RADIOLOGICAL SAFETY ASSESSMENTS FOR FUSION NEUTRON SOURCE IN ENGINEERING DESIGN ACTIVITIES UNDER IFMIF/EVEDA PROJECT

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Abstract

International Fusion Materials Irradiation Facility (IFMIF)-like fusion neutron source (FNS) produces fusion-like neutrons by injecting 125 mA/40 MeV deuteron beam into a liquid lithium target in order to elucidate fusion neutron irradiation effects on fusion reactor materials. Under the IFMIF/Engineering Validation and Engineering Design Activities (EVEDA) project, the present study (1) evaluated the production and migration amounts of radionuclides and (2) assessed worker and public doses under steady-state operation and incidental/accidental cases.

1. INTRODUCTION

EU and Japan are conducting the design activities for their own International Fusion Materials Irradiation Facility (IFMIF)-like fusion neutron sources (FNSs), namely IFMIF/DONES [1] and A-FNS [2], respectively, based on the achievements of the IFMIF/ Engineering Validation and Engineering Design Activities (EVEDA) project [3]. The project is being performed as international collaboration between EU and Japan in the Broader Approach (BA) Agreement [4] in order to develop common technologies for the IFMIF-like FNSs. The project includes development of the prototype accelerator LIPAc, engineering design activities for FNS, and R&D activities for Lithium (Li) Target Facility. The engineering design activity performed by Japan, launched in 2020, is divided into five tasks: (1) tritium migration estimation, (2) erosion/deposition modelling in the Li loops, (3) accident analysis in safety, (4) study on the optimization of the Li-Oil Heat Exchanger (HX), and (5) use of LIPAc as testing facility. The tasks (1)–(4) were completed in Mar. 2025 and only the last one continues until Jan. 2028. Since radionuclides are inevitably produced in the FNS facility, one of the crucial issues in the engineering design activities is the implementation of radiological safety measures. This paper presents the results of tasks (1)–(3) that aim to evaluate the production (Section 2) and migration (Section 2 and 3) amounts of the radioactive products and impacts on the worker (Section 2) and public (Section 4).

2. ACTIVATION CALCULATION AND TRANSFER MODELLING IN THE LIQUID LITHIUM LOOP

Various radionuclides are produced in the liquid Li target loop by reactions of the deuterons and neutrons with Li, impurities in Li, and structural materials of the loop. The major nuclides are tritium and Be-7, mainly produced by D-Li reaction. The production amount of tritium and Be-7 are 2.7×10^{15} Bq/year (7.5 g) and 5.02×10^{15} Bq/year, respectively, under two deuteron beams condition. Most tritium is trapped in Cold trap (C.T.) or Hydrogen Trap (H.T.) (1.4×10^{11} Bq) [5]. The tritium migration in the FNS facility is extensively investigated in the next section. Be-7 deposits on C.T. (1.6×10^{13} Bq) and HX (2.0×10^{15} Bq) as cold parts [6], forming Be_3N_2 that is thermodynamically stable [7]. It is proposed to trap Be-7 in C.T. to avoid accumulation in HX [1]. Unbonded Be-7 remain in liquid Li at a low concentration.

Although these major nuclides are evaluated so far, other nuclides have not been studied extensively due to lack of experimental knowledge. Therefore, the amounts of the activated products from impurity in liquid Li and

structural materials were estimated, as well as their migration through the liquid Li loop was evaluated [8]. Initially, we calculated produced amounts of nuclides in the liquid Li target loop using the inventory code FISPACT-II [9] with the impurity concentrations in Li measured with the Li samples obtained from the EVEDA Li test loop (ELTL) after the 3850-h validation operation [10]. Then, in order to obtain the deposition amounts of the nuclides on Li-loop components, we constructed a transfer model of the radioactive products through the loop. This model identified the dominant nuclides and their radioactivity in each Li-loop component for the first time. The effective doses due to deposited nuclides (Fig. 1) were assessed as one of the A-FNS engineering design activities [11]. This revealed feasibilities of hands-on maintenance for the pipe, HX, and electromagnetic pump, contributing to developing maintenance strategies of these components.

