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FUSION BLANKET APPLICATIONS

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FUSION BLANKET APPLICATIONS

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ABSTRACT

Beryllium is an excellent non-fissioning neutron multiplier for fusion breeder and fusion electric blanket applications. This report is a compilation of information related to the use of beryllium with primary emphasis on the fusion breeder application. Beryllium resources, production, fabrication, properties, radiation damage and activation are discussed. A new theoretical model for beryllium swelling is presented.

PREFACE

The Fusion Breeder Program has been interested in the use of beryllium as a non-fissioning neutron multiplier for fission-suppressed blanket applications since 1979. During this period, a number of beryllium-related studies were completed. Most of these studies are described in this report. Among those which are not included here are those which consider the nuclear properties of beryllium and its chemical compatibility in liquid metal systems. Two recent papers which relate to these subjects are as follows:

- J. C. Doyle and J. D. Lee, "Monte Carlo Calculations of 14 MeV Neutron Multiplication in Thick Beryllium Assemblies and Comparison With Experimental Results," UCRL-89597, Lawrence Livermore National Laboratory (1984).
- J. H. DeVan and P. Tortorelli, "Special Topics Reports For The Reference Tandem Mirror Fusion Breeder: Materials Compatibility Issues and Experimental Results," UCID-20166, Vol. 6, Lawrence Livermore National Laboratory (1984).

In addition, R. Haight and his colleagues at LLNL have recently conducted experiments aimed at further resolving nuclear data issues. These results will be published in the proceedings of the Sixth Topical Meeting on the Technology of Fusion Energy to be held in San Francisco on March 3-7, 1985.

I. INTRODUCTION AND MOTIVATION FOR USING BERYLLIUM

The primary motivation for using beryllium in a fusion breeder reactor is its ability to enhance breeding by providing additional neutrons via the $n,2n$ reaction. As shown in Fig. I.1, beryllium can produce more neutrons per fusion than any other non-fissioning multiplier. When used to replace a fissioning multiplier in the blanket of a fusion breeder reactor (also known as the fusion-fission hybrid reactor), beryllium can provide a comparable level of breeding per fusion with much less fission power generation. As a result, the number of fission reactors per unit power in the fusion breeder is increased several-fold and the issues involving fission product generation (i.e., activation and decay afterheat) are reduced several-fold.

When used in a fusion breeder to replace lead, lithium or other non-fissioning multipliers, beryllium can provide higher breeding, but results in a more complicated design. This occurs for two reasons. First, because of its nuclear properties (a very strong neutron moderator/reflector), it must be distributed throughout the fertile breeding zone in a near-homogeneous mixture of beryllium, fertile breeder (i.e., thorium or uranium), or tritium breeder (i.e., a material containing ^6Li) as possible. To obtain maximum breeding efficiency, the amount of structure (parasitic absorber) in the beryllium/fertile fuel/tritium breeder matrix must also be minimized. Second, the beryllium tends to swell due to insitu helium production (the product of the $n,2n$ reaction is two helium atoms) and provision must be made to remove it from the blanket at an interval which is less than the blanket structural lifetime.

These requirements have led to the study of mobile beryllium spherical pebble fuel forms similar to those shown in Fig. I.2 (1,2,3). Recent estimates of the lifetime of beryllium pebbles in fusion breeder blankets (4) indicate that an acceptable lifetime of several years can be expected. Alternatively, the use of a canned beryllium form has been proposed (5) for fusion-electric applications, this multiplier configuration has not been selected for fusion-breeder applications because it introduces more structural absorption into the design than desirable for an efficient fusion breeder blanket.

Figure I.1. A comparison of neutron multiplication for candidate metals. Strictly speaking, ^7Li is not a neutron multiplier but behaves as a neutron multiplier due to the $^7\text{Li} (n,n')\alpha$ reaction which produces tritium and preserves the incident neutron.

(A) SOLID PIN WITH STEEL SPRINGS

(B) SCROLL PIN

(C) SOLID SPHERE WITH SNAP-RING

Figure 1.2. Beryllium/thorium composition fuel form options.

Other attractive aspects of beryllium are its high thermal conductivity, low density, and chemical compatibility with many of the coolants considered for use in fusion breeder reactors. It is compatible with reactor grade helium up to temperatures approaching 700°C, as the very high affinity for O₂ guarantees that a protective oxide coating will be present. At higher temperatures, however, run away oxidation may result in excessive embrittlement. Beryllium and lithium are by themselves very compatible; however, other metals in contact with the lithium must be considered before definitive conclusions can be drawn for a specific system. Recent experiments, conducted for beryllium-lithium-ferritic-steel systems, indicate that a temperature limit ~ 450°C will acceptably limit mass transfer (1). Beryllium and nitrate molten salts (e.g., draw salt) have recently been shown to be compatible to temperatures up to 560°C (6), whereas beryllium and FLiBe are compatible to even higher temperatures (> 650°C). High purity beryllium is known to be compatible with low temperature high purity water, but corrosion is observed if the water is slightly ionized or the beryllium purity approaches commercial grade. Given the lack of conclusive data with high temperature water, it is reasonable to conclude beryllium must be protected from contact with this coolant. Finally, it is known that solid beryllium is not directly compatible with thorium/ uranium bearing molten salts because the beryllium will react with UF₄ in the salt to create BeF₂ and solid uranium-beryllium precipitates in the blanket which can lead to thermal runaway.

Given the motivation described above for considering beryllium, this report provides an overview of beryllium properties and technology as they relate to fusion breeder applications. Section II provides a review of beryllium resources while Section III reviews the fabrication technology for sintered beryllium materials. Section IV is a review of the mechanical, physical, and thermal properties of unirradiated beryllium and Section V provides a review of neutron irradiation effects. A new irradiation swelling model is presented, and irradiation creep is discussed. Section VI provides an assessment of radio-activation of the chemical impurities present in beryllium.

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- 6) Memo from Ralph Moir at Lawrence Livermore National Laboratory, LLNL-BCSS-84-11, 1984.

II. BERYLLIUM RESOURCES FOR FUSION BREEDER APPLICATIONS

II.1 Beryllium Usage

The current and expected usage of beryllium is summarized in Table II.1 (1). The principal current use of beryllium is for beryllium-copper alloys in communications, computers, switching, and other electrical and electronic applications. Such usage is expected to dominate the demand through the year 2000. The annual domestic usage that year is projected to continue at about 275 MT/yr, but is quite uncertain (between 100 and 380 MT/yr in the U.S. and roughly twice that level world-wide). Projections indicate that the nuclear and aerospace usage of beryllium will decline while electrical and electronic usage increases. Through the year 2000, the cumulative usage of beryllium is expected to be quite modest—between 3500 and 6600 MT in the U.S. and about 1.5 times that level world-wide. These projections do not include any possible use of beryllium in fusion reactors (e.g., engineering test reactor).

TABLE II.1. BERYLLIUM DEMAND.

Percent U.S. (1979) Usage

	<u>MT/yr</u>	<u>%</u>
Electrical Uses	102	37
Nuclear Reactors	54	19
Aerospace	50	18
Electronic Components	44	16
Other	<u>24</u>	<u>10</u>
	274	100

Projected (2000) Usage

	<u>U.S.</u>	<u>MT</u> <u>World^a</u>
Annual	100 - 380	170 - 560
Cumulative	3490 - 6610	5470 - 9330

^aIncludes U.S.

II.2 Beryllium Resources

The availability of beryllium is summarized in Table II.2 where U.S. Bureau of Mines (U.S.B.M.) and U.S. Geological Survey (U.S.G.S.) estimates are shown (1,2). These estimates differ primarily in the types of ore bodies which are considered in a given category with the Bureau of Mines data being more conservative in estimating the U.S. reserves and resources and the Geological Survey data being more conservative in the world-wide estimates. In general, the term "resource" describes a quantity of naturally occurring materials in such form that economic extraction is currently or potentially feasible. The term "reserve" refers to those resources which are "known" (as opposed to "not yet discovered") and economically recoverable at the time of classification. It should be noted that the great majority of beryllium reserves in the U.S. were discovered during a period of 10 years (2). Consequently, it is reasonable that future exploration will, doubtless, yield additional deposits. It should also be noted that resources often include estimates of not yet discovered deposits. For example, the U.S.G.S. U.S. resource estimate below assumes that roughly five times the known quantity of reserves can ultimately be identified for economic extraction.

TABLE II.2. BERYLLIUM AVAILABILITY DATA.

	(METRIC TONS)			
	United States		World ^b	
	Reserves	Resources ^a	Reserves	Resources ^a
U. S. Bureau of Mines	25,000	73,000	380,000	1,105,000
U. S. Geological Survey	55,000	282,000	84,000	678,000

^aIncludes Reserves

^bIncludes U.S.

In our analysis, we have used only the Bureau of Mines estimates as these data are both more recent and more conservative with respect to U.S. resources (clearly, the U.S.G.S. data would yield more optimistic results). The U.S.B.M. defines the domestic resource base as follows:

"Known domestic resources of beryllium consist primarily of bertrandite located at Spor Mountain and Gold Hill, Utah, and the Seward Peninsula, Alaska, and of beryl located in the coarse-zoned pegmatites of New England, South Dakota, and Colorado."

In addition to the above resources, sub-economic domestic resources of beryllium occur in fine-grained Spodumene-bearing pegmatites that are being mined for lithium. The U.S.C.S. (2) estimates that about 725,000 MT of beryl (about 30,000 MT beryllium content) are available in such deposits in North and South Carolina alone. These resources are currently uneconomical because the beryl exists in fine particles (i.e., too small to hand sort) and at low concentration (~ 0.5% beryl or 0.020% beryllium). Due to these characteristics, recovery of the contained beryl is considered to be impractical, even in conjunction with floatation of the other minerals (2).

The bertrandite ($4\text{BeO} \cdot 2\text{SiO}_2 \cdot \text{H}_2\text{O}$) which has been mined in Utah since 1969 is found in ~ 7% concentration in the parent ore. The ore is mined, milled, and converted to beryllium hydroxide for later processing into the metal and oxide. Although some beryl ($3\text{BeO} \cdot \text{Al}_2\text{O}_3 \cdot 6\text{SiO}_2$) is imported and converted to beryllium hydroxide, the U.S. is projected to be self-sufficient in beryllium by the year 2000 (2). Currently, hand-sorted Brazilian beryl accounts for ~ 40% of the imports. Other sources of imported beryl are mainland China, India, and Argentina. The world resources of beryllium are broken out by country in Table II.3. These resources consist entirely of beryl, but firm data are scant and the resource projections are considered to be much less certain than the domestic resource projections (1).

In 1981, the domestic price for beryllium metal was about 310 \$/kg (3). About 425 persons are actively employed in the domestic beryllium mining and primary refining industries.

