6.8.4 Primary Shield

6.8.4.1 **Introduction** – Two types of shielding materials have been considered for the bulk shield and the shielding of the beamlines of the Prometheus-L and -H reactor designs. The first type is a concrete-based shield consisting of 87% concrete, 8% carbon-steel, and 5% water. The other type is a composite material shield made of 20% Pb, 20% B_4C, 25% SiC, 30% water, and 5% aluminum. The motivation behind considering the second shielding type is the low-activation level attainable in this composite material after reactor shutdown as compared to the concrete-based type. Additionally, long-term activation with this composite shield is 2-3 orders of magnitude lower than with the concrete-based shield. (see Section 6.8.5.2 for more details).

To address the shielding requirements in the Prometheus design, the nuclear field attenuation characteristics of these two types of shielding materials were examined and are discussed below. Based on results from this 1-D comparative analysis, the required shield thicknesses for the bulk shield, the beamlines shield, and the reactor building wall were estimated. Some of these estimates were compared to the 3-D calculation results performed for the shielding design of the SIRIUS-M reactor.[1]

Shielding design depends largely on the criteria for personnel access to components inside and outside the reactor building. Thhree alternative design criteria wwould determine the required shield around the cavity and outside the reactor building. These are:

(a) Shield design allows for personnel access during operation. In this case, the design requirements call for reducing the biological dose rate outside the cavity and around beamlines during operation to values below 2.5 mrem/hr. This limit ensures that the reactor personnel will receive no more that 1.25 rem per quarter in restricted areas around beamline and cavity. The purpose of the reactor building in this case is merely to encompass the cavity chamber and other equipment (e.g. cranes, etc.).

(b) Shield design allows for hands-on maintenance of the cavity chamber and beamlines one day after shutdown where the dose rate should not exceed 2.5 mrem/hr.

(c) Shield design does not allow for personnel access to the interior of the reactor building.

Attenuation characteristics for the two types of shielding materials (concrete and composite) are given in Section 6.8.4.2. In the final Prometheus reactor design, the composite shield was adopted for the bulk and beamline shielding as well as for other penetration ducts. Concrete-based material was chosen for the reactor building in both laser and heavy ion reactor designs. The personnel access criterion adopted in the design allows for hands-on maintenance 48 hours after shutdown, although the more conservative estimates given below are for the case of permitting personnel access inside the reactor building one day following shutdown. The final configuration of the integrated system, including the cavity, bulk shield, beamlines, and their associated shield for the laser reactor, can be found in Section 6.3.1. Configuration of the shielding for the beamlines and other penetrations in the heavy ion reactor design is also shown in that section and general discussions on the design are cited in Sections 2.4 and 2.5. The final thicknesses for the required shield in both designs were decided based on the analysis given here and on expert judgement.[2]

6.8.4.2 Nuclear Field Attenuation Characteristics of the Shielding Materials Considered – The 1-D calculations using the ANISN transport code were performed using a spherical model for the first wall system, blanket/reflector/plena system, vacuum vessel, and bulk shield system shown in Figures 6.8.3-4 and 6.8.3-5. The 1-D calculations for the shield assumed a preliminary shield composition of 20% B_4C, 20% Pb, 30% SiC, and 30% H_20. The final shield composition was determined to be 20% B_4C, 20% Pb, 25% SiC, 5% Al, and 30% H_2O, but the calculations were not iterated for this composition. In the two cases (concrete and composite), the bulk shield is located at a distance of 9.53 m from the cavity center and has a tentative thickness of 2.1 m. The objective of analyzing this thick shield is to examine the attenuation of the neutrons and gamma rays fluxes, the nuclear heating rates, and the biological dose rate during operation as a function of depth inside the bulk shield. Based on these attenuation profiles, the required shield thickness to meet the adopted personnel access criterion is determined for both types of shielding materials.

Figure 6.8.4-1 shows the neutron and gamma ray fluxes in the concrete shield as a function of location inside the bulk shield. The neutron and gamma fluxes at the beginning of the bulk shield is ~2.8×10^{13} n/cm^2•sec and 7×10^{12} γ/cm^2•sec, respectively, based on incident neutron source strength of 9.83×10^{20} n/sec (average neutron energy is ~12.87 MeV) corresponding to neutron power of 2027 MW (see Section 6.8.1). The first wall of the model, shown in Figures 6.8.3-4 and 6.8.3-5, is located at a distance of 510 cm from the cavity center. In the final laser reactor design, the cavity radius is 500 cm and thus a normalization factor of $(510/500)^2 \sim 1.04$ would be applied to the local values shown in Figures 6.8.4-1 and 6.8.4-2. In the heavy ion reactor design, the cavity radius is 450 cm, the neutron source strength is 8.82×10^{20} n/sec (corresponding to a neutron power of 1818 MW), and the normalization factor in this case is $(510/450)^2\ (8.82/9.83) \sim 1.15$.

