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Adapting an X-Ray/Debris Shield to the Cascade ICF Power Plant

Michael T. Tobin

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ADAPTING AN X-RAY/DEBRIS SHIELD TO THE CASCADE ICF POWER PLANT:

NEUTRONICS ISSUES

Michael T. Tobin, Lawrence Livermore National Laboratory
P.O. Box 5508 L-459
Livermore, CA 94550
(415)-423-1168

ABSTRACT

A neutronics analysis has been carried out to determine the effects on the Cascade ICF reactor concept of adding a solid-lithium x-ray and debris shield to each ICF capsule. Results indicate that tritium breeding in LiAlO_2 is possible with a modest isotopic enhancement in ^6Li (to 15%). The shallow-burial index is greater than 1 (indicating that deep burial may be required) if the blanket is kept in the reactor for more than 2.5 yr. Nine per cent of the total thermal power is unrecoverable. Parts of the chamber wall may require replacement once during the reactor life due to radiation damage. Part of the SiC chamber end cap must be replaced annually. The reactor may not require any nuclear-grade construction.

INTRODUCTION

The Cascade ICF reactor concept^{1,2} has the potential for high-efficiency operation, a competitive cost of electricity, and much less environmental risk than fission or coal plants. The Cascade concept has evolved in aspects such as different blanket options have been examined. J. Pitts, inventor of the Cascade concept, recently suggested³ surrounding each ICF capsule with a solid lithium shield to protect the first granule layer of the blanket from x rays and to lengthen the x-ray/debris pulse by a few orders of magnitude. This allows the first surface to avoid ablating, thereby extending the granule lifetime. The shield modifies the neutron spectrum somewhat, so

neutronics analysis has been conducted to determine the effects of this modification.

THE MODEL

We carried out calculations using the TARTNP Monte Carlo transport code and the Livermore Evaluated Nuclear Data Library (ENDL).^{4,5} Activation calculations were done using TARTNP, ORLIB,⁶ and FORIG.⁷ ORLIB is used to create spatially dependent, energy-averaged activation cross sections using the library ACTL, a subset of ENDL. FORIG, a fusion-modified version of the well-known fission code ORIGEN, uses those cross sections to compute the production and decay of nuclides, shallow-burial indices, thermal power, and other information. We used the neutronics model for the Cascade chamber shown in Figure 1. The vertical axis is a plane of symmetry; it is treated as a reflecting boundary in the neutronics calculations. The horizontal axis is the chamber axis of rotation. The target is centered at the origin, which is the center of the chamber. The source is modeled as a hemispherical shell in which 14.1-MeV neutrons are uniformly distributed in a region of fuel burn with an areal density of 3 g/cm². This region is surrounded by unburned fuel with the same areal density. The shield is a 1-kg Li sphere with inner and outer radii of 2 and 7.7 cm.

The flowing granular blanket is 1-m thick and contains two distinct material layers. The graphite surface layer is 2-cm thick and has an effective density of 1.05 g/cm³. (The effective density accounts for both the porosity of the granules and the packing fraction of the flowing material.) The LiAlO₂ region is 98-cm thick and has an effective density of 1.274 g/cm³. The lithium in the LiAlO₂ is enriched to 15 at.% ⁶Li, twice the natural isotopic composition. The total blanket thickness is roughly that required to achieve an adequate tritium breeding ratio, keep the neutron leakage low, and shield the supporting tendons from excessive nuclear heating.^{8,9} The distance from the target to the surface of the blanket is 3.09 m at the closest point.

Outside the breeding blanket is a 5-cm-thick region of SiC (3.2 g/cm^3), representing the rotating chamber wall. This is followed by a 3-cm-thick region that contains an Al/SiC fiber composite (Nicalloy) representing the tendons that hold the chamber in compression. The density of the tendons is 2.85 g/cm^3 . A stationary end plate (5 cm thick) with the chute for introducing the granules to the chamber caps the end of the chamber. The beam port radius is 77 cm. The two ports have a combined solid-angle fraction of 1.5%.