II.3 Fusion Breeder Beryllium Requirements

To investigate the potential usage of beryllium in fusion breeder reactors, we have considered the amount of material which would be required to support a fusion-fission electrical generation system including the reference tandem mirror fusion breeder (3000 MW_f , 1.3 MW/m^2) of reference 4. The

beryllium inventory and throughput which have been estimated for the reference fusion breeder are shown in Table II.4. Recall that this fusion breeder would provide makeup fissile fuel for about 21 GW_e of LWR capacity. Therefore, although each in-core beryllium inventory requires 1.2% of the U.S.B.M. resource, the resource fraction per unit electrical generation is reduced to about 0.05% per GW_e.

TABLE II.3. WORLD BERYLLIUM RESOURCES^a
(Thousand Metric Tons of Contained Beryllium)

	Reserves	Other	Total
North American			
United States ^b	25	48	73
Canada	—	23	23
Mexico	—	2	2
South America			
Argentina	25	47	72
Brazil	140	256	396
Asia: India	64	119	183
Africa			
Mozambique	6	11	17
Rwanda	11	20	31
Republic of South Africa	15	29	44
Uganda	15	26	41
Zaire	7	14	21
Europe: U.S.S.R.	61	111	172
Oceania: Australia	<u>11</u>	<u>19</u>	<u>30</u>
World Total	380	725	1105

^aDerived in collaboration with U.S. Geological Survey.

^bIncludes beryllium content of domestic bertrandite resources.

If we do not recycle (i.e., recover and refabricate) the irradiated beryllium which is discharged from the breeder blanket (5), then the annual requirement for new beryllium is dependent only upon the rate of introduction of new fusion breeders and beryllium replacement requirements for existing plants. Assuming a 1, 2, or 5 year beryllium lifetime due to irradiation damage (6) and reasonable growth in fusion breeder capacity (e.g., on the order of 10-20%/yr), the annual makeup requirement to replace existing inventories will dominate. Without recycle, the resource fraction equivalent to the makeup requirement per fusion breeder reactor can vary between 0.25%/yr (5-year lifetime) and 1.2%/yr (1-year lifetime). Clearly, even the higher beryllium lifetime would not be acceptable for more than the first few fusion reactors unless a large fraction of the beryllium were to be imported or the irradiated beryllium were to be recycled with minimal loss after use in the breeder blanket.

TABLE II.4. BERYLLIUM REQUIREMENTS FOR THE REFERENCE FUSION BREEDER.

Total Fuel Zone Volume ^a	845 m ³
Pebble Volume ^b	507 m ³
Pebble Quantity ^c	36 · 10 ⁶
Beryllium Volume ^d	480 m ³
Beryllium Mass ^d	885 MT
Average Beryllium Lifetime ^e	~ 2 yr
Annual Pebble Throughput	18 · 10 ⁶ yr ⁻¹
Annual Beryllium Mass Throughput	442 MT/yr

- a) 200 m central cell
- b) ~ 60% packing
- c) 3 cm diameter pebbles
- d) 1.84 g/cm³ product density
- e) conservative assumption

With recycle, losses can be limited to the sum of recycle process losses and beryllium burnup losses. Currently, powder metallurgical fabrication process for sintered metal products result in an approximately 7% recycle process loss rate (4). Experts believe that, with sufficient incentive, this loss rate could be reduced to about 1% (7).

The beryllium burnup losses will be small, but will become significant in comparison with the recycle losses when recycle losses are very low. Although the details of beryllium burnup are somewhat design dependent, varying with the blanket geometry and composition, a representative idea of beryllium burnup and its spatial profile can be deduced from existing studies of beryllium multipliers in fission-suppressed fusion breeder applications.

For instance, the beryllium helium production rate as a function of position for the thick multiplier blanket considered in reference 5, is shown in Fig. II.1. For a 1 MW/m^2 neutron wall loading, the helium production rate of reference 5 (in appm per full power year) is given by:

$$\text{He}(x) = 3750 \exp(-0.0841 x)$$

where, in slab geometry, x is the distance from the first wall into the 53 v/o beryllium blanket. The beryllium burnup is half the helium production rate (neglecting any beryllium n, α reactions), and we can express the average percentage of beryllium burned in a reactor per calendar year as:

$$B (\%/yr) = \frac{\Gamma C_f}{10,000 x_1} \int_0^{x_1} \frac{\text{He}(x)}{2} dx = \frac{2.2 \Gamma C_f}{x_1} [1 - e^{-0.08x_1}]$$

where Γ is the neutron wall loading (MW/m^2), C_f is the average plant capacity factor, and $x_1(\text{cm})$ is the beryllium zone thickness. If we simulate the reference fusion breeder (4) as $\Gamma = 1.3 \text{ MW/m}^2$, $C_f = 70\%$, and $x = 40 \text{ cm}$, the resulting beryllium burnup is estimated to be $0.048\%/yr$. This represents 430 kg/yr of beryllium or $0.0006\%/yr$ of the U.S.B.M. estimated resource.

The several contributions to the beryllium requirements for each reference fusion breeder are summarized in Table II.5. The total mass of beryllium required to support a given reactor economy is then:

$$M = NI \left[(1 + f) + T \left(\frac{\epsilon}{\lambda} + \frac{B}{100} \right) \right]$$

where N (GW_e) is the size of the reactor economy, I (MT/GW_e) is the in-core beryllium inventory averaged over both the fusion breeder and its fission reactor clients, f is the beryllium ex-core inventory fraction, λ (yr) is the beryllium lifetime, T (yr) is the duration of the reactor economy, ϵ is the recycle loss fraction, and B (%/yr) is the beryllium burn-up rate (as defined above). For the purpose of this discussion, we will take $f = 1$ to match the optimized fuel cycle described in reference 4.

TABLE II.5. SUMMARY OF CONTRIBUTIONS TO BERYLLIUM REQUIREMENTS
FOR THE REFERENCE FUSION BREEDER.

	Annual (MT/yr)	30-Year Life Cycle (MT)	As Fraction Of U.S.B.M. Resource (%)
Initial Inventory	—	884	1.2
Burnup	0.43	13	0.018
Makeup (2 yr life)			
no-recycle	442	13,260	18.2
recycle (7% loss)	31	928	1.27
recycle (1% loss)	4.4	133	0.18

Figure II.1. Typical beryllium burnup profile.

II.4 Results and Conclusions

Before developing results which include recycle, it is of interest to propose the resource life definitions. The following definitions apply to the combined fusion-fission electricity generation system consisting of fusion breeders and their LWR clients:

- short term service - $60 \text{ GW}_e \times 30 \text{ years}$ (first generation),
- intermediate term service - $500 \text{ GW}_e \times 60 \text{ years}$ (second generation),
- long term service - $1250 \text{ GW}_e \times 200 \text{ years}$.

The short term service requirement is a direct analogy to the first generation of light water fission reactors currently operating in the U.S. The intermediate term service requirement anticipates the impact upon electricity generation in the U.S. which would be required to represent a large fraction (say 30-50%) of the U.S. electrical generation in the 2040 timeframe. The long term service requirement would satisfy an electric economy which is considerably larger than the current U.S. electrical demand (but is similar to the demand which might be expected towards the middle of the next century) for a period roughly equivalent to that of the industrial revolution (i.e., the past 200 years). The intermediate period is considered to be long enough to develop low beryllium inventory fusion breeders, fusion fuel cycles which relax tritium breeding requirements (e.g., semi-catalized D-D fuel cycles), or economical fusion-electric power plants which do not require beryllium for neutron multiplication.

Results which consider initial inventory and make-up requirements (with and without recycle) and assume that only 50% of the U.S.B.M.* domestic resources (i.e., 36,500 MT) could be utilized for U.S. fusion applications are shown in Table II.6.

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* Multiply results by a factor of $282/73 = 3.9$ using U.S.G.S. data of Table II.2.

TABLE II.6 RESULTS OF THE BERYLLIUM RESOURCE ASSESSMENT.
[U.S. Electrical Capacity Which Can Be Supported(GW_e)]

Beryllium Lifetime (yr)	Fusion Economy Duration (yr)	Without Recycle	With Recycle Loss of		
			7%	1%	0%
1	30	29	226	401	461
	60	15	149	353	458
	200	5	58	227	443
2	30	55	303	429	461
	60	29	225	399	458
	200	9	102	300	443
5	30	116	381	448	461
	60	66	324	432	458
	200	22	190	372	443
10	30	185	417	454	461
	60	116	379	445	458
	200	42	266	404	443

Using the above analysis, which considers only the domestic beryllium resource, we reach the following conclusions:

- The short term service criteria requires an achieved beryllium lifetime greater than 2 years if the beryllium is not recycled following use. With recycle, the short term service criteria can be satisfied regardless of the beryllium lifetime.

- The intermediate term service criteria can be nearly satisfied with very efficient recycle ($\sim 1\%$ loss per pass) and a beryllium lifetime greater than 2 years.
- The long term service criteria cannot be satisfied regardless of the lifetime and/or process loss because the beryllium inventory per unit electric generation is too large.

In addition to adopting a presumably conservative estimate of the U.S. beryllium resource, it must be recognized that the above analysis is conservative in several additional respects. First, imported beryllium is not considered in the reactor support calculations. Clearly, some level of imported beryllium is relevant, but economic and political conditions during the latter half of the next century remain unclear. As a first guess, we might assume that the U.S., with about 5% of the world population, might use about 15% of the world beryllium resource, or about 166,000 MT (according to the U.S.B.M. estimate). If, again, half this quantity were available for fusion applications, the resulting 83,000 MT resource would increase the resource lives by 2.3 fold. In this case, with a 2 year beryllium life and a 7% recycle loss, over 500 GW_e can be supported for 60 years and about 230 GW_e can be supported for 200 years. With a 5 year life and a 1% recycle loss, these support values can be increased to 980 and 850 GW_e , respectively. Therefore, considering the usage of a small fraction of the world beryllium resource, the intermediate service life criteria can be easily satisfied and the long term service life criteria can be nearly satisfied.

A second area of conservatism is reflected in the beryllium inventory requirement as related to the very low reference fusion wall loading of 1.3 MW/m^2 (4). More recent fusion breeder designs (8) would operate at wall loadings of about 3 MW/m^2 and would require beryllium inventories which are lower by the ratio of the wall loadings, or $3/1.3 = 2.3$ in the above case. If some beryllium is imported, U.S. resources are more consistent with the U.S.G.S. estimate than the U.S.B.M estimate or another 2.3-fold decrease in the inventory can be achieved, then the long term service life criteria can be satisfied if either efficient recycle ($\sim 1\%$ loss per pass) or a long beryllium lifetime (> 5 years) can be achieved. With a 5 year life and a 1% loss, about

1700 GW_e of electrical capacity can be supported for 200 years. This would require about 75, 3000 MW_f fusion breeders.

Finally, it is important to note that fusion breeder blanket designs and fuel management schemes to date have not emphasized beryllium conservation relative to improved fissile breeding. Minor modifications to existing designs could be expected to reduce the total beryllium inventory by about 30% without an excessive impact on overall breeding. More dramatic changes in blanket design (e.g., substituting ²³²Th for beryllium in the blanket) could lead to further reductions in the beryllium inventory.

