NASA TECHNICAL MEMORANDUM

NASA TM X-2835

SHIELD MATERIALS RECOMMENDED
FOR SPACE POWER NUCLEAR REACTORS

by Leonard J. Kaszubinski

Lewis Research Center
Cleveland, Ohio 44135

1. Report No. NASA TM X-2835	2. Government Accession No.	3. Recipient's Catalog No.
4. Title and Subtitle SHIELD MATERIALS RECOMMENDED FOR SPACE POWER NUCLEAR REACTORS		5. Report Date July 1973
		6. Performing Organization Code
7. Author(s) Leonard J. Kaszubinski		8. Performing Organization Report No. E-7347
9. Performing Organization Name and Address Lewis Research Center National Aeronautics and Space Administration Cleveland, Ohio 44135		10. Work Unit No. 503-25
		11. Contract or Grant No.
12. Sponsoring Agency Name and Address National Aeronautics and Space Administration Washington, D.C. 20546		13. Type of Report and Period Covered Technical Memorandum
		14. Sponsoring Agency Code

15. Supplementary Notes

16. Abstract

   Lithium hydride is recommended for neutron attenuation and depleted uranium is recommended
for gamma-ray attenuation. For minimum shield weights these materials must be arranged
in alternate layers to attenuate the secondary gamma rays efficiently. In the regions of the
shield near the reactor, where excessive fissioning occurs in the uranium, a tungsten alloy is
used instead. Alloys of uranium such as either the U-0.5Ti or U-8Mo are available to accom-
modate structural requirements. The zone-cooled casting process is recommended for lithium
hydride fabrication. Internal honeycomb reinforcement to control cracks in the lithium hydride
is recommended.

17. Key Words (Suggested by Author(s)) Nuclear shields; Space nuclear reactors; Lithium hydride; Tungsten; Uranium alloys	18. Distribution Statement Unclassified - unlimited		
19. Security Classif. (of this report) Unclassified	20. Security Classif. (of this page) Unclassified	21. No. of Pages 35	22. Price* $3.00

*For sale by the National Technical Information Service, Springfield, Virginia 22151

SHIELD MATERIALS RECOMMENDED FOR SPACE POWER NUCLEAR REACTORS

by Leonard J. Kaszubinski

Lewis Research Center

SUMMARY

The shielding materials recommended for space power nuclear reactors are lithium hydride for neutron attenuation and either depleted uranium or a tungsten alloy for gamma-ray attenuation. For minimum shield weights, these materials are arranged in alternate layers to attenuate secondary gamma rays in the shield efficiently. It appears that these materials have been developed sufficiently for use in space flight shields.

The zone-cooled casting process is recommended for lithium hydride fabrication. In this process, the lithium hydride is melted in a hydrogen atmosphere and cast directly into its stainless-steel container. This container protects the chemically reactive lithium hydride and inhibits hydrogen loss. The advantages of casting are close fit to the container and ease of fabricating intricate shapes. The disadvantage of casting is the formation of cracks during the freezing of lithium hydride. But an internal honeycomb reinforcement has been developed to control the sizes and propagation of these cracks within tolerable limits.

INTRODUCTION

The use of nuclear-reactor-type power systems is being considered for a wide variety of applications for space. These applications include direct-broadcast communications satellites, electric propulsion for deep-space probes, and manned orbiting space stations. All these applications of nuclear power require radiation shields to protect the payload and men. The shield weight is a major factor in the overall system design. The shield may comprise from 20 to 60 percent of the total nuclear power system weight. Because launch costs are sensitive to weight, minimization of shield weight is a prime concern in the design of nuclear space power systems.

To attain low-weight shields, the materials that are most efficient to attenuate the radiation from the reactor must be used. In the late 1950's, the Aircraft Nuclear Propulsion (ANP) program began developing lithium hydride as an efficient neutron-

attenuation material. This development was continued through the 1960's in connection with the Systems for Nuclear Auxiliary Power (SNAP) program. Historically, gamma-ray-attenuating materials did not get the amount of attenuation that the neutron-attenuating materials received. Tungsten was regarded as the most efficient (from a minimum-shield-weight standpoint), although costly, material for gamma-ray attenuation. But recently, depleted uranium has been considered for use. Uranium's efficiency as a gamma-ray-attenuation material is similar to that of tungsten. The high abundance of depleted uranium as tailings from diffusion plants has made it attractive costwise.

This report (1) reviews the rationale for the selection of efficient space flight shield materials; (2) gives an overview of the recent lithium hydride (LiH) fabrication development efforts conducted by NASA; (3) recommends the materials that appear to be the best compromise among shielding efficiency, low cost, and engineering simplicity; and (4) provides pertinent material properties and recommendations to assist the shield designer.

Depleted uranium is the material recommended for gamma-ray attenuation. However, its use near the reactor may not be desirable. Near the reactor, excessive fissioning of the uranium may cause an appreciable increase of heat generated in the shield. This excessive fissioning could also cause intolerable swelling of the uranium. Thus, for regions in the shield near the reactor a tungsten alloy is recommended.

Alloys of uranium may be needed to accommodate structural requirements in the high-temperature regions of the shield. The low-concentration uranium-titanium (U-Ti) alloys may be suitable for the temperature range from 478 to 811 K (400° to 1000° F). The uranium-molybdenum (U-Mo) alloys are recommended for use at temperatures greater than 811 K (1000° F).

The work in this report was done in the English system of units.

SHIELD MATERIALS

To minimize the shield weight required for the desired attenuation of all radiation sources emanating from the power system of a space power nuclear reactor, the shield is constructed of discrete layers of gamma- and neutron-attenuating shield materials. The size, number, and location of the shield layers depend on the allowed radiation dose levels, the power level of the reactor, and the design of the power system. Shield complexity and weight will vary widely depending on these factors. The unmanned low-power reactors require very simple shields. For instance, the shadow shield for the thermoelectric power system of the unmanned SNAP-10A reactor consisted of only a 50.8-centimeter (20-in.) thick layer of LiH (no discrete gamma layer used) weighing about 100 kilograms (220 lb) (ref. 1). On the other hand, the 4π shields for nuclear

reactor power systems for manned spacecraft become very complex (ref. 2). These 4π shields consist of multiple alternating layers of both types of shield materials (neutron and gamma) that weigh from 9070 to 65 800 kilograms (20 000 to 145 000 lb).

This section of the report describes the general requirements of both the neutron- and gamma-attenuating shield materials. Also recommendations are made for the materials to be used in the shielding of nuclear reactors for electrical power production in space.

Neutron-Attenuating Materials

An efficient neutron-attenuating material consists of low-atomic-number nuclei with a large number of nuclei per unit volume. These material characteristics are effective in slowing down (reducing neutron energy) the highly energetic neutrons released from the reactor. During or after the slowing-down process, the neutron shield material must also capture the neutrons without producing excessive gamma radiation.

Hydrogen, being the lowest-atomic-number nuclide, is the best slowing-down material. Materials having the highest hydrogen atom concentration per unit volume are sought. Use of these materials would minimize the thickness of the neutron shielding layers, which would, in turn minimize the size (and hence weight) of the gamma shielding layers that are located outside of the neutron shielding layers. Water, which contains about 11.1 weight percent of hydrogen (6.65×10^{22} hydrogen atoms/cm^3), is the material usually chosen for ground-based nuclear reactor shields. Its abundance and low cost make it attractive. Either water or a hydrocarbon could also be used for space nuclear reactor shields. However, the danger of losing the neutron shield through leakage is greatly increased with the use of a liquid or a gas. The construction of reliable containers for liquids would impose severe weight penalties on the power system and would increase the complexity of the shields. For these reasons, solid neutron-attenuating materials are sought for use in space nuclear reactor shields.

Table I lists several neutron-attenuating materials that have been considered for use in space nuclear reactor shields. Only the materials having melting points greater than 811 K (1000° F) are included because the shields for space power nuclear reactors may have peak temperatures in the range from 422 to 711 K (300° to 800° F) (ref. 2). Three types of materials are presented in table I:

    (1) The metal hydrides (LiH, CaH, TiH)

    (2) High-temperature materials (Li$_2$O and BeO-B)

    (3) Intermediate-temperature composites (Be-B and BATH)

Titanium hydride is considered to be a combination neutron and gamma-ray attenuator.

TABLE I. - HIGH-TEMPERATURE NEUTRON SHIELD MATERIALS

Material	Melting point		Thermal conductivity[a]		Density		Reference
	K	°F	W/(cm)(K)	Btu/(hr)(ft)(°F)	g/cm^3	lb/in.^{3}	-
LiH	925	1265	0.147	8.5	0.73	0.026	24
CaH	1256	1800	.031	1.8	1.9	.069	25
TiH$_{1.8}$	[b]908	1175	-----	----	3.8	.137	26
Li$_2$O	1533	2300	-----	----	1.8	.065	25
BeO-B	2728	4450	.55	32	2.9	.105	25
Be-B	1622	2460	1.04	60	1.8	.065	25
BATH[c]	867	1100	1.21	70	2.8	.101	27

[a]At a temperature of 294 K (70° F).

[b]Dissociation pressure, 9.3×10^4 N/m^2 (27.6 in. Hg) at indicated temperatures.

[c]Composite material made of boron, aluminum, and titanium hydride.

Lithium hydride has received the most attention to date in the development of SNAP reactor power systems (ref. 3). Although it is very brittle and its thermal conductivity is very low, which cause some engineering problems, it has other attractive properties, such as high hydrogen content, relatively low dissociation pressure, and the ability to capture neutrons without releasing gamma rays. For these reasons, LiH is recommended for use as the neutron-attenuating material for space power nuclear reactors.

The fabrication development of LiH is discussed in the section LITHIUM HYDRIDE. Appendix A includes the pertinent engineering properties of LiH.

