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Analytical estimation of accessibility to the activated lithium loop in IFMIF

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Abstract

Beryllium-7 (^{7}Be) is the dominant nuclide affecting accessibility and maintenance scenario of the lithium loop of the International Fusion Materials Irradiation Facility. The dose equivalent rate around typical components of the lithium loop was calculated employing a code QAD-CGGP2R. Deposition of ^{7}Be on the components was assumed to be proportional to their surface area wetted by liquid lithium. As result, the most severe condition was around the heat exchanger with surface area of 576 m^2. The dose equivalent rate was about 10^7 μSv/h, several orders of magnitude higher than 10 μSv/h limit considering ICRP recommendation. The dose rate can be reduced below the limit by a 22 cm-thick iron shield or a 6.5 cm-thick lead shield. Also, a possible method of reduction of the dose rate and the shield thickness by employing a cold trap was shown.

1. Introduction

For testing the effects of neutron irradiation on the properties of candidate materials for fusion reactors, an accelerator-based deuteron-lithium (D-Li) neutron source concept was established by an international collaboration, and is referred to as the International Fusion Materials Irradiation Facility (IFMIF). The work was under the auspices of the International Energy Agency (IEA) [1]. Development of this type of neutron source was started in the Fusion Materials Irradiation Test (FMIT) facility project [2] and its technical basis was succeeded by the IFMIF activity, starting in1995. Fig. 1 shows the arrangement of the IFMIF main lithium loop components. To provide the intense neutron flux of 9×10^{17} neutrons/m^2/s with a peak energy around 14 MeV in the test volume of 500 cm^3, two deuteron beams with total current of 250 mA and energy of 40 MeV are injected into a flowing liquid lithium target. It is operated at maximum flow speed up to 20 m/s for removal of the 10 MW heat deposited by the deuteron beams and to suppress the excessive increase of lithium temperature.

There are two sources of radioactivity contained in the lithium, one generated by the interaction of the deuterons with lithium and the other contained in material dissolved from the inner surface of loop components activated by neutrons. At typical operating conditions of the lithium loop, the corrosion rate is less than 1 μm/year [1], and the radioactivity

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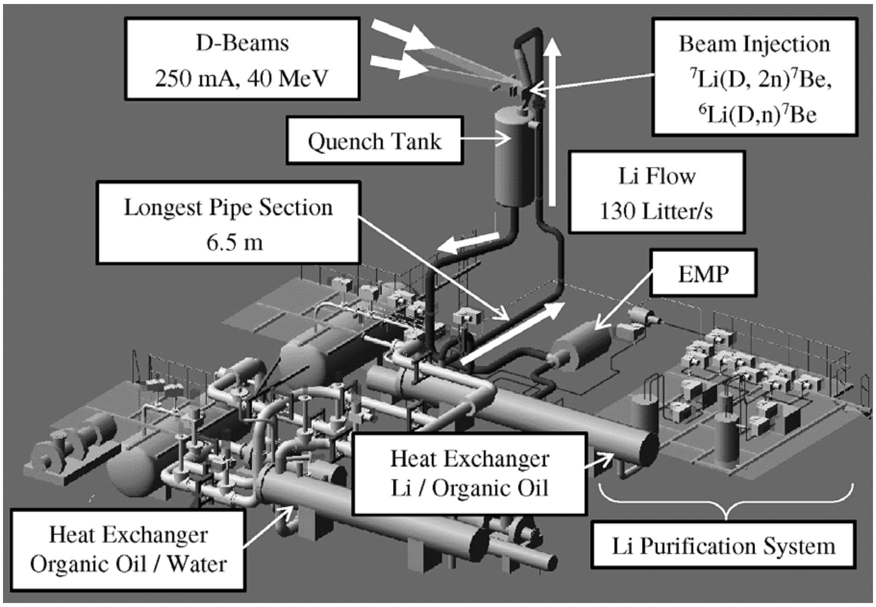

Fig. 1. Layout of IFMIF main lithium loop.

of the corrosion products is transferred by the lithium circulation and deposited either on inner surfaces of the lithium loop or at an impurity cold trap of the purification loop. The effect of the corrosion products upon accessibility to the lithium loop was estimated earlier [3] and the result was acceptable when considering a limit of dose equivalent rate 10 μSv/h derived from an ICRP-60 recommendation (100 mSv during 5-years) and assumed working time of 10000 h. Tritium (^3H) is generated through the D-Li and neutron induced tritium production (^6Li(n,t), etc.) reactions. Total amount of tritium radioactivity in the 9 m^3 lithium inventory is 4.9×10^{14} Bq and tritium permeation rate through the entire loop component walls is 1.0×10^6 Bq/h, much less than the capacity of the tritium treatment system in the current IFMIF design [4]. From these results little concern is identified for tritium from the viewpoints of accessibility and radioactive safety.

