Sensitivity Analysis of Actinide Decay Heat Focused on Mixed Oxide Fuel

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Nuclear power plants in the world accumulate a vast amount of spent fuels. In Japan, the spent fuel is reprocessed and is not disposed directly. Therefore utilization of this reprocessed fuel, i.e. Mixed Oxide (MOX) Fuel, will eventually expand. Under this background, hereafter, the MOX fuel will play a role of greater importance. It is, then, essential to understand the characteristics of the spent MOX fuel. In this work, the uncertainty of the fission-product and actinide decay heats was studied introducing the uncertainty of the nuclear data and the prediction accuracy of isotopic generation. At 100 year after discharge, for example, 7% error of the decay heat will be introduced by 10% uncertainty of Pu-241 generation calculation through Am-241 production.

1. Introduction

There are 440 nuclear power plants in the world. They hold a vast amount of spent fuels. In Japan, the spent fuel is reprocessed and is not disposed directly. Therefore utilization of this reprocessed fuel, i.e. Mixed Oxide (MOX) Fuel, will eventually expand.

Under this background, hereafter, the MOX fuel will play a role of greater importance. It is, then, essential to understand the characteristics of the spent MOX fuel. Above all, the radioactivity and decay power of spent MOX fuel are very crucial in storage or disposal terms. In this work, the uncertainty of the fission-product and actinide decay heats was studied introducing the uncertainty of the nuclear data and the prediction accuracy of isotopic generation.

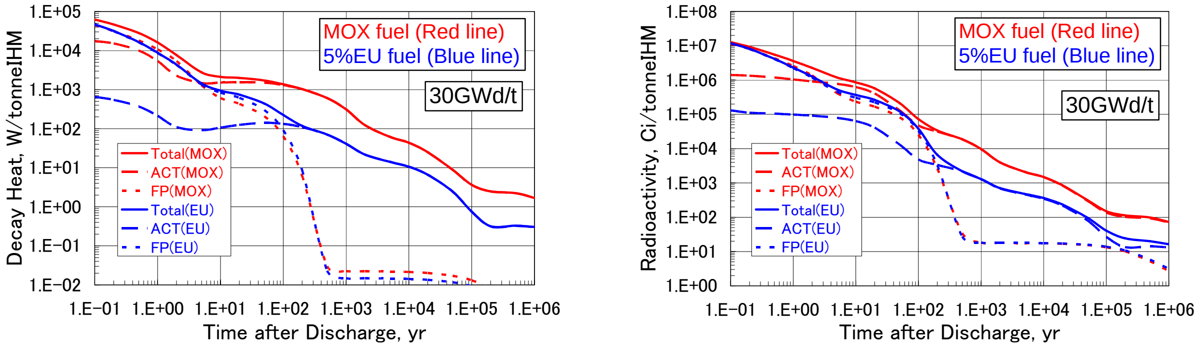
2. Procedure of Analyses

In this study, we made use of SWAT code system[1] developed by Japan Atomic Energy Research Institute (JAERI) and ORIGEN2 code[2] developed by Oak Ridge National Laboratory (ORNL). Burnup calculation is performed by SWAT code, and decay heat calculation is carried out by ORIGEN2 code. In SWAT, typical PWR pin cell model is selected as calculation model. Component of MOX fuel is specified as PWR33G-MOX[3] (enrichment of 0.22% depleted uranium and 5.3% Pu). Calculation conditions are as follows: Thermal power is 37.9MW/t, normal discharged burnup is 30GWd/t, and library is JENDL3.3. Isotopic inventory calculated by SWAT is translated into the input of ORIGEN2 code.

3. Calculation Results

3.1 Spent fuel characteristics in perspective

For keeping the decay heat and radioactivity of spent fuel in perspective, Fig.1 shows the results of standard calculation of this study. On these figures, red line means MOX fuel, while blue line means 5% enriched uranium (EU) fuel. We should pay attention to the Actinide component. The difference that exceeds one decade is seen between MOX fuel and EU fuel in the actinide component. As a result, total decay heat (ACT and FP) has been raised. In each cooling time, the nuclides that mainly show contribution are as follows: Cm-242 (~1yr), Pu-238 (~100yr), Am-241 (~500yr), Pu-240 (~10^4yr), Pu-239 (~10^5yr), and nuclides of neptunium series (>10^5). Characteristics of radioactivity also indicate similarity between MOX fuel and EU fuel (Fig.1 (b)).



(a) Absolute of Decay heat of spent fuel (b) Absolute of Radioactivity of spent fuel

Fig.1 Characteristics of spent fuel at 30 GWd/t

Figure 2 shows each decay heat by deference of discharged burnup per unit amount of power generation[4]. The representation of decay heat per unit amount of power generation is appropriate for the viewpoint of the economy. On MOX fuel the curve per unit amount of power generation of this figure leads the advantage of higher burnup approach clarify.