Gamma-Ray-Attenuating Materials

The requirements of an efficient gamma-attenuating material are just the opposite of those for the neutron shield. An efficient gamma shield consists of a material with a high electron density because the gamma rays react primarily with the electrons of atoms as they pass through a material. Thus, the high-density materials containing nuclei of high atomic number are the candidates for space power gamma shielding.

Table II lists the materials that could be considered for gamma-ray shields. This table excludes those elements having melting points lower than 811 K (1000° F), with the exception of lead. Lead is included in the table for comparison purposes because of its common use as X-ray and gamma-ray shielding in other applications. Table II also excludes the less-abundant elements that would incur excessive costs. The effectiveness of these materials in attenuating gamma rays can be ranked from either a weight or thickness standpoint. Table II shows the relative weight and thickness of a slab of the material required to attenuate gamma rays (4 to 10 MeV) by a factor of 10. Uranium is

TABLE II. - COMPARISON OF GAMMA-ATTENUATING MATERIALS

Material	Atomic number	Approximate relative cost	Melting point		Density		Thickness[a]		Relative weight[a]
			K	°F	g/cm^3	lb/in.^{3}	cm	in.	
U	92	4	1406	2070	18.7	0.675	2.9	1.14	1.00
Pb	82	2	600	620	11.3	.408	4.9	1.93	1.05
W	74	10	3683	6170	19.3	.697	3.0	1.18	1.08
Ta	73	8	3272	5430	16.6	.599	3.5	1.38	1.09
Mo	42	7	2897	4755	10.2	.368	6.7	2.64	1.27
Nb	41	7	2744	4480	8.6	.310	8.0	3.15	1.28
Fe	26	1	1812	2800	7.9	.285	9.4	3.70	1.39

[a]Based on attenuation of 4- to 10-MeV gamma rays by factor of 10.

the lowest in weight. If iron were used instead of uranium, it would be 39 percent heavier. Uranium is also the thinnest, requiring only 2.8 centimeters (1.2 in.) thickness, while iron would require 9.4 centimeters (3.7 in.).

There are other engineering aspects to consider in selecting one of these materials for space reactor shielding. The interaction of neutrons with the gamma shielding material will affect its selection. For instance, the use of uranium near the reactor may not be desirable. Even though "depleted" uranium is used (containing less than 0.3 wt. % U^{235}), significant fissioning in the uranium could occur. This could cause a substantial increase of heat generated in the shield and could also cause uranium swelling. These considerations usually exclude the use of uranium near the reactor. However, the use of depleted uranium is recommended in regions of the shield away from the reactor. The section entitled URANIUM discusses the characteristics of uranium. Appendix B includes the pertinent engineering properties of uranium and its alloys.

Another area of concern is the production of additional gamma rays (called secondary gamma rays) caused by neutron absorption and inelastic neutron scattering in the shield material. All the materials in table II will produce secondary gamma rays when they are bombarded with neutrons from the reactor. This production of secondary gamma rays is a distinct disadvantage because additional gamma shielding is required to attenuate them. Lead produces very weak secondary gamma rays. This property alone makes lead a very desirable candidate for gamma shields surrounding a reactor. However, because of its relatively low melting point, lead would be molten in a high-temperature shield if special cooling were not provided. Cooling the lead to prevent melting would require a low-temperature coolant not normally available in most space power systems. Thus, separate special low-temperature cooling loops must be provided. On the other hand, if molten lead is used, it must be reliably contained. The containment of high-temperature molten lead in steel containers has been investigated

(ref. 4). The molten lead decarburized its steel container, weakening it severely. Thus, the use of lead in a high-temperature shield presents engineering complexities that usually eliminate it from consideration.

If lead is eliminated for use in the shield and uranium cannot be used near the reactor, tungsten is the gamma shield material recommended for use near the reactor. Because tungsten is difficult to form and fabricate, the heavy-metal alloys are recommended for use (see the section TUNGSTEN HEAVY-METAL ALLOYS and appendix C).

Finally, there may be unique structural requirements imposed on the gamma shield layers. For instance, if the shield layer near the reactor must provide ground-impact protection for the reactor (after accidental reentry), high-energy-absorbing materials such as tantalum could be used. Although table II indicates that the use of tantalum instead of tungsten is not a serious compromise, the activation of tantalum by the $Ta^{181}(\eta, \gamma)W^{182}$ reaction must be considered.

LITHIUM HYDRIDE

The physical forms of LiH that have been considered for use in shielding range from vibration-packed granules (as high as 70 percent theoretical density) to hot-pressed compacts (greater than 99 percent density). Cold pressing and casting usually result in a density range from 93 to 95 percent of theoretical density. The advantage of using vibration-packed granules is low cost. The disadvantages are low hydrogen densities, very low thermal conductivity, and no strength. On the other hand, hot-pressed compacts are the most expensive. The high density that can be achieved with hot pressing will result in a minimum-weight shield. However, from the radiation-damage standpoint a very high density will be a disadvantage. It appears that a 3 to 5 percent void in the LiH is required to accommodate the radiation-induced swelling of LiH (see appendix A). Thus, it appears that cold pressing and casting are the prime candidates of fabricating LiH for space flight shields.

This section of the report first describes the cold pressing and casting fabrication processes. The casting process is recommended for use. Then a brief review of the development of LiH casting is followed by a description of more recent experiences, with the fabrication of the experimental shield for the NASA Lewis Research Center Zero Power Reactor and the fabrication of a LiH thermal conductivity test disk.

Fabrication Methods

Regardless of the method of fabrication, LiH must be protected from water vapor

at room temperature. At elevated temperatures, water vapor, nitrogen, carbon dioxide, and many other elements and compounds must not contact LiH.

Cold pressing. - The size of a LiH compact that can be prepared by the isostatic cold-pressing technique is limited only by the size of the pressure vessel that can be built. Granular LiH is vibration packed into a thick-walled flexible rubber or plastic bag. This bag is supported by a plastic form. The form is constructed in the geometry of the desired shape. The bag is sealed and evacuated. About 95 percent of theoretical density is attained with a pressure of 2.07×10^8 N/m^2 (30 000 psi). Sizes as large as 63.5 centimeters (25 in.) in diameter and 91.4 centimeters (3 ft) in length have been produced with very uniform density (ref. 3). Sizes as large as 1.32 meters (52 in.) in diameter and 1.2 meters (4 ft) in length are possible with existing isostatic chambers.

After compaction, the cold-pressed LiH is machined to fit its container. These containers are required to protect the LiH from chemical reactions and to limit hydrogen loss to acceptable values. The machining operations are conducted in a dry nitrogen atmosphere. This is necessary to protect the LiH from air and moisture contamination and to avoid spontaneous ignition of the chips and powder.

These machining operations can be quite expensive, especially when the LiH shield sections are either irregular or contain ducts. Furthermore, the fit between the LiH and the container may be too loose. This could cause neutron streaming, heat-transfer problems, and structural difficulties. The casting of LiH directly into its containment avoids these drawbacks. The following section describes the casting fabrication process.

Casting. - The zone-cooling technique has been used to produce LiH castings. The LiH is melted in a hydrogen atmosphere and poured directly into its containment (usually an austenitic stainless steel). The LiH is then progressively frozen from the bottom upward to accommodate the 20 percent shrinkage of freezing LiH. The casting size is limited only by the capacity of the casting facility. LiH castings as large as 2.13 meters (7 ft) in diameter and weighing as much as 773 kilograms (1700 lb) have been produced (ref. 5). Castings 2.44 meters (8 ft) in diameter and weighing as much as 1818 kilograms (4000 lb) are possible using an existing facility.

The zone-cooled casting process produces LiH densities in the range from 93 to 95 percent of theoretical. The 5 to 7 percent void in the cast material consists mainly of cracks. These cracks occur as the result of the internal stresses developed during freezing. Because LiH is very weak and brittle (see appendix A) cracks are formed to relieve the stresses. Large cracks in the LiH may reduce its shielding efficiency. For this reason, reinforcing structures have been successfully put into the cast material to control the size and propagation of the cracks (see next section entitled Casting Development).

In general, casting is a versatile method of producing large and intricate shapes of LiH. The advantages of the casting process are

(1) Close fit to the container (no machining tolerances) and thus better heat-transfer characteristics

(2) Ease of fabricating shapes directly into intricate containers

(3) No expensive machining equipment required

For these reasons the casting process is recommended for use in the fabrication of LiH shields for nuclear reactor space power systems.

Casting Development

The Atomic Energy Commission has sponsored lithium hydride shield development in the Aircraft Nuclear propulsion (ANP) and the Systems for Nuclear Auxiliary Power (SNAP) programs. This section gives a brief overview of the lithium hydride shield development in the SNAP programs.

Fabrication. - Nine SNAP LiH shields were cast (refs. 5 and 6). The shapes of these shields were either cylindrical or a conical frustum containing from 91 to 773 kilograms (200 to 1700 lb) of LiH. Each of these shields was unique. Not only were their shape and size varied, but the container wall thicknesses were also varied from 0.051 to 0.279 centimeter (0.02 to 0.110 in.) to study distortion problems. Furthermore, reinforcing structures such as honeycomb and stainless-steel wool were used in attempts to control the size and propagation of cracks. Also tube ducts and internal structures such as ribs and truss-type tubular members were incorporated into the castings.

