Transactions of the Korean Nuclear Society Autumn Meeting
Changwon, Korea, October 24-25, 2024

Preliminary Calculations of Fission Product Release and Off-Site Dose Exposure Under Depressurized Conduction Cooldown Accident of a VHTR

Nam-il Tak*, Sung Nam Lee, Tae Young Han
Korea Atomic Energy Research Institute, 111, Daedeok-daero 989 Beon-gil, Yuseong-gu, Daejeon 34057, Korea
**Corresponding author: takni@kaeri.re.kr*

Keywords : VHTR, off-site dose, fission product, MENTAS

1. Introduction

Mechanical calculations of fission product transport and off-site dose exposure under licensing basis events are considered challenging issues in the development of advanced non-light water reactors, including very high temperature reactors (VHTR) [1]. These calculations are particularly crucial as they are critically linked with a technology-inclusive, risk-informed, and performance-based (TI-RIPB) process of licensing of advanced non-light water reactors in the U.S. [2]. A computer code named MENTAS is under development at the Korea Atomic Energy Research Institute (KAERI) as one of the key tools for the mechanical source term approach [3,4].

In this paper, the MENTAS code is applied to a depressurized conduction cooldown (DCC) accident of a 350 MW_{th} VHTR. Preliminary results of fission product release from the fuel to the environment and off-site dose exposure at the exclusion area boundary (EAB) are presented and discussed.

2. Numerical Models and Assumptions

The VHTR analyzed in reference [5] was chosen for this work. It is a prismatic type reactor cooled by helium. A conceptual layout of the core is shown in Fig. 1. The main design parameters of the VHTR are summarized in Table I.

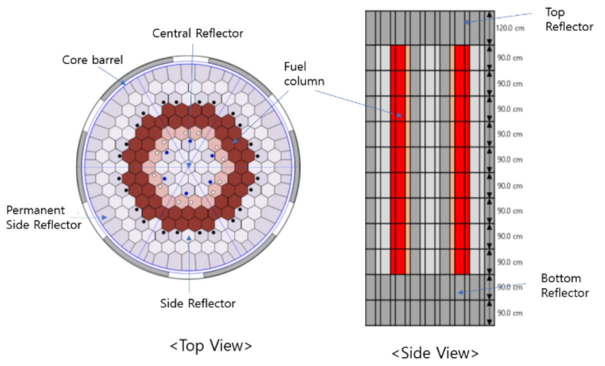

Fig. 1. Core layout of the VHTR

Reference [5] reported the thermo-fluid analysis results under various accident scenarios. Among the investigated scenarios, the DCC accident induced by a medium-sized pipe break was chosen for this work. Using the same input file as reference [5], the GAMMA+ calculation was repeated to obtain the fuel temperature histogram, which was used as the boundary condition of the MENTAS calculation. Fig. 2 shows the histogram of the fuel temperature under the DCC. The calculated maximum fuel temperature was 1408 °C at ~42 hr.

Table I: Main Design Parameters of the VHTR Considered

Parameter	Value
Thermal power (MW_{th})	350
Fuel (enrichment, %)	UO_2 (≤15.5)
No. of fuel columns	66
No. of axial fuel layers	9
Active core height (m)	8.1
Inlet/outlet temperature (°C)	490/950
Coolant pressure (MPa)	7.0

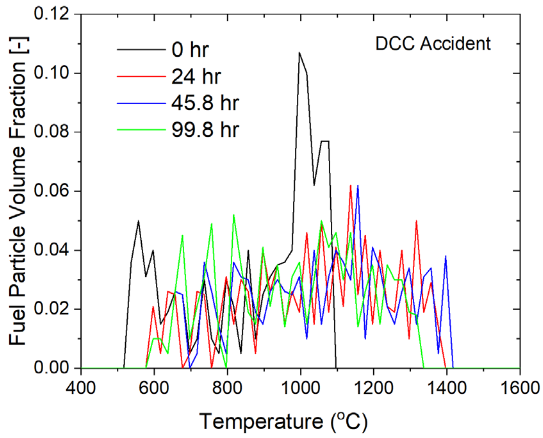

Fig. 2. Fuel temperature histogram under DCC accident.

