For hydrogen, A is taken as $4/3$ so that $\bar{\xi} = 1$.

Since the compounds or materials considered contain hydrogen and/or combinations of relatively light and heavy atoms, the slowing effect of the heavy atom(s) is neglected. Their presence is included in the scattering cross section for the compound.

For the hydrogen coolant in the homogeneous shield, the values of ξ and μ_o were selected as 1.0 and 0.6667.

The gamma ray energy release of high- and low-energy gamma rays from neutron capture is calculated on the basis of the absorption cross sections in the shield materials. The average energy release for high- and low-energy gamma rays is:

$$\begin{aligned} \bar{E}_{HE} &= \frac{(\sum_{an}^t - \sum_{an}) E_{nHE} V_n + (\sum_{ag}^t - \sum_{ag}) E_{gHE} V_g}{\sum_{an}^t V_n + \sum_{ag}^t V_g} \\ \bar{E}_{LE} &= \frac{\sum_{an} E_{nLE} V_n + \sum_{ag} E_{gLE} V_g}{\sum_{an}^t V_n + \sum_{ag}^t V_g} \end{aligned} \quad (6)$$

Where subscripts

HE = high energy gamma ray energy release from neutron radiative capture in the neutron (n) or gamma ray (g) shield material

LE = low energy gamma ray energy release from neutron radiative capture in a parasitic absorber in the shield materials
 NOTE: A ground rule established in this study was that 90 percent of all neutron absorptions would occur in parasitic absorber or poison (normally boron additive).

and, $\sum_a^t$ = total macroscopic absorption cross sections in shield material including the absorption cross section for parasitic absorber or poison in shield material
 $\sum_a$ = macroscopic neutron absorption cross section for parasitic absorber in shield material
 $\sqrt{V}$ = volume fraction of each constituent in the solid shield region.

The calculation of the spatial position of neutron absorption following slowing down in the slab shield model is solved using a simplified slowing down model (equation 2).

In an infinite slowing medium with no capture, the displacement of neutron absorption is calculated from the neutron removal cross section and the Fermi age of the medium. This slowing density then becomes a gamma ray source distribution for neutron capture.

The secondary gamma ray source distribution due to capture of incident thermal neutrons is given by $\exp\left(-t/L\right)$, which for strong absorbers (i.e., $\sum_a \gg \sum_s$) is approximately $\exp\left(-\sum_a t\right)$.

The shield attenuation model solved is represented as a seven-term equation for the separate radiative components. The seven terms are:

1) Fast (removal theory) neutrons

$$\phi_1 = \phi_N \exp\left(-\bar{\sum}_R t\right); \quad (7)$$

2) Slowed neutrons (Fermi age theory)

$$\phi_2 = \bar{\sum}_R \phi_N \exp\left(\bar{\sum}_R^2 \tau\right) \exp\left(-\bar{\sum}_R t\right); \quad (8)$$

3) Incident Gamma Ray

$$\phi_3 = \phi_g \exp\left(-\bar{\mu} t\right); \quad (9)$$

4) High-energy gamma ray from neutron capture following neutron slowing down

$$\phi_4 = \bar{E}_{HE} \bar{\sum}_R \phi_N \exp\left(\bar{\sum}_R^2 \tau\right) F\left(\bar{\sum}_R, \bar{\mu}_{SHE}\right); \quad (10)$$

- 5) Low-energy gamma rays from neutron capture following neutron slowing

$$\phi_5 = \bar{E}_{LE} \bar{\Sigma}_R \phi_N \exp \left(\bar{\Sigma}_R^2 \tau \right) F \left(\bar{\Sigma}_R, \bar{\mu}_{SLE} \right); \quad (11)$$

- 6) High-energy rays from incident thermal neutron capture

$$\phi_6 = \bar{E}_{HE} \left(\bar{\Sigma}_a^t - \bar{\Sigma}_a \right) \phi_{NT} F \left(\bar{\Sigma}_a^t, \bar{\mu}_{SHE} \right); \quad (12)$$

- 7) Low-energy gamma rays from incident thermal neutron capture

$$\phi_7 = \bar{E}_{LE} \bar{\Sigma}_a \phi_{NT} F \left(\bar{\Sigma}_a^t, \bar{\mu}_{SLE} \right); \quad (13)$$

where the function $F(a, b)$ is defined as

$$F(a, b) = \frac{1}{2 \cdot a} \left\{ E_1(bt) + \exp(-at) \cdot \left[E_1[(b-a)t] - \ln \left| \frac{b}{b-a} \right| \right] \right\} \quad (14)$$

where

E_1	=	exponential integral function, $E_1(b)$
ϕ_1	=	incident fast neutron (removal theory) flux
ϕ_2	=	incident thermal neutron flux
ϕ_3	=	incident gamma ray energy flux
K_i	=	importance function (heat deposition or dose rate) for each radiation component
$\bar{\Sigma}_R$	=	neutron removal cross section for two-material shield
t	=	the linear thickness of shield
τ	=	Fermi age of two-material shield including hydrogen coolant
μ	=	linear gamma-ray absorption coefficient for incident gamma rays
$\bar{\mu}_{SHE}$	=	linear gamma-ray absorption coefficient for high-energy shield secondary gamma rays
$\bar{\mu}_{SLE}$	=	average energy release with neutron capture for low-energy shield gamma rays
$\bar{\Sigma}_a, \bar{\Sigma}_a^t$	=	total neutron capture cross section for parasitic and parasitic plus shield materials, respectively

The program solves the above equations for the total radiation levels exiting the shield, and in combination with the respective importance functions a total shield performance is calculated. By making successive exploratory and pattern search changes in the variables, the constrained optimum solution is obtained.

The program was modified to obtain a graphic display of the characteristic shield weight changes for non-optimum cases. The modified program (SOC-DS Plot) was limited to a single variable, linear thickness; and for a fixed volume fraction, the constraint equation is evaluated. This provides shield weight with variations in shield compositions.

1.1.2 Optimization Method

The optimizing function used in the program is calculated from the shield mass thickness (including penalty for reactor vessel envelope) and from the inequality constraint that the shield performance must achieve a preselected value. The user has the option of selecting the magnitude of the constraint constants so that a desired level of accuracy is achieved at the optimum solution. The total optimizing function, S_n , is defined as follows:

$$S_n = \left[t_m^* + C_2 \cdot (1.0 - P)^R \right]$$

where

$$\begin{aligned} t_m^* &= \text{penalized shield mass thickness (g/cm}^2\text{)} \\ P &= \text{shield performance function} \\ C_2 \text{ and } R &= \text{input constants used to determine the magnitude of the inequality constraint.} \end{aligned}$$

The shield performance function used in the program has been derived from a gross slab shield model that considers the attenuation of radiation incident upon the shield and the radiation flux at the exit surface of the shield from sources internal to the shield. The total shield performance is calculated from the fast neutron (removal theory), thermal neutron, and gamma-ray energy flux incident upon the shield, and total radiation energy at the exit surface of the shield. The user of the program can specify an importance function for each radiation component in the shield attenuation model, making it possible to weight the neutron events

occurring external to the shield (neutron kinetic energy deposition, neutron radiative capture, or neutron dose) according to their importance in tank heating, or total dose rate calculations. The shield performance function, P , is the ratio of the fractional performance of the shield Δ

$$\text{where } \Delta = \frac{\sum_{i=1}^I K_i \phi_i}{\sum_{m=1}^M K_m \phi_m}$$

to the constraint, Δ^* . In the program,

$$P = \frac{\Delta}{\Delta^*} \text{ or } \frac{\Delta^*}{\Delta}$$

so that the inequality,

$$P > 1.0$$

is satisfied.

This constraint then provides a functional relationship between shield performance and the optimization parameters so that minimization of the optimizing function will provide the desired shield performance, P .

The basic operation of the search routine is described in detail in reference 6. The routine performs the exploratory and pattern search moves of each of the variables and moves the solution toward the constrained optimum value of the shield weight. The search routine is designed to operate in an alternating pattern-exploratory search mode with acceleration and deceleration parameters supplied by the user. The search routine continues to operate until the differences in the optimizing function for successive calculations and/or the search variable step sizes are reduced below a preselected convergence criteria.

1.1.3 Optimization Code

a. Input Description

The input to the SOC-DS Plot routines is patterned to minimize the amount of data required for each problem and to allow the user some flexibility in using the programs. There

are five types of data cards required for a complete problem. The initial data card for each problem deck is a fixed point data card which establishes the search routine operational limits. The second data card is applicable only to the SOC-DS Plot routine and is alphanumeric data for titling of the plot data output. The final three data cards are required for each problem and contain:

- 1) The title of the specific problem deck
- 2) The materials title card
- 3) The floating point data.

The following sections discuss each data card type and the deck setup shown in figure 27 indicates a typical problem deck.

b. Fixed Point Data Card

Fixed point data to the program are input only once in a set of problems. The user must specify the number of variables, search options, the maximum number of calculations to be performed, and the print control option for the program. Fixed point data are entered on a single card (Fortran Format 24I3), which is the first data card of an input deck. A fixed point data input is tabulated in table 15 with a brief description of each quantity.

c. Plot-Title Card (SOC-DS Plot Only)

The plot-title card is input only once for a set of problems to be plotted in the SOC-DS Plot routine. The data card identifies all plotting information with the user's name and scales the ordinate (Y-axis) by input quantities. The name specification is limited to six characters in columns 1-6 of the plot title card. The scaling values (YX1, YX2) in columns 13-24 and 25-36 of the card are determined by the user. If the plot on grid paper is desired, the user must determine the relative scaling units per inch of the ordinate (Y-axis) values. The standard plot paper is 10 inches vertically with a 10 by 10 per inch grid and the SOC-DS Plot routine is programmed to plot an eight-inch high figure. Scaling data must fit this limit (80 grid lines). For example, if the range of ordinate values is 0.0 - 160.0, the user would specify a starting Y-axis value (YX1) of -20.0 and a final Y-axial value (XY2) of 180.0. The figure

* Delete plot title card from deck setup for the SOC-DS routine since no plot data are obtained.

Figure 27. SOC-DS Sample Problem Deck Setup

TABLE 15
SOC-DS FIXED POINT DATA INPUT

Card Type 1

Fortran Symbol	Fortran Format	Description
NV	(2413)	Number of independent variables in search For SOC-DS, use $NV = 2$ For SOC-DS Plot, use $NV = 1$
LA		LA is input as 1 for all problems; primary branching control for search routine. LA is an internally controlled trigger that designates mode in which the program is operating. LA = 1 for first entry into search routine (initialization) LA = 2 for initial move of any variable during exploratory search mode LA = 3 for reverse move of any variable during exploratory search mode LA = 4 for pattern move in pattern search mode LA = 5 for initial move of any variable during pattern search mode LA = 6 for reverse move of any variable during pattern search mode LA = 7 for temporary optimum (base point) calculation LA = 8 search routine has minimized function and/or decreased variable step sizes to within the convergence criteria LA is printed output if MOUT > 1.
LIM		Maximum number of search move calculations to be performed by search routine
LT, LSN		Control quantities for alteration of normal search routine operation. (See reference 6 for explanation). LT and LSN are normally = 0.
MOUT		Control quantity for printing of intermediate values of search variables and calculation results (see output description for output available). MOUT = 1 for no intermediate output MOUT = 2 for intermediate output at base points (LA = 7) only MOUT = 3 for intermediate output at all calculation points.
KS		Search variable at which search is to be started. The value of KS must be equal to or greater than 1 and less than or equal to NV. KS is normally equal to 1.
NTYPE		Control word for calculation type; if NTYPE = 0, SOC-DS with two variables is used; if NTYPE = 1, SOC-DS Plot with one variable search plus plotting is used.

would then appear as shown in figure 28. The user must keep in mind that the grid lines are 0.1 inch and a scale must be selected to match these lines. The user may use any scaling unit if the plot is placed on plain paper; the figure labeling will appear as in figure 28.