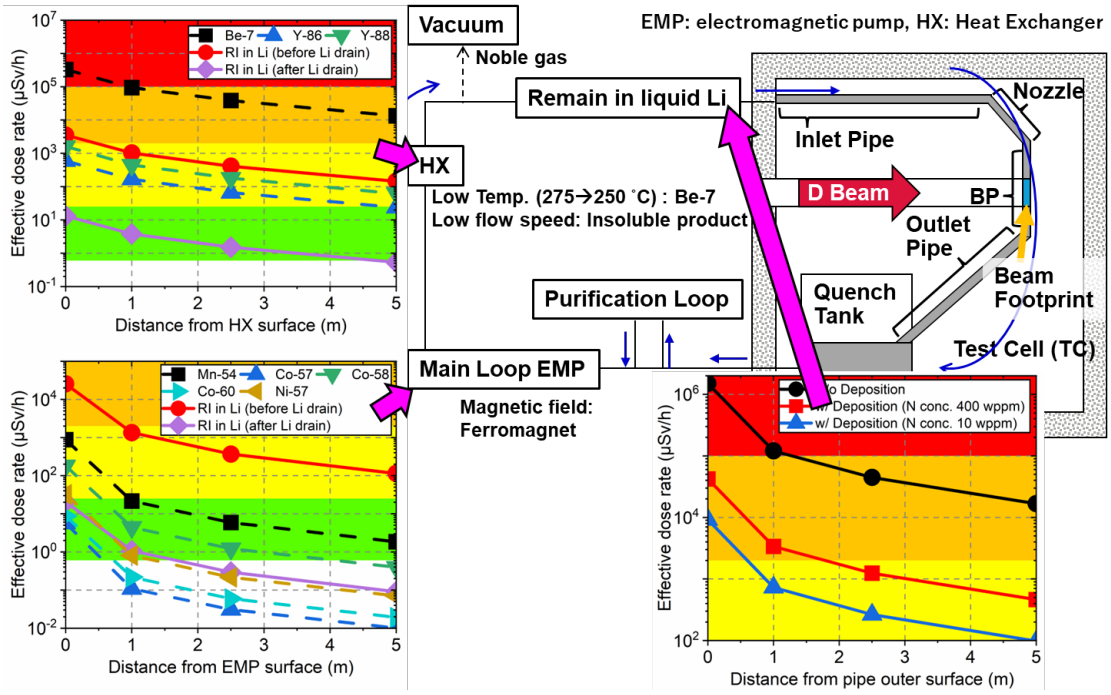

FIG. 1. Effective doses around the Li-loop components due to the radionuclide in the liquid Li and deposits.

3. TRITIUM MIGRATION ESTIMATION

Quantitative evaluation of radioactive material migrations in the FNS facility is essential especially for tritium as movable one. Our previous study constructed a tritium migration model to calculate tritium migration amounts around the liquid Li loop with considering permeations through structural materials, precipitations, evaporations, and so on [5]. This study aimed to upgrade the model to estimate the release amounts from the facility through the stack.

3.1. Tritium migration model and calculation condition

Fig. 2 shows the tritium migration model. Tritium permeates from the liquid Li target loop to the Li Loop Cell and Test Cell (TC). Tritium also migrates from the loop to the purification system. The two purification phases are assumed in the A-FNS. In the first phase, tritium is trapped in the C.T. and does not migrate out of the Li Purification Cell; in the second phase, most tritium trapped in H.T. and tritium that permeates into the vacuum insulation for the Nitrogen Trap (N.T.) is exhausted to the vacuum system. Tritium evaporates from the liquid Li and transfers to the vacuum system as well. Tritium produced in the TC and TM are removed by the TC Gas Purification System (Circulative He Purifier) and TS Low Pressure Helium Cooling System, respectively. Tritium produced around the acceralator (RIR/BTR) send to the stack.

Four Detritiation System (DSs); (1) High-level Tritium Processor; (2) VDS (Exhaust gas Detritiation System); (3) TC gas purification system (Circulative He Purifier); and (4) TS Low Pressure Helium Cooling System, are installed, corresponding to the blocks 8, 9, 10, and 11 in Fig. 2. The oxidation rates of tritium and water removal ratios using molecular sieve at High-level T Processor and VDS are 99.9% and 99%, respectively [12]. In the case of the unexpected increase of the tritium concentration in LLC, the tritium is sent to VDS. All tritium is removed

at TC Gas Purification System and TS Low Pressure Helium Cooling System, leading to no contribution to the release amounts through the stack.