Based upon these observations, it appears reasonable to consider beryllium multipliers for the first and subsequent generations of fusion breeder reactor service, but close attention to beryllium exploration, beryllium lifetime, and recycle losses will be required. Since beryllium will activate in the fusion environment (primarily due to impurities), a remote refabrication technology will be required. Without recycle, thick beryllium multipliers should only be considered for use in the first generation of fusion breeder reactors.

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III. BERYLLIUM PRODUCTION AND FABRICATION*

Beryllium, first isolated in 1828, was not used in any practical application until the exceptional properties of beryllium - copper alloys were realized in 1928. Beryllium metal did not go into production until the 1940's, when it became desirable to utilize its nuclear properties.

About 45 beryllium minerals have been identified, but only two are mined: beryl, or $3\text{BeO} \cdot \text{Al}_2\text{O}_3 \cdot 6\text{SiO}_2$, and bertrandite, or $\text{BeO} \cdot 2\text{SiO}_2 \cdot \text{H}_2\text{O}$. Two methods of extracting beryllium hydroxide from beryl have been commercially used: a fluoride process and a sulfide process. The fluoride process is suitable for lower grade ores, and is not currently in use. The sulfide process is used for high grade beryl ($\geq 10\%$ BeO), and is currently used by Brush-Wellman at their Delta, Utah, facility. In the sulfide process, beryl is melted at 1625°C to increase the reactivity to acids and then quenched to form a glass. The glass is heated again in a rotary kiln to further increase the reactivity. The powder product is then heated in sulfuric acid to extract beryllium and aluminum sulfates. The aluminum is removed by neutralization with ammonia. Sodium hydroxide is added to the beryllium compound to form sodium beryllate, which undergoes hydrolysis when steam sparged, leaving beryllium hydroxide.

The Delta mill also processes beryllium hydroxide from bertrandite. The ore is converted to an aqueous slurry and beryllium sulfate is leached out by adding sulfuric acid. An organic solution is then added to produce a water immiscible extractant. The solution is then stripped of beryllium with aqueous ammonium carbonate. Heating then produces beryllium hydroxide.

Beryllium hydroxide is an intermediate material used to produce beryllium metal, alloys, or ceramic. A number of conversion processes are known for producing the metal. The one used in production involves converting the hydroxide to BeF_2 , and then reducing the compound with magnesium to produce beryllium metal pebbles. These pebbles are then vacuum melted in an induction furnace to remove trapped slag and volatile impurities.

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* Much of the general information of this section and Section IV were taken from references 1 and 2.

III.1 Beryllium Processing

After beryllium metal is produced it is melted and casted. Powder processing, the best approach towards enhancing the mechanical properties of beryllium, consists of two steps: comminution and consolidation. Comminution begins by lathe-turning the VIM ingots into chips. The chips are then attrition milled into a metallic powder of the desired mesh. The impact attrition mill is currently used for powder production. A schematic of the process is shown in Fig. III.1. The chips in the feed hopper are accelerated by a gas stream and strike a beryllium target. The chips are comminuted and the gas borne powder goes to the primary classifier. Coarse particles drop out in the secondary classifier. Ultrafine particles, which are high in oxides and intermetallic contaminants, are separated using mesh screens. The powder product in the secondary classifier is used for consolidation, and the coarse particles in the primary classifier are reprocessed along with fresh chips.

The fabrication process selected to produce beryllium pebbles for fusion breeder applications involves the development of an automated line which will cold press pebbles, vacuum sinter them, hot forge them to full density, and vacuum anneal them. Brush-Wellman currently uses the first three steps of this process to produce aircraft brake segments - the only difference is that the process is manually operated because current Be powder is not free flowing and amenable to automated operations. In order to automate this process, a free flowing Be powder is required.

The brake parts produced by Brush have properties as good or better than S-200-E.

Ultimate tensile strength	40 ksi
Yield strength	30 ksi
Elongation (%)	1
Density	1.84 g/cc

We do not expect a free flowing powder to degrade these mechanical properties. In fact, using some of the manufacturing processing currently being discussed with Brush, both the tensile strength and elongation would be expected to increase.

Figure III.1. A schematic of an impact attrition mill.

The equipment needed for production of spherical beryllium pebbles is simple: mechanical presses and powder feeders to make cold press compacts, automated vacuum sintering furnaces for pressureless sintering, mechanical presses for hot sizing the sintered compacts, an automated vacuum furnace for annealing the forged compact and an automated mill to machine the snap ring groove (required to hold the thorium metal fuel pin) to final dimensions.

For fabrication of pebbles too damaged to reinsert into the blanket after an irradiation period, the hot pebbles would be vacuum melted and an automated atomization process (helium atomization of beryllium is currently being developed by Brush Wellman) would be used to first re-manufacture the beryllium powder prior to the cold press step. The entire process would require provision for shielding and remote maintainability. In addition, hooding requirements as per OSHA limits ($2 \mu\text{g}/\text{m}^3$) must be maintained to limit airborne contamination. A beryllium decontamination step (e.g., electro-refining) is not assumed, but might simplify the shielding and remote requirements.

Given a two-year beryllium lifetime, and assuming an automated plant which operates 24-hours/day, 7-days/week, 85% of the time, with no rejects, the production rate required for the reference tandem mirror fusion breeder of reference 3 (see Table II.3) would be 240 balls per minute. While this is a very high production rate for Re parts, it is very low for some powder metal industries (e.g., Ta capacitor manufacturers produce thousands of parts/minute). The beryllium reprocessing line is estimated to lose 7-10% of the beryllium throughput, so a small feedstream is required. A lower loss rate ($\sim 1\%/yr$) appears possible, but will require some development.

As an alternative, lower density parts can be made by cold pressing and sintering. If the parts are sintered to 90% of the theoretical density they are predicted to have 70-80% of their full density conductivity. Some 90% dense Be parts made in this manner have been tested at LLNL. These were characterized with respect to mechanical properties, porosity, etc. It is possible that the lower density parts will have lower swelling and, consequently, longer life.

III.2 Fabrication Cost Estimate

Preliminary cost estimates for the beryllium fabrication process components required to satisfy the requirements of the reference tandem mirror fusion breeder were developed during 1982 (3) and were slightly modified (to include product inspection lines) during 1983. These components, and their estimated direct costs, are listed below:

<u>Processes Components</u>		<u>Direct Cost</u>
8	Presses with feeders	3.4
4	Sintering furnaces	1.7
4	Sizing presses	1.7
4	Annealing furnances	1.7
4	Milling machines	1.7
12	Inspection lines	4.8
	Transfer lines	2.5
	Air handling, filters, etc.	4.3
	Sub Total	21.8
	Remoting Total (100%)	21.8
	TOTAL DIRECT COST	\$43.6M

The above equipment is sufficient to produce or recycle 18 million, 3 cm O.D. pebbles per year, or the entire beryllium inventory in two years. The remoting cost estimate is a guess pending further design work and the inspection line cost shown above anticipates that an inspection test assembly performing a relatively simple mechanical operation (e.g., a compression test or radiography) could be developed to test one pebble per 15-second interval for a capital equipment cost of \$400,000. Approximately 12 such assemblies would be required for the reference blanket (18 million pebbles per year).

These costs can be roughly adapted to alternative blanket requirements by scaling the above component costs by throughput and the characteristic parameter of the pebble (diameter, area, or volume) which drives the magnitude of a specific process and/or cost. For example, the amount of beryllium makeup and the cost of volumetric furnaces can be assumed to scale directly with the pebble volume at a fixed process rate (pebbles/yr). Similarly, the

cost of processing, air handling systems, and plant staff (assumed to be 2 \$M/yr for the above components) can be assumed to scale as the pebble area at a fixed process rate. The cost of transfer lines can be assumed to scale as the pebble diameter, and the cost of inspection can be assumed to scale directly as the process rate (pebbles/yr). For the following discussion, the cost of milling machines (which would scale as the pebble diameter) was deleted as the pebbles for more recent beryllium pebble applications (e.g., blanket designs of reference 4) do not have the cut groove of the Be/Th snap ring design the reference fusion breeder (3).

Using the above approach, the cost of beryllium fabrication as a function of the blanket pebble inventory (grams), lifetime (years), and radius (cm) can be estimated to be

$$C (\$/\text{yr}) = .13 \frac{I}{L} \left[.221 + \frac{.184}{r} + \frac{.054}{r^2} + \frac{.153}{r^3} \right]$$

This cost estimate assumes a total capital cost of 1.8 times the direct cost, a capital cost of 15%/yr and includes makeup of 7% of the beryllium throughput at 350 \$/kg. If the loss can be reduced from 7% to 1%, the leading term of the sum (0.221) is reduced to 0.059 (5).

The above model is illustrated for the molten salt blanket of reference 4 ($I = 715$ MT, $L = 5$ yr) in Fig. III.2. For the selected pebble radius of 0.5 cm, the annual cost becomes 37.5 \$M/yr. This relatively large cost represents roughly 5% of the overall annual cost. It is dominated by inspection and would be reduced to less than 15 \$M/yr if the inspection lines can be eliminated.

III.3 Development Issues

The only special equipment needed for the process line is the air handling system needed to contain the Be powder. The die life should be comparable to other powder metallurgy products (500,000 - 1,000,000 parts/die with punches redressed every ~ 50,000 parts).

Production of a free flowing Be powder suitable for automated operations requires further development. This might be achieved by a new powder manufacturing technique at Brush-Wellman (spherical powder) or the use of binders which can be totally removed during a bake-out prior to sintering.

Development of a pressing technique which produces a uniform dense sphere which can be pressureless sintered to 90-95% of theoretical density is also required. The pressed sphere must be strong enough to permit automated handling.

An efficient inspection/test capability may need to be developed if an adequate product uniformity cannot be achieved without such a capability. The technology for removing the fabrication line is expected to be straightforward, but further study is required.

References

- 1) D. R. Floyd and J. N. Lowe, editors, Beryllium Science and Technology, Plenum Press (1979).
- 2) H. H. Hausner, editor, Beryllium: Its Metallurgy and Properties, University of California Press, Berkley (1965).
- 3) D. H. Berwald, et al., "Fission-Suppressed Hybrid Reactor - The Fusion Breeder," UCID-19638, Lawrence Livermore National Laboratory (1982).
- 4) 1983 Molten Salt Report.

Figure III.2. Annual beryllium cost versus pebble size for the molten salt blanket.

IV. BERYLLIUM PROPERTIES*

Any discussion of beryllium properties must be accompanied with the caveat that the properties measured by experiment are unique to a particular specimen, and care should be exercised when applying the measured value to applications. This caveat is emphasized strongly for beryllium because of the sensitivity to variables such as grain size and orientation, impurity content, thermal history, and testing conditions. Beryllium specimens are available in a variety of forms, the most common being:

- normal purity powder block,
- high purity powder block,
- fine grain powder block,
- high oxide powder block,
- *normal purity ingot sheet*,
- normal purity powder extrusions,
- high purity ingot foil,
- high purity ingot wire.