In addition, the SOC-DS Plot routine assumes the Y-ordinate to start at 0.0. This procedure allows for titling of the illustration in the area between 0 and 1 inch and 9 and 10 inches of the figure.

The input data on the card are entered in a Fortran format (A6, 2E12.5) as shown in figure 28.

d. Problem Title Card

The user may specify any alphanumeric information in the problem title card. These alphanumeric data appear on the printed output for the problem for each case within a given problem set. The data are entered in columns 1-72 of the card.

e. Materials Title Card

The user may specify three titles for all plot data. The first two titles will title the abscissa and the figure. The third title identifies each curve of the plot. The title data are entered in a Fortran format (5A6) with the first two titles being 12 characters long and the third title 6 characters. The examples in figures 28 and 29 indicate the relation between input and plot information.

f. Floating Point Data Cards

The bulk of input data to the programs is entered in the floating point data cards. Each card is constructed as illustrated in figure 30. The input routine contains an addressable input format that allows the user the flexibility of entering single data words in one, two, or three dimensional arrays of data. The routine assigns all data to a common data vector (A1), computes the address of the quantity, and equates these data to the variable used within the program. This routine is useful in problems where sequential or serial problems use similar data with only minor data changes. The data on a given card are assigned addresses beginning with the address specified on the card and sequentially loads all data on the card up to and including the last piece of data. The user specifies the number of pieces of data on the card

Figure 28. SOC-DS Sample Plot Title and Scaling Operation

Figure 29. SOC-DS Plot Title Operations

FORTAN FORMAT (I2, I1, I5, 6E10.3)

Columns 1-2

NOF, No. of pieces of data on card ($1 \leq \text{NOF} \leq 6$)

Column 3

LF, if LF is blank or 0, the program expects the following card to be a floating point data card. If LF is 1, the program assumes that all data are read and proceeds to calculation.

Columns 4-8

NAF, address of the first piece of data (Columns 9-18) on the card.

Columns 9-18

Floating point data for addresses beginning at NAF.

19-28

29-38

39-48

49-58

59-68

Figure 30. SOC-DS Floating Point Data Card

and also specifies the last data card.

The data entered in a given addressable location will remain in this location until changed by the user. Hence, the user can use previously loaded data in serial problems, but must also change the data word if it is not required in serial problems. In addition, the user must exercise caution in specifying the number of pieces of data per card so that data words are not written over in serial problems. Input data and a brief description of each input quantity are shown in table 16.

Three quantities, XMAX, XMIN, and X, are the input values of the search move variables for the program. The first variable, X(1) is the linear shield thickness, T_1 , and the range of values it may assume during the search is determined by XMIN (1) and XMAX (1). Correspondingly, the second variable, X(2), is the volume fraction, V_n , of the first or neutron shield material in the two-constituent shield. (Hydrogen coolant channel volume fraction, V_h , is the separate input quantity, hence $V_n + V_g = 1.0$).

The third variable, X(3), is not presently used in the search program, but is used as the initial value of X(1) for serial problems. This procedure is used to provide a common starting point, $X(1) = X(3)$, for all serial problems. If the user does not enter X(3) on serial problems the initial value of X(3) is used for all problems.

g. Output Description

Two types (intermediate and final) of printed output are available with the SOC programs. The intermediate output (figure 31) is under input control as specified by the user in the data word, MOUT. These output data are primarily checkout or debug data for determining the search move pattern and also the relationship between the inequality constraint function and the total optimizing function. The data obtained at each calculation or temporary optimum (base) calculation point are shown in figure 32.

These data should be obtained only in checkout problems since the amount of intermediate data can become proliferous if obtained for each calculation point for all problems.

The final output data obtained at the completion of each problem differs between the SOC-DS and SOC-DS Plot routine because of the magnitude of printed output for stacked

TABLE 16
SOC-DS FLOATING POINT DATA INPUT

Address	Fortran Symbol	Term	Description
1	TN	t_n	Mass thickness (g/cm^2)
(20)	(TG)	(tg)	
2	DN	ρ_n	Density (g/cm^3) of shield material
(22)	(DG)	(ρ_g)	
3	SRN	$\sum_{rn}/\rho_n$	Mass neutron removal cross section (cm^2/gm)
(23)	(SRG)	($\sum_{rg}/\rho_g$)	
4	SUN	μ_{1n}/ρ_n	Mass gamma ray absorption coefficient for incident gamma rays (cm^2/gm)
(24)	(SUG)	(μ_{1g}/ρ_g)	
5	SUNS	μ_{2n}/ρ_n	Mass gamma ray absorption coefficient for high-energy secondary gamma rays born in shield (cm^2/gm)
(25)	(SUGS)	(μ_{2g}/ρ_g)	
6	SSN	$\sum_{sn}/\rho_n$	Mass neutron scattering cross section for neutrons. This quantity is representative of the slowing power of shield material and is used only in Fermi age calculations.
(26)	(SSG)	($\sum_{sg}/\rho_g$)	
7	YSN	$\bar{E}_{sn}$	High-energy gamma-ray energy release from neutron capture event following neutron slowing (Mev/event)
(27)	(YSG)	($\bar{E}_{sg}$)	
8	YISN	$\bar{E}_{in_n}$	High-energy gamma-ray energy release from neutron inelastic scatter event (Mev/event) (not presently used)

TABLE 16 (CONTINUED)

Address	Fortran Symbol	Term	Description
(28)	(YISG)	$(\bar{E}_{in_g})$	
9	YCN	$\bar{E}_{cn}$	High-energy gamma-ray energy release from thermal neutron capture event (Mev/event)
(29)	(YCG)	$(\bar{E}_{cg})$	
10	SCN	$\sum_{an}^T / \rho_n$	Total neutron capture cross section including parasitic or suppressor material (cm ² /gm)
(30)	(SCG)	$(\sum_{ag}^T / \rho_g)$	
11	SIN	$\sum_{in_n} / \rho_n$	Neutron inelastic scatter cross section for production of gamma rays (cm ² /gm) (not presently used)
(31)	(SIG)	$(\sum_{in_g} / \rho_g)$	
14	AN	An	Atomic weight of shield material. This quantity is used in the calculation of ξ and μ_o and should reflect the consideration of $\sum_{sn} / \rho_n$ and ξ in the slowing power of the material.
(34)	(AG)	Ag	
15	SACN	$\sum_{cn} / \rho_n$	Neutron capture cross section for parasitic or suppressor material (cm ² /gm)
(35)	(SACG)	$(\sum_{cg} / \rho_g)$	
16	YACN	$\bar{E}_{cn}$	Low-energy gamma-ray energy release from parasitic or suppressor material thermal neutron capture (Mev/event)
(36)	(YACG)	$(\bar{E}_{cg})$	

TABLE 16 (CONTINUED)

Address	Fortran Symbol	Term	Description
17	YASN	$\bar{E}_{sn}$	Low-energy gamma-ray energy release from parasitic or suppressor material neutron capture following slowing (Mev/event)
(37)	(YASG)	$(\bar{E}_{sg})$	
18	SUNSI	μ_{3n}/ρ_n	Mass gamma-ray absorption coefficient for low-energy gamma ray (cm^2/gm)
(38)	(SUGSI)	(μ_{3g}/ρ_g)	

Quantities are repeated for the second or gamma ray shield material with the addresses, FORTRAN names, and nomenclature shown in parentheses.

12	RKN	K_1	Importance function of neutrons (removal theory) in shield performance evaluation
13	RKT	K_2	Importance function of thermal or intermediate energy neutrons in shield performance evaluation
19	AW	C_1	Shield weight penalty constant for interstage, pressure vessel, and other nuclear engine components that are affected by shield height or thickness. The penalty constant in units of (g/cm^3) is added (or subtracted) from the shield weight if the linear shield thickness deviates from the input quantity, RT, reference shield thickness.
20	VOID	V_h	Volume fraction of shield materials in shield, or (1.0-VOID) is volume fraction of hydrogen coolant channels in shield
32	RKG	K_3	Importance function of incident gamma rays in shield performance evaluation
33	RKGS	K_4	Importance function of high-energy gamma rays born in shield in shield performance evaluation
39	EPST	ϵ_T	Not presently used, not required in input

TABLE 16 (CONTINUED)

Address	Fortran Symbol	Term	Description
40	EPS	ϵ	Not presently used, not required in input
41	PHIA	Δ^*	Shield performance criteria or optimization constraint
42	PHIN	ϕ_N	Incident fast neutron flux (n/cm^2 -sec)
43	PHINT	ϕ_{NT}	Incident thermal neutron flux (n/cm^2 -sec)
44	PHIG	ϕ_g	Incident gamma ray energy flux (Mev/cm^2 -sec)
45	RT	R_T	Linear shield thickness allowed in reactor pressure vessel envelope (cm)
46	RS	R_S	Shield radius (cm)
47	RKGST	K_5	Importance function of low-energy (parasitic or suppressor) gamma rays in shield performance evaluation
48	ALPHA	α	Acceleration parameter for search routine moves (α is normally input as 2.0)
49	BETA	β	Deceleration parameter for search routine moves (β is normally input as 0.2)
50	DEL	δ	Initial search move size parameter; $D(n) = \delta * (XMAX(n) - XMIN(n))$ where $D(n)$ = initial search move step size for n^{th} variable.
51	TOL	ϵ_S	Convergence criteria of search routine. The search is converged when the optimizing function, S_n and the last base point optimizing function, S_{nb} , are within tolerance, TOL, of each other or $D(n) \leq TOL$ for all variables.
52	TOL1	R	Constraint equation exponent which defines the form of the constraint function. TOL1 is normally 2.0.
53	PK	C_2	Constraint penalty constant. C_2 is selected on basis of accuracy desired in achieving the optimization constraint, Δ^* , and should be proportional to the final optimizing function. C_2 is normally input as $10. * (tn + tg)$.