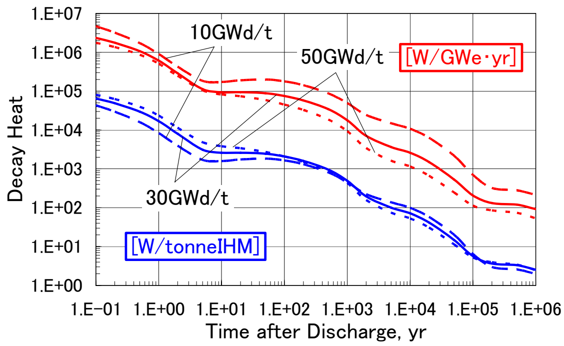
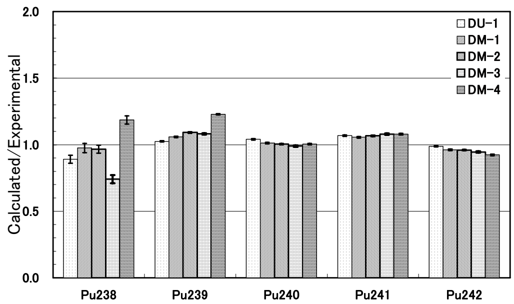


Fig.2 Decay heat per gigawatt (electric) year versus time after discharge

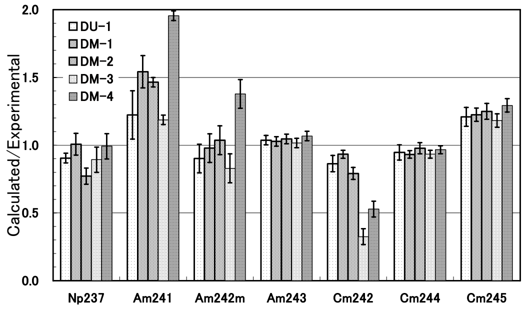
3.2 Impact of uncertainty of prediction accuracy of isotopic generation

Figure 3 indicates an example[5] of discrepancy of C/E ratio for Pu vector and MAs. These figures means that Pu isotopes have a few percent disagreement and that miner actinide isotopes have tens of percent disagreement. Therefore, evaluation of decay heat containing their uncertainty is required. On Pu-238, -239, -240, -241, Am-241, Cm-242, the ratio to the standard calculation result is shown about decay heat when the combustion calculation result contains the uncertainty from 2 to 20 percent as follows (Fig.4 (a) - (d)).

These figures show the variations of decay-heat behavior with respect to the intentional charge in the discharge amount of each isotope, for example ^{239}Pu in Fig.4 (a). The peak in Fig.4 (c), is not coming directly from Pu-241 itself but through Am-241. Pu-241 decays into Am-241 with the half-life of 14.35 years. It is Pu-241 that dominates the amount of Am-241 long after discharge. Am-241 is an important nuclide for decay heat for an important period for storing spent fuels from ten-odd years to thousands of years. According to this figure, at 100 year after discharge, 7% error of the decay heat will be introduced by 10% uncertainty of Pu-241 generation calculation through Am-241 production.

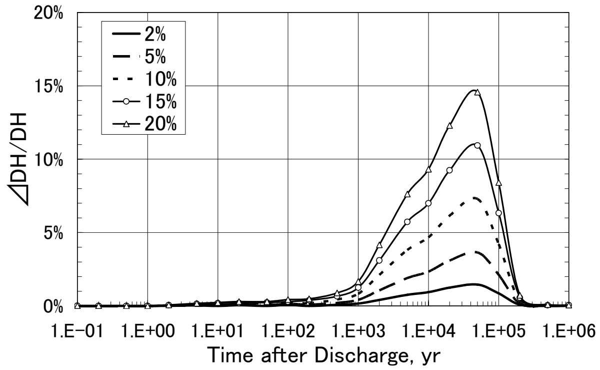


(a) Pu vector

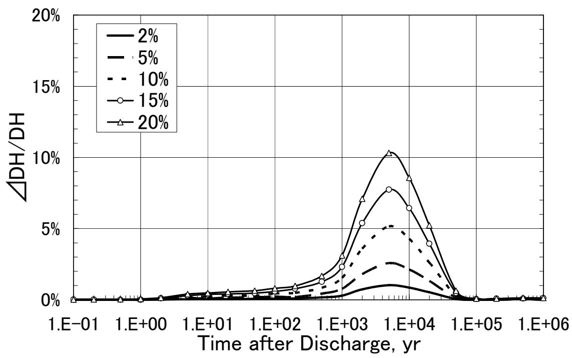


(b) MAs

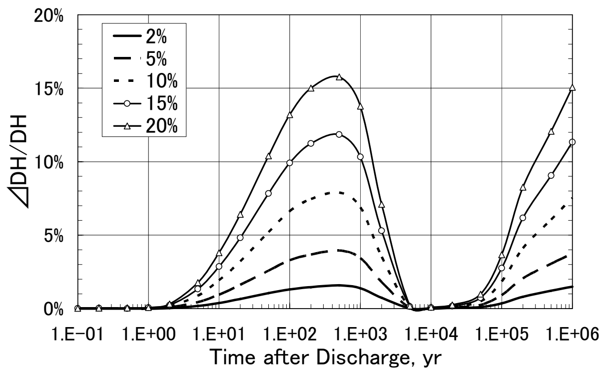
Fig.3 C/E ratio for Pu-isotopes and minor actinides[5]