There are also over a dozen non-standard products.

The most commonly used form of beryllium is normal purity powder block made according to specification S-200 QMV, and consequently most available data is for this product. In the discussion to follow, this product is implied and other products will be called out where applicable. The acronym QMV refers to powder produced from vacuum melted beryllium. The number 200 designates the mesh size of the powder. The mesh size determines the surface area of the powder, which in turn affects the oxide content. S-200 QMV is typically characterized by 1-2% BeO, but lower oxide content can be attained by oxygen control measures.

Typical levels of beryllium contaminants are listed in Table IV.1. A more detailed list of impurities in S-65 beryllium powder is provided in

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* Much of the general information of this section and Section III were taken from references 1 and 2.

Table IV.2 (3). In general, impurity levels depend upon both the or source and its processing. In some cases, the entries of Table IV.2 represent the threshold of detectability.

TABLE IV.1. DENSITIES OF COMMERCIAL GRADES.

	TYPICAL CONTAMINANT LEVELS					DENSITY
	BeO (%)	C (ppm)	Fe (ppm)	Al (ppm)	Si (ppm)	(g cm ⁻³)
Normal-purity block, sheet, extrusion	1.0	1200	1200	400	250	1.857
High-purity block	0.9	200	250	70	60	1.855
Instrument-grade block	4.75	3000	1800	300	1000	1.882
High-purity ingot wire, foil	0.1	200	150	35	55	1.848

IV.1 Physical and Thermal Properties

Some physical and thermal properties of beryllium are listed in Table II.2. Natural beryllium is 100% Be⁹, but other isotopes will be formed during irradiation. The crystal structure at any reasonable operating temperature is hexagonal close packed, giving rise to anisotropic thermal and mechanical properties. The theoretical density is 1.8477 gm/cm³, but measured densities, shown in Table IV.1, are somewhat higher. The remaining thermodynamic properties listed in Table IV.2 are the generally accepted values. Worthy of note is the high specific heat, which is the highest of any metal.

Interpretation of data measuring thermal expansion is made difficult by the anisotropy of the beryllium crystal structure. Typical values for extruded (measured along the extrusion direction) and hot pressed specimens are shown in Fig. IV.1. The hot pressed specimen had a more isotropic grain orientation.

TABLE IV.2. IMPURITIES IN S-65 BERYLLIUM POWDER

Z-No.	Element	Wt.% (10^{-4})
3	Li	3
4	Be	[99.37%]
5	B	6
6	C	200
7	N	50
8	O	7000
9	F	100
11	Na	5
12	Mg	100
13	Al	200
14	Si	300
15	P	10
16	S	200
17	Cl	30
19	K	10
20	Ca	50
21	Sc	2
22	Ti	20
23	V	10
24	Cr	25
25	MN	20
26	Fe	600
27	Co	7
28	Ni	100
29	Cu	40
30	Zn	10
31	Ga	0.2
32	Ge	20

TABLE IV.2. IMPURITIES IN S-65 BERYLLIUM POWDER (con't)

Z-No.	Element	Wt. % (10^{-4})
33	As	0.1
34	Se	1
35	Br	0.5
37	Rb	2
38	Sr	50
39	Y	N/A*
40	Zr	50
41	Nb	50
42	Mo	10
43	Tc	N/A*
44	Ru	5
45	Rh	0.1
46	Pd	2
47	Ag	5
48	Cd	5
49	In	0.05
50	Sn	20
51	Sb	.5
52	Te	5
53	I	0.5
55	Cs	0.1
56	Ba	30
57	La	0.1
58	Ce	1
59	Pr	0.5
60	Nd	3
62	Sm	5
63	Eu	0.1
64	Gd	5

* Not Available

TABLE IV.2. IMPURITIES IN S-65 BERYLLIUM POWDER (con't)

Z-No.	Element	Wt.% (10^{-4})
65	Tb	0.05
66	Dy	0.05
67	Ho	0.2
68	Er	2
69	Tm	0.5
70	Yb	0.1
71	Lu	0.05
72	Hf	0.1
73	Ta	1
74	W	5
75	Re	0.05
76	Os	0.05
77	Ir	0.01
78	Pt	1
79	Av	1
80	Hg	0.2
81	Tl	5
82	Pb	5
83	Bi	5
90	Th	0.05
92	U	15

TABLE IV.3. PHYSICAL PROPERTIES.

Atomic number:	4
Atomic weight (physical scale):	9.013
Isotopes:	Be ⁶ Be ⁷ Be ⁸ Be ⁹ Be ¹⁰ Be ¹¹
Mass:	6.0219 7.01916 8.00785 9.01504 10.1677 11.0277
Crystallographic structure	
to 1254°C (2290°F):	hexagonal close-packed
above 1254°C (2290°F):	body-centered cubic
Lattice parameters:	$a = 2.2854 \text{ \AA}$, $c = 3.5829 \text{ \AA}$
Density:	1.8477 g/cm ³
Melting point:	1283°C (2341°F)
Boiling point:	2484°C (4503°F)
Latent heat of fusion:	2.8 kcal/g atom
Latent heat of sublimation:	76.56 kcal/mol
Latent heat of evaporation:	53.55 kcal/mol
Specific heat:	$C_p \text{ (cal/°K mol)} = 5.068 + 1.361 \times 10^{-3}T + 0.23 \times 10^{-6}T^2 - 1.404 \times 10^{-5}T^{-2} \text{ (25-1254°C)}$
Vapor pressure:	$P \text{ (atm)} = 6.186 + 1.454 \times 10^{-4}T - 16,734/T$
Entropy 25°C:	2.28 cal/°C/mol
Enthalpy:	$\Delta H \text{ (cal/mol)} = 4.32T + 1.09 \times 10^{-3}T^2 - 1490$

Figure IV.1. The thermal expansion of beryllium

The thermal conductivity of beryllium is one of the more sample dependent properties. Some typical measurements are shown in Fig. IV.2 for block and sheet. The high conductivity of beryllium is advantageous in applications where thermal stress must be minimized, but it can be significantly reduced by void or bubble formation. A variety of correction factors with varying degrees of complexity have appeared in the literature, but in most case, the conductivity can be corrected by multiplying by $(1-P)$, where P is the porosity.

The electrical resistivity may be of interest if liquid metal coolants are used. The electrical resistivity is sensitive to both purity and thermal history. Some typical measurements are shown in Fig. IV.3.

IV.2 Mechanical Properties

Up to the 1970's, one of the principal reasons for the restricted use of beryllium was its low ductility and its unspectacular strength. Room temperature yield and ultimate strengths of 200 MN/m^2 and 275 MN/m^2 , respectively, were typical, and ductility no greater than 1% could be expected. A development program initiated in 1972 began to improve these properties dramatically. The breakthroughs were made possible by incorporating impact attrition milling and hot isostatic pressing into the fabrication process, and bringing other elements of the process under tighter control. Coincident with these improvements in fabrication was a detailed study of beryllium on the atomistic and single crystal scale. The theoretical effort has been successful in identifying some of the inherent limitations of beryllium, and is worthy of attention, but for brevity only bulk properties will be considered here. Currently, room temperature ductility of 6 to 7% is commonplace in laboratory specimen, and isotropic ultimate strengths of 550 MN/m^2 are attained. Despite these improvements, beryllium must still be considered as a relatively brittle material. Applications will be limited accordingly.

Figure IV.2. Thermal conductivity of beryllium block and sheet.

Figure IV.3. The electrical resistivity of beryllium.

The tensile properties of beryllium have been measured as a function of fabrication technique, grain size, thermal history, testing condition, surface finish, and strain rate. Prior to the development program mentioned above the data contained much scatter and was not always reproducible. However, with the manufacturing process better controlled much of the scatter has been eliminated. Fig. IV.4 shows the dependence of the ultimate and yield strength on temperature for normal purity hot pressed block manufactured by current techniques. The decrease in strength with increasing temperature is explained by the increasing mobility of dislocation networks. Fig. IV.5 illustrates the improvement attained by careful control of impurities. Iron impurities act as solution strengthening agents in the beryllium matrix. However, when aluminum is present Al-Be-Fe precipitates form and the strengthening effect is sharply reduced. Hence, specimens with more free iron have greater strength. Grain size is another variable to be considered. As shown in Fig. IV.6, strength properties increase with decreasing grain size. The reason is that individual crystals are harder along the boundaries than in the interior, so smaller grains result in greater micro-hardness in a polycrystalline material.

The compression strength is generally the same as the tensile strength, although some differences have been noted. The main difference occurs in the amount of plastic strain attained before failure. Whereas the strain in tension at room temperature is typically 1-2%, the strain in compression may be 10%. One possible explanation is that different slip mechanisms are activated. Also, cracks do not open in compression.

Ductility has been studied more than any other mechanical property of beryllium. A number of attempts have been made to identify methods of increasing the room temperature ductility so as to take advantage of beryllium's excellent thermal characteristics in a broader range of applications. The brittleness at room temperature is due to a strong predominance of basal slip as opposed to other slip mechanisms. Basal dislocations tend to pile up and initiate cleavage cracks, and because the stress required to initiate other slip mechanisms is high, the cracks tend to propagate instead of blunting. As a result, cleavage fracture (i.e., fracture directly through grains) is observed at low temperature.

Figure IV.4. Elevated-temperature tensile strength and tensile yield strength of normal-purity beryllium block.

Figure IV.5. Elevated-temperature tensile strength and tensile yield strength of high-purity beryllium block.

Figure IV.6. Hall-petch plot for high-purity block showing correlation between grain size and strength of beryllium.

Figure IV. 7. Effect of temperature on the critical stresses for slip, twinning, and fracture.

A considerable amount of effort has gone into investigating methods of reducing the activation energy for slip along non-basal planes. The most obvious method is to increase the temperature. Fig. IV.7 illustrates how the critical stress for (1010) prismatic slip in a single crystal approaches that for (0001) basal slip as temperatures increases. As the additional slip mechanism is activated, the critical stress for cleavage fracture increases and the material becomes more ductile. At temperatures greater than 200°C beryllium has an elongation greater than 15%. Fig. IV.8 shows how the ductility varies with temperature for three grades of beryllium. Two lines are shown for the hot pressed block because it is anisotropic, whereas the isopressed block is isotropic. Cleavage failure predominates up to 200°C, ductile fibrous fracture between 200°C and 350°C, and intergranular failure at temperatures above 350°C.

The desire to increase the room temperature cleavage stress has motivated a number of studies concerning the effects of impurities. The research has fairly conclusively shown the only effect of impurities is to increase the work hardening rate rather than decrease the cleavage stress and, thus, purification results in only limited improvement in ductility. It is currently believed the ductile-to-brittle transition temperature cannot be brought down to room temperature or below by controlling impurities. At temperatures above about 500°C, aluminum impurities at the grain boundary initiate fracture by melting.