Fig. 2 includes the tritium inventory at each block under the one-beam condition at the first stage purification phase conditions, where tritium migration is larger, as a conservative evaluation. The four calculation cases when steady-state operations and off-normal case in DS are summarized in Table 1. The chemical forms of tritium in the areas with low humidities are assumed to be all tritium gas (HT), since it is not converted to tritiated water (HTO). The atmosphere around the accelerator was conservatively assumed to be HTO, considering standard air. The dry air atmosphere for the LLC was proposed in our previous study [13]. However, there is a possibility that a part of the tritium could be converted to HTO, and two cases of HT and HTO in the LLC were calculated (cases 1 and 2 in Table 1). By assuming that all tritium forms in the LLC are HTO, the event that sending tritium to the VDS due to an unexpected increase in the tritium concentration in the LLC was simulated (Case 3). We assumed that both the High-level T Processor and VDS in series were off-normal and tritium was sent directly to the VDS (Case 4). The highest amount of tritium was released when all DSs failed and all tritium was sent directly to the stack, but this amount was almost the same as that in Case 4. The released amounts of tritium were evaluated by comparing the limitation in the exhaust, defined in the Japanese law “Act on the Regulation of Radioisotopes, etc.” as 7×10^{1} and 5×10^{-3} Bq/cm^3 for HT and HTO, respectively.

TABLE. 2. Four calculation cases considered in this calculation, including normal operation and DS off-normal events.

Case	Category	Tritium form in LLC	Condition
1	Normal	HT	Normal operation
2	Normal	HTO	Normal operation
3	Normal	HTO	Unexpected increase of tritium concentration in LLC
4	Off-normal	HTO	DSs (1) High-level T Processor and (2) VDS not working

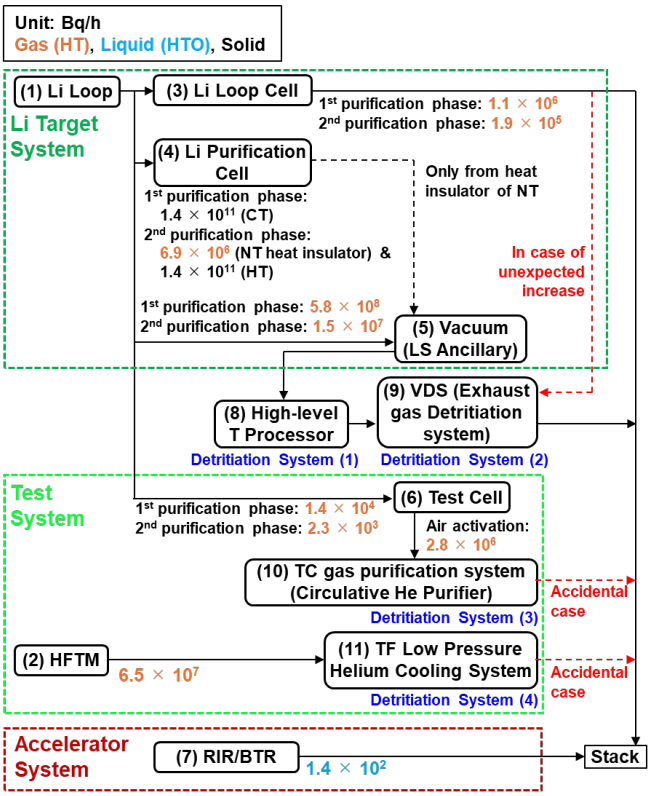

FIG. 2. Tritium migration model for the IFMIF-like FNS facility with the HT and HTO inventories. Tritium amounts are under one-beam condition.

3.2. Result and discussion

The tritium release in each calculation case is summarized in Table 2. In Case 1, the major migration sources of HT and HTO are from the LLC and the vacuum in the liquid Li target systems, respectively. When the chemical form of tritium in the LLC is HTO, the HTO amount is two orders of magnitude larger than that in Case 1. However, this can be reduced to the same amount as in Case 1 by processing in the VDS (Case 3). In Case 4, the primary sources of HT and HTO are the vacuum system of the liquid lithium target system and LLC, respectively.

The stack flow rates calculated from the amount of tritium and the limitation in the exhaust are summarized in Table 2 as well. These values include a safety margin of 30 times [14]. The results show that the stack flow rate should be either (1) approximately 4×10^2 m^3/h from Cases 1 and 3 or (2) approximately 7×10^3 m^3/h from Case 4. In the former case, in the event of the DS off-norma event, the irradiation operation must be stopped to prevent tritium release. On the other hand, in the latter case, although the larger ventilation is required, the irradiation operation can be continued even when the DS is in off-normal, thus the operation availability is ensured. The public dose from released tritium is evaluated in Section 6.