Container wall distortion: To minimize the weight penalty associated with the LiH containment, thin container walls are desirable. However, other considerations such as hydrogen permeation, micrometeoroid protection, structural requirements, and container distortion occurring during the casting process may control the selected thickness of the container wall. This latter effect (distortion during casting) was studied. All containers with walls thinner than about 0.102 centimeter (0.04 in.) were found to undergo moderate to severe distortion. The only successful attempt at limiting the distortion of a thin container wall (0.051 cm (0.02 in.) thick), to acceptable limits occurred when stainless-steel screens were used to disrupt the adhesion between the LiH and the container wall. This disruption of the bond would, however, tend to increase the resistance to heat flow and worsen the already poor heat-transfer properties of LiH. The very thick (0.279-cm (0.110-in.) thick), container wall was used in a shield where the container wall served as the only structural member supporting the reactor. The distortions in the container wall of this shield were nil.

8

It appears that container-wall thicknesses from 0.102 to 0.204 centimeter (0.04 to 0.08 in.) will result in acceptable distortions. Wall thicknesses approaching 0.204 centimeter (0.08 in.) may be required from the hydrogen permeation and micrometeroid damage standpoints (ref. 2).

Crack control: The tendency of LiH to crack during casting and when subjected to vibrational loading requires internal reinforcement in the LiH. Two types of reinforcing structures have been used. The first and most successful is an "egg crate" honeycomb with 2.54-centimeter (1-in.) square cells. The honeycomb walls are 0.0025-centimeter (1-mil) thick stainless steel. Perforations 1.27 centimeters (1/2 in.) in diameter on 1.91-centimeter (3/4-in.) centers in the honeycomb walls allow the molten LiH to flow freely among the cells to eliminate void formations. The length of the square honeycomb cells is oriented normal to the direction in which maximum shielding effectiveness is desired. Thus, the maximum crack depth would be only about 2.54 centimeters (1 in.) - the distance across one cell - through the thickness of the LiH shield layer.

The second type of reinforcing structure that has been used is stainless-steel wool. To avoid the relatively costly honeycomb, stainless-steel wool was substituted for the honeycomb in two of the SNAP castings. The amount of steel wool inserted into these shields was comparable to the honeycomb volume (i.e., ~0.2 percent). This amount of steel wool proved generally ineffective in the control of cracks. This ineffectiveness was, in part, the result of lack of uniformity that was aggravated by shifting of the wool caused by the flowing molten LiH. A larger amount of wool may prove more effective; however, the shifting problem must be solved.

At this time, it appears that the egg-crate honeycomb would be more appropriate for use as a LiH reinforcing structure in space nuclear reactor shielding.

Testing. - The SNAP LiH shields were subjected to structural and thermal environmental tests that are expected during launch and operation in space.

Structural testing: Internal structures have been used in the SNAP shields. One shield had perforated ribs welded to the inner surface of the container to secure the LiH to the container wall. This was done for two reasons: first, to eliminate the ringing or rattling of the LiH in the container when it was vibrated; and second, to reduce the thermal resistance at the LiH-wall interface. Structurally, the ribs were successful in eliminating the ringing. However, some of the weld joints failed during the vibration testing (ref. 6). The results of the thermal tests are included in the following section.

Another shield had internal perforated tubes used as tension structural members to support the reactor. The LiH was used as the compression structure. Again a weld connecting a perforated tube to the side wall of the container failed during vibration testing. The test was otherwise successful since the LiH withstood the compressive loading with no apparent failure.

Thermal testing: Five of the SNAP shields were thermally cycled from 294 to 811 K (70° to 1000° F) (ref. 6) to determine the effectiveness of the reinforcing structures in controlling crack propagation. Severe cracking occurred as a result of thermal cycling in those shields without the honeycomb. Bulk thermal conductivity measurements were conducted on three shields (see appendix A).

Zero Power Reactor Test Shield

Four shield segments were designed and built and were filled with LiH by the zone-cooling casting process (ref. 7). The segments were sectors of a right-circular cylindrical annulus intended for use in neutron shield tests at the NASA Lewis Research Center Zero Power Reactor Facility. The cylindrical annulus, 0.747 meter (29.4 in.) in diameter and 0 925 meter 36.4 in.) high, contained about 160 kilograms 352 lb) of high-purity (>98 percent) natural LiH in the 0.133-meter (5.22-in.) thick walls. Three of the four segments contained penetrations through the LiH that are intended for use in tests of radiation streaming through ducts. Figure 1 shows the assembly of the four
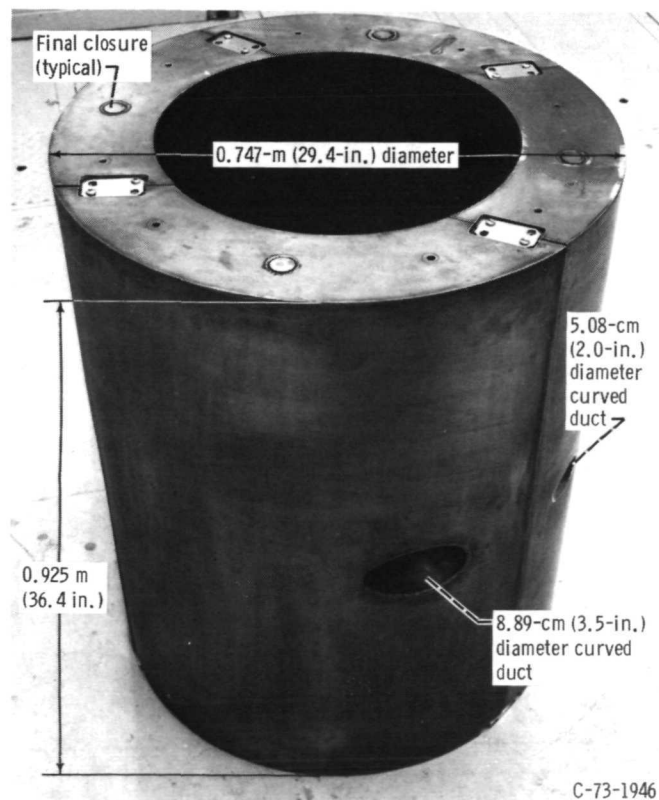

Figure 1. - Assembled segments of ZPR test shield.

segments. Two curved penetrations 8.9 and 5.08 centimeters (3.5 and 2 in.) in diameter are visible. Figure 2 shows the four segments. The containers are all made of welded type-304 stainless steel. The segments contained the egg-crate honeycomb (see previous section entitled Casting Development). The honeycomb cells' axis was oriented parallel with the centerline of the cylindrical annulus.

Geometry. - The segmented annulus design of the ZPR LiH test shield was dictated by the interface requirements with the existing ZPR test facility and the desire to reorient and replace sectors of the shield. Moderate distortion problems were anticipated because of the nonsymmetrical geometry of the segments. Two types of distortion can occur as a result of the casting thermal cycle:

(1) The relief of residual welding stresses, which causes gross movements of the container structure

(2) A pull-in of the container side wall caused by the contracting LiH as it cools Because inward movements of the side walls would not affect the assembly of the segments into the required annulus, no attempt was made to prevent these movements. But gross movements of the container structure were not permissable. Thus, a rigid support structure was constructed in which the segment containers were constrained during the casting thermal cycle. This structure successfully controlled the gross distortions (ref. 7).

The thickness of the segment container wall was 0.198 centimeter (0.078 in.). This thickness was selected as representative of space flight LiH shields according to reference 2. The convex walls of the segments generally pulled inward from 0.051 to 0.102 centimeter (0.02 to 0.04 in.). Surprisingly, two segments exhibited local bulges as large as 0.109 centimeter (0.043 in.) in the convex walls. This latter type of distortion was not anticipated. All the concave faces and the side plates were drawn inward, producing a concavity. The side plates pulled inward from 0.102 to 0.204 centimeter (0.04 to 0.08 in.), and the concave surfaces pulled inward from 0.152 to 0.304 centimeter (0.06 to 0.12 in.).

Crack control. - The internal honeycomb was effective in preventing excessive propagation of individual cracks. Crack depths (in the shielding direction) were successfully controlled within the 2.54-centimeter (1.0-in.) width of a honeycomb cell. However, some agglomeration of cracks in the LiH was noted in each segment. The reduced shield efficiency resulting from these crack agglomerations is as yet unknown.

