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Impact of Fuel Type and Discharge Burnup on Spent Fuel Properties

Pauli Juutilainen, Silja Häkkinen
VTT Technical Research Centre of Finland
Kivimiehentie 3, 02150 Espoo, Finland
pauli.juutilainen@vtt.fi, silja.hakkinen@vtt.fi

ABSTRACT

As a part of the project to formulate rigorous guidelines to characterize the spent nuclear fuel source term with the computational tools used at VTT Technical Research Centre of Finland, the Serpent Monte Carlo code was employed to perform burnup calculations for four different fuel assemblies. The purpose of the calculations was to determine the impact of fuel type and discharge burnup on the spent fuel properties that are significant in spent fuel management, first in design of interim storage pools and later in final disposal. Decay heat production is the main parameter studied in the present article, but the Serpent calculations provide data also on photon emission rates and spontaneous fission rates as well as the radiotoxicity of the spent fuel, but these are not considered in the present study.

1 INTRODUCTION

The spent nuclear fuel (SNF) characteristics such as decay heat and reactivity determine the number of assemblies that can be loaded in a final disposal canister and hence have a great impact on the volume needed in the underground facility. Also other components of the source term such as e.g. nuclide inventory determining the radioactivity of SNF, are essential in the safe handling and disposal of the SNF. However, the computational characterization of spent fuel assemblies involve several sources of uncertainty such as e.g. uncertainties in nuclear data, impurities in fuel and structural materials, choice of calculation parameters, uncertainties in operation history etc. Also different kind of fuel assemblies and different levels of burnup have a significant impact on the SNF source term. These uncertainty components and their effects are studied in the KYT project KÄRÄHDE.

The commercial global nuclear fleet includes many different kind of reactor types such as e.g. PWRs, BWRs, VVERs CANDUs and AGRs. Even SFRs are currently employed in commercial power production in Russia. In Finland alone two types of reactors, BWR and VVER-440, are presently operated. Additionally, an EPR unit is expected to start operation in the near future and a VVER-1200 is in preparation to be constructed in the 2020s. Even within one reactor type different kind of fuel assemblies have been and are planned to be used. For example in the Olkiluoto reactors several fuel assembly types such as GE14, Atrium-10XM and SVEA-96 Optima have been used [1], [2]. In the Loviisa reactors TVEL and BNFL manufactured assemblies have been used with different enrichments with and without burnable absorber rods [3]. Fuel development has also enabled higher burnup levels compared to the earlier fuel types. This paper focuses on the effect of different fuel types and burnups on the source term concentrating on the reactor and fuel types used and planned to be used in Finland.

2 METHODS

The calculations were performed with the continuous-energy Monte Carlo code Serpent [4] that has been developed at VTT over the last 15 years for various reactor physics applications, such as spatial homogenization, criticality calculations and fuel cycle studies. As a later topic, radiation shielding tools have been added to the code. In the present work, neutron transport and burnup calculation functionalities were employed to provide source term information for spent fuel management. The code version 2.1.31 was used and the nuclear data were obtained from JEFF-3.2 cross-section library and JEFF-3.1.1 fission yield and decay libraries.

In all simulations, the fuel assemblies were irradiated up to 80 MWd/kgU burnup as an anticipation for extended discharge burnups in the future. The burnup step lengths slightly varied between the simulations, but in all cases, the xenon equilibrium was traced with very short steps in the early phase and somewhat longer steps - 0.5 MWd/kgU - were applied from 1 to 25 - 30 MWd/kgU to ensure full depletion of the burnable absorbers. The step of 2.5 MWd/kgU was generally applied for higher cumulative burnup, but a few modifications were done to set a burnup point at the average discharge burnup of the assembly type.

Linear extrapolation and quadratic interpolation with 10 substeps were used in the predictor and corrector, respectively.

Every pin was handled as a separate depletion zone in burnup calculation. Pins containing burnable absorber were additionally divided to 10 equally large depletion zones. The automatic zone division tool of Serpent was utilized for these.

At each step, 10 million neutron histories were simulated. It yielded the statistical uncertainty of roughly 15…20 pcm, however, indicating growing trend up to ~25 pcm when the multiplication factor k_{eff} decreased along with higher burnup.

2.1 Studied fuel assemblies

The studied fuel assemblies include typical BWR (GE14) [5] [6], VVER-440 (TVEL 2nd gen) [3] [7] [8], EPR [9] [10], and VVER-1200 [8] [11] assemblies. Some assembly characteristics including the boron concentration used in the calculations are presented in Table 1. The uranium mass has been calculated for a 1 cm thick slab. The actual active length of the assembly is insignificant since the calculations have been performed in two dimensions.

Fresh natural uranium -based UOX fuel was assumed in all cases excluding the VVER-1200 assembly for which recycled uranium was assumed. In practise, the recycled uranium assumption adds a small amount of U-236 - 0.7 wt-% in these calculations - into the fresh fuel. Additionally, 10 ppm impurities of both N-14 and Cl-35 were assumed for all modelled assemblies.

