

Preliminary Assessment of Accident Source Terms for a Heat Pipe-Cooled Microreactor

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1. Introduction

The Korea Atomic Energy Research Institute (KAERI) is developing a heat pipe-cooled microreactor (HP-MR) concept to provide reliable electricity generation for remote applications [1,2]. The HP-MR has notable advantages in reliability and simplicity due to the passive heat transport capability of heat pipes.

This work presents a preliminary assessment of accident source terms for the HP-MR design concept developed by KAERI. Taking advantage of its micro-scale thermal power, the most bounding analyses have been performed to investigate its fast and definite licensing potential.

2. Assessment Method

KAERI has developed a 5 kWth reactor concept for space exploration [1]. In addition, the development of a 100 kWth concept for remote applications including the Moon is currently in progress [2]. Fig. 1 shows the conceptual design of the 5 kW HP-MR core [1]. High-assay low-enriched uranium (HALEU) with an enrichment of 19.5 wt% is used as nuclear fuel.

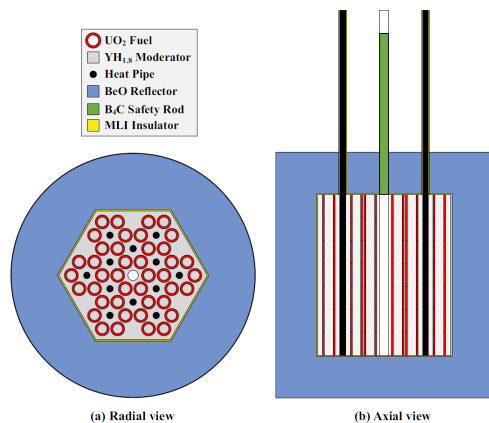


Fig. 1. A concept of the 5 kW HP-MR core.

Table I summarizes the assessment method and assumptions for accident source terms for the HP-MR concepts of KAERI. For nuclide data, including decay chains and dose conversion factors, ICRP publications [3-5] are adopted. The following two criteria, which

practically bound all meaningful radionuclides, are used to select the nuclides for off-site dose calculation.

(1) Calculated nuclide activity by McCARD [6] is positive.

(2) Dose conversion factor of the nuclide is available in ICRP publications ([4] or [5]).

Based on the criteria, the total number of the nuclides considered in this work is 400. As the most bounding accident scenario, it is assumed that the entire inventory of 400 radionuclides in fuel are promptly released into the containment. The possibility of the containment rupture is practically excluded since there is no physical/chemical mechanism to cause pressure buildup in case of the HP-MR.

Table I. Assessment Method and Assumptions for Accident Source Terms

Radionuclide inventory in fuel calculation	McCARD
Off-site dose calculation	MENTAS [7]
Containment leak rate	0.1% vol/day
Number of considered nuclides	400
Exclusion area boundary (EAB) x/Q factor	Gijang (200m)
Release scenario into containment	Prompt 100% release into containment
Decay and decay chain at containment	Considered

3. Calculation Results

Two reactor operation periods (1-month and 1-year) are considered to calculate the initial fuel inventory of radionuclides using McCARD. Relatively short operation periods are selected because it is assumed that the considered HP-MR is built as a technology demonstration reactor. Table II lists the top 10 nuclides and those activities in fuel. It was found that more than 200 nuclides have activities larger than the activity of U-238 (= 8.06 E+8 Bq).

Figure 2 shows the result of the MENTAS calculation for the accumulated total effective dose equivalent (TEDE) at the EAB after the most bounding accident scenario. It is obvious that the acceptance criterion for the off-site dose is satisfied with the worst accident conditions even if the 30-day TEDE criterion is used.

This means that the low population zone (LPZ) can coincide with the EAB. It is also clear that the 5 kW HP-MR design satisfies the acceptance criterion for the off-site dose as well, since it contains a much lower radionuclide inventory in fuel.