Tritium Process Laboratory (TPL) in JAEA, as a one of the tritium handling facilities, processed tritium of 4×10^9 Bq/h with the stack flow rate of 3.88×10^4 m^3/h [15]. The release amounts through the stack were 2×10^5 Bq/h (HT) and 2×10^6 Bq/h (HTO) in average [16]. By comparison with these values, the tritium emissions and stack flow rates calculated for the FNS facility were considered reasonable.

TABLE. 2. HT and HTO release amounts from the stack for the cases corresponding to Table 1. Stack flow rates calculated based on the limitations of the concentrations in exhaust are also summarized.

Case	HT (Bq/h)	HTO (Bq/h)	Stack flow rate (m3/h)
1	1.1×10^6	$6.4. \times 10^4$	3.84×10^2
2	5.8×102^2	1.2×10^6	6.98×10^3
3	5.8×10^2	6.9×10^4	4.15×10^2
4	5.8×10^8	1.1×10^6	6.85×10^3

4. ASSESSMENT FOR THE PUBLIC DOSES

In our previous study, the preliminary assessment for the public impact was conducted assuming most conservative scenario [17]. In order to elucidate the public impacts when releases of tritium or Be-7 under steady-state and accidental cases, accidental scenario was postulated in addition to the scenarios described in the above.

4.1. Hypothetical accident scenario

Two hypothetical accident scenarios were considered: (1) Li fire at CT, HT, or HX and (2) off-normal event in DS. Be-7 will not be released outside the facility under normal operation; on the contrary, Be-7 has a risk to be released in accidents. The accidents at CT, HT, or HX were selected because of the highest source term emission in the FNS facility, as described above, although those accidents are basically prevented and mitigated with

multiple defence layers. One of the initiating events are Li fire at CT, HT, or HX. The major chemical form of aerosol from Li fire are Li_2O [18]. Other chemical forms of the aerosol of Li fire are reported as LiOH and Li_2CO_3 depending on the conditions such as humidity [19], and a part of tritium would be released as lithium hydroxide. Be-7 in the liquid lithium would be released as aerosol in the event of Li fire [20]. The released form of tritium and Be-7 deposited in CT, HT, and HX when exposed to a Li fire is unknown, but these are presumed to be tritium gas, tritiated water, and aerosols.

The released tritium and Be-7 from the Li loop components (first boundaries) need to be removed properly. Tritium gas or tritiated water are treated by DS [21], whereas tritium and Be-7 contained in aerosols are trapped in an aerosol filtering such as HEPA filter installed in HVAC system [22,23]. In addition, different measures other than HEPA filter were considered, but not included in the current FNS designs [24]. Only if the aerosol filtering fails to work, Be-7 will be released into the atmosphere. In the present study, the effective doses due to the released tritium and Be-7 when Li fire with and without working DS and aerosol filtering were evaluated.

As one of the other accidental cases, failures of DS resulting in the increase in tritium release amounts (discussed in Section 3) were selected as well.

4.2. Calculation condition

The effective doses were calculated with the atmospheric dispersion calculation code ROPUCO [25]. The chemical form of tritium was assumed to be tritiated water with a high dose conversion coefficient for conservative. Only one DS with the Detritiation factor as 100 was assumed. Based on the performance of the HEPA filter, the removal efficiency of the aerosol containing Be-7 was assumed to be 99.99% [26].

Public dose limit in normal condition is 1 mSv/year (ICRP60) [27]. The IFMIF Intermediate Engineering Desing Report (IIEDR)set more strict value as 10 μSv/year under the steady-state operation [28] and initiating event likelihoods based on public doses (Table 3). The calculated public doses were evaluated with these criteria.

TABLE. 3. Initiating event likelihood based on public dose for the FNS facility.

Operating conditions	Initiating event likelihood	Public dose
Normal	-	< 10 μSv/year
Incidental	$10^{-3} < 10^{-2}$/year	10 μSv < 100 μSv
	$10^{-4} < 10^{-3}$/year	100 μSv < 1 mSv
Accidental	$10^{-5} < 10^{-4}$/year	1 mSv < 2.5 mSv
	$< 10^{-6}$/year	2.5 mSv <

4.3. Results and discussion

Table 4 summarizes the radiation dose at the site boundary (500 m) during normal operation and during the hypothetical accident. Under the steady-state operation, the public doses are much less than the criteria in normal operation defined in the previous IFMIF design (10 μSv). Even in the case of DS was out of work, the radiation doses were below 10 μSv, clearly indicating that DS failure alone cannot be an accidental event.