Void control. - The technique of the zone-cooling casting process eventually produced void-free castings after some initial difficulties. In the first two segments that were cast, voids from 98.3 to 163.9 cubic centimeters (6 to 10 in.[3]) formed beneath the fill tubes. The last two segments were void free after an adjustment was made to the trim heater configuration around the fill tube.
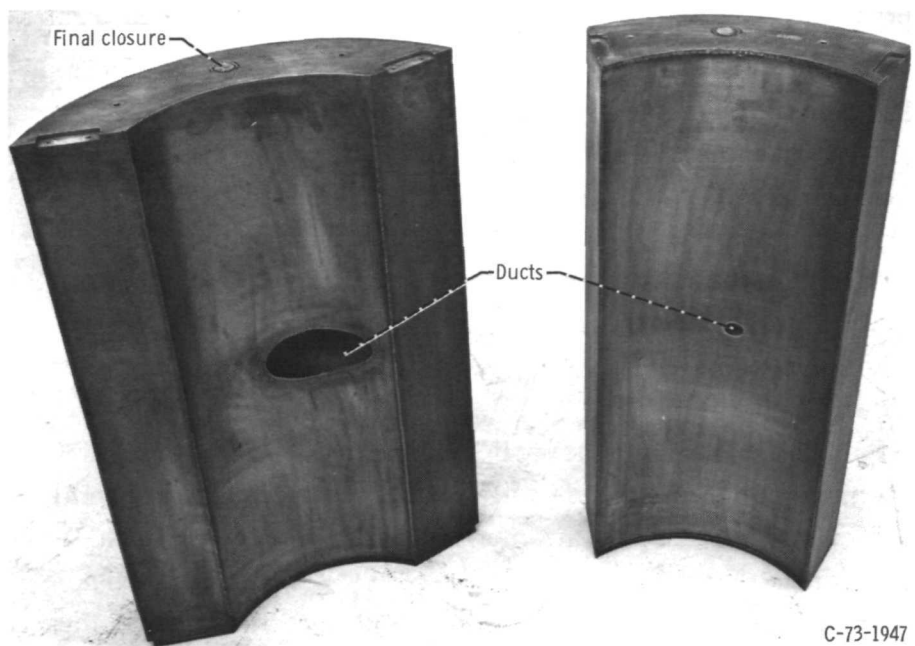
(a) Segment containing 8.89-centimeter (3.5-in.) diameter curved duct and segment containing 2.54-centimeter (1.0-in.) diameter straight duct.
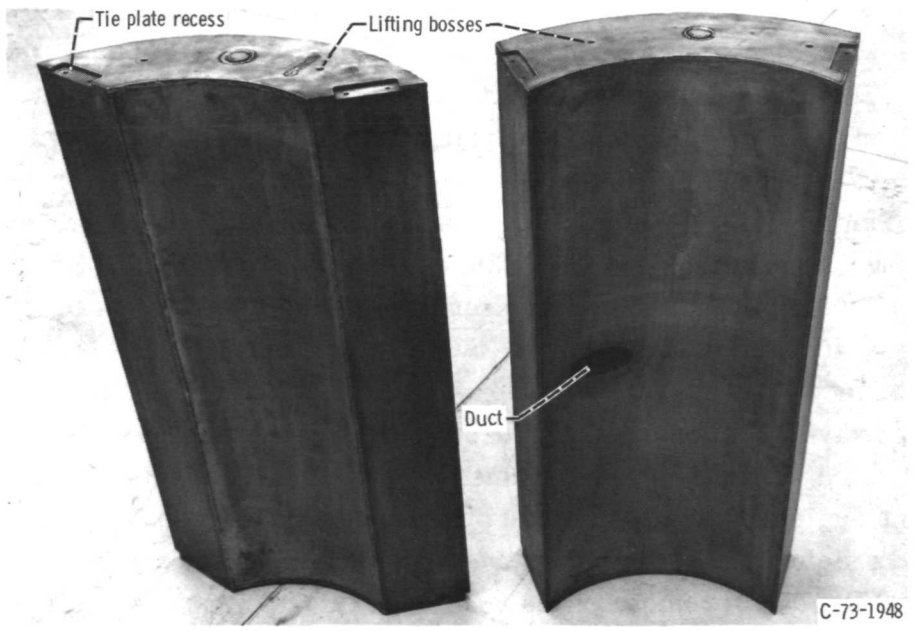
(b) Solid segment and segment containing 5.08-centimeter (2-in.) diameter curved duct.

Figure 2. - ZPR test shield segments.

Recommendations. - It appears that nonsymmetrical LiH shapes can be cast without excessive distortions. However, the distortions that do occur are greater than those experienced in symmetrical castings. Thus, looser fits between shield parts may be necessary. This looseness of fit may cause undesirable radiation streaming and thus contribute to shield inefficiencies. Therefore, it is recommended that the shield designer

    (1) Minimize the number of LiH castings in the shield

    (2) Strengthen or stiffen the thin flat walls

    (3) Avoid nonsymmetrical shield parts if possible

Thermal Test Disk

A flat disk-shaped container made of stainless steel was built and filled with lithium hydride by the zone-cooled casting process. The container was 0.762 meter (30 in.) in diameter by 0.152 meter (6.0 in.) thick. This disk will be subjected to a series of thermal conductance and vibration tests. This section of the report describes the fabrication of the thermal test disk.

Geometry. - The LiH disk that was cast could not be oriented horizontally because (1) discontinuities in a flat-face surface near the final closure, (2) slight voids near the filler tube, and (3) possible void formations underneath the flat end if it were positioned at the top would interfere with the thermal pattern of the planned thermal conductance test. Thus, the disk was cast on edge. That is, the flat ends of the disk were positioned vertically and the filler tubes were located on the cylindrical surface of the disk.

The two filler tubes straddled the exterior flange and an internal ring, as shown in figure 3. The flange, internal ring and internal ribs were required for the structural vibration tests. Also shown in figure 3 are the thermocouple support structures. Figure 4 shows the egg-crate honeycomb installed. Only the perforations of the top layer of cells are visible. Figure 5 shows the completed thermal test disk. The bulged center of the disk (on both ends) was not anticipated. It was expected that the flat side plates of the casting fixture, shown in figure 6, would restrain any bulging caused by the hydrostatic pressure of the molten LiH. However, these side plates of the fixture were not sufficient to oppose the oil canning of the thermal disk wall that occurred during freezing of the LiH.

Recommendations. - Deviations from a tested orientation or configuration in the casting of LiH requires very careful analysis of the forces developed during the casting cycle. Distortions can easily occur at the 978 K (1300° F) peak temperature in the casting cycle. In the case of the thermal test disk, it was recognized that forces caused by the hydrostatic pressure induced by the molten LiH could bulge the flat ends of the disk.
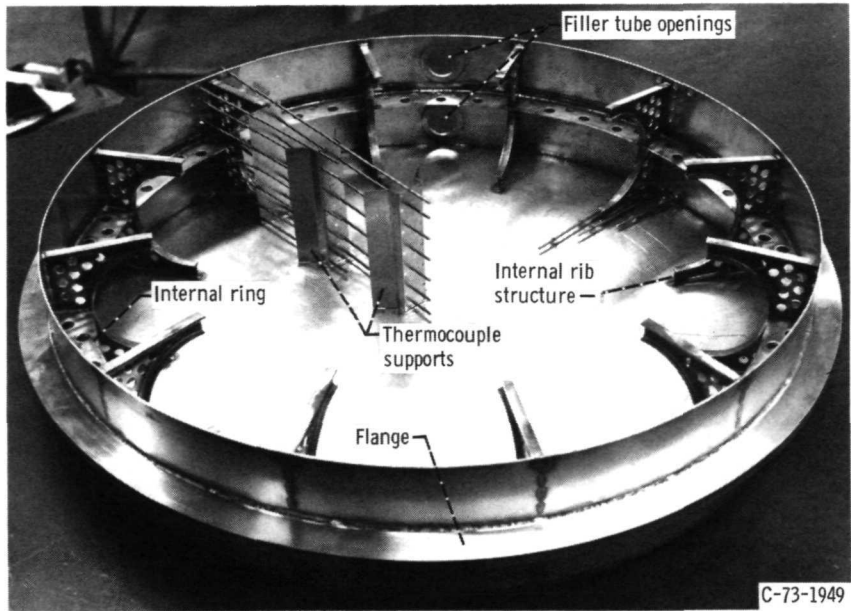
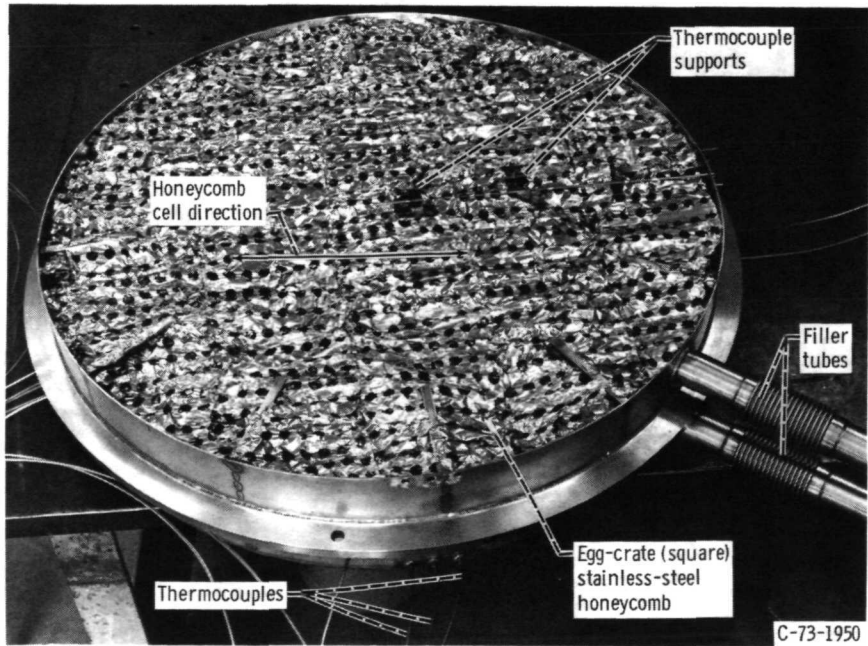Figure 3. - Thermal test disk partially assembled.