Grain size is another variable affecting ductility. Finer grains compensate for the unavailable slip systems and at the same time add strength. As seen in Fig. IV.9, the elongation increases inversely with the square root of the grain diameter. The limit to which ductility can be improved by controlling grain size is not yet known, but some believe ductile beryllium at room temperature may be possible with submicron-sized grains.

Figure IV.8. Elevated-temperature elongation of beryllium block.

Figure IV.9. The influence of grain size and BeO content on tensile elongation of beryllium block.

The elastic modulus of beryllium is fairly independent of microstructural variables, and has been measured as 290 GPa at room temperature. It remains nearly constant up to temperatures around 450°C, and then drops off rapidly at higher temperatures. As seen in Fig. IV.10, the proportional limit depends on grain structure, impurity content, and thermal history. The yield point exhibited by the high oxide and normal purity block is generally attributed to rapid multiplication of mobile dislocations brought on by pinning at impurities. The precise behavior at the yield point (type A, B, or C) has been explained by heat treatment effects. At room temperature notch weakening is observed but at temperatures above the ductile-to-brittle transition notch strengthening is observed. The shear modulus has been estimated as 145 GPa.

Poisson's ratio at low stress before plastic deformation sets in is 0.06 to 0.08 in beryllium block at low temperature. At larger stress plastic deformation may set in before 0.2% strain is attained, and Poisson's ratio can increase to 0.5. At higher temperatures Poisson's ratio does not appear to change much. Figures IV.10 and IV.11 show typical stress-strain curves.

If beryllium is to be used at temperatures below the ductile-to-brittle transition temperature then the fracture toughness is of interest in design. In a brittle material the behavior of pre-existing flaws is of paramount importance. The fracture toughness K_{IC} relates the maximum stress σ to the maximum flaw size a by the relation:

$$K_{IC} > \sigma (\pi a)^{1/2}. \quad (1)$$

If either the stress or flaw size cause the right hand side of eq. (1) to exceed the fracture toughness, then failure is predicted to occur. Given that the stress must be a factor n below the yield stress σ_y to avoid yielding, the critical flaw size is $(K_{IC}n/\sigma_y \pi)^2$. If the fracture toughness of a material is low, crack inspection can become painstakingly difficult.

Figure IV.10. Typical stress-strain curves for beryllium block (transverse specimens).

Figure IV.11. Typical stress-strain curve for beryllium sheet.

Figure IV. 12. The effects of temperature on the fracture toughness of S200 grade beryllium.

Attempts have been made to study the effect of microstructural and environmental variables on fracture toughness. Fig. IV.12 illustrates the discrepancy between data taken by three researchers on the effect of temperature. Apparently the ductile-to-brittle transition temperature was exceeded by one of the lots tested, but not by the others at the same temperature. A commonality is in the almost linear increase in fracture toughness with temperature at low temperatures. At room temperature a value of $10 \text{ MPa}\cdot\text{m}^{1/2}$ is often used.

Studies of impurity effects on fracture toughness indicate that an increase in fracture toughness accompanies an increase in purity. In high purity samples, values as high as $18 \text{ MPa}\cdot\text{m}^{1/2}$ have been recorded. Matrix hardeners such as iron tend to decrease the fracture toughness, whereas the effect of BeO is not clear. As with ductility, only a limited improvement in fracture toughness can be attained by purification.

There appears to be a lack of grain size effect on fracture toughness, but as seen in Fig. IV.12, there is an orientation effect. The effect is due to planes of weakness aligning perpendicular to the pressing direction. In isostatically pressed material the fracture toughness is generally isotropic.

The effect of porosity on the fracture toughness of beryllium is opposite that on the strengths and moduli. The fracture toughness increases to twice the fully dense value as the porosity increases to 5%, and then decreases again as the porosity increases further. The mechanism for toughening is thought to be either crack blunting, or that the pores cause multiple crack nucleation and, hence, additional surface area to absorb energy.

When beryllium is used at temperatures above about 500°C , thermal creep must often be considered. The thermal creep rate $\dot{\epsilon}$ is empirically related to stress σ and temperature T by

$$\dot{\epsilon} = K \sigma^n \exp (-Q/RT), \quad (2)$$

where K is a structure sensitive constant, n , is the stress exponent and Q is an activation energy. The parameter R is the gas constant. The empirical constants are highly dependent upon the microstructural variables, with measured values of n ranging from 2 to 4 and measured values of Q ranging from 44

to 276 kcal/mole, respectively. Unfortunately, the available data is still too sparse to construct deformation mechanism maps for beryllium.

Despite the lack of a complete data base, some observations can be made with regard to creep. Fig. IV.13 shows the creep rate for various grades of beryllium at 1000°C. The acronym HIP refers to hot isostatically pressed beryllium. Low melting temperature impurities like aluminum increase the creep rate by promoting grain boundary sliding. On the other hand, a high BeO content reduces creep by inhibiting grain boundary sliding. The effect of grain size has also been studied. For grains larger than about 25 μm , the creep rate decreases inversely with the grain diameter. However, at smaller grain sizes the creep rate increases with the grain diameter for reasons not yet completely understood. Given the high variability observed between different types of beryllium, it can only be suggested that creep measurements on the type of interest be made before use in an application.

Figure IV.13. Creep rates for various grades of beryllium

References

- 1) D. R. Floyd and J. N. Lowe, editors, Beryllium Science and Technology, Plenum Press (1979).
- 2) H. H. Hausner, editor, Beryllium: Its Metallurgy and Properties, University of California Press, Berkley (1965).
- 3) Burt Rowlands, private communication, Royal Ordnance Factory, Cardiff, UK (1984).

V. BERYLLIUM CHARACTERISTICS IN A RADIATION ENVIRONMENT

The lifetime of beryllium in fusion breeder blanket applications is expected to be determined by the interaction between irradiation induced dimensional changes such as swelling and creep and irradiated materials properties such as ductility and fracture toughness (1). In Section V, theoretical models for radiation induced swelling and creep are presented and compared with experimental data, an estimate of irradiation creep is made, and a discussion of beryllium lifetime considerations is given.

V.1 Radiation-Induced Swelling of Beryllium in Fusion Reactors

At the present time, only limited data (2,3) on the effects of fast neutron, high-fluence ($> 10^{22} \text{ n/cm}^2$) irradiation on the dimensional changes of beryllium exists. However, data on beryllium specimens irradiated in both the ETR and ATR have been reported by Beeston and coworkers (4,5). The specimens in ETR were irradiated at about 100°C to fluences up to about $3.5 \times 10^{22} \text{ n/cm}^2$ ($E > 1 \text{ MeV}$), and some were subsequently annealed to higher temperatures to study the formation of helium bubbles. These data provide an important source of information to test the models of radiation-induced dimensional changes in beryllium which were developed during the course of this study. However, it is important to first discuss some basic aspects of diffusion, displacement damage, helium production, and radiation-induced dimensional changes.

Since beryllium is an anisotropic metal with an hexagonal closed-packed crystal structure, the following dimensional changes can be produced by neutron irradiation:

- a) void swelling,
- b) bubble swelling,
- c) anisotropic or irradiation growth,
- d) irradiation creep,
- e) microcrack formation.

Void swelling has been observed to occur in most metals subject to fast-neutron irradiation at elevated temperatures. Two conditions must, in general, be met for void swelling to occur. First, the irradiation temperature must be higher than the temperature for vacancy migration but lower than the

temperature for significant self-diffusion. Second, the ratio of He/dpa must be small or else swelling will be gas-driven (i.e., bubble swelling will be the dominant mechanism for swelling at high He/dpa ratios).

During void swelling, self-interstitials form dislocation loops or are absorbed at edge dislocations. If the dislocation loops and the edge dislocations have a preferred crystallographic orientation, as expected for anisotropic metals such as beryllium, void swelling is accompanied by anisotropic growth or irradiation growth. The latter may even occur in the absence of void formation if vacancies are absorbed preferentially at dislocations with Burgers vectors of different orientation than those for dislocations which capture self-interstitials. Anisotropic growth in polycrystalline materials will eventually lead to microcrack formation, and hence, swelling. At the same time, the fracture strength will be reduced.

Radiation-enhanced creep requires displacement damage, and it increases with the void swelling rate.

For an assessment of all these dimensional changes, three processes are of crucial importance: the vacancy diffusion rate, the displacement rate, and the helium production rate. It might be expected that the helium production rate in beryllium dominates all other considerations.

V.1.1 Diffusion Properties of Beryllium

Self-diffusion data in pure metals have recently been reviewed by Peterson (6). For self-diffusion parallel to the C-axis in Be

$$D_{\parallel} = 0.62 \exp \{-1.71 \text{ eV/kT}\} \text{ cm}^2/\text{s},$$

and perpendicular to the C-axis,

$$D_{\perp} = 0.52 \exp \{-1.63 \text{ eV/kT}\} \text{ cm}^2/\text{s}$$

In comparison to the melting point of Be, $T_m \approx 1550 \text{ K}$, the activation energies for self-diffusion are low. For example, the activation energy for nickel is 2.8 eV, so that at about half of its melting point or at 873 K, where self-diffusion is important for nickel, the self-diffusion coefficient is equal to

$2.2 \times 10^{-17} \text{ cm}^2/\text{s}$. In comparison, $D_{\parallel} = 2.1 \times 10^{-17} \text{ cm}^2/\text{s}$ and $D_{\perp} = 1.03 \times 10^{-16} \text{ cm}^2/\text{s}$ in Be at 523 K. As a result of the low activation energy, self-diffusion becomes a significant process in Be at about 250°C.

V.1.2 Displacement and Helium Production Rates

The energy of primary recoil atoms (PKA) produced in beryllium by fast neutrons is large compared to other metals as a result of the small atomic weight. As seen from Table V.1, the average primary recoil energy produced

Table V.1 Displacements in Beryllium.

Neutron Energy (MeV)	Average Primary Recoil Energy (MeV)	Damage Efficiency	Number of Displacements
1	0.18	0.075	355
14	2.52	0.010	663

by 1 MeV and 14 MeV neutrons is 0.18 MeV and 2.52 MeV, respectively. For these high PKA energies, the electronic stopping power is dominant for beryllium, which results in the very low damage efficiencies shown in the third column of Table V.1. The number of displacements were obtained by estimating the displacement energy from the empirical relationship $E_d = k_B T_m$, where k_B is the Boltzmann constant and the T_m the melting point. For beryllium, this estimate gives $E_d = 19 \text{ eV}$. The elastic scattering cross section of neutrons with beryllium is about 1 barn and 2.5 barn for 1 MeV and 14 MeV neutrons, respectively. As a result, a neutron fluence of 10^{22} n/cm^2 ($E = 1 \text{ MeV}$) produces about 3.5 dpa, whereas a fusion fluence of 10^{22} n/cm^2 ($E = 14 \text{ MeV}$) generates about 16.6 dpa.