Table II lists the major assumptions used for the present calculation. All the key nuclides important to dose calculations need to be included. In this work, 200 nuclides were considered in the MENTAS analysis. The nuclides include major fission products as well as activation products such as Ag-110m. The existing work by Lee et al. [6] was used for the initial inventory of the nuclides. Fission product release from the fuel was calculated using the SORS empirical model developed by General Atomics [7]. It was assumed that the nuclides released from the fuel are directly released into the containment. It means that the fission product barrier by the primary loop was completely neglected. A single section of graphite dust was used to speed up the calculation. The dose exposure was calculated using the atmospheric dispersion parameter of the EAB of Kori Unit 1.

Table II: Major Assumptions for MENTAS Analysis

Parameter	Value
Release model from fuel	SORS
Nuclide deposition on structural surface	Neglected
Containment leak rate	0.12 %/day (or 100 %/day)
Graphite dust - No. of sections - Particle size - Generation rate	 1 0.65~1.2 μm 1.0e-4 kg/m³/s
EAB x/Q dispersion parameter	0.001 s/m³

3. Calculation Results

Fig. 3 shows the nuclide release rate into the containment under the DCC accident. Only several nuclides are shown in Fig. 3 among the 200 nuclides. Since the barrier by the primary loop is neglected in this work, the nuclide release into the containment occurs even while the core temperature decreases.

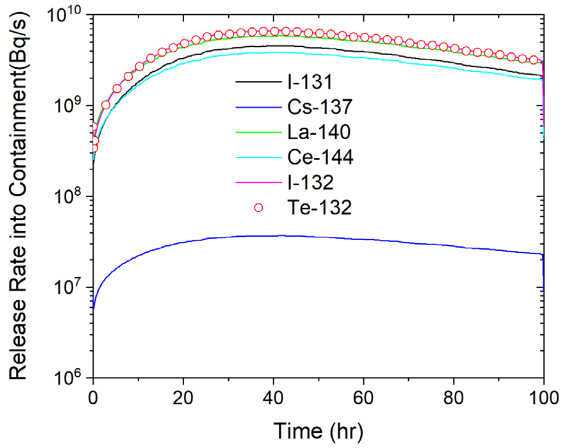

Fig. 3. Nuclide release rate into containment under DCC accident of VHTR.

Figs. 4 & 5 show the cumulative nuclide release to the environment and the accumulated dose at the EAB. Due to continuous supply into the containment, the environmental release and the accumulated dose continuously increase with time.

Two kinds of the leak rates (i.e., 0.12 %/day and 100 %/day) were applied in Figs. 4 & 5. It can be seen that the environmental release and the dose exposure are reduced with the smaller leak rate. However, the amount of the reduction is not as large as expected.

Tables III & IV list the top 5 nuclides contributing to the EAB dose. In the whole-body dose, the top 5 nuclides contribute 59.1% of the total summed dose. On the other hand, the top 5 nuclides contribute 98.4% of the thyroid dose.

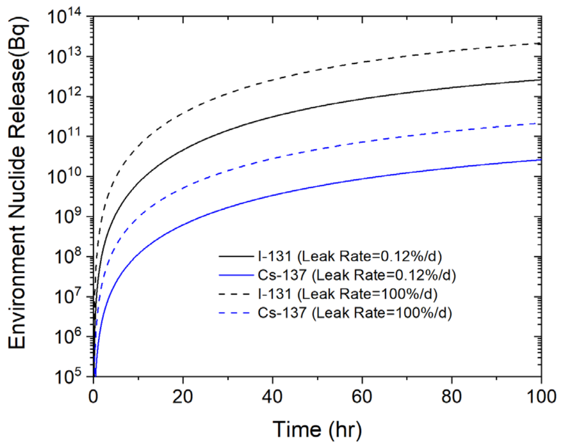

Fig. 4. Nuclide release to environment under DCC accident of VHTR.

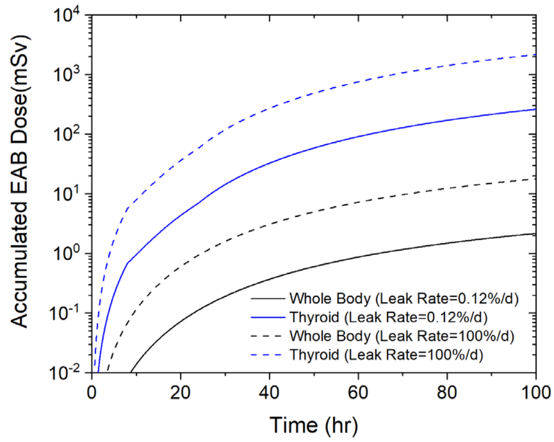

Fig. 5. Accumulated dose at EAB under DCC accident of VHTR.