TABLE 16 (CONTINUED)

Address	Fortran Symbol	Term	Description
54	XMAX	$X_{\max}$	Maximum value that search variables may assume in search moves
54	XMAX (1)		
55	XMAX (2)		
56	XMAX (3)		
57	XMIN	$X_{\min}$	Minimum value that search variables may assume in search moves ($X_{\min} > 0.0$)
57	XMIN (1)		
58	XMIN (2)		
59	XMIN (3)		
60	X	X	Actual value of search variables during search. The only required input is X (2) and X (3). X(2) is the initial or starting point of volume fraction of neutron shield material and is required for SOC-DS only. X(3) is the initial value of X(1) for serial problems; X(3) may be entered only once for a set of problems. Program SOC-DS Plot assumes the values of XVFN for X(2), XMAX (2), and XMIN (2) and varies only X(1).
60	X (1)		
61	X (2)		
62	X (3)		
Input Quantities 69 and 70-84 are required for SOC-DS Plot			
68	PDATA		Number of data points XVFN to be evaluated for each curve
69	PLTN		Number of separate curves to be plotted on each illustration

TABLE 16 (CONTINUED)

70	XVFN	V_n	Volume fraction of neutron shielding material in shield. There must be quantities of V_n as input and for plotting. These values must monotonically decrease in the range $(0.0 \leq V_n \leq 1.0)$
70	XVFN (1)		
71	XVFN (2)		
72	XVFN (3)		
73	XVFN (4)		
74	XVFN (5)		
75	XVFN (6)		
76	XVFN (7)		
77	XVFN (8)		
78	XVFN (9)		
79	XVFN (10)		
80	XVFN (11)		
81	XVFN (12)		
82	XVFN (13)		
83	XVFN (14)		
84	XVFN (15)		

Note: All cards are loaded with the format shown in figure 30.

N	WEP	SN	TN	TNL	TG	TGL
	R	RATO	TX			
	X(1)	XMAX(1)	XMIN(1)			
	X(2)	XMIN(1)	XMIN(2)			
	X(3)	XMAX(3)	XMIN(3)			

where

N	=	search calculation number
WEP	=	penalized shield weight (SOC-DS Plot) or penalized shield mass thickness (SOC-DS)
SN	=	optimizing function including WEP and the inequality constraint function
TN, TNL, TG, and TGL	=	mass and linear thicknesses of neutron or gamma ray shield material excluding coolant channel volume fraction
TX	=	total shield mass thickness excluding penalty for shield linear thickness
R, RATO	=	shield performance evaluation parameter Δ/Δ^* or its inverse Δ^*/Δ ; R and RATO = 1.0 indicates the constraint has been met
X, XMAX, and XMIN	=	search variables and ranges; see input description for discussion

FIGURE 31. SOC-DS INTERMEDIATE OUTPUT DATA

DIRECT SEARCH PARAMETERS AND RESULTS

NO. OF VARIABLES*****	1	← $D\delta$
INITIAL STEP SIZE*****	1.0000E-01	← TOL; G_s
TOLERANCE ON VARIABLES*****	1.0000E-02	← TOL1 or R
TOLERANCE ON FUNCTION*****	1.0000E-05	
STEP SIZE CONTROLS (ALPHA)*****	1.6667E 00	← α
(BETA)*****	2.2222E-01	← β
SEARCH CONTROL (LA)*****	2	
SEARCH CONTROL (LSN)*****	0	
SEARCH CONTROL (TRUNCATE)*****	0	← LT
NO. OF CALCULATIONS*****	41	
OPTIMUM FUNCTION (N)*****	1.0077E 06	← S_n
OPTIMUM FUNCTION (N-1)*****	1.0057E 06	
OPTIMUM FUNCTION (N-2)*****	1.0057E 06	
VARIABLES(MIN.)*****	1.00E 00	← XMIN(1)
VARIABLES*****	1.23E 01	← X(1)
VARIABLES(MAX.)*****	2.00E 02	← XMAX(1)
VARIABLES(N-1)*****	1.23E 01	
VARIABLES(N-2)*****	1.23E 01	
DELTA VARIABLES*****	-2.40E-03	← D(1)

Figure 32. Direct Search Parameters and Results

problems. The SOC-DS routine provides a complete output package at the completion of each problem. This output data consist of:

- 1) The search routine input parameters and calculation results;
- 2) The program input and calculation results;
- 3) If the problem has converged to the specified tolerance, the optimum values of each search variable and the optimum function.

The search routine output is shown in figure 32 and for output variables noted according to the input specifications. The following quantities $S_n(n)$, $S_n(n-1)$, and $S_n(n-2)$, are calculated values of the optimizing function of the last calculation, the last temporary optimum base point, and the next to last base point. The search routine is programmed to accept the last base point as the optimum function if there has been 15 exploratory search moves following a successful pattern move. This logic is self-contained in the search routine in order to eliminate many successive search moves in the areas of very shallow valleys of the optimum solution. The quantities, $D(i)$ are also used in convergence tests and are the search move step sizes for each variable at the completion of the problem.

The program output is shown in figure 33 for a typical problem. Symbolism is included in the illustration, identifying quantities that are not adequately described in the input specifications and description of the calculation equations.

Output obtained from the SOC-DS Plot routine is the final program input and calculation results, and the plotting information is in graphic and tabular form. The data shown in figure 34 are the calculation results for the curve 1 data plotted in figure 35. The first column of printed data is the volume fraction of the first or neutron shield material in the two-constituent shield. The second column is the calculated penalized shield weight. The final calculation results in figure 33 are obtained for each data point in the plot information.

1 ALUMINUM BLOCK SHIELD WITH BEO INSIDE
 SHIELD RADIUS *****
 SHIELD PARAMETERS(P.C.,S.V.F.,REF.SH.T.)*****
 NEUTRON SHIELD *****

THICKNESS(MASS,LINEAR)*****
 DENSITY *****
 REMOVAL CROSS SECTION,MASS *****
 ABSORPTION COEFF.(PRI.,SEC.,SEC.),MASS*****
 SCAT. CROSS SECTION,MASS*****
 IN. SCAT. CROSS SECTION,MASS *****
 CAPTURE CROSS SEC. (TOTAL,SUPPRESS.) *****
 ATOMIC WEIGHT *****
 GAMMA RAY YIELDS,(CAP.,CAP.)*****
 ,(SLOW. DOWN,SLOW. DOWN)****
 ,(IN. SCAT.) *****

GAMMA RAY SHIELD *****

THICKNESS(MASS,LINEAR)*****
 DENSITY *****
 REMOVAL CROSS SECTION,MASS *****
 ABSORPTION COEFF.(PRI.,SEC.,SEC.),MASS*****
 SCAT. CROSS SECTION,MASS*****
 IN. SCAT. CROSS SECTION,MASS *****
 CAPTURE CROSS SEC. (TOTAL,SUPPRESS.) *****
 ATOMIC WEIGHT *****
 GAMMA RAY YIELDS,(CAP.,CAP.)*****
 ,(SLOW. DOWN,SLOW. DOWN)****
 ,(IN. SCAT.) *****

INCIDENT FLUXES,(NEUTRON-FAST) *****
 (NEUTRON-THERMAL) *****
 (GAMMA RAY) *****

FLUX CONVERSION CONSTANTS,(NEUTRON-FAST) *****
 ,(NEUTRON-THERMAL) ***
 ,(GAMMA RAY-PRI.) ****
 ,(GAMMA RAY-SEC.) ****
 ,(GAMMA RAY-IN.SC.) **

CONVERGENCE CRITERIA-THICKNESS*****
 CONVERGENCE CRITERIA *****
 OPTIMIZATION CONSTRAINT *****
 RICK'S BORON FACTOR (MASS,VOLUME)*****
 TOTAL SHIELD THICKNESS (MASS,LINEAR) *****
 SHIELD THICKNESS (PEN.),V.F.N.*****
 NEUTRON SHIELD THICKNESS *****
 GAMMA RAY SHIELD THICKNESS *****
 SHIELD WEIGHT(1.ACTUAL,2.PENALIZED)*****
 GAMMA RAY FLUXES (EXITING)*****
 NEUTRON FLUXES (EXITING) *****

Figure 33. Sample Output

```

PLOTING INFORMATION IS COMPLETE FOR CURVE 1
0.95000E 00      0.37233E 02
0.90000E 00      0.37965E 02
0.85000E 00      0.38729E 02
0.80000E 00      0.39514E 02
0.70000E 00      0.41184E 02
0.60000E 00      0.43063E 02
0.50000E 00      0.45272E 02
0.45000E 00      0.46565E 02
0.40000E 00      0.48056E 02
0.35000E 00      0.49810E 02
0.30000E 00      0.51951E 02
0.20000E 00      0.58289E 02
0.10000E 00      0.71170E 02
0.50000E-01      0.83785E 02

```

Figure 34. Sample Plotting Data Output

1.2 TAPAT SYSTEM ONE-DIMENSIONAL TRANSPORT ANALYSIS

Table 17 lists the shield configurations that were analyzed by the TAPAT code system. The volume percents of the shield materials are given in the first column of the table. Note that the total volume percentages of the shields equals 80 percent. The remaining 20 percent by volume of the shield region is comprised of steel structure (1.46 percent) and hydrogen coolant (18.54 percent). The 1.46 percent volume fraction of steel is an estimate, based on an early design concept, of the nuts, bolts, screen, etc. needed to hold the shield together. Note this exception: the liquid hydrogen shield employed an aluminum structure as shown in table 17. The table also lists shield densities, mass, and linear thicknesses.

In most cases, the penalized total shield weight was held constant at 54.2 g/cm². In cases where the shield material thicknesses varied from the reference of 9 inches, a weight penalty or weight reduction was made in the shield material thicknesses. The difference is especially apparent for the low density materials such as water, LiH, and liquid hydrogen.

Figure 35. SOC-DS Sample Plot

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TABLE 17
PHYSICAL DESCRIPTION OF SHIELDS ANALYZED
BY ONE-DIMENSIONAL TRANSPORT CODE (TAPAT)

Shield Materials*	Shield Density (gm/cc)	Shield Thickness (gm/cm ²)	Shield Thickness (cm)
BeO (30.4 percent), Al (49.6 percent)	2.37	54.2	22.9
BeO (49.6 percent), Al (30.4 percent)	2.43	54.2	22.1
BeO (80 percent)	2.48	54.2	21.8
BeO (80 percent)**	2.48	54.2	21.8
LiH (56 percent), SS (24 percent)	2.46	54.2	22.0
LiH (33 percent), SS (47 percent)	4.10	54.2	13.2
LiH (80 percent)	0.74	41.9	56.7
TiH ₂ (72 percent), Al (8 percent)	3.05	54.7	18.0
TiH ₂ (29.6 percent), Al (50.4 percent)	2.60	54.2	20.8
TiH ₂ (8 percent), Al (72 percent)	2.36	54.2	22.9
LH ₂ (80.1 percent)*** Al (19.9 percent)	0.598	45.2	75.6
Be (80 percent)	1.60	52.4	32.7
H ₂ O (80 percent)	0.841	49.0	58.2
Al (80 percent)	2.28	54.2	23.8
<p>* Numbers in parentheses are volume percents. The shielding materials occupy only 80 percent of the shield region volume. The other 20 percent of the volume contains steel structure (1.46 percent) and coolant (18.54 percent)</p> <p>** This shield contains 20 times the amount of boron in the previously listed BeO shield.</p> <p>*** This shield contains no steel structure.</p>			

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Each shield contains a different boron atom density. The boron atom density was computed in such a way that 90 percent of the total thermal captures in the shield occurred in the boron. In the case of BeO, however, it was found that the large neutron reflection to the support plate caused substantial secondary gamma production in the plate. Hence, the absorption of boron was increased a factor of 20 to 99.5 percent for this case. Figure 36 shows the dimensions of the axial regions employed in the one-dimensional transport analysis.