Figure 4. - Honeycomb reinforcement installed in thermal test disk.
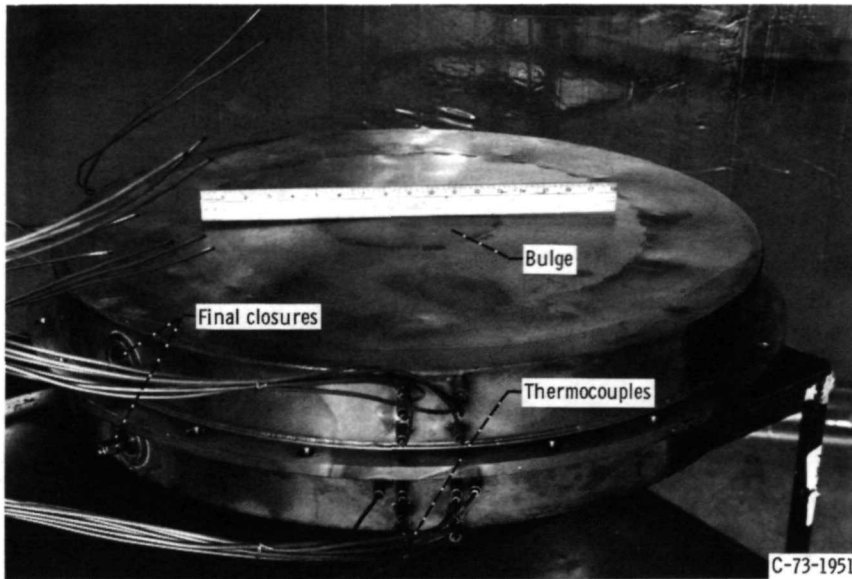
14

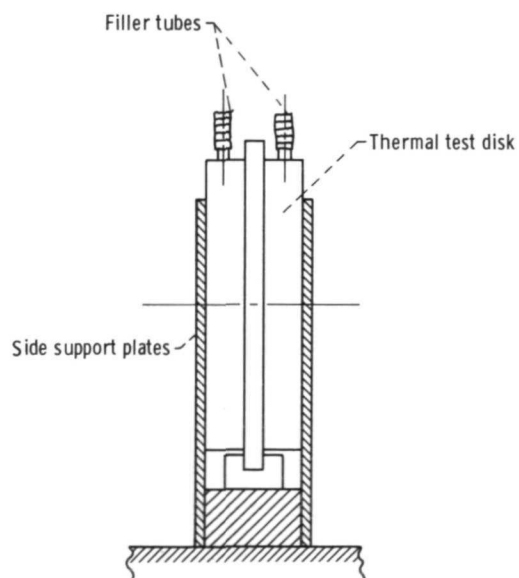Figure 5. - Completed thermal test disk.

Figure 6. - Thermal test disk in casting support fixture.

But the magnitude of other thermally induced forces was underestimated.

Another problem area is that small voids continually occur below the filler tubes. The wraparound filler tube heaters are not always successful in controlling the last stages of LiH freezing in the casting. Other approaches such as immersion heaters should be investigated.

URANIUM

The use of either unalloyed uranium or an alloy of uranium will depend on the temperature and the desired strength and fabrication properties. For high-temperature applications ($>$ 933 K (1220$^{\mathrm{O}}$ F)), a cubic-gamma-phase-stabilized alloy such as the U-8Mo alloy would be required. The dilute low concentration U-Ti or U-Zr alloys may be suitable for use at temperatures below 933 K (1220$^{\mathrm{O}}$ F). Unalloyed uranium may be suitable for temperatures below about 533 K (500$^{\mathrm{O}}$ F).

Thermal Stability

For temperatures below 933 K (1220$^{\mathrm{O}}$ F), unalloyed uranium or uranium alloys with alloying additions in relatively low concentrations can be used. These materials can be heat treated to attain random grain orientation for thermal stability under thermal cycling conditions in the alpha temperature zone (i.e., $<$933 K (1220$^{\mathrm{O}}$ F)). For temperatures above 933 K (1220$^{\mathrm{O}}$ F), alloying additions to the uranium must be made in sufficient concentrations (such as the U-8Mo alloy) to stabilize the cubic gamma phase either partially or completely, thereby eliminating the intrinsic instability of the orthorhombic alpha phase when the material is thermally cycled above 933 K (1220$^{\mathrm{O}}$ F).

Strength

The strength properties of unalloyed uranium rapidly deteriorate with increasing temperature. At 533 K (500$^{\mathrm{O}}$ F), the ultimate strength of unalloyed uranium may be as low as 2.07×10^8 N/m^2 (30 000 psi) - see appendix B. Although unalloyed uranium can be thermally stabilized by heat treatment for use to 933 K (1220$^{\mathrm{O}}$ F), its low strength at the higher temperatures would probably not allow its use as a structural material in the shield. For this reason, the U-Ti and U-Mo alloys are considered for use (see appendix B).

Fabrication

The fabrication properties and weldability of unalloyed uranium and the low-concentration U-Ti alloys are similar. Welds in uranium and in the U-0.5Ti alloy have strengths equivalent to their base metals. However, the ductility of the welds in both uranium and the U-Ti alloys may be one-half to one-third of the base metal ductility (see appendix B). Full-penetration welds in 1.27-centimeter (1/2-in.) thick uranium have been successful (ref. 8). Successful full-penetration welds in thicker plates have yet to be demonstrated.

The fabrication of the U-8Mo alloy is considerably more difficult. Higher temperatures (several hundred degrees Fahrenheit higher than required for unalloyed uranium) are required during wrought forming (ref. 9). The U-8Mo alloy exhibits "memory" characteristics; that is, it tends to return to its original shape after forming. Welds in the U-Mo alloys having molybdenum concentrations greater than 4 weight percent are very brittle and are plagued with stress-corrosion failures (refs. 10 and 11). This latter problem can be eliminated by prompt stress relief after welding (ref. 11). Because the U-8Mo alloy is difficult to fabricate, its use is not recommended where the strength and thermal stability of uranium and the U-Ti alloys are sufficient to meet the structural requirements.

Heat Treatment

Uranium and its low-concentration alloys must be heat treated to attain random grain orientation for thermal stability. Two types of heat treatment are available. The first is a beta heat treatment that consists of quenching or furnace cooling from the beta phase - between 933 and 1044 K (1220° and 1420° F). The second type consists of quenching from the gamma phase - above 1044 K (1420° F).

The U-Ti alloys age harden. Prolonged exposure (>1 hr) of these alloys at 672 K (750° F) could decrease their room-temperature elongation by one-half, while exposure to 756 K (900° F) may decrease the room temperature elongation to one-quarter of its normal value (ref. 12). This age-hardening characteristic may restrict the use of the U-Ti alloys to temperatures below 672 K (750° F) if retention of low-temperature ductility is required.

Corrosion

Unalloyed uranium is susceptible to rapid atmospheric corrosion (oxidation). At

ambient temperatures in air, a black tenacious oxide film covers its surface. At temperatures above 478 K (400° F) in air, the oxide film spalls from the surface, allowing the oxidation to progress deeper into the material. Above 867 K (1100° F), the uranium may ignite in air regardless of its thickness.

The U-Ti alloys provide increased oxidation resistance. Reference 13 indicates that the U-0.5Ti alloy offers a 25-fold improvement in resistance to corrosion when compared to unalloyed uranium. The U-8Mo alloy can provide even better oxidation resistance. Reference 9 indicates that the corrosion resistance of U-8Mo at room temperature, as opposed to unalloyed uranium, is improved by a factor of 1000.

Although the uranium layers in the shield will be in a vacuum during its space flight operation, there may be situations when the uranium may be either exposed to air at temperatures above ambient or subjected to corrosive atmospheres. Various protective coatings have been applied to uranium with varying success. Nickel plating has been partially successful. However, slight deviations in the plating process may result in pinholes and thin spots (ref. 14). Thin spots in the nickel plating have been observed in the less accessible surfaces of irregular shapes. Depending on the environment to which the nickel-plated uranium is subjected, these imperfections can lead to undercutting, peeling of the nickel plate, and the subsequent severe corrosion of the uranium. The nickel plating exposed to a nonsalt general weathering environment has remained intact for years. A high-temperature air environment, however, may cause quick failure of the nickel coating if small pin holes or thin spots are present.

The best organic coating was an acrylic enamel sprayed on a zinc chromate primer that was also applied by spraying. This coating withstood a salt spray test at room temperature for over 700 hours.

For maximum protection the uranium layers of the shield could be encased in all-welded stainless-steel cans. This procedure, however, must allow for the incompatibility of uranium with stainless steel at temperatures in excess of 978 K (1300° F).

TUNGSTEN HEAVY-METAL ALLOYS

A heavy-metal tungsten alloy was selected to avoid the difficult forming and machining characteristics of tungsten. One of the biggest advantages of the heavy-metal alloys is that their machinability is similar to that of stainless steel.

There are several commercially available heavy-metal tungsten alloys. The tungsten content in these alloys ranges from 89.5 to 95.0 weight percent. The W-3.5Ni-1.5Fe alloy is chosen for its high density (high tungsten content) and because it possesses the highest elongation properties of all the heavy-metal alloys. Appendix C contains some properties for this alloy.

Fabrication

Powder-metallurgy techniques are used to make the heavy-metal tungsten alloys. Alloy parts are produced by blending the component powders, pressing the blended powders into the required shape, and densifying by liquid-phase sintering.

Two-phase alloys. - The heavy-metal alloys are two-phase alloys. In the W-3.5Ni-1.5Fe alloy, grains of the tungsten-rich phase ($\sim$99 percent tungsten) are surrounded by a second or matrix phase ($\sim$30-wt.% W - 50-wt.% Ni - 20-wt.% Fe). This matrix phase melts at 1739 K (2670° F), and the tungsten-rich phase has a melting point in excess of 2772 K (4530° F).

Size limitations. - The size of heavy-metal parts that can be produced is limited by the pressing facility and sintering furnace. It appears that current commercial facilities can produce parts as large as about 30.5 centimeters (12 in.) diameter and about 91.4 centimeters (3 ft) long. The pressing operation is usually performed by explosive compaction in the breech of a naval gun. A large 2.07×10^{8}-N/m^{2} (30 000-psi) isostatic chamber has been used to produce a tungsten crucible weighing 773 kilograms (1700 lb) (ref. 15). This crucible was 33 centimeters (13 in.) in diameter and 61 centimeters (2 ft) long and had a wall thickness of 2.54 centimeters (1 in.). The ability to produce larger compacts of the tungsten heavy-metal alloys has not yet been demonstrated. Another problem is that the existing sintering furnaces may not be able to handle the weight of a large tungsten piece. Also, very massive parts have a tendency to slump during the sintering operation (ref. 15). Controlling this slumping may require a development program.