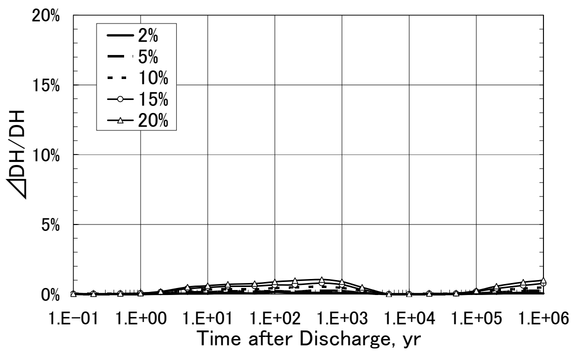
(a) Pu-239



(b) Pu-240



(c) Pu-241



(d) Am-241

Fig.4 Uncertainty of the FP and actinide decay heats introduced by uncertainty of the prediction accuracy of isotopic generation of each nuclide

3.3 Impact of uncertainty of one-group cross section

On the spent fuel storage term (several 100 years from 10 years), Am-241 is dominant nuclide. Importance of the amount of Pu-241 has already been described. In a reactor in operation, generation of Pu-241 is remarkable. Evaluation value[6] of uncertainty of Pu-240 capture cross sections is shown in Figure 5. In this figure, capture cross section of Pu-240 has 20% from 10% uncertainty in whole energy range. Then the influence on decay heat when capture cross sections of Pu-240 and Pu-239 were changed from 10 to 50 percent was examined.

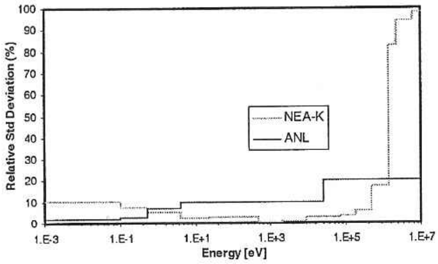
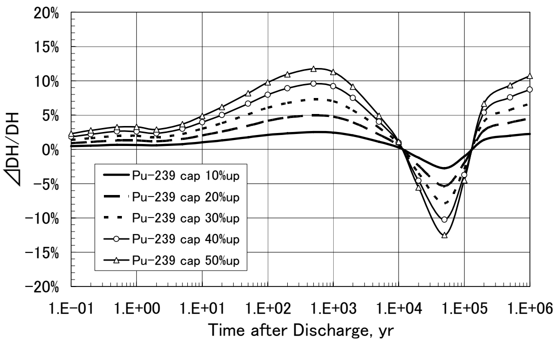
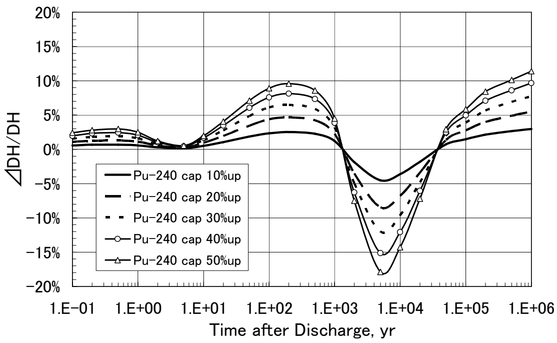


Fig.5 Evaluation value of uncertainty of Pu-240 capture cross sections[6]

Figure 6 shows the peak caused by Am-241 around 500 year cooling. Fig.6 (a) indicates that decay heat increases by 3% when the capture cross section of Pu-239 is increased by 10%. Fig.6 (b) shows that decay heat increases by 5% when the capture cross section of Pu-240 is increased by 20%. The dip of about 10^4 year results from reduction of Pu-239 and Pu-240, respectively.



(a) Pu-239



(b) Pu-240

Fig.6 Uncertainty of the total decay heats introduced by uncertainty of the one-group capture cross section

4. Summary and Future Plan

It is important to evaluate the characteristics of spent MOX fuel, especially decay heat, appropriately. In this study, sensitivity analyses of prediction accuracy of isotopic generation and uncertainty of one-group cross section were performed. Between 100 and 1,000 years after discharge of the spent MOX fuel, the total decay heat is dominated by Am-241. We should pay attention to the generation of Pu-241 which decays into Am-241. The amount of Pu-241 has a close relation to the capture cross section of Pu-239 and -240. The uncertainty of the burnup calculation is discussed. At 100 year after discharge, for example, 7% error of the decay heat will be introduced by 10% uncertainty of Pu-241 generation calculation through Am-241 production.

We will make sensitivity analyses of capture cross section of notable isotopes on each energy group.

Reference

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