Note: Drawing not to scale. The shield thickness, t , for each shield analyzed is given in table 17.

* Axial position for which data are given in table 17.

Figure 36. One Dimensional Transport Code Region Descriptions

The method used in this analysis is described schematically in figure 37. The NSDM-III computer code is used to generate eight (three fast and five thermal) neutron group cross section data on tape. This tape is used as input to the TAPAT code system. The output cross sections from NSDM-III were also used to generate gamma-ray production cross sections, which are also input to the TAPAT code.

The codes in the TAPAT system employed in this analysis were the MIST neutron and photon transport codes and the FLUX Edit code. The FLUX Edit code computes the spatial and group dependent gamma ray sources from the neutron fluxes for use in the photon transport code. Results of the analysis are summarized in tables 18 and 6.

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* The TAPAT codes employed in this analysis were the flux edit code and the MIST neutron and photon transport codes.

Figure 37. Flow Diagram of One-Dimensional Transport Method Analysis

TABLE 18

COMPARISON OF SHIELDED AND UNSHIELDED GAMMA RAY AND NEUTRON FLUXES
COMPUTED BY THE ONE-DIMENSIONAL TRANSPORT CODE (TAPAT)

Shield* Description (Percent)	Gamma Ray Flux (Mev cm ² -sec-n)			0.821 ≤ E < 10 Mev		Neutron Flux Data		E < 1.86 ev	
	Total γ Flux	Total Flux Minus Secondary Gamma Source Generated in Shield	Ratio of Total γ Flux to Unshielded Flux	Flux n/cm ² -sec	Ratio of Shielded to Unshielded	Flux n/cm ² -sec	Ratio of Shielded to Unshielded	Flux n/cm ² -sec-n	Ratio of Shielded to Unshielded
Unshielded	1.04 (-1)	1.04 (-1)	1.0	3.74 (-3)	1.0	2.34 (-2)	1.0	2.83 (-2)	1.0
BeO (30.4), Al (49.6)	1.15 (-2)	9.14 (-3)	0.111	3.36 (-4)	0.089	4.35 (-3)	0.186	3.22 (-3)	0.114
BeO (49.6), Al (30.4)	1.18 (-2)	9.65 (-3)	0.113	2.18 (-4)	0.058	2.89 (-3)	0.123	2.72 (-3)	0.096
BeO (80)	1.20 (-2)	1.03 (-2)	0.115	1.03 (-4)	0.028	1.45 (-3)	0.062	2.06 (-3)	0.073
BeO (80)**	9.69 (-3)	8.80 (-3)	0.093	1.02 (-4)	0.027	1.38 (-3)	0.059	5.79 (-4) ⁺	0.020
LiH (56), SS (24)	6.62 (-3)	6.14 (-3)	0.064	2.26 (-4)	0.060	4.44 (-4)	0.018	1.24 (-4) ⁺	0.0044
LiH (33), SS (47)	7.76 (-3)	6.31 (-3)	0.075	7.20 (-4)	0.192	2.39 (-3)	0.102	7.12 (-4) ⁺	0.025
LiH (80)	1.15 (-2)	1.15 (-2)	0.111	4.71 (-6)	0.0013	6.52 (-6)	0.00028	2.80 (-4) ⁺	0.0099
TiH ₂ (72), Al (8)	6.93 (-3)	5.47 (-3)	0.067	1.67 (-4)	0.045	2.23 (-4)	0.0095	8.57 (-5) ⁺	0.0030
TiH ₂ (29.6), Al (50.4)	8.52 (-3)	6.44 (-3)	0.082	3.17 (-4)	0.085	7.40 (-4)	0.032	3.42 (-4) ⁺	0.012
TiH ₂ (8), Al (72)	1.07 (-2)	7.57 (-3)	0.103	5.58 (-4)	0.149	3.64 (-3)	0.155	1.92 (-3)	0.068
LH ₂ (80.1), Al (19.9)	1.00 (-2)	8.91 (-3)	0.096	3.40 (-5)	0.0091	4.69 (-5)	0.0020	3.70 (-5)	0.0013
Be (80)	1.40 (-2)	1.21 (-2)	0.134	3.28 (-5)	0.0088	3.07 (-4)	0.013	4.41 (-4)	0.016
H ₂ O (80)	8.58 (-3)	7.80 (-3)	0.082	5.29 (-6)	0.0014	7.28 (-6)	0.00031	5.99 (-6)	0.00021
Al (80)	8.42 (-3)	6.19 (-3)	0.081	7.77 (-4)	0.207	9.18 (-3)	0.393	4.82 (-3)	0.170

* See table 17 for more detailed description.
** This shield contains 20 times the amount of boron in the previously listed BeO shield.
+ Negative thermal flux values in the shield region were encountered for these cases; thus, these fluxes are unreliable.

2. NUCLEAR PARAMETERS

The nuclear parameters as used in the Shield Materials Optimization Code are tabulated in table 19. The table contains the parameters for the molecule or mixture rather than for each element. The parameters were determined as follows:

- ν_i = number of atoms of element
- ρ = density of the compounds or elements in grams per cc
- M = molecular weight of material
- Z = atomic number of element
- Σ_R/ρ = values of the macroscopic removal cross section divided by density, taken principally from NDA-2143-6⁽⁴⁶⁾. The cross section for the mixtures or molecules were calculated by the formula $\Sigma_R/\rho = .602 \sum_i (\nu_i \sigma_i)/M$
- μ/ρ = values for the gamma mass attenuation coefficient in cm^2/g , taken from XDC-59-10-19⁽⁴⁷⁾.

These data are obtained in a similar manner for the gamma rays generated within the material as well as the incident gamma rays from the reactor. A single energy of 2 Mev is assumed for the incident gamma rays since this is the point at which the mass attenuation coefficient is approximately minimum for heavy or high Z materials. The attenuation coefficient for the secondary gammas produced within the shield materials are based on the average energy given off in the capture. This again is a cross section weighted gamma energy yield and is obtained from the relation $\bar{E}_\gamma = \frac{\sum_i (\nu_i \sigma_{\alpha_i} E_{\gamma_i})}{\sum_i (\nu_i \sigma_{\alpha_i})}$

- Σ_s/ρ = Mass scattering cross section obtained principally from reference 9. The scattering cross section, which is necessary for determining the epithermal neutron age in each material, was taken for an energy range of about 100 Kev to 1 Mev where the cross section is essentially flat. The value for the molecule was determined in the same manner as the removal cross section.

TABLE 19
NUCLEAR PARAMETERS

Material	ρ_g/cc	Σ_r/ρ	$(\mu/\rho)_1^*$	$(\mu/\rho)_2$	$(\mu/\rho)_3$	Σ_s/ρ	Σ_a/ρ	E_γ	E/capt.
Al	2.7	0.029	0.0432	0.0282	0.084	0.03123	0.00479	5.0	7.72
Al B ₂	3.19	0.0401	0.0431	0.0271	0.084	0.183	0.00930	5.0	9.80
Al (OH) ₃	2.42	0.0562	0.0473	0.0408	0.08926	0.5571	0.00764	2.7	4.42
B	2.54	0.058	0.0430	0.0258	0.0842	0.2227	0.0078	5.0	0.382
B ₄ C	1.81	0.0514	0.04332	0.0261	0.0848	0.227	0.00613	5.0	11.4
Be	1.84	0.072	0.0349	0.0211	0.0773	0.40008	6.67-4	6.0	6.82
BeO	3.03	0.0495	0.0428	0.0238	0.0835	0.2332	2.45-4	5.94	6.76
BeH ₂	0.75	0.168	0.0554	0.0482	0.0947	2.499	0.0366	2.25	2.29
Be (OH) ₂	1.91	0.0707	0.0481	0.0454	0.0889	0.7441	0.00938	2.25	2.29
Be ₂ B	2.41	0.0647	0.0407	0.02159	0.07099	0.2978	0.00333	5.12	10.83
Be(BH ₄) ₂	0.604	0.0932	0.0476	0.0358	0.08777	0.7754	0.0175	2.48	3.12
CaH ₂	1.7	0.05119	0.0489	0.03249	0.09168	0.6119	0.01558	3.69	4.54
Ca(OH) ₂	2.34	0.04517	0.0470	0.03037	0.0896	0.4077	0.00895	3.69	4.54
H ₂	0.0706	0.600	0.0876	0.0826	0.173	11.9	0.198	2.23	2.23
Fe(OH) ₃	3.9	0.0478	0.0459	0.02885	0.0872	0.4579	0.0196	5.99	6.21
Li H	0.78	0.151	0.0506	0.04451	0.0876	1.608	0.0252	2.24	2.23
Li OH	1.43	0.0754	0.0465	0.04455	0.0872	0.6266	0.00837	2.24	2.23
Li NH ₂	1.18	0.107	0.0501	0.04132	0.0909	1.340	0.01947	2.73	3.15
Li(BH) ₄	0.66	0.165	0.049	0.03662	0.0896	1.079	0.0404	2.48	
Mg H ₂	1.4	0.0736	0.0504	0.04148	0.0926	0.9897	0.0165	2.44	2.88
Nylon-66	1.14	0.0953	0.0430	0.0433	0.0954	1.385	0.001	2.2	2.428
Pb	11.34	0.0103	0.0457	0.0451	0.145	0.0328	4.93-4	7.0	7.4
PbO	8.0	0.01219	0.0455	0.0435	0.1408	0.0464	4.59-4	7.99	7.39
TaB ₂	12.6	0.01524	0.043	0.0406	0.1233	0.1003	0.0641	5.0	6.1
Ti H ₂	3.76	0.0455	0.0476	0.03022	0.0897	0.6657	0.0538	6.49	7.39
WB ₂	13.1	0.0141	0.0436	0.0393	0.1207	0.0825	0.0566	5.0	6.87
YH ₂	3.958	0.0357	0.0444	0.0333	0.0862	0.4124	0.01245	4.83	4.62
Zr H ₂	5.61	0.0287	0.0432	0.0367	0.0855	0.3041	0.00513	3.23	3.42

$*(\mu/\rho)_1$ = for reactor gamma rays
 2 = for shield hard secondary gamma rays
 3 = for shield soft secondary gamma rays

$\bar{E}_{\gamma_i}$ = Average gamma energy release per capture in each element taken from references 10 and 11.