Thus, it appears that the largest tungsten heavy-metal alloy part that can be produced with existing facilities is about 30.5 centimeters (12 in.) in diameter by 91.4 centimeters (3 ft) long. If plates wider than this are required, the maximum-size billet can be forged and rolled. These forming operations must be conducted at temperatures greater than 1367 K (2000° F) (ref. 16). If cylindrical parts are required, these plates can then be roll formed to produce the desired shape. These operations are quite costly. It may be more practical to produce individual pieces that conform to the maximum billet size and join them together to form the desired shield layer.

Joining

Joining by welding does not appear to be feasible. Welding of the W-Ni-Fe alloys results in either vaporized matrix material or matrix melting only (ref. 17). In the former instance, no joint is made; in the latter, the joint is much less dense than the rest of the material and has completely different properties. This applies to the other

heavy-metal alloys as well because they are all characterized by the two-phase alloy composition.

Brazing is another possible means of joining, but the temperature resistance of the assembly is reduced depending on the braze alloy selected. The heavy-metal alloys can be joined by hot pressure welding; however, this requires close tolerance fits and very accurate temperature control.

Heat Treatment

The heavy-metal alloys are not really heat treatable. However, they are susceptible to hydrogen embrittlement. Because maximum sinterability is usually obtained in hydrogen atmospheres, it is desirable to heat treat the parts either in vacuum or in an inert atmosphere to remove the adsorbed and/or absorbed hydrogen. Approximately 3/4 hour per centimeter (2 hr/in.) of thickness is required at 1367 K (2000° F) to remove the hydrogen.

CONCLUDING REMARKS

The shielding materials recommended for space nuclear reactors are lithium hydride for neutron attenuation and either depleted uranium or a tungsten alloy for gamma-ray attenuation. For minimum shield weights, these materials are arranged in alternating layers to attenuate efficiently the secondary gamma rays produced in the shield.

It appears that these materials have been sufficiently developed to be used in the design and construction of space flight shields. The SNAP program has provided a broad technology base for the use of lithium hydride as a shield material. Further recent experiences of lithium hydride fabrication in connection with the shielding used for the NASA Lewis Research Center Zero Power Reactor have extended the fabrication technology of lithium hydride.

The use of uranium or tungsten alloys in shields should offer no insurmountable problems. Although some properties of these materials may appear troublesome, such procedures as proper alloy selection, heat treatment, and use of protective coatings are available to the shield designer to overcome these problems.

Lewis Research Center,
    National Aeronautics and Space Administration,
        Cleveland, Ohio, May 1, 1973,
            503-25.

APPENDIX A

PROPERTIES OF LITHIUM HYDRIDE

The properties included herein are those pertinent to space nuclear shields. Reference 18 contains an extensive description of the properties of lithium hydride.

Physical Properties

Lithium hydride (LiH) is a unique compound of lithium and hydrogen possessing the following properties:

Crystal structure . . . . Face-centered cubic
Molecular weight g/mole; lbm/mole 7.936; 0.0175
Density, g/cm^3; lb/ft^3 . . . . 0.775; 48.3
Hydrogen content (natural LiH), wt. % 12.68
Hydrogen density (natural LiH), H atoms/cm^3; H atoms/$in.^3$ . . . 5.86×10^{22}; 9.3×10^{23}
Melting point, K; ^oF 959; 1267
Plateau dissociation pressure at melting point, N/m^2; in. Hg . . . . . 3.31×10^3; 0.98
Thermal conductivity at 811 K (1000° F), W/(cm)(K); Btu/(hr)(ft)(^oF) . . . 0.05; 2.9
Thermal expansion in range of 294 to 811 K (70° to 1000° F), percent 2.62

Lithium hydride is a salt-like white crystalline substance. The polycrystalline cast material is blue-gray in color. Exposure to moisture results in the formation of lithium hydroxide (and evolving hydrogen), which is white.

Thermal Properties

Table III shows the linear thermal expansion of LiH. The thermal expansion of

TABLE III. - LINEAR THERMAL EXPANSION OF LiH

Temperature range		Linear thermal expansion[a], percent		
K	^oF	LiH	Uranium	Type 304 stainless steel
294 to 533	70 to 500	1.06	0.33	0.45
294 to 700	70 to 800	2.00	.61	.77
294 to 867	70 to 1100	2.94	.98	1.11

[a]Ref. 28.

uranium and type-304 stainless steel are shown for comparison. At 700 K (800° F) the LiH expands thermally about $2\frac{1}{2}$ times as much as the type-304 stainless steel. The mechanical integrity of LiH is degraded by thermal cycling and thermal shock. Since LiH is salt-like (very brittle), it cracks and breaks up when it is thermally cycled. Reinforcing structures in the LiH are required to control gross cracking during thermal cycling.

The thermal conductivity of cast LiH in helium and in a vacuum is shown as the solid curves in figure 7. The dashed curves in figure 7 represent "bulk" thermal con-
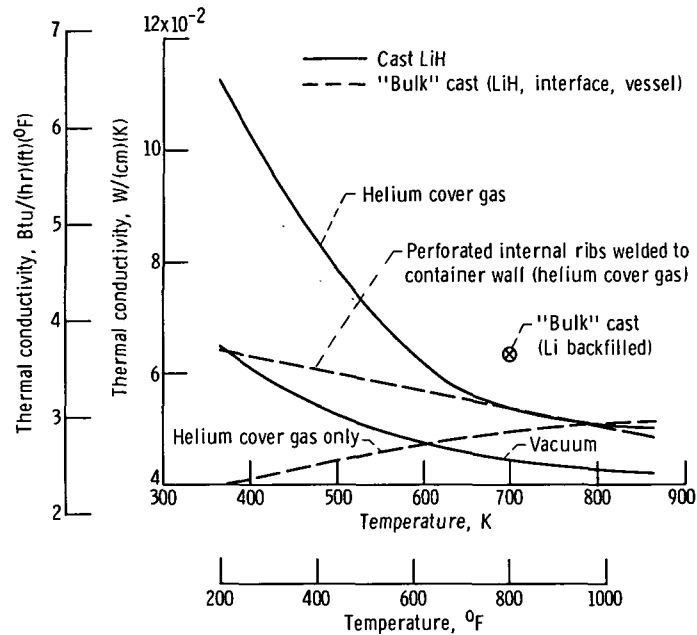

Figure 7. - Thermal conductivity of cast lithium hydride.

ductivity values of cast LiH inside stainless-steel containers. The bulk thermal conductivity is defined as the thermal conductivity of the system: LiH, interface, and vessel. The differences between the two dashed curves and the upper solid curve (cast LiH in helium) are attributed to the thermal resistance at the LiH-wall interface. At 867 K (1100° F) there appears to be no difference between the bulk thermal conductivity and the LiH thermal conductivity. Reference 6 postulates that at this temperature the LiH is contacting the vessel wall. This contact can occur because the thermal expansion coefficient of the LiH is much greater than that of its stainless-steel container. Figure 7 shows that the use of internal perforated ribs welded to the vessel wall substantially enhances the bulk thermal conductivity at the lower temperatures. The one data point at 700 K (800° F) for a Li-backfilled LiH shield (ref. 6) is also shown in figure 7.

Chemical and Thermodynamic Properties

Lithium hydride is chemically unstable (pyrophoric) in oxidizing atmospheres. It reacts with moist air exothermically, forming LiOH and Li_2O and evolving hydrogen. In moist air the powder may ignite spontaneously. It burns fiercely, forming a mixture of products that includes some nitrogenous compounds. The lump material reacts with humid air, forming a superficial coating which is a viscous fluid. This coating inhibits further reaction. Both the powder and lump material ignite when moistened. The moistened powder may cause a violent dust explosion.

The equilibrium hydrogen pressure over LiH varies with both temperature and composition in the manner characteristic of most metal-hydrogen systems. Table IV

TABLE IV. - VARIATION OF PLATEAU

DISSOCIATION PRESSURE WITH

TEMPERATURE

Temperature		Plateau dissociation pressure[a]	
K	°F	N/m^2	in. Hg
589	600	4.35×10^{-3}	1.29×10^{-6}
700	800	2.26	6.69×10^{-4}
811	1000	99.6	2.95×10^{-2}
922	1200	1.32×10^3	0.39
959	1267	3.31×10^3	0.98

[a]Ref. 19.

shows the variation of plateau dissociation pressure with temperature. These dissociation pressures require that a hydrogen overpressure be maintained over the LiH or that the LiH be sealed in containers. It has been estimated (ref. 19) that a solid shape of LiH at 700 K (800° F) would release all its hydrogen in about 1000 hours. Thus, the LiH must be sealed in containers that are impervious to hydrogen.

Austenitic (300 series) stainless steels are being considered (see compatibility and corrosion below) as the container material. Hydrogen will, however, diffuse through container walls made of stainless steel. An estimate based on the hydrogen diffusion data of reference 20 was made of the hydrogen loss in the LiH inner layer of the SNAP-8 reactor shield. Table V shows the results of this estimate. If the temperature is 811 K (1000° F) or less, the hydrogen loss by diffusion is negligibly small. If the container is damaged by launch loads or micrometeroid penetration, the amount of hydrogen loss can be much greater.

TABLE V. - ESTIMATE OF HYDROGEN LOSS IN

10 000 HOURS BY DIFFUSION THROUGH

CONTAINER WALLS

[Material, type 304 stainless steel 0.2 cm
(0.08 in.) thick.]

Maximum temperature[a]		Hydrogen loss, percent
K	°F	
922	1200	2.1
811	1000	.15
700	800	3×10^{-3}
589	600	3×10^{-5}

[a]Temperature at inside radius (56 cm (22.04 in.))
of inner LiH layer (10 cm (3.94 in.) thick) of a
passively cooled SNAP-8 reactor shield.