The production rate of helium in the ATR and ETR was estimated by Beeston (1) to be about 4700 appm He per 10^{22} n/cm^2 ($E > 1 \text{ MeV}$). Hence, in

these fission reactors, the He/dpa ratio is on the order of 1300 appm He/dpa, where dpa indicates that only displacements by neutrons and excluding those by the decay of the nucleus are taken into account.

In the beryllium moderated fusion breeder blanket, Beeston (1) estimates the helium production rate to be in the range of 5500-7000 appm per 10^{22} n/cm² ($E > 1$ MeV). The higher value assumes complete burnup of the Li daughter into helium. In this case, the He/dpa ratio is expected to be substantially greater than 420 He/dpa in the front of the blanket (where the neutron spectrum is fastest, but not 14 MeV) and somewhat less than 1570 He/dpa in the back of the blanket (where the uncollided fusion source strength is low).

To put these helium production and displacement rates in proper perspective, we note that helium ion implantation for blistering studies results in a ratio of about 1000 appm He/dpa (7). As these studies have shown, swelling in this regime is dominated by gas-driven or bubble swelling for all irradiation temperatures. We conclude, therefore, that the predominant damage mechanism in both fission and fusion neutron irradiations of Be is helium bubble swelling. All other processes mentioned above will play an insignificant role in determining the radiation lifetime of beryllium components in fusion reactors.

V.1.3 Helium Bubble Swelling of Neutron Irradiated Beryllium

The formation and growth of helium bubbles in metals can take place by both athermal and thermal processes, depending on the irradiation temperature. When self-diffusion is insignificant, the growth proceeds by the emission of self-interstitials which remained trapped near the bubble, or by the punching of interstitial loops. For both cases there exists a pressure in the bubble which greatly exceeds the bubble equilibrium pressure of $2\gamma/r$, where r is the bubble radius and γ the surface energy. In fact, the helium density within the bubble remains nearly equal to the solid packing density, and swelling is simply equal to

$$\frac{\Delta V}{V_0} = \frac{V_{\text{He}}}{V_0} C_{\text{He}} \quad (1)$$

where Ω is the atomic volume of the host metal, V_{He} is the molar volume of dissolved helium, and C_{He} is the atomic fraction of helium.

Equation (1) has been fitted to the swelling data reported by Beeston (4) for beryllium irradiated at about 100°C, with the result that

$$\frac{\Delta V}{V_0} (\%) = [0.52 \pm 0.23] \phi t, \quad (2)$$

where ϕt is in units of 10^{22} n/cm^2 ($E > 1 \text{ MeV}$). Per unit of this fluence, about 0.9 at % of helium is produced, so that equation (2) can also be written

$$\frac{\Delta V}{V_0} (\%) = [0.58 \pm 0.25] C_{\text{He}} (\text{at } \%). \quad (3)$$

This equation implies a molar volume of helium in Be of

$$V_{\text{He}} = [0.58 \pm 0.25] \Omega \quad (4)$$

which is in excellent agreement with the value of 0.62Ω obtained from computer simulation studies of helium clusters in Cu by Baskes and Holbrook (8).

Equation (3) is expected to be applicable for irradiation temperatures $\leq 250^\circ\text{C}$ based on the arguments given above on self-diffusion.

At higher temperatures self-diffusion allows bubbles to grow by absorption of thermal vacancies. In this case, the bubble pressure remains close to the equilibrium pressure

$$P = 2 \gamma / r \quad (5)$$

If this equation is multiplied by the bubble fractional volume then the left hand side becomes equal to $C_{\text{He}} kT$ if an ideal gas law is assumed. Since the bubble fractional volume can also be written as $\Delta V/V_0 = (4\pi/3)Nr^3$, where N is the bubble density, we obtain

$$\frac{\Delta V}{V} (\%) = 100 \left(\frac{3}{4\pi N} \right)^{1/2} \left(\frac{C}{2\gamma} kT \right)^{3/2} \quad (6)$$

where C = atoms He per cm^3 of beryllium.

Beeston (4) has experimentally determined the bubble density in post-irradiation annealed beryllium samples. For samples irradiated at about 100°C no visible bubbles were found until the annealing temperatures exceeded 300°C, which is close to the estimated temperature where self-diffusion becomes important.

The measured bubble densities for annealing temperatures between 400°C and 600°C are given by the empirical equation

$$N = 1.4 \times 10^{14} \exp (0.41 \text{ eV/kT}) \text{ cm}^{-3}. \quad (7)$$

The reported measurements of the surface energy γ of beryllium are 1 J/m² (9) and 1.6 J/m² (10). However, Muir (11) lists a value of 2 J/m², which would be in better agreement with empirical correlations based upon melting temperature and cohesive energy.

Accordingly, bubble swelling predictions were made for surface energy values of 1 J/m² and 2 J/m² by using equations (6) and (7). At low temperatures, equation (6) predicts swelling values below the values obtained with equation (3). Hence, it is assumed that the actual bubble swelling is determined by whichever equation provides the largest value. The results are shown in Figs. V.1 and V.2 for a surface energy of 1 J/m² and 2 J/m², respectively. Measured swelling values for the annealed samples are also indicated in these figures together with the estimated helium concentrations.

It is seen that better agreement between the theoretical predictions and the measured data is obtained when a value of 2 J/m² is assumed for the surface energy of beryllium. It is also found that the transition temperature where equilibrium bubble swelling becomes equal to the "solid helium" swelling shifts to lower values with the increase in the helium concentration.

V.1.4 Strength and Swelling of Porous Beryllium

The strength and swelling of porous beryllium upon irradiation is a question of some interest. Lower density beryllium (70 to 90% of theoretical) has been suggested to reduce swelling with the possibility of longer service life in fusion reactors. As seen in Fig. V.3, the strength of lower density

Figure V.1. Results of the swelling model assuming a surface energy of 1 J/cm^2 .

Figure V.2. Results of the swelling model assuming a surface energy of 2 J/cm^2 .

beryllium is less than higher density beryllium (12), thus, a reduction in irradiation swelling would have to compensate for the lower initial strength during irradiation in order that any failure mechanism based on strength would predict a longer service life than fully dense beryllium.

To examine the question of a reduction in irradiation swelling, the properties of a porous beryllium (13) will be taken as

Powder* Particle Size (μm)	Sintered Density	Average Gross Permeability (in^2)	Uni- formity (%)	Yield Strength (ksi)	Mean Pore Size (μm)	Open Porosity (%)
-74 + 10	1.607 g/cm ³ (86% of theoretical)	2.9×10^{-11}	± 10	22	2.5	22

* Elimination of fines and controlled particle distribution were necessary for optimum permeability and uniformity. Powder would fall through a 74 μm screen but stops on a 10 μm screen.

For porous beryllium, the swelling will depend upon the loss of helium to the open pores, the helium trapped in the closed pores, and the helium which agglomerates in the grains and grain boundaries. From the above data and microstructure (11), an estimate of the times for a diffusion path length, L, of 10 microns (conservative distance from the center of a particle to an open pore) can be obtained from the self-diffusion equation

$$D_L = 0.52 \exp (-1.63 \text{ eV/kT}) \text{ cm}^2/\text{s}$$

at 250°C (523 K) and at 400°C (673 K). At 250°C, $D_L = 1.0 \times 10^{-16} \text{ cm}^2/\text{s}$, and at 400°C, $D_L = 3.2 \times 10^{-13} \text{ cm}^2/\text{s}$.

Since the time, t , is given by $t = L^2/D$, the time at 250°C is 9.71×10^9 s (310 years) and at 400°C is 2.3×10^6 s (36 days) for a helium atom to migrate a diffusion path length of 10 microns. Thus, for temperatures approaching 400°C, some helium can reach the open pores and be removed from the material.

However, since helium will also collect in the closed pores, it is instructive to estimate the fluence at which the closed porosity is filled with helium. Assuming spherical pores, the volume of a pore 2.5 microns in diameter is $8.2 \cdot 10^{-12}$ cm³. The number of gas atoms in a bubble this size at equilibrium is given (14) by

$$= \frac{8\pi r^2}{3kT} = \frac{8\pi \times 2000 \times (1.25 \cdot 10^{-4})^2}{3 \times 1.38 \times 10^{-16} \times 673} = 2.82 \times 10^9 \frac{\text{Helium Atoms}}{\text{Pore}}$$

The number of helium atoms per cm³ of beryllium is obtained as the number of atoms per pore divided by the volume of a pore times the porosity times the fraction of closed pores.

$$\frac{\text{Atoms He}}{\text{cm}^3} = \frac{2.82 \times 10^9}{8.2 \cdot 10^{-12}} \times 0.14 \times 0.78 = 3.75 \times 10^{19}$$

From the helium production rate of 5500 appm per 10^{22} n/cm² ($E > 1$ MeV) from Section V.1.2 for a fusion blanket, the production of helium atoms/per cm³ per 10^{22} n/cm² fluence is calculated as

$$\frac{0.005 \times 6.023 \times 10^{23} \times 1.607}{9.01} = 5.4 \times 10^{20} \frac{\text{Atoms Helium/cm}^3}{10^{22} \text{ n/cm}^2}$$

Hence, the closed pore porosity is filled from the production of helium due to a fluence of

$$\frac{3.75 \times 10^{19}}{5.4 \times 10^{20}} \times 10^{22} = 6.9 \times 10^{20} \text{ n/cm}^2$$

The large diameter of the closed pores will produce increased swelling at equivalent helium contents (as can be deduced from Fig. V.1). Therefore, for lifetime expected fluence $> 10^{22}$ n/cm², pressure buildup in the closed pores may produce even larger swelling than for higher density beryllium.

Thus, it can be reasonably expected that increased closed pore swelling would counter the loss of helium due to the increased surface area from open pores, making it questionable whether porous beryllium with reduced strength will give a longer service life than fully dense beryllium.

Figure V.3. Yield strengths of porous high-purity beryllium compacts produced using optimum isopressing and sintering methods.

V.2 Estimate of Irradiation Creep

Creep under neutron irradiation is composed of three contributions,

$$\dot{\epsilon} = \dot{\epsilon}_i + \dot{\epsilon}_s + \dot{\epsilon}_{th},$$

where $\dot{\epsilon}_{th}$ is the thermal creep rate, $\dot{\epsilon}_s$ the creep rate occurring in conjunction with void or bubble swelling, and $\dot{\epsilon}_i$ is the component which is independent of swelling. The latter component is generally linear in the stress and directly proportional to the displacement rate. It exceeds the thermal creep rate $\dot{\epsilon}_{th}$ only when self-diffusion is insignificant. Based on the above discussion, $\dot{\epsilon}_i < \dot{\epsilon}_{th}$ ($T \approx 250^\circ\text{C}$), and it is negligible in comparison to $\dot{\epsilon}_{th}$ above 250°C .