$\Sigma\alpha/\rho$ = Mass absorption cross section in cm^2/g determined for radiative capture. In other words, the cross section for the yield of capture gamma rays for each molecule is determined in the same manner as the removal cross section. These values are also taken from references 48 and 49.

The atomic weight of the moderator in each molecule is taken to be the atomic weight of the lightest atom, in most cases hydrogen. Where hydrogen is used as the moderator, in order to obtain a correct value for ξ from the code, an atomic weight of $4/3$ was used in the equation $\xi \approx \frac{6}{3A + 2}$. For all other moderators the atomic weight of the lightest atom such as lithium, beryllium, boron, or carbon is used. It is justifiable to use this single atom value since the heavier atoms contribute very little to the slowing down. The effect of the mixture on the slowing down is accounted for in the average scatter cross section. These values are required in the code for obtaining slowing down power or for obtaining the age (τ).

APPENDIX B

MATERIALS PROPERTIES

APPENDIX B

MATERIAL PROPERTIES

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Following is a compilation of the physical, chemical, and radiation resistance characteristics of most of the materials considered in this study. Detailed data are presented for 24 of the 25 materials considered in the nuclear analysis.

1. Aluminum Diboride (Al B_2)

a. Structural Characteristics

Aluminum diboride is a refractory material sometimes used in high-temperature structural applications. It is thought to lack sufficient thermal stress resistance to withstand the extreme temperature differential that will be encountered in the NERVA application. It would probably be used as hot-pressed pellets enclosed in capsules.

b. Chemical Characteristics

The density of aluminum diboride is 90 lb/cu ft. Its melting temperature is 3000°F (reference 12).

c. Thermal Characteristics

Thermal conductivity is estimated to be 10 Btu/hr, ft, °F; specific heat is estimated to be 0.15 Btu/lb, °F.

d. Radiation Damage Resistance

The radiation damage resistance characteristics of aluminum diboride are unknown.

e. Manufacturing and Cost

Although aluminum diboride is not an expensive raw material it is costly to form it into hot-pressed pellets and to encapsulate it into a shield element.

2. Aluminum Hydroxide (Al(OH)₃)

a. Structural Characteristics

Aluminum hydroxide occurs as a white crystalline powder or granular material. It lacks the necessary structural characteristics for use as a self-supporting shielding element⁽¹³⁾.

b. Chemical Characteristics

Aluminum hydroxide decomposes chemically at 300°C to give off water^(14, p. 215). The density of aluminum hydroxide is 2.4 gms/cc^(14, p. 1517).

c. Thermal Characteristics

Thermal conductivity of aluminum hydroxide is estimated to be on the order of 3 Btu/hr, ft, °F; the specific heat of aluminum hydroxide is 0.22 Btu/lb, °F⁽¹⁴⁾.

d. Radiation Damage Resistance

The expected radiation dosage for the NERVA application is approximately 2.5×10^9 rads in carbon. Disintegration starts at approximately 10^8 rads and is almost complete by the time a dose of 10^{10} rads is reached⁽¹⁵⁾. The radiation damage resistance is insufficient for NERVA application.

e. Manufacturing and Cost

Aluminum hydroxide is an inexpensive raw material, but since it must be encapsulated in small diameter cylinders it is expected to be costly in finished form ready for use in a shield. No toxicity hazards are associated with this material.

3. Beryllium

a. Structural Characteristics

Data on the structural characteristics of pure beryllium in the unirradiated condition are plentiful. Data on beryllium with boron additives in the unirradiated and irradiated condition are scarcer.

The ultimate tensile strength of beryllium seems adequate for use as a self-supporting structural element in the shield. Values for the ultimate tensile strength for unirradiated pure beryllium between 60°R and 530°R vary between 29,000 and 55,000 psi depending upon the method of fabrication and the impurity content (16, p. 2.1.6.211; 10, p. 179). Beryllium containing 1 weight percent dispersed boron in the unirradiated condition for one group of 18 samples averaged about 40,000 psi. Samples of the same material irradiated to 2.6×10^{19} nvt at room temperature had an ultimate tensile strength averaging 33,000 psi (17, p. 121 and 125).

Irradiated beryllium is essentially brittle. Pure beryllium at room temperature in the unirradiated condition has elongation to rupture between 1 and 4 percent (16, p. 2.1.6.214). Beryllium containing 1 weight percent boron at room temperature and irradiated to 2.6×10^{19} nvt had elongation to rupture less than 1 percent. Most of the samples tested had considerably lower ductility, and almost half of them failed before the 0.2 percent elongation was reached (17, p. 125).

The modulus of elasticity of pure beryllium in the unirradiated condition between 60°R and 460°R varies from 33,000,000 to 47,000,000 psi (16, p. 2.1.6.51 and 10, pp. 205 and 206).

Poisson's ratio for pure beryllium is quite low. Measurements of unirradiated material in the temperature range from -250°C to room temperature gave values of Poisson's ratio ranging between 0.01 and 0.05 (10, p. 179).

Consideration of the above data has led to the selection of structural characteristics for borated and irradiated beryllium (10^{19} nvt) at room temperature as follows:

<u>Ultimate Tensile Strength</u>	<u>Elongation to Rupture</u>	<u>Modulus of Elasticity</u>	<u>Poisson's Ratio</u>
35,000 psi	0.2%	45×10^6 psi	0.03

b. Chemical Characteristics

The basic compatibility of beryllium with the fluids and structural materials used in these reactors has been established by extensive testing under actual conditions in Nevada. The compatibility of borated beryllium at operating and higher temperatures can be evaluated from the data given in the following paragraphs.

Reactions of pure beryllium with hydrogen, nitrogen, and CO_2 are negligible below 1570°R . (10, pp. 239 to 242). Reactions of beryllium plus 1 wt. percent dispersed boron with air are negligible below 1400°R (11, p. 66).

Compatibility of the borated beryllium with structural materials such as aluminum, steel, titanium, etc. is good at moderate temperatures. At high temperatures (2350°R), the borated beryllium has severe reactions with materials such as nichrome, hastelloy, and Inconel (11, p. 59). Beryllium has no phase changes below 2290°R . (10, p. 167).

c. Thermal Characteristics

The thermal conductivity of unirradiated beryllium and that of unirradiated beryllium containing 1 wt. percent boron are almost identical. Typical values as a function of temperature are:

<u>Room Temperature</u>	<u>260°R</u>	<u>60°R</u>
K, (Btu/hr, ft, $^\circ\text{F}$)	120	100

The above data are contained in reference 16, page 2.1.5.32 and reference 11, page 58.

Irradiation of borated beryllium to 10^{19} nvt probably will cause substantial reduction in the thermal conductivity particularly at low temperature. This is expected because most high conductivity metallic materials have a basic thermal resistance that is low compared to the defect resistance caused by irradiation to high nvt. In lieu of actual experimental data on the thermal conductivity of irradiated beryllium, it will be assumed for this study that in the

temperature range 60°R to 530°R that the thermal conductivity is 50 Btu/hr, ft, °F.

Data on the coefficient of linear thermal expansion for beryllium containing 1 weight percent boron is given in reference 11, page 57. The value given for room temperature is $2 \times 10^{-6}(\text{°F})^{-1}$. The coefficient of thermal expansion decreases with decreasing temperature. Consideration of these data led to the selection of a mean coefficient of thermal expansion for cryogenic temperatures of about $1 \times 10^{-6}(\text{°F})^{-1}$.

The density of pure beryllium at 250°R is approximately 116 lb/ft³. (16, p. 2.1.5.143). The specific heat of pure beryllium at 250°R is approximately 0.12 Btu/lb, °F. (16, p. 2.1.5.31).

d. Radiation Damage Resistance

Dimensional changes in both borated and unborated beryllium are slight for the radiation doses that will be encountered in the NERVA reactors. Irradiation of samples of pure beryllium to a dosage of 7×10^{21} nvt at 70°C caused a swelling less than 0.1 percent in the samples (10, p. 347). Irradiation of samples of beryllium plus 1 weight percent boron dispersion caused increases in dimensions of less than 0.1 percent. These samples were irradiated to 10^{18} nvt (11, p.70).

Metallographic examination of samples of beryllium plus 1 weight percent boron irradiated to 2×10^{19} nvt showed very slight changes in the microstructure of the material (11, p. 71).

e. Manufacturing and Cost

Beryllium is costly; the cost of borated beryllium in small lots has been estimated at \$500 per pound. Perhaps for the large lots that would be used in a NERVA flight shield, the price would come down to about \$100 per pound. The cost of unborated beryllium ranges from \$60 to \$100 per pound, depending on the specifications, size, and shape of the material. It is believed that material in the size, shape, and quality for shield elements would cost approximately \$65 per pound. The toxic hazards of beryllium must be controlled during its manufacture; however, these problems have been already solved.

4. Beryllium Hydride (Be H₂)

a. Structural Characteristics

Beryllium hydride occurs as a white, crystalline material⁽¹⁸⁾ and it would require encapsulation for support.

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b. Chemical Characteristics

Beryllium hydride is a non-volatile solid^(19, p.50) that decomposes at 125°C. It also decomposes in contact with water^(18, p. B-156) and liberates hydrogen. It does not react appreciably with dry air at room temperature^(19, p. 50).

c. Thermal Characteristics

The thermal conductivity is estimated to be 10 Btu/hr, ft, °F; specific heat is estimated to be 0.2 Btu/lb, °F.

d. Radiation Damage Resistance

The radiation damage resistance of beryllium hydride is unknown.

e. Manufacturing and Cost

Presumably, beryllium hydride is costly raw material and in addition would require encapsulation. Thus it is expected that the cost of finished inserts made of this material would be quite expensive.

Because beryllium hydride acts as a local irritant on the skin^(20, p. 502), inhalation or ingestion may cause permanent injury or death, special manufacturing precautions must be observed.

5. Beryllium Hydroxide (Be (OH)₂)

a. Structural Characteristics

Beryllium hydroxide occurs in crystalline or amorphous powder form^(20, p. 503) or as a gelatinous material^(21, p. 12). Thus it cannot be used as a self-supporting structural element.

b. Chemical Characteristics

Beryllium hydroxide decomposes when heated to 138°C^(20, p. 504). Its density is 2.0 gm/cm³.

c. Thermal Characteristics

Thermal conductivity was estimated to be 3 Btu/hr, ft, °F and specific heat to be 0.3 Btu/lb, °F.

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d. Radiation Damage Resistance

The radiation damage resistance of beryllium hydroxide is unknown.

e. Manufacturing and Cost

Presumably, the cost of finished shield inserts containing beryllium hydroxide would be high since it contains beryllium, an expensive raw material, and requires expensive encapsulation. Special precautions are needed in manufacturing since this material acts as a local irritant and inhalation or ingestion may cause permanent injury or death^(20, p.502).