In contrast, beryllium-7 (^7Be) produced through the reactions $^7Li(D,2n)^7Be$ and $^6Li(D,n)^7Be$ becomes a major concern, because its production rate is 5.02×10^{15} Be/s, using the production ratio of ^7Be to 40 MeV deuteron of 0.00322 (±12%) Be/D [5,6], and ^7Be emits a 0.478 MeV γ-ray with a probability of 0.105 and a half-life of 53.3 days. The radioactivity of ^7Be can be considered to approximately reach equilibrium of 5.02×10^{15} Bq after one year of IFMIF operation. This paper presents an analytical estimation of accessibility to a typical component of the IFMIF lithium loop contaminated by the transported ^7Be.

2. Analysis condition

2.1. Radioactive source

The analysis was performed with two cases of deposition assumption for equilibrium ^7Be. In the most conservative case of 100% deposition, all ^7Be of 5.02×10^{15} Bq was assumed to be uniformly deposited on wet surfaces of the lithium loop. Under this assumption, the radioactivity of each component is proportional to its surface area wetted by the liquid lithium as shown in Table 1.

In the case of 10% deposition, the deposition was assumed to fall to 10%. A cold trap and two hot traps are planned to be installed in a lithium purification system of the IFMIF lithium loop, to reduce impurities; including hydrogen (including its isotopes), carbon, oxygen, nitrogen and others; to less than 10 wppm. The ratio of the equilibrium beryllium (5.02×10^{15} Bq) to the lithium inventory of 4.5 ton is about 80 appb (equal to 80 wppb). The temperature gradient between the IFMIF cold trap (200 °C) and most other parts (250–285 °C) causes localization of ^7Be deposition in the form of $^7Be_3N_2$ as reported [7] in the FMIT project. The

Table 1
Radioactive source due to deposited ^{7}Be in components and outer dimensions of components

Component	Length (m)	Inner diameter (m)	Thickness (mm)	Wet area (m^2)	Radioactivity (Bq)	
					100% deposition	10% deposition
Pipe	6.5	0.1999	0.82	4.1	3.17×10^{13}	3.17×10^{12}
Quench tank	2.095	1.2	12.0	9.0	7.01×10^{13}	7.01×10^{12}
EMP	2.9	0.4778	15.1	9.7	7.56×10^{13}	7.56×10^{12}
HX	7.9	1.1	15.0	576.4	4.48×10^{15}	4.48×10^{14}
Others	–	–	–	47.3	3.63×10^{14}	$4.55 \times 10^{15*}$
Total	–	–	–	646.5	5.02×10^{15}	5.02×10^{15}

* In case of 10% deposition, most of ^{7}Be was assumed to exist in the cold trap.

solubility of $^{7}Be_3N_2$ in liquid lithium is about 0.5 appb, and thus the concentration of ^{7}Be in liquid lithium will be kept at 1 appb in IFMIF [8]. Therefore, local confinement of ^{7}Be in the IFMIF cold trap will account for most of the ^{7}Be inventory.

2.2. Component dimension

The longest pipe with length 6.5 m was chosen as a typical component. Other typical components were a quench tank, an electro-magnetic pump (EMP) in the main loop and a heat exchanger (HX) between the lithium loop and the organic-oil loop. The largest source component is the HX which includes 434 sets of U-tubes.

In the case of the pipe and the quench tank, their radioactivity was assumed to deposit on their cylindrical wet surfaces with 'inner diameter' shown in Table 1. The main EMP in the IFMIF lithium loop is a center-return type having three cylindrical wet surfaces with diameters of 199.9 mm, 390.6 mm and 477.8 mm respectively. In case of the EMP, the radioactivity was assumed to uniformly deposit on these three surfaces. As mentioned above, the HX includes many U-tubes, which are almost uniformly arranged in the HX vessel. In the case of the HX, the radioactivity of 4.48×10^{15} Bq (4.48×10^{14} Bq for the case of 10% deposition) was assumed to uniformly deposit in the volume with length 7.9 m and diameter 1.1 m.

2.3. Calculation code and mesh

The calculation code employed was QAD-CGGP2R to deal with the three-dimensional (3-D) problem of a radioactive source deposited on each component wall, the buildup factor within each component wall, and the estimation points (detector locations). This code was revised from QAD-CGGP2 [9] to provide a dose equivalent rate. Cylindrical layers of the radioactive source were divided into 24 (in cases of the pipe and the EMP) or 96 (the quench tank) elements in the circumference direction (θ) and more than ninety elements in length (L). In the case of the HX, its volumetric source was divided into 96 elements in θ, 98 in L and 55 in the radial direction (R). Each small radioactive source is assumed by the code to be a point source, the code then calculates dose equivalent rate as the sum of those due to the point sources. This discrete method caused estimation error less than 10% (at a location 1 cm from each component wall) and less than 1% (5 cm) in dose equivalent rate.