The creep contribution, $\dot{\epsilon}_s$, which is coupled to swelling, is significant only at temperatures where near-equilibrium bubbles form. In the present case, this requires irradiation temperatures in excess of about 250°C . As a result, $\dot{\epsilon}_s$ can be considered as an enhancement of thermal creep or as the effect of stress on equilibrium bubble growth. For any given direction of the applied stress

$$\dot{\epsilon}_s \lesssim \frac{1}{3} \left[\frac{\dot{\Delta V}}{V_0} (\sigma) - \frac{\dot{\Delta V}}{V_0} (0) \right] \lesssim \frac{1}{3} \frac{\dot{\Delta V}}{V_0} (\sigma).$$

V.3 Beryllium Lifetime Considerations

Failure of beryllium in the reactor blanket can occur in various ways depending on the beryllium form (pebbles, rods, etc.), the strength and ductility of beryllium, the stress in the member, and the environment. The loading stresses on the beryllium pebbles in a breeder blanket are produced from thermal expansion, differential swelling from irradiation, and external forces (weight, hydraulic pressure, etc.).

For components containing pebbles or rods, a maximum-stress theory of failure, modified by adequate ductility and by fracture mechanics considerations, can be used to conservatively determine the lifetime for a particular design concept (1). On the basis of maximum-stress theory, failure

can occur when one of the principal stresses from loading reaches the minimum yield strength (in tension or compression) of the irradiated beryllium. However, surface cracks and defects that may propagate must be unstable according to the stress intensity factor and fracture toughness of the irradiated beryllium. As indicated in Section V.2, the swelling and creep may be coupled such that stress relaxation may reduce some of the differential swelling stresses. These effects are important since the differential swelling stresses reduce the calculated lifetime of the beryllium in present calculational methods.

Section V presents our current understanding of irradiation effects for beryllium, but reliable lifetime estimates will require additional beryllium property data at high-fluence and at 350°C-600°C. The properties measured should include strength, ductility, swelling, and effect of creep on differential swelling stresses.

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VI. BERYLLIUM ACTIVATION

Although irradiated beryllium itself does not pose a gamma emission hazard, low levels of impurities in beryllium will become activated to the extent that glove box re-fabrication immediately following irradiation will not be possible. Also, tritium generated in the beryllium matrix provides a potential radiological hazard.

VI.1 Induced Activation of Beryllium Impurities

Neutron activation of trace amounts of impurities in beryllium can have significant effects on waste classification and remote handling requirements for components containing beryllium. The assays of fourteen samples of S.65 beryllium powders (1) were used to define a reference composition of a beryllium component. This composition is as shown in Table IV.2. The activation of this material in a reference neutron environment was calculated to determine the waste disposal rating (WDR) and the remote maintenance rating (RMR) (2,3,4). The WDR is based upon the limits set in 10CFR61 for concentrations of certain long-lived radionuclides for near-surface waste disposal. The WDR is defined as the sum of the ratios of the concentrations of those nuclides in the waste to the corresponding limits. If $WDR < 1$, then the waste can be disposed according to the waste classification defined by the limits 10CFR61. The RMR is defined as the radiation dose rate in mrem/hr at the surface of a uniformly activated, thick, infinite slab with the same composition and density as the material being considered. The reference neutron environment is the one used for the MARS (5) first wall zone, two year exposure, and 5 MW/m² wall loading. The FORIG code was used to do the calculations (6).

The results for the WDR are shown in Table VI.1. From this it can be seen that niobium and silver are the impurities causing the $WDR > 1$ and thereby preventing near-surface waste disposal in class C. From Table IV.2, the assumed amounts of niobium and silver are 50 and 5 ppm, respectively. However, the assays (1) indicate that it is possible to have beryllium with less than 5 ppm of niobium and 1ppm of silver. When these smaller concentrations are used for the composition, the total WDR for beryllium decreases to 0.8. This would permit near-surface disposal of the beryllium

Table VI.1 Waste Disposal Ratings For Beryllium^a

Nuclide	Time After Removal From Reactor		
	Shutdown	10 years	100 years
C14	0.03	0.03	0.03
N159	0.0004	0.0004	0.0004
N163	0.03	0.03	0.03
Sr90	1×10^{-5}	1×10^{-5}	1×10^{-6}
Nb94	4.9	4.9	4.9
Tc99	1.3×10^{-3}	1.3×10^{-3}	1.3×10^{-3}
Nb92	6×10^{-4}	6×10^{-4}	6×10^{-4}
Ag108m	1.00	1.00	0.62
As26	0.05	0.05	0.05
Bi207	6.6×10^{-5}	5.5×10^{-5}	1.8×10^{-5}
Bi208	0.03	0.03	0.03
Total	6.0	6.0	5.7

a) Based on proposed limits in reference 4.

without any processing. Except for niobium and silver, there are no other elements in any of the assays with concentrations that would cause the beryllium to exceed class C waste ratings.

Even though activated beryllium can be treated as class C waste for near-surface disposal, it still presents problems in handling because of its gamma radiations. This is qualitatively shown by the RMR. The FORIG output for RMR is attached as Table VI.2. Table VI.2 illustrates the complexity of the radiation dose spectrum. The total dose levels shown in the last line are quite high. Even after 100 years of decay, the dose from a semi-infinite slab would be 2.5×10^3 mrem. This is one thousand times larger than the 2.5 mrem occupational limit for handling without protective measures. Even with corrections to the values of RMR to reflect values for finite sized components (3), protective measures and/or remote handling will be required to remove beryllium components from a reactor for maintenance and repair operations.

VI. 2 Insitu Tritium Generation

Nuclear reactions in beryllium produce a small amount of tritium. This is of no consequence in blankets where the principal tritium breeding material and beryllium are together (e.g., a liquid lithium cooled packed bed of beryllium). In helium cooled blankets, in which the tritium breeding material is isolated from the helium coolant to minimize tritium contamination, tritium generation in the beryllium and, thus, tritium contamination of the helium may well provide a significant radiological hazard.

Tritium is generated in the beryllium directly by the $\text{Be}(n,t)^7\text{Li}$ reaction and indirectly by the $\text{Be}(n,\alpha)\text{Li}^6$ reaction followed by $\text{Li}^6(n,t)^4\text{He}$ reactions. In an early study of a beryllium multiplier blanket (7), the principal beryllium reactions per D-T source neutron were calculated to be:

Be (n,2n)	1.61
Be (n, γ)	0.0025
Be (n, α)	0.159
Be (n,t)	0.023

Table VI.2. FORIG Results

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dose rate for activation products

output unit = 4

decay of 0.5 cm first wall - bas
1.00000e+02 cm thickness of 0.1 with a density of 1.35000e+00 g/cc. Volume is 1.00000e+00 m³.

principal dose rate sources in mR/hr
basis=

nuclide	discharge	1.0hr	1.0d	7.0d	30.0d	180.0d	1.0yr	5.0yr	10.0yr	25.0yr	50.0yr	100.0yr
b 13	9.914e+02	0	0	0	0	0	0	0	0	0	0	0
n 13	1.561e+05	2.409e+03	0	0	0	0	0	0	0	0	0	0
n 14	4.925e+08	0	0	0	0	0	0	0	0	0	0	0
na 23	5.612e+04	4.481e+25	0	0	0	0	0	0	0	0	0	0
na 24	2.695e+04	2.698e+04	2.696e+04	2.694e+04	2.639e+04	3.66e+04	2.067e+04	7.121e+03	1.850e+03	4.56e+01	4.28e-02	7.269e-08
na 26	6.761e+07	6.456e+07	2.230e+07	2.674e+04	2.402e-07	0	0	0	0	0	0	0
np 27	9.264e+04	6.089e+14	0	0	0	0	0	0	0	0	0	0
np 27	5.526e+06	6.053e+04	0	0	0	0	0	0	0	0	0	0
si 28	8.580e+00	8.580e+00	8.580e+00	8.580e+00	8.580e+00	8.580e+00	8.580e+00	8.580e+00	8.580e+00	8.580e+00	8.580e+00	8.580e+00
si 28	2.633e+07	1.354e+00	5.759e-03	3.700e-03	3.56e-11	0	0	0	0	0	0	0
si 29	2.202e+06	5.738e+03	0	0	0	0	0	0	0	0	0	0
si 30	2.104e+06	0	0	0	0	0	0	0	0	0	0	0
si 31	6.118e+02	4.851e+02	1.109e+00	3.237e-17	0	0	0	0	0	0	0	0
cl 35	1.821e+04	9.274e+03	6.351e-05	0	0	0	0	0	0	0	0	0
er 41	1.129e+04	3.510e+04	5.695e+00	1.067e-23	0	0	0	0	0	0	0	0
k 42	1.539e+04	1.474e+04	4.893e-03	1.695e+00	4.513e-01	2.276e-01	4.231e-01	1.890e-01	502e-01	556e-01	512e-01	285e-02
k 43	2.049e+03	1.957e+03	9.375e-02	1.185e+01	2.262e-02	0	0	0	0	0	0	0
ca 44	2.468e+04	2.452e+04	2.118e+04	8.467e+03	2.519e+02	2.795e-08	1.419e-20	0	0	0	0	0
sc 44	1.584e+05	1.379e+05	4.417e+04	7.76e+03	1.130e+01	3.22e-16	0	0	0	0	0	0
sc 46m	6.036e+05	5.267e+05	4.245e+03	8.369e+02	1.201e+00	2.47e-09	0	0	0	0	0	0
sc 46	5.612e+05	5.618e+05	5.370e+05	5.300e+05	4.382e+05	3.67e+05	2.737e+04	1.545e-01	4.248e-03	8.838e-28	0	0
sc 46m	1.720e+03	0	0	0	0	0	0	0	0	0	0	0
sc 47	4.612e+05	4.792e+03	4.129e+01	1.629e+03	4.727e+01	5.190e-05	2.635e-21	0	0	0	0	0
sc 48	9.179e+05	9.034e+05	6.278e+05	6.431e+04	1.055e+01	1.888e-24	0	0	0	0	0	0
sc 50	1.191e+04	1.204e+04	1.204e+04	1.204e+04	1.204e+04	1.204e+04	1.204e+04	1.204e+04	1.204e+04	1.204e+04	1.204e+04	1.204e+04
vi 51	2.192e+04	1.603e+01	0	0	0	0	0	0	0	0	0	0
v 52	2.628e+02	5.025e+01	5.038e-02	3.884e+02	432e+02	2.139e-01	9.927e-05	2.081e-32	0	0	0	0
v 53	1.191e+04	1.204e+04	1.204e+04	1.204e+04	1.204e+04	1.204e+04	1.204e+04	1.204e+04	1.204e+04	1.204e+04	1.204e+04	1.204e+04
cr 59	1.241e+04	1.260e-07	0	0	0	0	0	0	0	0	0	0
cr 59	1.241e+04	1.260e-07	0	0	0	0	0	0	0	0	0	0
cr 59	1.241e+04	1.260e-07	0	0	0	0	0	0	0	0	0	0
mn 52m	3.037e+04	3.034e+04	2.962e+04	2.962e+04	2.549e+04	1.434e+02	3.268e+00	4.381e-16	4.321e-36	1.898e-95	0	0
mn 52m	1.191e+04	1.204e+04	1.204e+04	1.204e+04	1.204e+04	1.204e+04	1.204e+04	1.204e+04	1.204e+04	1.204e+04	1.204e+04	1.204e+04
mn 52m	3.244e+03	4.519e+02	9.265e-18	0	0	0	0	0	0	0	0	0
mn 56	1.241e+04	1.260e-07	0	0	0	0	0	0	0	0	0	0
mn 56	1.241e+04	1.260e-07	0	0	0	0	0	0	0	0	0	0
mn 56	1.241e+04	1.260e-07	0	0	0	0	0	0	0	0	0	0
fe 59	2.695e+07	2.551e+07	4.555e+04	7.307e-13	0	0	0	0	0	0	0	0
fe 59	2.695e+07	2.551e+07	4.555e+04	7.307e-13	0	0	0	0	0	0	0	0
fe 59	2.695e+07	2.551e+07	4.555e+04	7.307e-13	0	0	0	0	0	0	0	0
co 57	2.016e+05	2.016e+05	2.017e+05	1.981e+05	1.868e+05	2.73e+05	7.922e+04	1.888e+03	1.768e+01	1.451e-05	0.044e-15	5.399e-36
co 58	5.065e+06	5.065e+06	5.065e+06	5.065e+06	5.065e+06	5.065e+06	5.065e+06	5.065e+06	5.065e+06	5.065e+06	5.065e+06	5.065e+06
co 60	5.065e+06	5.065e+06	5.065e+06	5.065e+06	5.065e+06	5.065e+06	5.065e+06	5.065e+06	5.065e+06	5.065e+06	5.065e+06	5.065e+06
co 60m	2.753e+03	4.893e+03	5.119e+01	6.222e-05	6.255e-05	6.255e-05	6.255e-05	6.255e-05	6.255e-05	6.255e-05	6.255e-05	6.255e-05
co 60m	2.753e+03	4.893e+03	5.119e+01	6.222e-05	6.255e-05	6.255e-05	6.255e-05	6.255e-05	6.255e-05	6.255e-05	6.255e-05	6.255e-05
co 60m	2.753e+03	4.893e+03	5.119e+01	6.222e-05	6.255e-05	6.255e-05	6.255e-05	6.255e-05	6.255e-05	6.255e-05	6.255e-05	6.255e-05
cu 64	4.187e+04	3.180e+04	5.688e+01	3.575e-16	0	0	0	0	0	0	0	0
cu 64	1.535e+04	1.845e+04	0	0	0	0	0	0	0	0	0	0
cu 64	8.788e+05	8.321e+05	2.371e+05	1.157e+01	7.544e-12	0	0	0	0	0	0	0