Table 1. Some characteristics of the studied assemblies

Param.	BWR	VVER-440	EPR	VVER-1200
U mass [kg/cm]	0.505	0.520	1.21	1.29
U-235 [%]	4.23	4.37	3.56	4.92
Normal rods	74	120	253	306
Gd rods	18	6	12	6
Boron [ppm]	0	500	600	600

3 RESULTS

Some total decay heat values per uranium mass of fresh fuel are presented in Table *2*. The BWR assembly seems to slightly differentiate with its lower heat emission. In contrast, the VVER-1200 fuel assembly with recycled uranium shows somewhat higher heat production, as expected. The heat production is one of the limiting factors in dimensioning the interim storage configurations and final disposal cavities. However, when the final disposal is considered, the maximum temperature is supposed to achieve its peak within 50 - 100 years after disposal [12], so the later heat production is of smaller interest.

The relative difference between the highest and lowest heat producing assembly per initial fuel mass at each time point is depicted in Figure 1. The difference is rather strongly dependent on the discharge burnup. It shows decreasing trend immediately after the discharge, but grows significantly during the period when the spent fuel is going to be moved to final disposal from interim storage. It is worthwhile to note, however, that the heating rates decrease to rather low levels towards the end of the depicted period, when even small inaccuracies in absolute values may cause large relative deviations.

Table 2. Total linear decay heat production per unit mass (in W/tU) at a few cooling times after discharge. Discharge burnup of 50 MWd/kgU was assumed for all cases. Time “0” refers to the end of irradiation

Time (years)	BWR	VVER-440	EPR	VVER-1200
0	1.745e06	2.352e06	2.062e06	2.604e06
5	2581	2870	2881	3032
100	395	473	458	560
500	109	144	136	149
1000	63	81	77	82

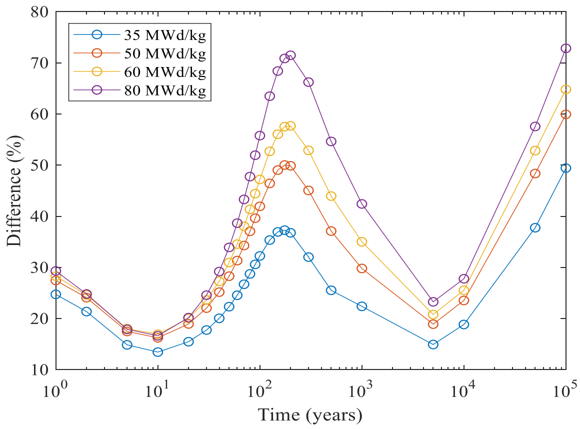

Figure 1. Relative difference between the highest and lowest heating power per unit mass of initial U in the studied assemblies as a function of time from discharge.

An example of the effect of discharge burnup on the decay heat is presented in Figure 2. The figure depicts the relative heat production of the spent

VVER-1200 fuel over the first 175 years after discharge with respect to the fuel discharged at 50 MWd/kgU. The period is assumed to sufficiently cover the period over which the decay heat can be a limiting factor in storage site dimensioning.

The other studied fuel assemblies show rather similar behaviour with the relative heat production. However, a difference between VVER-1200 assembly and the other studied assemblies can be observed with the highest calculated burnup, 80 MWd/kgU, whose relative peak heat production in the beginning is similar or higher than that of VVER-1200, but drops faster. In the end of the portrayed period, BWR and EPR spent fuel produce heat less than 40 % above the reference, whilst the respective ratio for the VVER-440 fuel is nearly 50 % and VVER-1200 fuel slightly less than 60 %.

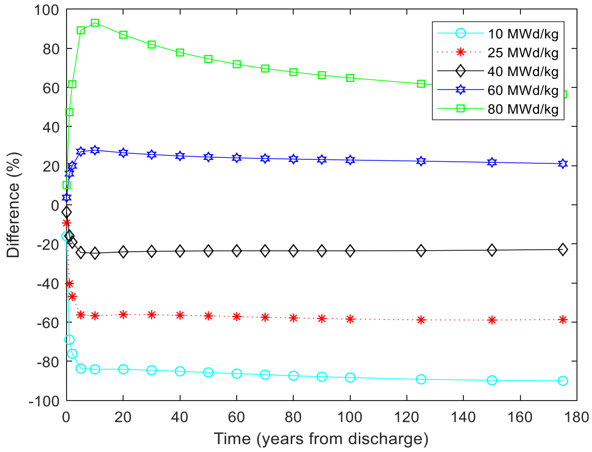

Figure 2. Relative heat production of spent VVER-1200 fuel as a function of time with different discharge burnups compared to 50 MWd/kgU

The heat production in the spent fuel at each calculated cooling time is largely dominated by the top heat-producing nuclides, even though their share somewhat drops in the beginning. Figure 3 presents the contribution of the five largest decay heat producers as the fraction of total heat production from each of the studied fuel assemblies, following the discharge burnup of 50 MWd/kgU. The top heaters play slightly larger role in VVER-1200 fuel, but otherwise these contribute similarly.