TRITIUM BREEDING

We determined an *allowable* tritium breeding ratio by considering the allowable release to the public. Chapter 10, Part 50 of the Code of Federal Regulations restricts the site-boundary dose from routine plant emissions to 5 mrem/yr.¹⁰ For a 1-km site boundary and reasonable release conditions (30-m-high, 1-m-wide stack, 10-m/s exit velocity), this dose results from the emission of 40 Ci of tritium per day. This calculation was done using AIRDOSE, an EPA-certified computer code that determines off-site doses from routine emissions. For 1875 MW (375 MJ at 5 Hz) of fusion power, a tritium breeding ratio (TBR) of exactly 1 implies the production of 2.6 MCi/day. The *minimum* TBR required is then ~ 1.00002 . For a TBR greater than this, tritium will begin to build up in the system. This is expected to be an important issue because the maximum allowable site-boundary dose after an accident is 25 rem.¹¹ We calculate that 1 kg of T, if entirely released as HTO in an accident, would result in a site-boundary dose of about 16 rem. We might therefore want to operate at as close to the minimum TBR as possible to avoid having to transport tritium away from the plant site to meet the CFR requirements. Continuous monitoring and isotopic tailoring (in ^6Li) of the blanket would then be necessary.

It is well known that LiAlO_2 as a blanket material cannot support a TBR greater than 1 without the use of Be as a neutron multiplier

and/or increasing the ${}^6\text{Li}$ isotopic fraction. So, here we seek to maximize tritium production by using an x-ray/debris shield made entirely of ${}^7\text{Li}$. This allows us to breed 0.075 tritons per source neutron and to conserve the total number of neutrons because $(n + {}^7\text{Li})$ gives $(n' + \text{T} + {}^4\text{He})$, where the incident neutron energy must be greater than 2.4 MeV. The density used is 0.54 g/cm^3 rather than 0.534 as for natural lithium. There is also a 6.5% increase in the number of neutrons leaving the capsule/shield assembly, primarily because of $(n, 2n)$ reactions with unburned deuterium in the capsule. For the capsule/shield configuration described above, the TBRs were calculated for a range of ${}^6\text{Li}$ isotopic concentrations; see Figure 2. (Densities were adjusted along with the isotopic composition.) A TBR of 0.986 is achieved when natural lithium is used in the LiAlO_2 . A TBR greater than 1 is first reached at 10% ${}^6\text{Li}$ content in the blanket (1.00024). At 15% enrichment, the TBR rises to 1.013. Over the entire range of enrichments (7.5 to 60%) the highest achievable TBR appears to be about 1.015. While a 15% enrichment is sufficient for breeding much more tritium than is burned, the ${}^6\text{Li}$ is consumed at about 0.3% per year (70% capacity factor). The 15% enrichment then gives at least 15 yr of operation before the TBR drops below 1. (Although this may not be desirable as discussed above due to excess tritium production, some 1.3 kg per year.)

The neutron leakage through the SiC wall/Nicalloy tendons is 0.069 neutrons per DT reaction. The neutron leakage rate through the beam ports is 0.106 neutrons per DT reaction. Overall reactor tritium breeding is 15% with ${}^7\text{Li}$ and 85% with ${}^6\text{Li}$. This is different from the C/BeO/ LiAlO_2 case, in which the tritium breeding was 5% with ${}^7\text{Li}$ and 95% with ${}^6\text{Li}$.¹²

ACTIVATION, DECAY HEAT, AND SHALLOW BURIAL

The x-ray/debris shield is of very high purity because of the recycling of vaporized Li to make the shields, so activation is not expected to be an issue.

The carbon layer is a high-purity (99.976%) pyrolytic graphite. The density is set at half that of solid carbon to account for the packing fraction of the granules. A composition for this graphite was taken from Ref. 12. The most notable impurity is lead at 120 wt ppm. The full-power flux in the zone is 3.51×10^{15} n/cm²-s. To obtain an accurate activation calculation, the flux must be adjusted by the capacity factor and by the ratio of the masses of carbon inside and outside the reactor. We assume 70% capacity factor and an "in-out" ratio of 1:20.¹² The *effective* flux is then 1.23×10^{14} n/cm²-s for the entire carbon inventory (63.8 Mg, 30.4 m³). For a 30-yr lifetime, the carbon has a shutdown activity of 1.76 MCi, which decays to 34.9 kCi after 1 min. Nearly 70% of this remaining activity is tritium. If we assume a steady-state inventory of 2 g of tritium in the carbon (similar to the BeO case previously considered¹³), then 4.43 kCi of T is collected from the carbon and the actual total inventory at 1 min is 30.5 kCi.