6. Beryllium Oxide (BeO)

a. Structural Characteristics

There is a considerable body of data concerning the modulus of rupture and ultimate tensile strength of beryllium oxide with and without additives, in the irradiated and unirradiated conditions, and covering a variety of temperatures. In all of these data, the effect of grain size is apparent. Fine-grain, dense specimens are stronger and more resistant to radiation damage than coarse-grained material. The modulus of rupture of pure, unirradiated beryllium oxide at room temperature ranges from 14,000 to 28,000 psi. The rather large spread in these data occurs because the specimens tested had widely varying grain size. Data for dense, fine-grained material are limited to the range of 20,000 to 28,000 psi. (These data are contained in reference 16, page 41.1.6.211; reference 10, page 66; and reference 22, page 843.) The room temperature modulus of rupture of unirradiated BeO containing 3 weight percent boron-carbide is 16,000 psi. Unirradiated BeO containing 5.5 weight percent zirconium-boride had a modulus of rupture between room temperature and 500°F of 28,000 psi (11, p.44; 23 p.13). BeO containing 3 weight percent boron-carbide and irradiated to a total nvt of 4×10^{19} had a modulus of rupture at room temperature of 15,600 lb/in². This was 97 percent of the values for samples of the same material in the unirradiated state^(23 p.13). Consideration of the above data leads to the conclusion that a conservative value for the modulus of rupture of BeO containing three weight percent boron-carbide additive and irradiated to the total NR-1 integrated dose is 15,000 psi.

Data contained in reference 16, page 41.1.6.221 show that the compressive strength of pure beryllium oxide at room temperature in the unirradiated condition is in the neighborhood of 175,000 psi. This is so far above the tensile strength that determines the modulus of rupture that it will never be closely approached. Compressive data for irradiated material with additives therefore is of little interest.

The modulus of elasticity of pure, unirradiated BeO is in the neighborhood of 5×10^7 psi^(10, p.69; 16, p.41.1.6.51; & 22, p.843). Irradiation has little or no effect on the modulus of elasticity until integrated doses causing micro-cracking of the material are reached^(24, p.81 & 82).

Beryllium oxide is a brittle material. Failure occurs by brittle fracture below 1000°C. Typical strain to fracture is about 0.06 percent at room temperature^(10, p.66; 24, p.351).

b. Chemical Characteristics

The chemical reactivity of beryllium oxide with hydrogen, nitrogen, dry air, helium, aluminum, stainless steel, and other structural materials is negligible throughout the entire temperature range in which the beryllia can be used as a self-supporting structural material^(10, p.63; 11, p.42; 25, p.333).

Tests have been made of the boron retention of beryllium oxide containing B₄C and other boron additives. Samples tested in air up to 1600°F retained essentially all of the boron additives^(11, p.45).

Beryllia reacts with water vapor in air at elevated temperatures. This reaction is not important at temperatures well below 1000°C^(10, p.55). The melting point of pure beryllia is 2570°C^(10, p.52). The vapor pressure of pure beryllium oxide is low. Reference 10 page 52 gives a formula for calculation of vapor pressure over the temperature range 1977°C to 2140°C. At 2000°C, the vapor pressure is 6 microns Hg and is decreasing at a high exponential rate with decreasing temperature.

c. Thermal Characteristics

Data on the thermal conductivity of pure beryllium oxide in the unirradiated condition as a function of temperature may be found in reference 11, page 41; reference 16, page 41.1.5.32; reference 26, page 1029; reference 10, page 57; and reference 27, Chapter 1.3. Thermal conductivity data for borated beryllium oxide as a function of

temperature in the unirradiated condition may be found in reference 11, page 41. Thermal conductivity of irradiated beryllium oxide at low and moderate temperatures is given in reference 24, page 208 and reference 28, page 175. Consideration of all of the above data led to the selection in reference 29 of estimated thermal conductivity values for borated beryllium oxide irradiated to the total integrated dose for NR-1. Some values from the curve in this memo are given below:

<u>Temperature ($^{\circ}\text{R}$)</u>	<u>100</u>	<u>200</u>	<u>460</u>
$k(\text{Btu/hr, ft, } ^{\circ}\text{F})$	30	55	70

Data on the thermal expansion coefficient of pure beryllium oxide in the unirradiated condition are given in reference 16, page 41. 1.5.33. The average value between 260°R and 460°R is approximately $2 \times 10^{-6} \text{ }^{\circ}\text{F}^{-1}$. Expansion coefficients for unirradiated beryllium oxide containing 5.5 weight percent zirconium boride are given as a function of temperature in reference 11, page 41. The average value between 260°R and 660°R is $2.5 \times 10^{-6} \text{ }^{\circ}\text{F}^{-1}$. It is assumed in the absence of data that irradiation does not produce any significant change in the thermal expansion coefficient of beryllium oxide.

In the light of the above data and assumptions, the average coefficient of thermal expansion between 260°R and 460°R for irradiated beryllium oxide with boron additives is approximately $2.5 \times 10^{-6} \text{ }^{\circ}\text{F}^{-1}$. The corresponding value for the single temperature 200°R is estimated to be $2 \times 10^{-6} \text{ }^{\circ}\text{F}^{-1}$. The density of hot pressed beryllia is 2.9 to 3.04 gm/cc^(11, p.40). The specific heat of hot pressed beryllia is approximately 0.08 Btu/lb, $^{\circ}\text{F}$ at 250°R ^(10, p.53) and increases to 0.24 at room temperature.

d. Radiation Damage Resistance

As has been mentioned previously, the radiation damage resistance of beryllium oxide is dependent upon its grain size. Dense, fine-grained material has much higher radiation damage resistance than the coarse-grained material. Data are available on the effects of radiation on dense, fine-grained material with and without boron carbide additives. Samples irradiated to total nvt between 3 and 5×10^{19} at elevated temperature (800 to 1000°C)

showed small changes in dimensions, modulus of rupture, modulus of elasticity, and micro-structure^(17, p.120; 23, p.11; 30, p.33-35; 31).

The thermal conductivity of beryllium oxide is seriously affected by irradiation to a total nvt of 10^{20} . It is shown in reference 29 that at 200°R , the thermal conductivity of irradiated material (10^{20} nvt) is approximately one-quarter of the thermal conductivity of the unirradiated material.

Doses significantly in excess of 10^{19} can produce substantial changes in the structural properties of beryllium oxide. Doses in excess of 2×10^{20} nvt at 100°C in pure beryllium oxide cause micro-cracking, which increases with irradiation. Material irradiated to 7×10^{20} nvt showed 2 to 3 percent volume changes and approximately 90 percent decrease in the modulus of rupture^(24, p.77; 31). Further irradiation to 1.5×10^{21} nvt at 100°C caused extensive powdering of test specimens^(24, p.87).

e. Manufacturing and Cost

Pure beryllium oxide rods of the dimensions suitable for use as inserts in the reference shield concept were quoted at \$30 per pound in 2000 pound lots. The cost of beryllium oxide rods with boron-carbide additive in the same dimensions is estimated to be between \$35 and \$45 per pound. In manufacturing beryllium oxide, toxic hazards must be controlled. Techniques for doing this have been fully developed.

7. Boron Carbide (B_4C)

a. Structural Characteristics

Because of its relatively low resistance to thermal stress and the large internal temperature difference that would be experienced it was felt necessary to consider boron carbide as hot-pressed, solid pellets enclosed in a steel tube. In this way, lack of adequate structural characteristics is not a deterrent to the use of boron carbide.

b. Chemical Characteristics

The density of boron carbide is 2.51 gm/cm^3 (theoretical density). The melting point is 2450°C ^(32, p.1057, 1058).

c. Thermal Characteristics

The thermal conductivity of boron carbide at room temperature is 15.7 Btu/hr, ft, °F. The specific heat is 0.22 Btu/lb, °F. The coefficient of thermal expansion is $2.5 \times 10^{-6} (\text{°F})^{-1}$. This is the average for the range 20 to 800°C (32, p.1057, 1058).

d. Radiation Damage Resistance

The radiation damage resistance of boron carbide is considered satisfactory for this application. Irradiation of crystalline boron carbide to a total dose of 3×10^{20} nvt resulted in volumetric expansion less than 1.4 percent. Most of this radiation induced change can be removed by annealing in the 700 to 900°C temperature range (32, p. 1104, 1105).

The helium produced by the n-α reaction is not expected to result in excessively high pressures within the capsule even if it is all released from the solid boron carbide.

e. Manufacturing and Cost

Two factors tend to make the cost of a boron carbide shield high. First, it is necessary that the boron carbide be used in a tubular support to obtain the necessary structural characteristics. Second, hot-pressing of boron carbide is an expensive operation. It is considered necessary to evaluate in the hot-pressed form so that the thermal conductivity would be high enough to permit the use of reasonable sized capsules.

8. Calcium Hydride (CaH₂)

a. Structural Characteristics

Calcium hydride cannot be used as a self-supporting structural element. It occurs as a powder or in weak, brittle, white, rhombic crystals (18, p.B 161; 19, p.567).

b. Chemical Characteristics

The density of hot pressed material is approximately 1.6 gm/cm³ (11, p.183). The melting point of calcium hydride is 1990°R if it is maintained in a hydrogen atmosphere (18, p. B-161). In an atmosphere other than hydrogen, chemical reactions and decomposition take place at lower temperatures. Calcium hydride decomposes in air at about 1670°R (18, p.B-161). In a nitrogen atmosphere,

calcium hydride reacts chemically with the nitrogen when heated above approximately 1210°R^(33, p.110). Calcium hydride decomposes under atmospheric pressure at approximately 1250°F^(11, p.184).

If ignited, calcium hydride will burn in air or oxygen^(19, p.42). It is hygroscopic readily reacting with moisture in the air to yield calcium hydroxide which is highly irritating to the skin by caustic action^(19, p.882). When heated or in contact with acids, an exothermic reaction evolving hydrogen may occur. Often enough heat is involved to ignite the calcium hydride. Powdered calcium hydride may form dust clouds that are explosive in contact with flames, sparks, heat, or oxidizers^(32, p.882).

Calcium hydride is compatible with series 300 and 400 stainless steels below 1500°F^(11, p.186).

c. Thermal Characteristics

The thermal conductivity of hot pressed calcium hydride is 6.0 Btu/hr, ft, °F^(11, p.185). The coefficient of linear expansion of cold pressed material at room temperature is $11 \times 10^{-6} \text{ } ^\circ\text{F}^{-1}$ (11, p.185).

d. Radiation Damage Resistance

Hot pressed samples irradiated to 6.6×10^{17} nvt showed negligible damage^(11, p.187).

e. Manufacturing and Cost

It is expected that the cost of the calcium hydride shield would be high since encapsulation and assembly in inert atmosphere is needed. In addition, calcium hydride is a hazardous material. It is toxic, flammable, and explosive in powder form.

9. Calcium Hydroxide (Ca(OH)₂)

a. Structural Characteristics

Calcium hydroxide cannot be used as a self-supporting structural element since it occurs in the form of powder or weak, brittle crystals^(18, p.B-161).

b. Chemical and Physical Characteristics

The melting point of calcium hydroxide is 580°C. At the melting point it decomposes

giving off water^(18, p.B-161). It is also hygroscopic.