2.4. Materials

All component walls were assumed to be made of 316 L stainless steel consisting of Fe (66%), Cr (16%), Ni (12%), Mn (2%), Mo (2%) and Si (1%). The remaining fraction 1% for other rare elements was ignored in the calculation. The atomic number and the partial density influence the buildup factor and the shielding performance.

Only the outer most walls shown in Table 1 were assumed for calculation of γ-ray attenuation and buildup, while the EMP has the inner cylinders and the HX has the many U-tubes. Existence of liquid lithium in the component was also ignored. These assumptions produce slightly conservative results, as their shielding effect is ignored.

3. Results and discussion

3.1. Dose equivalent rate around each component

Fig. 2 shows calculated results of dose equivalent rate ($\dot{H}$) around the pipe in the case of 100% deposition, the quench tank, the EMP and the HX, just after shutdown of D-beam injection. The rates at 1 cm from each component are respectively

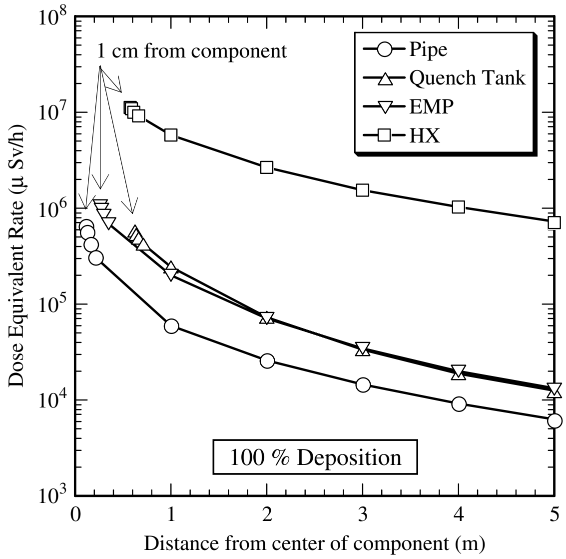

Fig. 2. Dose equivalent rate around each component.

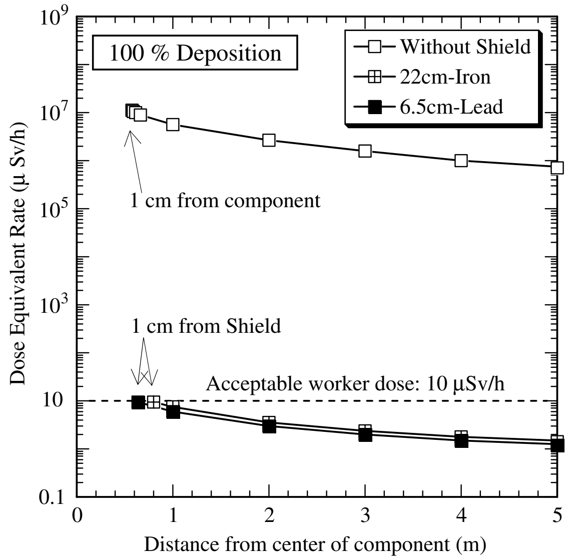

Fig. 3. Effect of iron/lead shield on dose equivalent rate around the heat exchanger for the case of 100% deposition.

6.5×10^5, 5.7×10^5, 1.1×10^6 and 1.1×10^7 μSv/h. The maximum value is given near the HX with wet-surface area of 576 m^2, which corresponds to 89% of the total wet area in the IFMIF lithium loop. In the case of 10% deposition, each dose equivalent rate is 1/10 of that in the case of 100% deposition. Any rate is far larger than the acceptable dose equivalent rate of 10 μSv/h for an assumed annual working time of 2000 h.

Access control even with distance (R) of 5 m from centerlines of component can reduce the rates only by one or two orders of magnitude. Especially in the case of long component, such as the 7.9 m-long HX, the rate is reduced almost in inverse proportion to the distance ($\dot{H} \propto R^{-1}$). The rates are 6.2×10^3, 1.2×10^4, 1.3×10^4 and 7.2×10^5 μSv/h at the locations $R = 5$ m from the quench tank, the EMP and the HX, respectively in the case of 100% ^{7}Be deposition. Furthermore, even a cooling time of 1-month during annual maintenance reduces the rates only by 1/1.5, since the half-life of ^{7}Be is 53.3 days. The rate is still large, for example 4.9×10^5 μSv/h at 5 m from the HX at 30 days after a beam-shutdown.

