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Source Term Analysis of Accident Tolerance Fuel

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1. Introduction

Accident tolerance fuel (ATF) is new fuel design which improve fuel performance such as low hydrogen generation under accident and high thermal conductivity of fuel to limit of fuel failure and progression of severe accident. Various concept of ATF, advanced cladding technology such as iron-based cladding, SiC ceramic cladding and fuel pellet technology such as U-Si type pellet, large grain pellet, metal microcell pellet, was suggested. Coating cladding and adopt pellet for large grain is considered as a near term technology. Westinghouse suggested Cr coated cladding with cold spray, and Framatome suggested Cr coated cladding with PVD. Both companies are conducting LTR / LTA at several NPPs [1].

In Korea, coating cladding and additive pellet / metal microplate is considered as ATF concept. KAERI developed CrAl coated cladding with AIP and Mo microcell fuel [2, 3]. KNF developed Cr coated cladding with AIP and LAS (La_2O_3, Al_2O_3, SiO_2 additive) added fuel [1]. Now we considered Cr coated cladding and LAS or MoLAS which combined with LAS and Mo-plate technology as near-term technology of ATF.

However, though new material not used previously in fuel is added, various properties such as nuclear, thermo-mechanical change is expected. In particular, due to large neutron absorption cross-section of Cr coating and Mo-plate activation, fuel cycle length decrease, source term change are expected. Therefore, effect of fuel change should be checked.

In this study, we have conducted basic research for analyzing the change of spent fuel criticality, shielding and decay heat. In general, spent fuel shielding and decay heat analysis are a two-step process. First, source term of fuel is generated by Origen code, part of scale code package [4]. Then shielding analysis is conducted using monte carlo code and decay heat calculation is conducted by fluid code. In this step, source term or decay heat generated by Origen code was used as an input data of each code. Major fuel assembly models (W14×14, W16×16, W17×17, CE16×16, etc.) are included in Origen code, but ATF model is not. Therefore, we made the ATF model for Origen code using Triton code. To compare the effect of each change, cladding and fuel, we modeled four types of considered ATF concept. Then, we conducted source term analysis and analyzed the effect of cladding and fuel changes.

2. Triton Modeling

Triton code, which is one part of scale code package, produce cross section data and calculate fuel depletion. To calculate model not included in Origen code, it is needed to model new fuel and produce cross section data.

In this study, we made four type of ATF models to evaluate the effect of cladding and fuel change. All ATF models are shown in Table 1. Type 1 is a reference model which is based on traditional Zr alloy – UO2 fuel. Cr coated Cladding was applied in type 2. Natural Cr isotope was used, and coating thickness was set as 15 ㎛. MoLAS fuel was applied in type 3. Mo content was set as 3 vol% and nature isotope was used. Although large neutron absorption cross-section of Mo-95 would affect source term, depleted Mo was not considered in this study. LAS content was set as 1000 ppm. Mo-plate was heterogeneously mixed in the fuel, but it was confirmed by MCS code that this did not affect the cross-section. Both cladding and fuel change were applied in type 4.

When modeling MoLAS fuel, it was assumed that the content of U-235 was the same for all types and that the Mo-plate and LAS replaced U-238. This is because, if the content of U-235 is changed, the effect by Cr coating and MoLAS cannot be confirmed.

Fig. 1 shows that schematic diagram of type 2 and type 4. HIPER16 fuel assembly was used as a basic fuel assembly design [5].

Table 1 Type of Triton Modeling

Model	Type 1	Type 2	Type 3	Type 4
Fuel Material	UO_2	UO_2	MoLAS	MoLAS
Clad material	HANA	HANA	HANA	HANA
Coating material	-	Cr	-	Cr
Enrichment (wt% U-235)	4.65%	4.65%	4.65%	4.65%
Coating Thickness (cm)	-	0.0015	-	0.0015
Guide tube material	ZIRLO	ZIRLO	ZIRLO	ZIRLO

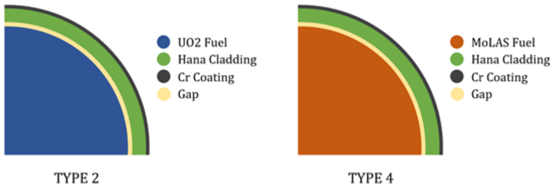

Fig. 1 Schematic diagram of each fuel model

3. Source Term Analysis

Before calculating source term, we selected major nuclide by referring to ISG-8 Rev 3 [6]. In this guidance, 28 nuclides that have a major impact on burnup credit were presented. These nuclides were listed in Table 2.

Table 2 Analyzed nuclides in this study

Nuclides			
Ag-109	Mo-95	Pu-241	Sm-151
Am-241	Nd-143	Pu-242	Sm-152
Am-243	Nd-145	Rh-103	Tc-99
Cs-133	Np-237	Ru-101	U-234
Eu-151	Pu-238	Sm-147	U-235
Eu-153	Pu-239	Sm-149	U-236
Gd-155	Pu-240	Sm-150	U-238

To compare the nuclides composition following burnup, 20, 40, 60 GWd/MtU cases were calculated. Effective full power day was assumed to 500 days and relative power following burnup was set as constant. Cooling time between each cycle was assumed to 30 days. After irradiation, 7 years of cooling time was considered. Source term analysis was conducted using Origen code.

The effect of Cr coating was negligible in case of source term. The difference of type 1 and type 2 was about 1%. Although all cladding was coated to Cr, total content of Cr was low to affect the fission product contents. However, MoLAS fuel affected to nuclides composition. In case of Actinide nuclides, Pu-238 and U-235 contents difference were about 4%. This is due to large neutron absorption cross-section of Mo-95. The neutron absorption of U-235 and Pu-239 was reduced due to the competitive effect, and it seems that the combustion of fissile material was reduced.

U-238 content of type 3 and type 4 is lower than that of type 1 and 2. This is not ATF effect, but low initial content of U-238.

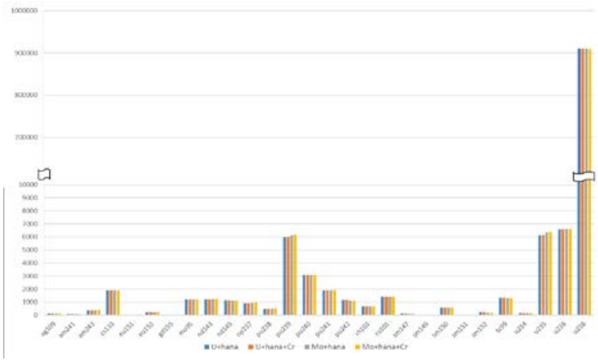

Fig. 2 Source term at 60 GWd/MtU

4. Conclusion and Future Works

In this study, ATF modeling and source term analysis were conducted. Cr coating does not affect to source term of spent fuel. However, due to Mo additive in fuel source term of spent fuel was changed. In future, we will verify the ATF model by comparing monte carlo code results. It could be changed according to nuclear design results After verifying, decay heat and shielding analysis will be calculated.

Acknowledgement

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