The decay thermal power, initially 67.5 kW, drops to 50 W at 1 min after shutdown; both powers are negligible. The shallow-burial index is 0.034 (8.34 Ci of ¹⁴C in 30.4 m³). This is calculated using full-density carbon (2.1 g/cm³) and a regulatory limit of 8 Ci/m³. Calculations by General Atomics¹³ (GA) gave 700 Ci of ¹⁴C in 25 m³, corresponding to a shallow-burial index of 3.5. This difference in results is probably due to the lack of N or O impurities in the carbon for the present calculation, and to N and O in the graphite in the GA calculation. (Shallow burial indices are per 10 CFR 61. Nuclides not included in 10 CFR 61 are taken from Fetter et al.¹⁴)

These results do not include activated heavy-ion target material such as lead. It is estimated¹⁵ that the total steady-state reactor blanket inventory of lead would be <1 mg. This represents <10⁻⁵ wt ppm in the carbon, a negligible addition to the lead already considered as a carbon impurity.

The LiAlO_2 layer is also of high purity (99.585%) and half density (50% packing fraction), and makes up the remainder of the 1-m blanket. The composition of this layer is taken from Ref. 16. The most notable impurity is strontium (1000 wt ppm). The full-power flux in this zone is 1.14×10^{15} n/cm²-s. The *effective* flux, adjusted from the full-power value as previously discussed, is 4.78×10^{14} n/cm²-s, corresponding to an in-out ratio of 1:1.67. The volume is 168 m³ (full density) and the mass is 428 Mg. For a 30-yr irradiation, the LiAlO_2 has a shutdown activity of 10^9 Ci, which decays to 800 MCi at 1 min. This activity is dominated by ^{24}Na (15-hr half life); it includes 110 g of tritium, which is the steady-state inventory bound up in the granules.

The shutdown decay thermal power is 23.4 MW, which drops to 15 MW in 1 min and to 70 kW in 1 week. These thermal powers present no risk to structures even without active cooling. The shallow-burial index is 14.2. (Here we use 80 Ci/m³ as the limit for ^{14}C , taking credit for being bound in the low-permeability ceramic matrix.) The major contributor is ^{26}Al (shallow-burial index = 12), which reaches an index of 1 at 2.5 yr.

The SiC wall (5 cm thick) has a full-power flux of 1.32×10^{14} n/cm²-s and an *effective* flux of 9.2×10^{13} n/cm²-s. The shutdown activity is 5.8 MCi, which decays to 4.9 MCi in 1 min. The thermal power is a negligible 94 kW at shutdown. The shallow-burial index is 0.006, all due to ^{14}C . Again, we should ensure that no N or O is actually in the SiC, because the composition taken from Ref. 12 (99.998% pure) has neither.

The tendon region is made from Nicalloy (50% Al, 50% SiC). An approximate composition was established by using a 50 at.% mix of the Al and SiC (with impurities) already referenced. The shutdown activity is 18 MCi, which decays to 13 MCi in 1 min. The shutdown decay thermal power is an insignificant 0.3 MW. The shallow-burial index is 0.14, due entirely to ^{26}Al . Again we

must ensure that the Nicalloy has no appreciable N or O impurities.

The Al liner of the external water shield is assumed to be 2 cm thick and to have the composition given in Ref. 10. This Al is relatively impure (98.05%) and therefore represents a worst-case activation situation. The shutdown activity is 106 MCi, dominated by ^{28}Al (2.25-m half life). The activity decays to 80 MCi in 1 min and to 4.8 MCi in 1 hr. The shutdown decay thermal power is 1.9 MW, which decays to 68 kW in 1 hr. The shallow-burial index is 0.56, due entirely to ^{26}Al .

The water shield is made up of 10 vol% Al. For these calculations, boron is *not* added, again presenting a worst-case activation scenario. [The production of ^{28}Al would be reduced with the presence of boron, as would most other (n, γ) reactions. This also assumes that the boron would be adding few impurities to the water.] The composition is again taken from Ref. 12. The shutdown activity is 68 MCi, which decays to 3 MCi in 1 hr. The shutdown thermal power is 1.2 MW, which decays to 48 kW in 1 hr. The shallow-burial index is 0.007, equally divided between ^{14}C and ^{26}Al .

Table 1 summarizes the activity, decay power, and shallow-burial indices.

ENERGY DEPOSITION

Table 2 gives the distributed energy deposition per DT reaction for the Cascade chamber. The total energy deposited is 18.87 MeV per DT reaction. Table 2 also lists the thermal power based on a fusion power of 1875 MW. The total available thermal power is then 2070 MW_t ($M = 1.105$).