The theoretical density of calcium hydroxide is 2.3 gm/cm^3 .

c. Thermal Characteristics

Experimental values of the thermal conductivity were not available. The conductivity was estimated to be on the order of $1 \text{ Btu/hr, ft, } ^\circ\text{F}$. The specific heat of calcium hydroxide is $0.26 \text{ Btu/lb, } ^\circ\text{F}$ ^(14, p.1521).

d. Radiation Damage Resistance

The radiation damage resistance of calcium hydroxide is unknown.

e. Manufacturing and Cost

The raw materials are inexpensive, but costly encapsulation is required. It is expected that the overall cost of fabricating inserts would be high. Special handling is required due to the fact that calcium hydroxide is caustic and is irritating to the skin and respiratory system^(19, p.567).

10. Ferric Hydroxide ($\text{Fe}(\text{OH})_3$)

a. Structural Characteristics

Ferric hydroxide is a brown, flocculent precipitant, which lacks the structural characteristics required for use as a self-supporting shield element⁽¹³⁾.

b. Chemical Characteristics

Ferric hydroxide decomposes chemically at 500°C ^(34, p.62). The density of the oxide varies from 3.4 to 3.9 gms/cm^3 depending upon its void fraction⁽¹³⁾. It is soluble in acids and insoluble in water, alcohol, and ether.

c. Thermal Characteristics

Thermal conductivity of ferric hydroxide is estimated to be $1 \text{ Btu/hr, ft, } ^\circ\text{F}$ and its specific heat is estimated to be $0.2 \text{ Btu/lb, } ^\circ\text{F}$.

d. Radiation Damage Resistance

The radiation damage resistance of ferric hydroxide is unknown.

e. Manufacturing and Cost

Presumably the cost of finished shield inserts containing ferric hydroxide would be fairly high since they require expensive encapsulation. No special precautions are needed in manufacturing this material.

11. Hydrogen - Steel

a. Structural Characteristics

The hydrogen will be continuously flowing through borated steel honeycomb or a nest of borated steel tubes. The structural characteristics of borated steel are the pertinent ones in this situation. Austenitic stainless steel containing 1 percent boron by weight have the typical as-irradiated (10^{19} nvt) properties listed below^(16, p. 25.3.6.212-214).

Tensile yield strength	45,000 psi
Elongation at rupture	5%
Modulus of elasticity	30×10^6 psi

These structural characteristics are adequate for this application.

b. Chemical and Physical Characteristics

Borated, austenitic stainless steel is compatible with hydrogen, nitrogen, helium, and air within the temperature ranges that would be encountered at the shield location. Hydrogen density is very strongly dependent on temperature. To maintain the density high enough to obtain effective moderation and attenuation of neutrons, it has been estimated that approximately 30 lb/sec flow through the shield will be required.

c. Thermal Characteristics

Thermal conductivity of austenitic stainless steel is between 8 and 9 Btu/hr, ft, °F. The coefficient of thermal expansion is between 9×10^{-6} and $10 \times 10^{-6} (^\circ\text{F})^{-1}$. Specific heat is approximately 0.12 Btu/lb, °F. Since the steel would be used in sheet form (0.015 to 0.025 inches thick) as tube or honeycomb walls, no serious thermal stress problems are anticipated.

d. Radiation Damage Resistance

Borated austenitic stainless steels have been tested to total dosages in excess of 10^{20} nvt and have demonstrated satisfactory radiation damage resistance. The major effect of irradiation is to reduce the ductility of the steel to the value listed under section d (14 p 25, 3, 6, 212-214).

e. Manufacturing and Cost

It is necessary to make only a borated stainless steel structure for hydrogen to flow through. The cost of this structure is expected to be less than \$150,000.

12. Lead Oxide (PbO)

a. Structural Characteristics

Lead oxide lacks the necessary structural characteristics for use as a self-supporting shield element. It is assumed that it would be used as a packed powder enclosed in capsules.

b. Chemical Characteristics

The density of lead oxide is 9.5 gms/cm^3 . The melting temperature is 1600°F .

c. Thermal Characteristics

Thermal conductivity of lead oxide is estimated to be $1 \text{ Btu/hr, ft, }^\circ\text{F}$; specific heat is estimated to be $0.1 \text{ Btu/lb, }^\circ\text{F}$.

d. Radiation Damage Resistance

The radiation damage resistance characteristics of lead oxide is not known.

e. Manufacturing and Cost

The cost of finished shield elements filled with lead oxide is expected to be moderately expensive since encapsulation is costly.

13. Lithium Amide (LiNH₂)

a. Structural Characteristics

Lithium amide occurs as a white crystalline solid or powder.⁽¹³⁾ It does not possess structural characteristics needed for use as a self-supporting element.

b. Chemical Characteristics

The melting point of lithium amide is approximately 400°C . It decomposes in water and is hygroscopic. It reacts with water or steam to produce toxic and flammable vapors. On contact with oxidizing materials it can react vigorously, and on contact with acid or acid fumes it can evolve a great deal of heat. On the addition of heat lithium amide decomposes chemically; the temperature of chemical decomposition is unknown. This information is taken from reference 19, pages 125 and 126, reference 20, and reference 13. The density of lithium amide is approximately 1.18 gm/cm^3 at room temperature.

c. Thermal Characteristics

Thermal conductivity of lithium amide is estimated to be $0.1 \text{ Btu/hr, ft, }^\circ\text{F}$ and the

specific heat is estimated to be 0.2 Btu/lb, °F. No thermal stress problems are expected with lithium amide since it would be used as an encapsulated powder.

d. Radiation Damage Resistance

The radiation damage resistance of lithium amide is unknown.

e. Manufacturing and Cost

Lithium amide is an inexpensive raw material but encapsulation will be costly.

14. Lithium Hydride (LiH)

The experience gained in the design of the B-5 shield answers most of the questions regarding lithium hydride in a flight shield. The B-5 shield was made of lithium hydride pellets encapsulated in stainless steel tubes. These stainless steel tubes were arranged in a triangular array with coolant flowing through the void area between the tubes. Tubes were supported by a structure at each end of the tubular capsule. The analysis, design, fabrication, and irradiation testing of the B-5 shield are covered in references 8, 35, and 36. For ease of comparison of shielding materials in this study, a summary of the characteristics of lithium hydride as a shielding material is presented below. These characteristics were obtained from the work done on the B-5 shield and in the ANP and SNAP projects.

a. Structural Characteristics

Because of its very low tensile strength and its brittleness, lithium hydride is unsuitable for use as a self-supporting structural element in the shield. Reference 11, page 97, gives the modulus of rupture of cold-pressed lithium hydride as 1800 psi. The modulus of elasticity is 3.7×10^6 psi^(11, p.98).

b. Chemical and Physical Characteristics

Lithium hydride is hygroscopic and must be protected from water vapor to prevent swelling. This characteristic combined with its lack of suitable structural properties dictates that lithium hydride must be encased in a capsule. The compatibility of lithium hydride with stainless steel is satisfactory throughout the temperature range of interest^(11, p.92-97). The melting point is 1730 °R^(11, p.75; 8).

c. Thermal Characteristics

The thermal conductivity of cold-pressed lithium hydride in the temperature range 200°R to 530°R is between 4 and 5 Btu/hr, ft, $^{\circ}\text{F}$ (16, p.40.1.5.32). These figures are for unirradiated lithium hydride. In view of its rather high basic thermal resistance, irradiation probably will not significantly reduce the thermal conductivity of lithium hydride unless microcracking, fracturing, or powdering occurs. One or more of these conditions probably would occur in shield capsules because of the extremely poor resistance to thermal stress fracture. Taking these factors into account, an effective thermal conductivity of lithium hydride pellets was chosen to be 1.0 Btu/hr, ft, $^{\circ}\text{F}$.

The mean coefficient of thermal expansion from 100°R to room temperature is $10 \times 10^{-6} \text{ }^{\circ}\text{F}^{-1}$ (16, p. 40.1.5.33). For the range of 100°R , to 1260°R , the mean coefficient of thermal expansion is $25 \times 10^{-6} \text{ }^{\circ}\text{F}^{-1}$. The solid material has a theoretical density of 48.3 lb/cu ft. Cold pressed pellets achieve 95 percent of theoretical density with reasonable pressing pressures (11, p. 80). The specific heat of lithium hydride at room temperature is approximately 0.65 Btu/lb, $^{\circ}\text{F}$ (16, p. 40.1.5.31).

d. Radiation Stability

Shield capsules for the B-5 shield were irradiated at 250°C to a total nvt of 3.5×10^{16} . This combination of irradiation and temperature was sufficient to cause 3.5 percent swelling in a diametral direction (36, p.12). This swelling was enough to cause noticeable permanent deformation of the cladding tube. The outline of the lithium hydride pellets could be plainly discerned through the clad. In NERVA flight shield, the expected maximum irradiation would be almost 300 times as much as the total received during this test. In addition, the temperatures at the center of the lithium hydride pellets would be significantly higher than the temperatures experienced in this test.

During the ANP project materials development, 25 specimens of cast and cold pressed LiH were irradiated up to 2×10^{19} nvt at temperatures ranging from 200° to 1000°F . The results apparently indicated satisfactory radiation damage characteristics for dosage up to 10^{18} nvt. Results for higher dosages appeared inconclusive (11, p.111-114).

These two widely differing results show how important the details of impurities, processing, etc., can be in determining radiation damage resistance. No way of testing for radiation damage resistance of specific batches of LiH is known except for actual irradiation.

It is thought that a satisfactorily reliable shield of LiH can be made, but it is expected to require considerable development.

e. Manufacturing and Cost

Raw materials cost for a lithium hydride encapsulated shield is low but the fabrication cost is extremely high. The cost of 5/8 inch diameter capsules 9 inches in length has been estimated, based upon the B-5 shield capsule costs and including the incorporation of manufacturing improvements for lower cost. The estimated cost per capsule is approximately \$40. On a weight basis, these capsules cost approximately \$100 per pound.

Lithium hydride is toxic, flammable, and hygroscopic. The manufacturing problems related to these characteristics have been solved but are reflected in the cost of fabrication.

15. Lithium Hydroxide (LiOH)

a. Structural Characteristics

Lithium Hydroxide occurs as colorless brittle crystals, which do not have the structural characteristics necessary for use as a self-supporting shield element.⁽¹³⁾ In the NR-1 applications, it is considered that the lithium hydroxide would be contained in metallic capsules.

b. Chemical Characteristics

The density of lithium hydroxide is 2.54 g/cm³. The melting point is 462°C.⁽¹³⁾ Lithium hydroxide is not chemically compatible with aluminum but can be contained in stainless steel capsules.

c. Thermal Characteristics

The thermal conductivity of lithium hydroxide estimated to be on the order of 1 Btu/hr, ft, °F. The specific heat of lithium hydroxide is 0.36 Btu/lb, °F^(14, p.1529).

d. Radiation Damage Resistance

The radiation damage resistance characteristics of lithium hydroxide are unknown.

e. Manufacturing and Cost

Lithium hydroxide is a strong base and makes a caustic solution in water. Large doses of lithium compounds can cause dizziness and prostration. These factors must be taken into account in the manufacturing of shields from this material⁽²⁰⁾. Lithium hydroxide is not an expensive raw material but requires encapsulation and is expected to be fairly expensive in finished form.