The effective doses in case of accidents at HX, CT, or HT while DS and AFS are in operation are less than the public dose limit defined by ICRP60 (1 mSv), as shown in Fig. 5. The maximum effective dose due to tritium appears around 400 m from the release point since it diffuses from the release heigh of 10 m to the ground. On the other hand, deposited Be-7 on the ground due to precipitation contributes to the effective doses as gamma

source, which causes maximum peak near the release point. According to Table 4, this event corresponds to Incidental condition and event likelihood must be kept below < 10^{-2}/year.

The effective doses in case of accidents at HX, CT, or HT without DS and AFS are beyond the public dose limit. These results indicate the potential radiological risks in the FNS facility can be reasonably reduced with the commonly used measures (installing DS and aerosol filtering) by preventing multiple failures, although their probabilities and mitigation measures have to be further investigated. Although there are some off-normal events that were not considered in this study, the amount of source term released in this event was several orders of magnitude larger than in other events, so the impacts of other events are much smaller.

TABLE. 4. Public doses due to tritium and Be-7 released outside the FNS facility during normal operation and a hypothetical accident under the most severe meteorological conditions (wind speed of 1 m/s, atmospheric stability class F, and no precipitation for tritium and precipitation rate of 5 mm/h for Be-7), and the initiating event likelihood shown in Table 3.

	Normal Operation	Hypothetical accident	
Tritium	3×10^{-3} μSv (< 10 μSv)	DS failure 5×10^{-2} μSv (< 10 μSv)	
		Li fire at CT/HT 65 μSv (10 μSv < 100 μSv, $10^{-3} < 10^{-2}$/year)	
		Li fire at CT/HT & DS failure 6.5 mSv (2.5 mSv <, < 10^{-6}/year)	
Be-7	0 Sv (< 10 μSv)	Li fire at CT/HX	
		CT 0.2 μSv (< 10 μSv)	HX 29 μSv (10 μSv < 100 μSv, $10^{-3} < 10^{-2}$/year)
		Li fire at CT/HX & aerosol filtering failure	
		CT 2.2 mSv (1 mSv < 2.5 mSv, $10^{-5} < 10^{-4}$/year)	HX 2.9×10^{2} mSv (2.5 mSv <, < 10^{-6}/year)

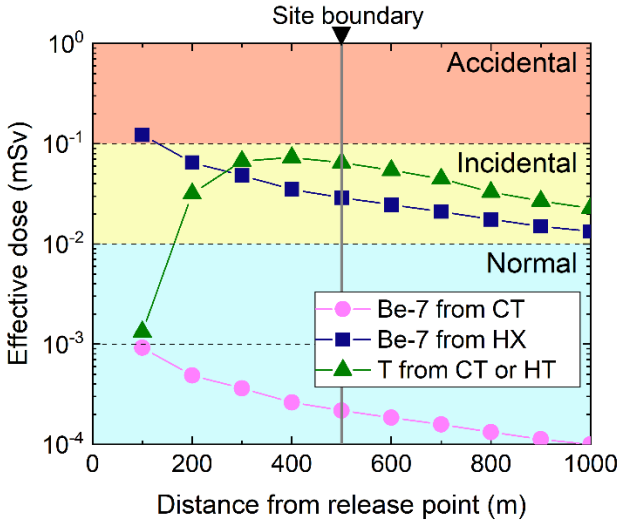

FIG. 3. Effective doses in the public due to released tritium (T) and Be-7 in case of accidents at HX, CT, or HT with DS and aerosol filtering in operation under most severe meteorological conditions (wind speed of 1 m/s, atmospheric stability class F, and no precipitation for tritium and precipitation rate of 5 mm/h for Be-7) at release height of 10 m.

5. CONCLUSION

The engineering design activities mainly focused on the assessments for the impacts of the radioactive products such as tritium and Be-7 have been performed by Japan under the IFMIF/EVEDA project. The achievements of the activities will advance the FNS design especially for the liquid Li target system, DS, and safety system, as well as contribute to the radiological safety assessments in FNS.

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