The thermal power in the carbon layer is 777 MW, about 40% of the total. The corresponding values for the LiAlO_2 layer are 1224 MW and ~60%. The thermal power in the SiC wall is 6 MW, just 0.25% of the total. In addition to the leakage through beam ports and wall, another loss is the

lithium vapor that is pumped from the chamber. Condensation of this vapor removes 115 MW of power from the system. Slightly more than 91% of the thermal power enters the power conversion cycle.

The peak heating will occur at the normal from the capsule to the blanket surface. Figure 3 shows the heating for a single 375-MJ shot for the C, LiAlO_2 , SiC, and Nicalloy. One advantage of using LiAlO_2 for a blanket material is its high operating temperature. To make the most of this advantage it is important that 'spikes' in the specific-energy profile be avoided. Figure 3 shows a fairly smooth energy distribution. This smoothness is not achievable for every blanket option: For a 30-cm-thick C layer over a 70-cm-thick Li_2O layer (one blanket option considered), the much softer neutron spectrum after the thick C layer results in considerable breeding in the ^6Li immediately behind the C. The associated energy release causes a spike in the energy deposition profile (see Figure 4), which results in reduced power conversion efficiency because the blanket must operate at a lower average temperature to avoid melting due to the spike. A solution to this problem may be to select a thinner carbon layer and run the Li_2O with only about 2 or 3% ^6Li in the blanket. This will allow a near-minimum TBR and avoid the spike due to many ^6Li reactions in a thin layer.

RADIATION DAMAGE

The use of SiC as a first structural wall (FSW) is a concern for two reasons: SiC may have a lower displacements-per-atom (dpa) damage threshold than steel, and helium is generated much more rapidly in SiC than in steel. But because SiC is so much more environmentally benign than steel and can operate at a much higher temperature, SiC remains the choice for the FSW. We therefore rely on the thick blanket to reduce dpa and He generation rates in the SiC to levels that allow the FSW to last an economical lifetime. We calculated these rates at the normal to the capsule, where they would be highest. A 1-D spherical geometry was

used for the neutronics model, in which the carbon blanket surface is 3.1 m from the capsule.

The dpa cross sections were taken from the library used for the code SPECTER, which is used to compute dpa rates.¹⁷ For the first centimeter of the SiC wall, the dpa rate is 1.5 dpa per full-power year, where the total dpa cross section is 239 barns. The rate averaged across the full 3-cm Nicalloy tendon region is 0.58 dpa per full-power year; the total cross section is a slightly higher 257 barns. The dpa damage-limited lifetime for SiC in a fusion neutron flux is uncertain. We use here a value based on work done with a fission spectrum (so He production is not considered) that indicates that 20 dpa might be a reasonable lifetime limit. This work¹⁸ indicates a 13.3-yr life (19 yr at 70% capacity factor) for SiC at the normal to the capsule. (The SiC that is protected by a greater effective thickness of blanket material will, of course, have a longer lifetime). The lifetime for the tendons is then 34 full-power years. These lifetimes must be validated by experiments using SiC with a fusion spectrum.

The He production rates for the first centimeter of SiC wall material and the tendon region are 27 at. ppm and 8.3 at. ppm per full-power year, respectively. Again, He at. ppm damage limits for SiC are very uncertain. Based on work done with other ceramics (most notably BeO), we use here a limit of 2000 at. ppm (which corresponds to 2% swelling) as a reasonable damage threshold.¹⁹ After 30 yr at 70% capacity, the SiC and tendons would have swelling due to He production of 0.6% and 0.2%, respectively. These results indicate that He production in the FSW will be far less than levels of concern, especially because this analysis assumes that none of the He is released over the 30 yr, an extremely conservative assumption because the wall is operating at high temperatures.

It is uncertain what effect He generation and dpa rates will have on granule lifetime. After 30 yr, the average LiAlO₂ granule consists of 2 at.% He. For the C granules, this value is 1.1 at.% (both for 70% capacity). The dpa rate in the first centimeter

layer of C is 46 dpa/yr. The dpa rate for the LiAlO_2 layer will average a few tens of dpa/yr.

Radiation damage to the end cap of the reaction chamber is a concern. As can be seen from Figure 1, this SiC wall (5 cm) has far less than a meter of blanket material for shielding toward the chamber centerline. Over the 20 cm next to the driver port, the dpa rates vary from greater than 23 to less than 6 dpa/yr. Over the same section, the He at. ppm rates vary from greater than 1950 at. ppm/yr to less than 260 at. ppm/yr. Beyond 20 cm from the driver port there is adequate shielding, so that radiation damage is about the same as that for the SiC chamber reported above. These calculations indicate that the last 20 to 25 cm of the end cap would have to be replaced at least annually. Replacement will be impeded by the dose rate from the LiAlO_2 , which will not decay to an accessible level (2.5 mrem/h) until 9 days after a shutdown.