When the importance of the main heaters is studied with different discharge burnups, the behaviour of their contribution is similar, but the differences between the fuel types increase with both smaller and higher discharge burnups than the 50 MWd/kgU depicted in Figure 3.

With all studied fuel types and 50 MWd/kgU discharge burnup, the five most heat producing nuclides are mainly the same, however, the order of the nuclides vary soon after the discharge. Later, that is, 150 years after the discharge, the top-5 lists are identical between the fuel types. An example of such top heater nuclides is presented in Table 3.

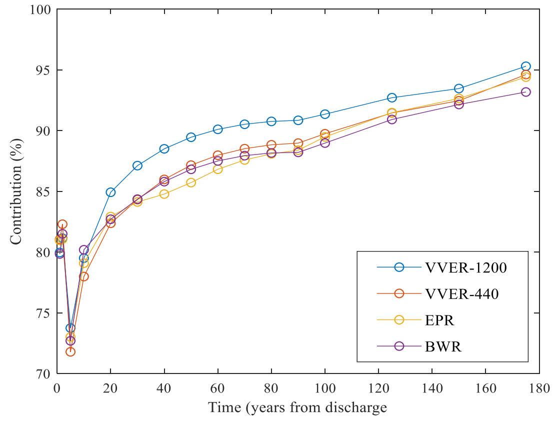

Figure 3. Share of the top-5 heat producers of total heat production in the spent fuel discharged at 50 MWd/kgU.

Table 3. The top-5 heat producing nuclides five years after discharge by fuel type and 50 MWd/kgU discharge burnup. The rightmost column presents the respective list of heat producers 150 years after discharge, when no difference between the fuel types exists.

After 5 years				**After 150 y**
BWR	**EPR**	**VVER-440**	**VVER-1200**	**All**
Y-90	Cs-134	Cs-134	Cs-134	Am-241
Ba-137m	Ba-137m	Y-90	Y-90	Pu-238
Cs-134	Y-90	Ba-137m	Ba-137m	Pu-240
Cm-244	Cm-244	Rh-106	Pu-238	Ba-137m
Rh-106	Rh-106	Cm-244	Rh-106	Y-90

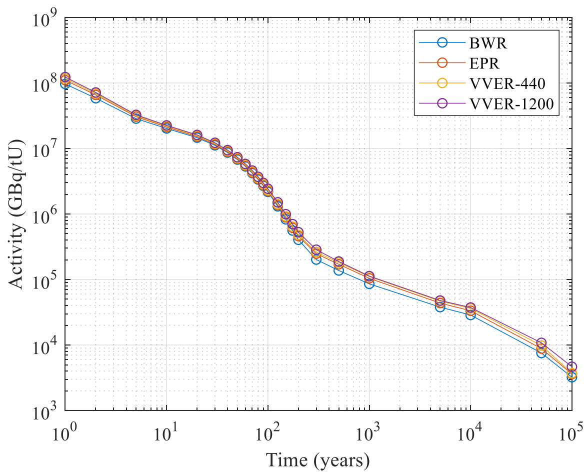

Figure 4. The total activity from 1 to 100,000 years cooling time of the studied fuel types with 50 MWd/kgU discharge burnup.

The total activity per initial uranium mass for all studied assembly types is depicted in Figure 4.

The discharge burnup of 50 MWd/kgU is assumed in the figure. Whilst the logarithmic scale does not make it easy to see the differences, the relative difference between the most and least active assemblies behaves similarly to the difference in heat production which was presented in Figure *1*. The heat production naturally depends on the activity, but due to the changes in the unstable inventory and thus the dominating decay reactions, the heat production decreases much more slowly than the activity over the first centuries.

4 CONCLUSIONS

The decay heat of the spent fuel is one of the key parameters to consider when designing a final disposal site for the spent nuclear fuel. As a part of the process to formulate a standardised methodology to define the source term with VTT's computational tools, four different types of fuel assemblies were simulated with Serpent Monte Carlo code and some of the results were presented in this article. The relative difference in the decay heat production between different assembly types may be significant, but on the other hand, the most important heating nuclides are approximately - if not completely - the same with all types.

The discharge burnup of 50 MWd/kgU was used as the reference in these calculations. For higher burnups, the increase in heat production is roughly in line with the increased amount of extracted energy in the long term, but for the interim storage period the relative increase in heat production of the spent fuel is much higher than the increase in obtained energy.

It is worthwhile to note that the calculations were performed for two-dimensional models with periodic boundary conditions in the absence of better information. Additional approximations are related to the assumptions of irradiation history, such as the boron concentration that was assumed constant.

ACKNOWLEDGEMENTS

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