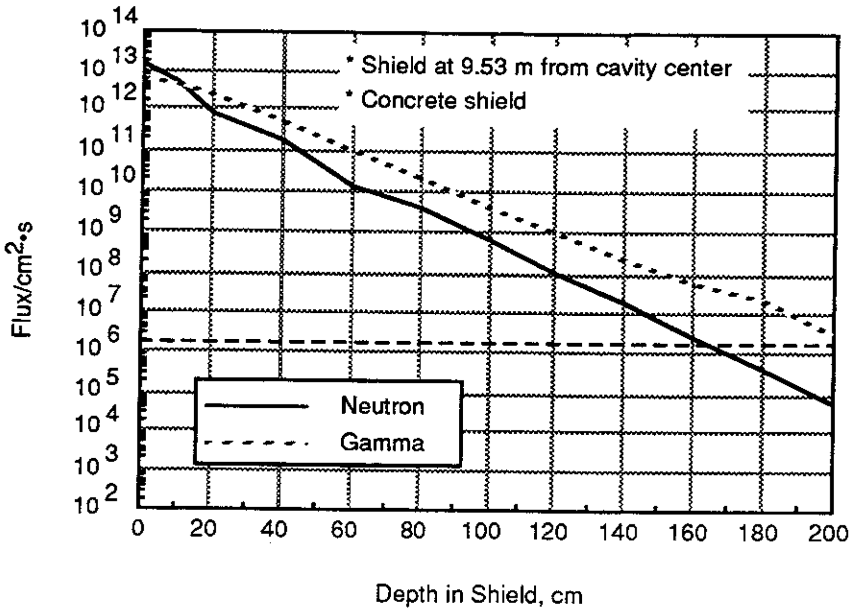

Figure 6.8.4-1 Neutrons and Gamma Ray Fluxes in the Concrete Bulk Shield as a Function of Depth Inside the Bulk Shield

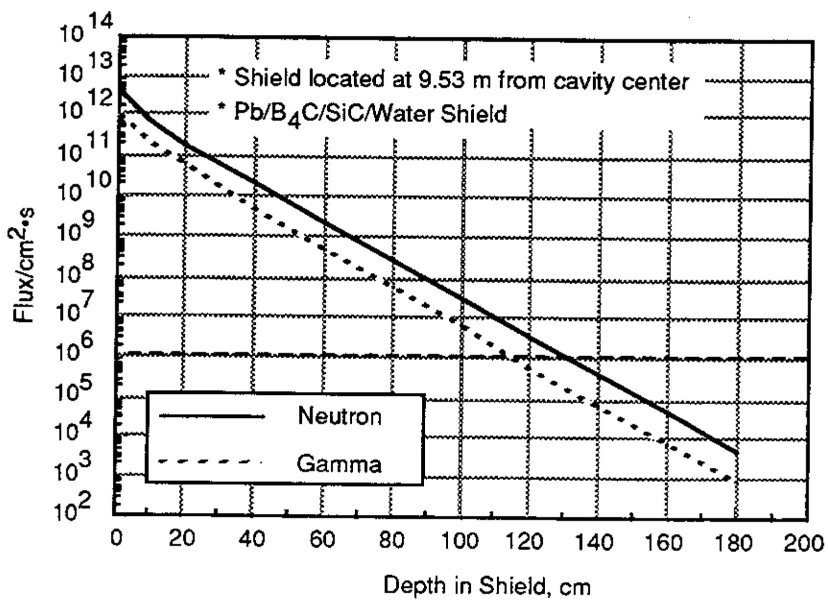

Figure 6.8.4-2. Neutron and Gamma Ray Fluxes in the Composite Bulk Shield as a Function of Depth Inside the Shield