16. Lithium Tetrahydroborane (Li (BH)₄)

a. Structural Characteristics

Lithium tetrahydroborane cannot be used as a self-supporting element in a shield since it occurs as white, greasy, orthorhombic crystals.⁽³⁷⁾

b. Chemical and Physical Characteristics

This compound has a molecular weight of 21.8 and a specific gravity of 0.67. It melts at 280°C. It is hygroscopic, reacting with water vapor to form lithium borate and hydrogen. This evolved hydrogen could cause problems. It is a strong base and reacts markedly with aluminum at moderate temperatures.

c. Thermal Characteristics

Thermal conductivity is estimated to be 1 Btu/hr, ft, °F and specific heat is estimated to be 0.3 Btu/lb, °F.

d. Radiation Damage Resistance

The radiation damage resistance of lithium tetrahydroborane is unknown.

e. Manufacturing and Cost

This material is expected to be costly in fabricated form since it will require encapsulation. Extreme manufacturing precautions would be required since it is a very lethal material.

17. Nylon 6-6

a. Structural Characteristics

The tensile yield strength of nylon is 9,000 psi at room temperature. The modulus of elasticity ranges between 4×10^4 psi and 4×10^5 psi. The elongation to rupture at room temperature is 50 percent.^(32, p.1090) Because of rapid degradation of the structural properties of nylon with increasing temperature it is necessary to encapsulate the nylon to obtain the desired structural reliability.

b. Chemical Characteristics

The density of nylon is 1.13 gm/cm^3 . The maximum temperature recommended for continuous service is 300°F . The melting point of nylon is 420°F . Nylon begins to boil at 690°F .^(37, 32, p.1090)

c. Thermal Characteristics

The thermal conductivity at room temperature is $0.23 \text{ Btu/hr, ft, }^\circ\text{F}$. The specific heat is $0.4 \text{ Btu/lb, }^\circ\text{F}$. The coefficient of thermal expansion is $6.6 \times 10^{-5} (^\circ\text{F})^{-1}$.

d. Radiation Damage Resistance

Neutron irradiation has little effect on nylon. In the presence of gamma radiation, nylon begins to off-gas at $10^{10} \text{ erg/gm, carbon}$.⁽³⁸⁾ The expected gamma dose at the shield is approximately $2.5 \times 10^{11} \text{ ergs/gm, carbon}$; thus the radiation damage resistance of nylon appears to be inadequate for this application.

e. Manufacturing and Cost

Nylon would be encapsulated in quite small diameter tubes because of its low thermal conductivity. It is therefore believed that a nylon shield would be quite costly to fabricate.

18. Tantalum Boride (TaB_2)

a. Structural Characteristics

Tantalum boride is a refractory material that is used in some high-temperature structural applications. It is thought to lack sufficient thermal stress resistance for this application and would require handling as encapsulated pellets.

b. Chemical Characteristics

Tantalum boride has a density of 12.6 gms/cm^3 . Its melting temperature is 5800°F .^(12 p.64)

c. Thermal Characteristics

Thermal conductivity of tantalum boride is estimated to be $3 \text{ Btu/hr, ft, }^\circ\text{F}$ and specific heat is estimated to be $0.15 \text{ Btu/lb, }^\circ\text{F}$.

d. Radiation Damage Resistance

The radiation damage resistance characteristics of tantalum boride are not known.

e. Manufacturing and Cost

Tantalum boride is not a particularly expensive raw material but it would be costly to form in pellets and encapsulate.

19. Titanium Hydride (TiH_x)

a. Structural Characteristics

The strength of titanium hydride is adequate for use as a structural element. Reference 31, page 240, gives the room temperature modulus of rupture of unirradiated titanium hydride plus 3 weight percent boron carbide as 19,000 psi. The corresponding figure for titanium hydride with 5 weight percent titanium boride is given as 18,000 psi. Titanium hydride is brittle at or below room temperature. Data on tensile tests of hydrided titanium at room temperature are given in reference 39, page 147. The room temperature modulus of elasticity of two specimens of titanium hydride containing approximately 3.1 weight percent hydrogen is given in reference 39, page 143. Numerical data on these items is **secret** and cannot be given in this report.

b. Chemical Characteristics

Titanium hydride is chemically stable with aluminum at room temperature. There is no reason to suspect reactions with it, or other structural materials, in the temperature range of interest. The hydrogen retention and oxidation resistance in air of titanium hydride is good up to 1,000 °F. (40, p.2; 31, p.238). There are no significant chemical reactions in the temperature range of interest between titanium hydride and nitrogen or helium.

c. Thermal Characteristics

The thermal conductivity of unirradiated titanium hydride is approximately 12 to 14 Btu/hr, ft, °F. (41)

The specific heat of titanium hydride was estimated to be approximately 0.15 Btu/lb, °F at room temperature, based on data given in reference 31, p.239. The density of hydrided material containing 3.9 weight percent hydrogen is 3.8 gm/cm³. (40) The room temperature coefficient of thermal expansion for titanium hydride is approximately 2.8×10^{-6} °F⁻¹ (40, p. 2).

d. Radiation Damage Resistance

Specimens of titanium hydride with additives of boron carbide and titanium boride were irradiated to a total integrated flux of 2.5×10^{19} nvt. Examination of the specimens indicated that they were essentially unchanged in dimension or appearance by irradiation. (31, p.239)

e. Manufacturing and Cost

Crack-free massive titanium hydride can be made. Careful control of processing variables and additives is needed. (31, p.234; 23, p.12) This situation is similar to that pertaining to zirconium hydride processing, and probably will lead to approximately the same costs. The cost of titanium hydride in approximately the size and shape needed for the flight shield was estimated at \$60 per pound. (42)

20. Titanium Hydride - Boron Carbide - Aluminum Composites

This material is not a part of present technology. It is thought that a desirable combination of structural, thermal, and radiation attenuating characteristics could be obtained in such a composite material. These materials can be obtained in powder form, mixed, and extruded. Only one sample of this material has been made.

a. Structural Characteristics

A tensile test of the single sample has been carried out. The results are as follows: Tensile yield strength is 28,000 psi; ultimate tensile strength is 30,000 psi; elongation to rupture is 2.9 percent and modulus of elasticity is 10×10^6 psi. The specimen used for this tensile test was a mixture consisting of 10 percent by volume titanium hydride, 1 percent by volume boron carbide, and the remainder aluminum. The sample was tensile tested as extruded. No annealing was done. Examination of microphotographs of the sample shows that the microstructure appears to be sound and dense (figures 38 and 39).

b. Chemical Characteristics

Titanium hydride, boron carbide, and aluminum are chemically compatible within the applicable range of temperatures. The temperature limit on this material used as a self-supporting structural member is set by the loss of strength of aluminum at approximately 800°F. The hydrogen retention of titanium hydride is not limiting since it is good up to 1,000°F. (40, p.2; 31, p.238)

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Figure 38. Longitudinal Section of Al-TiH₂-B₄C Material (100 x)

Figure 39. Longitudinal Section of Al-TiH₂-B₄C Material (450 x)

The density of the mixture is approximately 236 lb/cu ft.

c. Thermal Characteristics

No experimental values are available for the thermal conductivity of this material. However, thermal conductivities of the component ingredients are known and the thermal conductivity of the composite has been estimated to be approximately 80 Btu/hr, ft, °F. The specific heat of the composite material has been estimated from the weight proportions and specific heats of the components to be 0.22 Btu/lb, °F. The coefficient of thermal expansion of this material is not known. It is presumed to be that of aluminum, $8 \times 10^{-6} (\text{°F})^{-1}$.

d. Radiation Damage Resistance

The radiation damage resistance of the composite mixture is not known experimentally. The radiation damage resistance of each of the individual components considered separately are known to be adequate. Boron carbide and aluminum mixed in the form of Boral have been subjected to total doses in excess of 10^{20} nvt without apparent change in the strength or the dimensions of the material. There is no reason to suspect that the addition of titanium hydride in mechanical mixture with the other ingredients would change the radiation damage resistance.

e. Manufacturing and Cost

The cost of the composite material has been estimated to be approximately \$12 per pound in a form ready to use in the shield. The raw materials are inexpensive in the powder form, and extrusion is an economical fabrication process for cylindrical rods.

21. Tungsten Boride (WB_2)

a. Structural Characteristics

Tungsten boride is a refractory material sometimes used for high-temperature structural applications. It is thought to lack the necessary thermal stress resistance to accomodate the extreme temperature differential that would be encountered in this application. Thus for this application it is assumed that it would be in the form of encapsulated pellets.

b. Chemical Characteristics

Tungsten boride has a density of 13.1 gm/cm³ and a melting point of 5300 °F. (12 p.64)

c. Thermal Characteristics

The thermal conductivity of tungsten boride is 20 Btu/hr, ft, °F. The specific heat is 0.05 Btu/lb, °F (12, p. 64).

d. Radiation Damage Resistance

The radiation damage resistance of tungsten boride is unknown.

e. Manufacturing and Cost

Presumably the cost of a finished shield insert containing tungsten boride would be high due to the expense of forming hot-pressed pellets and the cost of encapsulation.

22. Yttrium Hydride (YH_x)

Most of the experimental data on the characteristics of yttrium hydride appears in reference 43, a secret document. It cannot be given in this report. Data classified confidential appears in references 31 and 40.

23. Zirconium Hydride (Zr H_x)

A great deal of the detailed experimental data on the properties of zirconium hydride appears in reference 43, a secret document. Because of this, these data cannot be presented in this report. Additional data, classified confidential, appears in references 40 and 44.

24. Magnesium Hydride (Mg H₂)

a. Structural Characteristics

Magnesium hydride cannot be used as a self-supporting structural element as it occurs in white, tetragonal crystals or powder form. (18, p.B-190)

b. Chemical-Physical Characteristics

Magnesium hydride is a non-volatile solid at room temperatures which decomposes at 280°C. If brought into contact with water at room temperature, it decomposes violently. The foregoing is from reference 18, page B-190. Magnesium hydride does not ignite spontaneously on exposure to air. (19)

The density of magnesium hydride is 1.4 gm/cm³.

c. Thermal Characteristics

The thermal conductivity of magnesium hydride was not known, but was estimated to be of the order of 4 Btu/hr, ft, °F from data on other hydrides. The specific heat was estimated to be 0.2 Btu/lb, °F.

d. Radiation Damage Resistance

The radiation damage resistance of magnesium hydride is not known by the writer.

e. Manufacturing and Cost

Magnesium hydride is an inexpensive raw material but costly encapsulation would be required so that it is anticipated that the overall cost of fabricated inserts would be high.

Magnesium hydride, if inhaled, is locally irritating and may cause irreversible but not severe damage. (19, p.951)

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