Table VI.2. FORIG Results (Cont'd)

	PAGE 271														
ca159m	3.276e+04	1.708e-15	0.	2.750e+02	2.420e+02	0.	1.452e+02	0.	4.053e+00	0.	1.666e-31	0.	4.460e-15	0.	4.261e-32
ca141	2.307e+02	2.007e+02	0.	2.750e+02	2.420e+02	0.	1.452e+02	0.	4.053e+00	0.	1.666e-31	0.	4.460e-15	0.	4.261e-32
ca147	1.549e+03	2.372e+03	1.449e+03	9.949e+02	2.354e+02	1.946e-02	1.767e-07	3.037e-47	0.	0.	0.	0.	0.	0.	0.
ca149	4.442e+03	2.276e+03	8.61e-01	2.536e-26	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
pn149	1.196e+02	1.93e+02	9.036e+01	1.379e+01	1.021e-02	3.926e-23	0.	0.	0.	0.	0.	0.	0.	0.	0.
sm153	1.331e+03	1.111e+03	9.321e+02	1.999e+02	3.037e-02	1.878e-28	0.	0.	0.	0.	0.	0.	0.	0.	0.
au152	5.624e+03	2.24e+03	5.523e+03	5.619e+03	5.601e+03	5.485e+03	0.	3.45e+03	4.359e+03	3.378e+03	1.573e+03	4.399e+02	3.447e+01	0.	0.
au152m	1.339e+03	2.241e+03	2.236e+02	2.013e+03	7.909e-21	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
au152m	2.705e+04	2.958e+04	2.235e+04	2.701e+04	2.682e+04	2.600e+04	0.	4.96e+04	0.	0.	0.	0.	0.	0.	0.
nd166	2.247e+01	1.190e+01	1.208e+01	2.520e+01	4.828e-04	2.539e-17	8.855e-54	0.	0.	0.	0.	0.	0.	0.	0.
nd166m	1.081e+01	1.081e+01	1.081e+01	1.081e+01	1.081e+01	1.080e+01	0.	1.078e+01	1.075e+01	1.065e+01	1.050e+01	1.020e+01	0.	0.	0.
tm170	2.468e+02	2.467e+02	2.455e+02	2.376e+02	2.099e+02	0.352e+01	3.445e+01	1.309e-02	6.945e-07	1.037e-19	4.358e-41	0.	0.	0.	0.
tu177	2.21e+01	8.256e+01	7.404e+01	4.015e+01	7.61e+00	1.668e-03	2.81e-03	0.059e-05	3.006e-09	6.879e-30	1.269e-37	0.	0.	0.	0.
nr181	2.291e+03	2.319e+03	2.833e+03	2.605e+03	1.788e+03	1.540e+02	2.448e+00	4.180e-10	3.397e-23	4.268e-62	0.	0.	0.	0.	0.
tu182	4.474e+05	6.332e+05	6.399e+05	6.168e+05	5.370e+05	2.174e+05	2.112e+04	1.967e+01	1.844e-04	7.247e-36	7.547e-06	5.46e-06	0.	0.	0.
w181	2.329e+01	2.326e+01	2.316e+01	2.238e+01	1.962e+01	8.318e+00	2.883e+00	6.768e-04	1.967e-08	4.829e-22	0.	0.	0.	0.	0.
w187	2.469e+05	3.399e+05	1.231e+03	1.590e+03	2.108e-04	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
re188	3.419e+04	5.336e+04	1.231e+03	1.742e+02	1.163e+02	2.644e+01	0.	1.155e+00	1.911e-06	2.288e-14	3.928e-38	0.	0.	0.	0.
fr192	3.215e+02	2.14e+02	5.185e+02	3.011e+02	2.428e+02	5.958e+01	1.051e+01	1.202e-05	4.491e-13	2.343e-35	0.	0.	0.	0.	0.
au196	1.080e+02	1.079e+02	6.732e+01	2.322e+01	1.489e+01	6.036e-16	0.00e-35	0.	0.	0.	0.	0.	0.	0.	0.
np203	8.273e+02	2.265e+02	6.191e+02	7.493e+02	5.294e+02	5.682e+01	2.610e+00	1.308e-09	2.069e-21	8.184e-57	0.	0.	0.	0.	0.
tf202	2.866e+04	2.661e+04	2.710e+04	1.929e+04	5.236e+03	1.064e+00	9.935e-05	3.207e-41	0.	0.	0.	0.	0.	0.	0.
pb203	1.043e+03	1.135e+02	4.2830e+01	2.049e-07	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
bi207	1.080e+05	1.080e+05	1.080e+05	1.080e+05	1.078e+03	1.078e+03	1.064e+03	9.858e+02	8.999e+02	8.445e+02	4.338e+02	1.431e+02	0.	0.	0.
bi208	5.661e+00	5.661e+00	5.661e+00	5.661e+00	5.661e+00	5.661e+00	5.661e+00	5.661e+00	5.661e+00	5.661e+00	5.661e+00	5.661e+00	5.661e+00	5.661e+00	5.661e+00
sum	6.952e+08	1.156e+08	4.711e+07	2.017e+07	1.591e+07	8.822e+06	4.503e+06	2.724e+06	1.383e+06	1.979e+05	1.114e+04	2.515e+03	0.	0.	0.
total	6.952e+08	1.156e+08	4.711e+07	2.017e+07	1.591e+07	8.822e+06	4.503e+06	2.724e+06	1.383e+06	1.979e+05	1.114e+04	2.515e+03	0.	0.	0.

The Be (n,t) reaction is the principal route to tritium production in beryllium but the other route can also become significant at higher exposures. For example, Maninger has calculated tritium accumulation to be 40% higher when the second route is included. The above calculation applied to a Be sample exposed at the first wall of a LiPb blanket for an exposure of 10 MW-yr/m².

The major question arising from the buildup of tritium in beryllium is the extent of tritium release. It has been speculated that tritium might be released in bursts when helium buildup causes the beryllium to swell and crack. Escaping helium could then carry tritium with it. Cracking due to temperature excursions may cause releases as well. Note that the helium production rate is about 150 times higher than the tritium production rate. Additional research is needed to determine how tritium will be released from beryllium.

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VII. CONCLUSIONS AND RECOMMENDED FURTHER WORK

Beryllium resource estimates have been reviewed. These resources appear to be adequate for use in fusion breeders which could provide fuel to support the world's fission reactors for well over a century. Strategies to improve beryllium resource utilization such as efficient recycle, have been identified.

Mass production of the beryllium parts by remote fabrication methods will be required because the recycled beryllium is radioactive. The requisite technology has been studied and its development was determined to be feasible. The development of free flowing powder for fabricating parts is directly relevant for the fusion breeder application and is already underway in the beryllium industry.

An extensive review and compilation of the properties of beryllium (both irradiated and unirradiated) was undertaken and a new model for beryllium swelling was formulated based upon a synthesis of theoretical and experimental results. These results are provided for general use in developing beryllium component designs. Swelling caused by helium generation will lead to stresses and loss of ductility. We predict that thermal creep may help extend the lifetime by relieving differential swelling caused stresses, but that irradiation induced creep may be a negligible effect. Nevertheless, there have never been any experiments to measure radiation enhanced creep with beryllium, and a set of radiation creep experiments should be carried out. Further irradiation experiments to measure other properties (e.g., ductility) should be carried out to help develop better predictions of lifetime.

The tritium generated in beryllium due to $\text{Be}(n,T)$ and other reactions, is small and unimportant from a breeding point of view but can be important from a safety point of view, specifically. A considerable amount of tritium might be released during a temperature excursion which might occur during a loss-of-cooling accident. Studies of release mechanisms are recommended.

Our studies show that, due to activation of impurities, irradiated beryllium will have to be handled remotely during recycle operations. However, once beryllium containing parts are to be disposed of, near surface burial should be allowed if the niobium and silver content are kept low, but within the assay limits of available grades of beryllium.

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