Table III: Nuclide Contribution to EAB Whole Body Dose

Rank	Nuclide	Contribution (%)
1	La-140	23.6
2	I-132	17.0
3	Nb-95	6.4
4	Zr-95	6.1
5	Kr-88	6.0

Table IV: Nuclide Contribution to EAB Thyroid Dose

Rank	Nuclide	Contribution (%)
1	I-131	67.0
2	Te-132	17.6
3	I-133	12.1
4	Te-131m	0.9
5	I-135	0.8

Fig. 6 shows the effect of the nuclide decay and the daughter-in-growth on the EAB dose. It is shown that the EAB dose increases when the nuclide decay is not considered.

Until 100 hours, the total cumulative release of graphite dust to environment was calculated to be 22.6 kg. Table V provides the top 5 nuclides attached to graphite dust and released to the environment together with the graphite dust. The amount of activity is much smaller than the direct release to the environment (See Fig. 4).

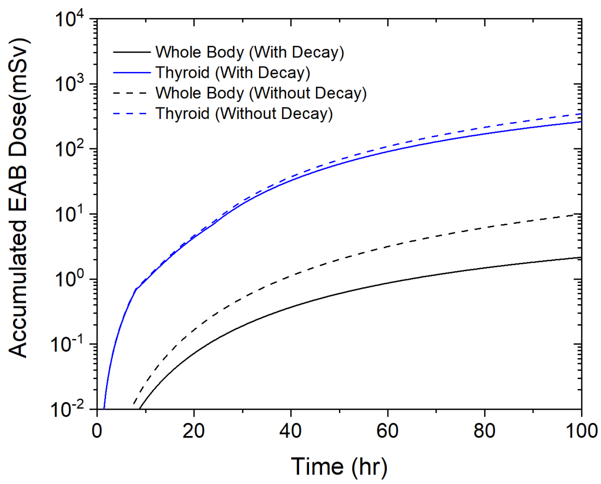

Fig. 6. Effect of nuclide decay and decay chain.

Table V: Nuclides Released Together with Graphite Dust

Rank	Nuclide	Activity (MBq)
1	Sr-89	79.0
2	Ba-140	55.8
3	La-140	50.2
4	Nb-95	42.1
5	Zr-95	41.2

4. Conclusions

The objective of this paper is to demonstrate the major capabilities of the MENTAS code. Using the preliminary parameters and boundary conditions, fission product release and off-site dose exposure under an accident condition of a VHTR were analyzed using the MENTAS code. In addition to the dose exposure results, key nuclides contributing to the dose were identified. Moreover, the effect of the decay chain and the amount of graphite dust released to the environment were investigated using the MENTAS code.

Future work should focus on the verification and validation of the MENTAS code.

Acknowledgements

This work was supported by the National Research Foundation of Korea (NRF) grant funded by the Korea government (MSIT) (No. 2020M2D4A2067322).

REFERENCES

[1] INL, Mechanical Source Terms White Paper, INL/EXT-10-17997, revision 0, 2010.
[2] NEI, Risk-Informed Performance-Based Technology Inclusive Guidance for Non-Light Water Reactor Licensing Basis Development, NEI Technical Report 18-04, 2019.
[3] N. I. Tak et al., "Verification and Validation of Aerosol Analysis Module in the MENTAS Code," *Transactions of the KNS Spring Meeting*, Jeju, Korea, May 18-19, 2023.
[4] N. I. Tak et al., "Verification of Gaseous Radionuclide Transport Module in the MENTAS Code," *Transactions of the KNS Autumn Meeting*, Gyeongju, Korea, Oct. 26-27, 2023.
[5] S. Yuk et al., Neutronics and Safety Analysis Coupling for VHTR Accidents: Design-Basis Accident and Beyond Design-Basis Accident, KAERI/TR-9287/2022, KAERI, 2022.
[6] H. C. Lee et al., "Decay Heat Analysis of VHTR Cores by Monte Carlo Core Depletion Calculation," *Ann. Nucl. Energy*, Vol. 37, pp. 1356-1368, 2010.
[7] L. M. Carruthers, C. E. Lee, LARC-1: A Los Alamos Release Calculation Program for Fission Product Transport in HTGRs During the LOFC Accident, LA-NUREG-6563-MS, 1976.