Table II. Calculated Radionuclide Inventory in Fuel for 100 kWth HP-MR (Top 10 Nuclides)

	1-Month Operation		1-Year Operation	
	Nuclide	Activity[Bq]	Nuclide	Activity[Bq]
1	U-239	5.00E+14	U-239	5.01E+14
2	Np-239	4.99E+14	Np-239	5.01E+14
3	I-134	2.35E+14	I-134	2.35E+14
4	Te-134	2.06E+14	Te-134	2.06E+14
5	I-133	2.04E+14	Xe-133	2.04E+14
6	Cs-138	2.04E+14	I-133	2.04E+14
7	Xe-135	1.98E+14	Cs-138	2.04E+14
8	Ba-139	1.95E+14	Xe-135	1.98E+14
9	Y-95	1.94E+14	Ba-139	1.95E+14
10	Cs-139	1.93E+14	Y-95	1.93E+14

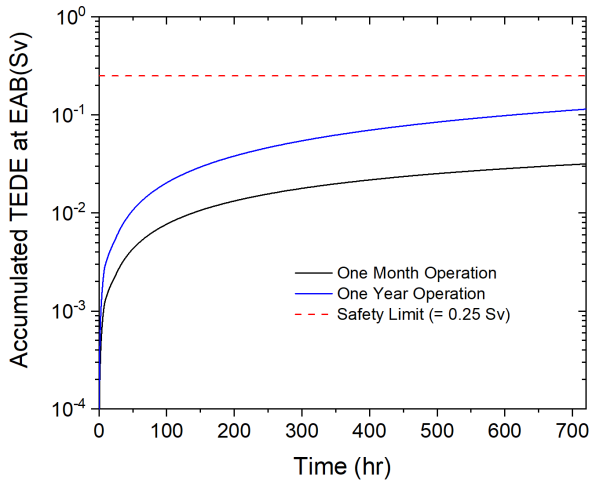


Fig. 2. Calculated off-site dose at the most bounding accident scenario of 100 kW HP-MR.

Table III presents the contribution of each nuclide to 30-day TEDE at the EAB. It shows that the importance of each nuclide changes with the reactor operation period. In particular, the importance of I-131 decreases with the operation period whereas those of Ce-144 and Pu-239 increase. The contribution of Cs-137 (which is known as a key nuclide in large PWRs) is calculated as 0.35% and 1.16% for 1-month and 1-year operation, respectively. The contribution of U-238 and U-235 is found to be less than 0.3%.

4. Conclusions

Preliminary assessment of accident source terms was carried out for the KAERI HP-MR design. The results of the calculation show that the acceptance criterion for the off-site dose is satisfied with the worst accident conditions when the containment limits the environmental release. It is emphasized that the assumed

accident conditions bound all of postulated accidents without the containment failure.

Further study may focus on a more realistic source terms analysis with reasonable assumptions.

Table III. Contribution of Each Radionuclide to 30 day-TEDE at EAB (Top 10 Nuclides)

	1-Month Operation		1-Year Operation	
	Nuclide	Contribution [%]	Nuclide	Contribution [%]
1	I-131	15.83	Ce-144	34.24
2	Ce-144	14.83	Pu-239	11.93
3	Ba-140	10.77	Y-91	9.04
4	Y-91	9.95	Zr-95	6.90
5	Sr-89	7.60	Sr-89	6.18
6	Zr-95	7.06	I-131	4.72
7	Ce-141	6.00	Sr-90	4.49
8	Pr-143	4.71	Ba-140	3.71
9	Pu-239	3.54	Ru-106	3.55
10	La-140	3.25	Ce-141	3.50

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REFERENCES

- [1] B. H. Park, "Key Technologies for Korean Space Heat Pipe Reactor," *Nucl. Technol.*, DOI: <https://doi.org/10.1080/00295450.2023.2289750>, 2024.
- [2] S. Oh et al., "Preliminary Study on Working Fluids for Power Conversion in Space Heat Pipe Reactors," *Transactions of the Korean Nuclear Society Autumn Meeting*, Changwon, Korea, October 30-31, 2025.
- [3] ICRP, ICRP Publication 107, Nuclear Decay Data for Dosimetric Calculations, 2007.
- [4] ICRP, ICRP Publication 119, Compendium of Dose Coefficients based on ICRP Publication 60, 2011.
- [5] ICRP, ICRP Publication 144, Dose Coefficients for External Exposures to Environmental Sources, 2020.
- [6] H. J. Shim et al., "McCARD: Monte Carlo Code for Advanced Reactor Design and Analysis," *Nucl. Eng. Technol.*, Vol. 44, No. 2, pp. 161–176, 2012.
- [7] N. I. Tak et al., Development of the MENTAS Code for Radionuclide Transport Analysis within the Reactor Building of a Very High Temperature Reactor, KAERI/TR-10501/2024, 2024.