For instance, for lithium hydride at 811 K (1000° F) in a container having a 0.363-centimeter (0.143-in.) thick wall, about 10 percent of the hydrogen would be lost in a 10 000-hour period in a near-earth orbit as a result of micrometeroid damage (ref. 21). At a temperature of 644 K (700° F), with a wall thickness of 0.20 centimeter (0.08 in.) the hydrogen loss would be much less than 1 percent. This substantial reduction of hydrogen loss is a result of the rapid reduction of the LiH dissociation pressure with lower temperatures (see table IV).

Mechanical Properties

The room-temperature properties (ref. 19) of cold-pressed and sintered LiH are

Ultimate compressive strength, N/m^2 (psi) . . . . 1.01×10^8 (14 600)
Ultimate tensile strength, N/m^2 (psi) . . . . 1.84×10^7 (2670)
Modulus of elasticity (tension), N/m^2 (psi) . . . . 7.24×10^{10} (10.5×10^6)
Proportional limit, N/m^2 (psi) . . . . 6.60×10^6 (962)
Bulk modulus, N/m^2 (psi) . . . . 4.55×10^{10} (6.6×10^6)

Reference 19 also gives the ultimate strength of LiH (cold pressed) at elevated temperatures, reproduced here in table VI. Lithium hydride is very brittle near room temperature. At elevated temperatures the elongation on fracture shows that LiH goes from a very plastic material at 533 K (500° F) to a material which readily tears apart at 727 K

Temperature		Compressive	Tensile
K	°F	Ultimate tensile strength, N/m^2 (psi)	
533	500	------------	1.3×10^7 (1916)
538	510	5.7×10^7 (8250)	------------
555	540	3.2×10^7 (4670)	------------
644	700	------------	5.7×10^6 (830)
688	780	1.5×10^7 (2190)	------------
727	850	------------	2.1×10^6 (310)

(850° F). The elongation at 533 K (500° F) was about 30 percent, compared to 10 percent at 644 K (700° F) and 2 percent at 727 K (850° F).

Compatibility and Corrosion

The successful containment of LiH during casting and at elevated temperatures for extended periods of time has been investigated (refs. 18, 19, and 22). The presence of impurities, including those caused by exposure of LiH to air, may profoundly affect the corrosive character. Lithium oxide and lithium chloride are extremely corrosive to most metals and ceramics. Lithium oxide is a common impurity resulting from undue exposure to air. Lithium chloride is present in most commercial LiH in varying amounts.

Reference 19 evaluates the LiH corrosion effects on austenitic stainless steels. The material used for these tests was high-purity LiH in the form of cold-pressed pellets. The impurities in the LiH were not given. Prior to exposure at temperature, the specimens were loaded in the stainless-steel capsules in a dry argon atmosphere, outgassed at test temperatures, backfilled with argon, and then sealed by welding. The capsules were subjected to temperatures of 755, 922, and 1033 K (900°, 1200°, and 1400° F) for time periods as long as 2000 hours. In general, the 300 series and 19-9 DL stainless steels reacted in a bulk manner to form a reaction layer or a zone within the exposed material. The specimens did not show evidence of intergranular penetration, intergranular corrosion, or major surface recession. The depth of the reaction zone, as measured from photomicrographs, ranged from nil to 0.127 millimeters (0.005 in.). However, microhardness traverses revealed that the reaction zones were deeper (a maximum depth of 0.457 mm (0.018 in.) for 2000-hr exposure at 922 K (1200° F)).

In reference 19, tensile test specimen blanks made of the austenitic stainless steels were exposed to LiH at elevated temperatures, 755 to 1033 K (900° to 1400° F). The effect of exposure to LiH was, in general, to increase the ultimate and yield tensile strengths and to decrease the reduction in area and elongation.

Several nickel- and cobalt-base braze alloys were exposed to molten LiH for periods as long as 2000 hours. The only braze alloy found to be compatible with molten LiH (for as long as 4000 hr) was Haynes Stellite 157 (a cobalt-base alloy). Most nickel-base braze alloys are unsuitable since nickel is very reactive with molten LiH (ref. 19).

Radiation Damage

Lithium hydride that is either not outgassed or has become contaminated with moisture may exhibit significant radiation-induced swelling (i.e., >10 vol. %) with even modest radiation dosages. For outgassed and noncontaminated LiH, volumetric swelling can be limited to less than 2 percent in a reactor radiation environment (neutrons and gammas) with fast neutron doses of 10^{20} nvt (E > 0.1 MeV) if the temperature of the LiH is maintained above about 589 K (600° F) (ref. 22). For a gamma dose of 25 gigarads from cobalt-60 (E ~ 1.25 MeV), the volumetric swelling can also be controlled to less than 2 percent if the LiH temperature is maintained above 589 K (600° F) (ref. 23).

APPENDIX B

PROPERTIES OF URANIUM

This appendix contains the pertinent mechanical properties of uranium and some of its alloys as they apply to nuclear reactor shielding.

The mechanical properties of all uranium materials can vary widely depending on strain rate, material purity, grain size, fabrication history, and so forth. The variations in ultimate strength are shown as the banded areas in figure 8. The lower values are, in general, caused by cast material, high impurities, and large grain size. These same factors also affect the elongation properties, as shown in table VII.
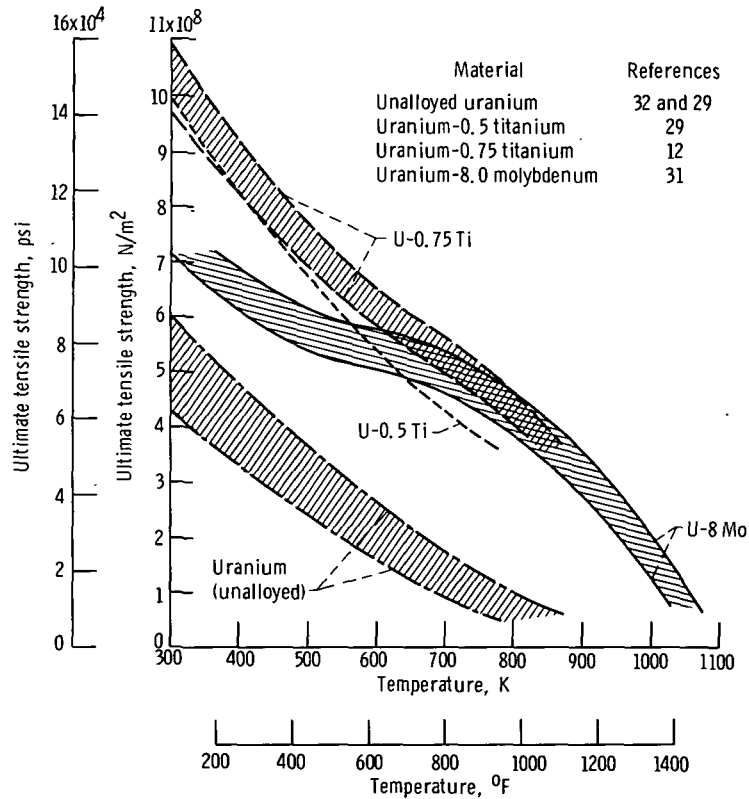

Figure 8. - Ultimate tensile strength of uranium and selected uranium alloys.

TABLE VII. - ELONGATION PROPERTIES OF URANIUM MATERIALS

(a) Room-temperature elongation

Material	Condition	Elongation, percent	Reference
Unalloyed uranium	Forged, beta quenched, and aged	6 to 16	29
	Forged, beta quenched, aged, and welded	2 to 5	29
	Rolled and alpha annealed at 895 K (1150° F)	13 to 20	30
	Rolled, alpha annealed at 895 K (1150° F), and welded	6 to 8	30
U-0.5Ti	Rolled and beta quenched	6	(a)
	Rolled and gamma quenched	15 to 19	(a)
	Forged and gamma quenched	0 to 4	29
	Forged, gamma quenched, and welded	1	29
U-0.72Ti	Rolled and gamma quenched	6	(a)
	Cast, gamma quenched, and welded	2	12
U-8Mo	Rolled and gamma quenched	1	31

[a]Private communication from J. E. Batch, Union Carbide Corp., Nuclear Division, Y-12 Plant, Oak Ridge, Tenn., Oct. 12, 1971.

TABLE VII. - Concluded. ELONGATION PROPERTIES OF URANIUM MATERIALS

(b) Elevated-temperature elongation

Material	Condition	Elongation, percent			Reference
		At 473 K (390° F)	At 673 K (750° F)	At 867 K (1100° F)	
Unalloyed uranium	Cast	24	28	42	32
	Rolled and alpha annealed at 895 K (1150° F)	32 to 50	-------	-------	30
	Rolled, alpha annealed at 895 K (1150° F), and welded	25 to 29	-------	-------	30
U-0.5Ti	Rolled and gamma quenched	-------	32	-------	(a)
U-0.72Ti	Rolled and gamma quenched	-------	25	-------	(a)
	Cast and gamma quenched	-------	16 to 25	10 to 15	12
	Cast, gamma quenched, and welded	-------	10	-------	12
U-8Mo	Rolled and gamma quenched	6	9	5	31

[a]Private communication from J. E. Batch, Union Carbide Corp., Nuclear Division, Y-12 Plant, Oak Ridge, Tenn., Oct. 12, 1971.

APPENDIX C

PROPERTIES OF TUNGSTEN HEAVY-METAL ALLOYS

Some of the properties of the W-3.5Ni-1.5Fe heavy-metal alloy are given in table VIII.