At a shield thickness of 200 cm, the neutron and gamma fluxes drop to $3x10^4 n/cm^2{\bullet}sec$ and $3.3x10^6 \gamma/cm^2{\bullet}sec$, respectively. Therefore, the flux attenuation coefficient $\lambda_{con,f}$ is ~10.9 cm; i.e., an order of magnitude reduction in the neutron flux level occurs every ~25 cm. As for gamma rays, the flux attenuation factor $\lambda_{con,f}$ is ~13.7 cm, i.e., an order of magnitude reduction in the flux takes place for every 31.5 cm. It was reported in Reference 1 that the biological dose during operation in this type of shield is dominated by the gamma ray dose, and that the dose level drops by an order of magnitude for every 32 cm. Also, it was shown in Section 6.8.1 that the nuclear heating during operation in the concrete shield is attributed mainly to gamma ray heating whose profile decreases by an order of magnitude for every 33 cm (see Figure 6.8.1-5). For gamma ray-related parameters, the attenuation coefficient is ~13.7-13.9.

The criterion adopted in the case of allowing personnel access inside the reactor building one day after shutdown is to attain a neutron flux level of $2x10^6 n/cm^2{\bullet}sec$ at the back of the (concrete) shield. In magnetic fusion reactors, it was shown[3] that by attaining this neutron flux level during operation, the biological dose one day after shutdown from activation of the shield and outlying components is ~2.5 mrem/hr. This will allow hands-on maintenance inside the reactor building. By examining Figure 6.8.4-1, a neutron flux level of $2x10^6 n/cm^2{\bullet}sec$ occurs at a depth of ~1.65 m, which is an estimate for the required thickness of the bulk concrete shield. Note that the actual neutron flux at the back of a 1.65-m thick bulk shield could be less than the limit of $2x10^6 n/cm^2{\bullet}sec$ (means thinner shield) since the flux shown in Figure 6.8.4-1 at a depth of 1.65 m has a reflected neutron component (~20%) from the rest of the shield beyond 1.65 m. Thus,

X(bulk) = 1.65 m (concrete shield).

Figure 6.8.4-2 shows the corresponding neutron and gamma ray fluxes in the case of the composite bulk shield. The neutron and gamma fluxes at the front edge of the bulk shield are $5x10^{12} n/cm^2{\bullet}sec$ and $1x10^{12} \gamma/cm^2{\bullet}sec$, respectively. At a depth of a 100 cm, they are ~ $5x10^7 n/cm^2{\bullet}sec$ and $6x10^6 \gamma/cm^2{\bullet}sec$. The neutron flux attenuation factor $\lambda_{comp,f}$ is ~ 8.69 cm; i.e., an order of magnitude reduction in flux level occurs for every 20 cm. The gamma ray attenuation factor is ~ 8.3 cm and an order of magnitude decline in gamma flux level is achieved for every ~19 cm. This shows that the attenuation characteristics of the composite shield is better than those of concrete. Note in this case that neutron flux is always larger than gamma ray flux throughout the bulk shield, a trend that is reversed to the one found for the concrete shield . Since the gamma flux in this composite shield is about an order of magnitude less than neutron flux, the dose during operation is dominated by the contribution from the neutrons whose flux declines by an order of magnitude for every 20 cm. Thus, the dose attenuation coefficient for this type of shield was considered to be the same as the flux attenuation coefficient; i.e., $\lambda_{comp,d} \sim 8.69$ cm.

Similar to the concrete bulk shield case, an assumption is made that by attaining a neutron flux level of 1×10^6 n/cm^2•sec at the back of the shield during operation, the biological dose one day after shutdown would be ~2.5 mrem/hr. This assumption needs validation for these composite shield materials. The results of activation level post shutdown (see Section 6.8.5.2) indicate that this could be a valid assumption since, at shutdown, the radioactivity level in the composite shield case is an order of magnitude less than that for concrete. By adopting this design criterion and by considering the results shown in Figure 6.8.4-2, the required bulk shield thickness is ~130 cm. Thus,

X(bulk) = 1.30 m (composite shield).

References for 6.8.4

1. B. Badger, et al., "SIRIUS-M: A Symmetric Illumination, Inertially Confined Direct Drive Material Test Facility," UWFDM-711, Fusion Technology Institute, University of Wisconsin, October 1986.

2. M. Abdou, private communication, February, 1992.

3. B. Logan, et al., "MARS: Mirror Advanced Reactor Study Final Report," UCRL-53480, Lawrence Livermore National Laboratory, (1984).