ACCIDENT SITE-BOUNDARY DOSE

Typically, the first step in analysis of the consequences of a worst-case accident is to determine the potential nuclide source term. The activation analysis already described provides this information. Next, a very conservative estimate of a release of these nuclides (1% of the total inventory is used here, except for gases and tritium, which are all released) under reasonable meteorological conditions is assumed. The code HOTSPOT²⁰ is used to estimate the site-boundary dose. According to 10CFR100, the public can receive no more than 25 rem in the first 2 hr of a worst-case accident.¹¹ Here we compute the dose the population would receive over a 50-yr period (for those nuclides that will remain in the body that long), for members of the public who remain at the site boundary (1 km) for the duration of the accident. Release conditions are 30-m release height, 1-m/s wind speed, and Pasqual stability condition D (stable). A source term that exceeds the 25-rem limit exists for the LiAlO_2 blanket, even if the target-factory tritium is not considered (see Table 3).

No release mechanism (such as meltdown due to decay heat after shutdown) is inherent to the power plant operation, however. More analysis is therefore required to determine an exact, credible source term for a credible accident scenario, so it can be determined whether any special equipment must be used to protect the public.

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Figure Captions

Fig. 1. Geometry of neutronics model for Cascade.

Fig. 2. The Cascade tritium breeding ratio (TBR) reaches local maxima for x-ray/debris shields enriched by 15 and 37% in ^6Li . The total TBR includes a TBR of 0.075 arising from all the ^7Li in the shield.

Fig. 3. The specific energy increases only slightly in the LiAlO_2 after the 2-cm C layer.

Fig. 4. The specific energy increases greatly in the Li_2O behind a thick (30-cm) C layer because of the exoergic tritium-breeding reaction. This creates problems in using Li_2O , whose melting temperature is already lower than that of LiAlO_2 .

Table 1. Activation, decay power, and shallow-burial indices for Cascade.

	<u>Shutdown</u>	<u>1 day</u>	<u>1 month</u>
Activity (MCI)			
Carbon	1.76	0.028	0.024
LiAlO ₂	1045	108	1.16
SiC	5.8	0.013	0.007
Nicalloy	19.5	0.31	0.03
Al wall	106	1.44	0.17
Water shield	<u>68.4</u>	<u>0.66</u>	<u>0.016</u>
	1246	110	1.41

Decay power (kW)			
Carbon	67.5	0.02	0.005
LiAlO ₂	23063	3065	26.1
SiC	93.6	0.15	0.10
Nicalloy	310	3.3	0.02
Al wall	1873	16.9	0.11
Water shield	<u>1241</u>	<u>9.63</u>	<u>0.18</u>
	26650	3095	26.5

Shallow-burial indices

Carbon	0.034
LiAlO ₂	14.2
SiC	0.064
Nicalloy	0.21
Al wall	0.56
Water shield	<u>0.007</u>
	15.1

Table 2. Cascade energy deposition.

	Energy (MeV)	Thermal power (MW)
Carbon layer		
Target ^a	2.55	272
Shield ^a	0.82	88
α energy ^a	3.46	367
Neutron	<u>0.47</u>	<u>50</u>
Layer total	7.30	777
LiAlO ₂ layer	11.52	1224
SiC wall	<u>0.05</u>	<u>6</u>
Blanket total	18.87	2007
Leakage	0.58	63
Lithium vapor condensing outside the chamber		<u>115</u>
Total losses	0.58	178 (8.6%)

^a Target, shield, and α -energy deposition have been reduced by the beam port solid angle fraction, 1.5%. Energy multiplication factor is $M = 1.1$, resulting in 2070 MW_t.

Table 3. Site-boundary doses from release of blanket nuclides and possible tritium inventory in a target factory.

<u>Nuclide</u>	<u>Quantity (Ci)</u>	<u>Dose (rem)</u>
$^3\text{H}^{(a)}$	9.7×10^6	16.0
^{24}Na	3.3×10^6	4.9
^{65}Zn	1.2×10^4	3.4
^{124}Sb	4.9×10^3	1.8
^{60}Co	8.5×10^3	20.0
^{76}As	8.0×10^4	4.3
^{123}Sn	5.0×10^2	0.3
^{210}Po	5.5	<u>0.7</u>
		51.4

(a) 1 kg from target factory; all HTO.