Under these conditions, workers would not be allowed to carry out maintenance work such as repair and replacement of components.

3.2. Effect of radiation shielding

Since radiation fields are too high for maintenance operations, shielding will be required. Iron (Fe) and lead (Pb) are candidate materials for radiation shielding to attenuate γ-ray with energy of 0.478 MeV emitted from ^{7}Be. An additional analysis using QAD-CGGP2R was performed to find the thickness of iron/lead shield to satisfy the acceptable dose equivalent rate of 10 μSv/h. In the code, the attenuation factor is 8.46×10^{-3} and 1.62×10^{-2} m^2/kg respectively for iron and lead in case of γ-ray with energy of 0.478 MeV. The densities of iron and lead used in the calculations were respectively 7.86×10^3 and 1.134×10^4 kg/m^3.

Fig. 3 shows the calculated results for dose equivalent rate around the HX without radiation shield, with iron shield and with lead shield, just after a beam-shutdown in the case of 100% ^{7}Be deposition. With a 22 cm-thick iron shield or a 6.5 cm-thick lead shield surrounding the HX, the rate can be reduced to the acceptable level of less than 10 μSv/h. The rates are 9.3 and 9.7 μSv/h at locations 1 cm from the iron and the lead shield respectively. The rate is reduced in inverse proportion to the distance ($\dot{H} \propto R^{-1}$) in a region $R > 1$ m, and thus the rate beyond the calculation region ($R > 5$ m) can be predicted.

3.3. Possible other measures for worker safety

The mass of the shields are 57 ton (the 22 cm-thick iron shield) and 22 ton (the 6.5 cm-thick lead shield). Either of these is larger than the HX at

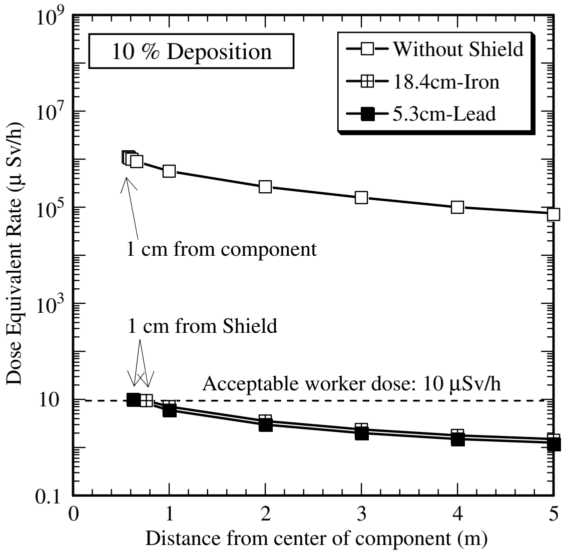

Fig. 4. Effect of iron/lead shield on dose equivalent rate around the heat exchanger for the case of 10% deposition.

19 ton. In case these heavy shields are not acceptable from viewpoint of reasonable design of the target system, other measures for worker safety should be investigated. Possible measures are employment of a cold trap and remote-handling systems. Fig. 4 shows calculated results in the case of 10% ^{7}Be deposition, assuming impurity removal by a cold trap. The needed thickness of the shield reduces to 18.4 and 5.3 cm respectively for iron and lead shields. Better estimation of ^{7}Be localization in a cold trap under temperature gradient condition is needed. Also, investigation of damage to remote-handling systems in a γ-ray field of 10^7 μSv/h is needed. These tasks for design of the target system are planned for the IFMIF Engineering Validation and Engineering Design Activity.

4. Conclusions

The dose equivalent rate due to ^{7}Be deposition in the IFMIF lithium loop was estimated as follows, using the code QAD-CGGP2:

(1) The dose equivalent rate at a typical component in the IFMIF lithium loop was several orders of magnitude higher than the acceptable level of 10 μSv/h. In the highest dose case assuming 100% deposition of ^{7}Be, the maximum value was 1.1×10^7 μSv/h at a location 1 cm from the heat exchanger. The high value is due to its large wet-surface area of 576 m^2.
(2) The dose equivalent rate around the heat exchanger can be reduced to the acceptable level by a 22 cm-thick iron shield or a 6.5 cm-thick lead shield.
(3) Employment of a cold trap reduces the dose equivalent rate and required shield thickness. For example, the shield thickness was 18.4 and 5.3 cm respectively for iron and lead shields if deposition of ^{7}Be in the heat exchanger is reduced to 10% of the amount produced.

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