TABLE VIII. - PROPERTIES OF 95W-3.5Ni-1.5Fe ALLOY

Property	Reference 17[a]	Reference 33[b]
Density at room temperature, g/cm^3 ($lb/in.^3$)	18.2 (0.657)	18.0 (0.651)
Thermal conductivity at room temperature, $W/(cm)(K)$ ($Btu/(hr)(ft)(^\circ F)$)	----	1.11 (63.9)
Coefficient of thermal expansion, $cm/(cm)(K)$ ($in./(in.)(^\circ F)$):		
At 293 to 673 K (67° to 752° F)	5.2×10^{-6} (9.4×10^{-6})	4.6×10^{-6} (8.3×10^{-6})
At 293 to 1073 K (67° to 1473° F)	----	5.8×10^{-6} (10.4×10^{-6})
Ultimate tensile strength, N/m^2 (psi):		
At room temperature	9.17×10^8 (132 700)	8.96×10^8 (130 000)
At 478 K (400° F)	7.45×10^8 (108 000)	----
At 700 K (800° F)	6.2×10^8 (90 000)	----
At 922 K (1200° F)	5.52×10^8 (80 000)	4.96×10^8 (72 000)
At 1143 K (1600° F)	4.14×10^8 (60 000)	3.1×10^8 (45 000)
Yield tensile strength (0.2 percent offset) at room temperature, N/m^2 (psi)	6.02×10^8 (87 300)	6.2×10^8 (90 000)
Elongation, percent:		
At room temperature	22	15
At 478 K (400° F)	37	----
At 700 K (800° F)	40	----
At 922 K (1200° F)	20	9.5
At 1143 K (1600° F)	12	5
Compressive yield at room temperature, N/m^2 (psi)	6.79×10^8 (98 400)	6.2×10^8 (90 000)
Shear strength at room temperature, N/m^2 (psi)	7.1×10^8 (103 000)	----
Modulus of elasticity at room temperature, N/m^2 (psi)	3.66×10^{11} (53×10^6)	----
Unnotched impact (Charpy) at room temperature, J(ft-lb)	19 to 58.2 (14 to 43)	54.2 (40)
Notched impact (Charpy) at room temperature, J(ft-lb)	1.9 to 2.4 (1.4 to 1.8)	4.7 (3.5)

[a]Heat treated for 2 hr per in. of thickness at 1273 K (1831° F) in vacuum.
[b]Heat treatment not given.

REFERENCES

1. Berg, G. E.: Final SNAPSHOT Performance Report. Rep. NAA-SR-11934, Atomics International, Aug. 15, 1966.

2. Kaszubinski, Leonard J.; Kacher, Henry F.; and Odar, Fred: Preliminary Designs of Space Power Nuclear Reactor Shields. NASA TM X-2038, 1970.

3. Welch, F. H.: Lithium Hydride Technology II. Fabrication of Shadow Shields for SNAP Applications. Rep. NAA-SR-9400, vol. II, Atomics International, Sept. 30, 1965.

4. Anon.: SNAP Reactor Programs Progress Report. May-July 1970. Rep. AI-AEC-12975, Atomics International, Sept. 15, 1970.

5. Anon.: SNAP Reactor Programs Progress Report. August-October 1970. Rep. AI-AEC-12982, Atomics International, Dec. 15, 1970.

6. Welch, F. H.: Lithium Hydride Technology V: Testing and Examination of SNAP Shadow Shields. Rep. NAA-SR-9400, vol. 5, Atomics International, Aug. 31, 1967.

7. Welch, F. H.: Lithium Hydride Shield Segment Fabrication. Rep. AI-71-40, Atomics International (NASA CR-120782), Sept. 3, 1971.

8. Clifford, C. B.: Design and Fabrication of a Prototype Laminated Uranium Metal Shipping Cask for Large Shipments of Cobalt-60. Rep. KY-521, Paducah Gaseous Diffusion Plant, Apr. 3, 1967.

9. Blasch, Earl B.; Stukenbroeker, George L.; Lusky, Robert J.; Bonilla, Charles F.; and Berger, Harold: The Use of Uranium as a Shielding Material. Nucl. Eng. Des., vol. 13, 1970, pp. 146-182.

10. Pridgeon, J. W.: Stress Corrosion Cracking in Uranium-Molybdenum Alloys. Rep. Y-1417, Union Carbide Nuclear Co., May 6, 1963.

11. Peterson, C. A. W.; and Vandervoort, R. R.: Stress-Cracking in the Uranium-10w/o Molybdenum Alloy. Rep. UCRL-7767, Lawrence Radiation Laboratory, Mar. 13, 1964.

12. Anon.: Y-12 Development Activities Progress Report, November 1967-March 1968. Rep. Y-1666, Union Carbide Corp., May 1, 1969.

13. Murphy, Daniel J.: Some Properties of Uranium-Low Titanium Alloys. Trans. ASM, vol. 50, 1958, pp. 884-904.

14. Nelson, R. M.; and Otey, M. G.: Uranium Coatings. Paducah Plant Laboratory, Aug. 23, 1966.

15. Neeley, A. C.; Googin, J. M.; and Asbury, J. J.: The Role of Powder Metallurgy in the Fabrication of Very Large Parts. Rep. Y-DA-1043, Union Carbide Nuclear Co., Feb. 18, 1966.

16. McIntyre, Ruluff D.: Tungsten-Nickel-Copper Ternary Alloys for High-Temperature Applications. NASA TN D-3015, 1965.

17. Watts, T. D.: Properties of Tungsten and Tungsten Alloys for Reactor Shielding Applications. Rep. Y-1675, Union Carbide Corp., June 19, 1969.

18. Smith, Roger L.; and Miser, James W.: Compilation of the Properties of Lithium Hydride. NASA TM X-483, 1963.

19. Welch, F. H.; Phillips, E. C.; Asquith, J. G.; and Davies, N. F.: Lithium Hydride Technology I: Properties of Lithium Hydride and Corrosion Studies. Rep. NAA-SR-9400, vol. I, Atomics International, May 8, 1964.

20. Jones, P. M. S.; Gibson, R.; and Evans, J. A.: The Permeation and Diffusion of Hydrogen Isotopes Through Stainless Steels. Part 2: EN58B Low Carbon Steel. Rep. AWRE-O-90/65, United Kingdom Atomic Energy Authority, Nov. 1965.

21. Schulz, R. B.: Optimum Lithium Hydride Shield Vessel Design for Protection Against Meteoroid Punctures. Rep. NAA-SR-Memo-11790, Atomics International, May 10, 1966.

22. Welch, F. H.: Lithium Hydride Technology III: Properties of Lithium Hydride for SNAP Shielding Applications. Rep. NAA-SR-9400, vol. III, Atomics International, May 1, 1967.

23. Souers, P. C.; Inai, T.; Blake, T. S.; Penpraze, R. M.; and Leider, H. R.: NMR and Electron Microscopy Studies on Irradiated Lithium Hydride. Rep. UCRL-71835, Lawrence Radiation Lab., 1969.

24. Messer, Charles E.: A Survey Report on Lithium Hydride. Rep. NYO-9470, Tufts University, Oct. 27, 1960.

25. Baxter, W. G.; Welch, F. H.; and Halterman, M. L.: Direct-Air-Cycle Aircraft Nuclear Propulsion Program. Rep. APEX-915, General Electric Co., Mar. 1, 1962.

26. Mueller, William M.; Blackledge, James P.; and Libowitz, George G.: Metal Hydrides. Academic Press, 1969.

27. Holden, A. N.: BATH Shield Material Development. Rep. WANL-TME-2737, Westinghouse Electric Corp., Dec. 1970.

28. Touloukian, Y. S., ed.: Thermophysical Properties of High Temperature Solid Materials. Vol. 5, Macmillan Co., 1967.

29. Anon.: Y-12 Development Activities Progress Report, May-July 1967. Rep. Y-1652, Union Carbide Corp., Jan. 20, 1969.

30. Nelson, V. G.: The Mechanical Properties of Wrought Depleted Uranium and of Welds Joining this Material. Rep. SCDR 58-61, Union Carbide Corp., Mar. 1961.

31. Waldron, M. B.; Burnett, R. C.; and Pug, S. F.: The Mechanical Properties of Uranium-Molybdenum Alloys. Rep. AERE-M/R-2554, United Kingdom Atomic Energy Authority, 1958.

32. Gittus, John H.: Uranium. Butterworth & Co., 1963.

33. Anon.: Data brochure from Mallory Metallurgical Co., Division of P. R. Mallory & Co., Indianapolis, Ind., 1970.

"The aeronautical and space activities of the United States shall be conducted so as to contribute . . . to the expansion of human knowledge of phenomena in the atmosphere and space. The Administration shall provide for the widest practicable and appropriate dissemination of information concerning its activities and the results thereof."
—NATIONAL AERONAUTICS AND SPACE ACT OF 1958

NASA SCIENTIFIC AND TECHNICAL PUBLICATIONS

TECHNICAL REPORTS: Scientific and technical information considered important, complete, and a lasting contribution to existing knowledge.

TECHNICAL NOTES: Information less broad in scope but nevertheless of importance as a contribution to existing knowledge.

TECHNICAL MEMORANDUMS: Information receiving limited distribution because of preliminary data, security classification, or other reasons. Also includes conference proceedings with either limited or unlimited distribution.

CONTRACTOR REPORTS: Scientific and technical information generated under a NASA contract or grant and considered an important contribution to existing knowledge.

TECHNICAL TRANSLATIONS: Information published in a foreign language considered to merit NASA distribution in English.

SPECIAL PUBLICATIONS: Information derived from or of value to NASA activities. Publications include final reports of major projects, monographs, data compilations, handbooks, sourcebooks, and special bibliographies.

TECHNOLOGY UTILIZATION PUBLICATIONS: Information on technology used by NASA that may be of particular interest in commercial and other non-aerospace applications. Publications include Tech Briefs, Technology Utilization Reports and Technology Surveys.

Details on the availability of these publications may be obtained from:

SCIENTIFIC AND TECHNICAL INFORMATION OFFICE

NATIONAL AERONAUTICS AND SPACE ADMINISTRATION

Washington, D.C. 20546