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Marjan Kromar
Jožef Stefan Institute

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Marjan Kromar, and Bojan Kurinčič

Determination of the NPP Krško Spent Fuel Decay Heat

Marjan Kromar[1, a)] and Bojan Kurinčič[2, b)]

[1] *"Jožef Stefan" Institute, Reactor Physics Division, Jamova 39, SI-1000, Ljubljana, Slovenia*
[2] *Nuclear Power Plant Krško, Engineering Division - Nuclear Fuel & Reactor Core, Vrbina 12, 8270 Krško, Slovenia*

[a)] Corresponding author: marjan.kromar@ijs.si
[b)] bojan.kurincic@nek.si

Abstract. Nuclear fuel is designed to support fission process in a reactor core. Some of the isotopes, formed during the fission, decay and produce decay heat and radiation. Accurate knowledge of the nuclide inventory producing decay heat is important after reactor shut down, during the fuel storage and subsequent reprocessing or disposal. In this paper possibility to calculate the fuel isotopic composition and determination of the fuel decay heat with the Serpent code is investigated. Serpent is a well-known Monte Carlo code used primarily for the calculation of the neutron transport in a reactor. It has been validated for the burn-up calculations. In the calculation of the fuel decay heat different set of isotopes is important than in the neutron transport case. Comparison with the Origen code is performed to verify that the Serpent is taking into account all isotopes important to assess the fuel decay heat. After the code validation, a sensitivity study is carried out. Influence of several factors such as enrichment, fuel temperature, moderator temperature (density), soluble boron concentration, average power, burnable absorbers, and burnup is analyzed.

INTRODUCTION

In a nuclear reactor, the fission of heavy atoms such as isotopes of uranium and plutonium results in the formation of highly radioactive fission products. Due to neutron capture higher actinides are formed, some of them being unstable. All these unstable isotopes radioactively decay and produce decay heat and radiation. Some radioactive atoms will decay while the reactor is operating and the energy released by their decay will be removed from the core along with the heat produced by the fission process. All radioactive materials that remain in the reactor at the time it is shut down and the fission process halted will continue to decay and release a significant amount of thermal energy. It is of high importance to precisely know the amount of decay heat in order to assess core and containment cooling capabilities during normal operation and postulated abnormal events. In addition, nuclear fuel is after its use in the reactor stored in the pool, where pool heat exchangers remove excessive heat. Accurate knowledge of the inventory is therefore important also during the storage of the fuel and subsequent reprocessing or disposal.

In this paper possibility to calculate the fuel isotopic composition and determination of the fuel decay heat with the Serpent code [1] is investigated. Serpent is a well-known Monte Carlo code used primarily for the calculation of the neutron transport in the reactor. It has been validated for the burn-up calculations [2]. However, in the calculation of the fuel decay heat, different set of isotopes is important than in the neutron transport case. For the purpose of this evaluation only the radioactive decay of fission products and actinides is considered. A typical case of the NPP Krško fuel is selected for comparison with the Origen code [3]. Origen is a well-known computer code system for calculating the buildup, decay, and processing of radioactive materials. Comparison with the Serpent code is performed to verify that the Serpent is taking into account all isotopes important to assess the fuel decay heat. A similar analysis comparing fuel radioactivity has been performed in [4] showing promising results. After the code validation a sensitivity study is carried out. Fuel isotopic composition is namely pretty dependent on the neutron spectrum and consequently on fuel operating conditions. Influence of several factors such as boron concentration, burnable poison presence, water density etc. is analyzed.

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BRIEF CODE DESCRIPTION

Origen

The Oak Ridge Isotope Generation (ORIGEN) depletion/decay code was developed at ORNL in 1973. Since then, many new versions have been created [3] and are available from RSICC and NEA data bank. It is a well-known point-depletion inventory code and has been used to model nuclide transmutation with capability to generate source terms for accident analyses, characterize used fuel (including activity, decay heat, radiation emission rates, and radiotoxicity), activate structural materials, and perform fuel cycle analysis studies. Origen uses a matrix exponential method to solve a large system of coupled, linear, first-order ordinary differential equations with constant coefficients (Bateman equations). Version 2.1 with “pwrue” library has been used in this paper.

Serpent

Sensitivity study of the fuel decay heat is performed with the Serpent code [1, 2]. Serpent is a three-dimensional continuous-energy Monte Carlo reactor physics burnup calculation code, developed at the VTT Technical Research Centre of Finland. It is not a typical Monte Carlo code. While the majority of the other codes use the ray-tracing algorithm as a transport model, Serpent uses the Woodcock delta tracking method. In this way the geometry routines get simplified and calculations are faster compared to conventional Monte Carlo codes. The improved matrix exponential method CRAM (Chebyshev Rational Approximation Method) for solving the Bateman equations [5] has been implemented for the burnup applications. Serpent uses a continuous energy neutron cross section library in an ACE format based on the ENDF/B-VII evaluated nuclear data library [6]. The code is specialized for two-dimensional lattice physics calculations, but the universe-based geometry description allows the modelling of complicated three-dimensional geometries as well. Detailed geometrical modelling of the NPP Krško fuel assembly in the Serpent code enables accurate determination of fuel isotopics and consequently determination of fuel decay heat. In this analyses Serpent 2, version 2.1.24, has been used.

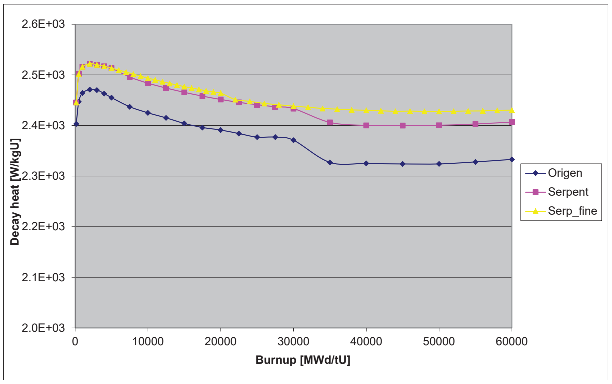
CODE COMPARISON

The NPP Krško is a 2-loop Westinghouse PWR that began electricity production in 1981. The plant uses standard 16 × 16 fuel with some VANTAGE+ features. A typical fuel assembly with 4.95 % enrichment and no IFBA (Integral Fuel Burnable Absorber) rods was selected as a starting point. A reference case scenario consists of the following reactor operational parameters:

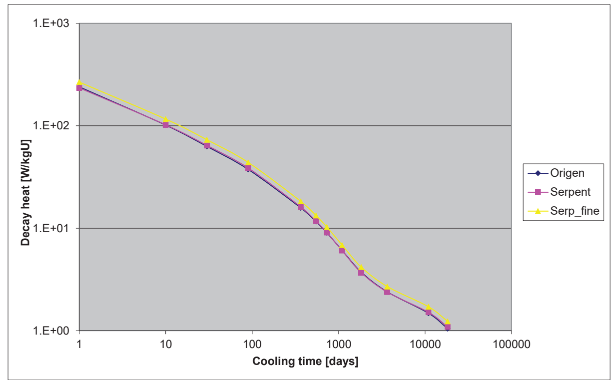
1. Fuel temperature 900 K,
2. Moderator temperature 580.46 K with density 0.70871 g/cm^3,
3. Soluble boron concentration of 1000 ppm.

Parameters are close to the average operational parameters applied in the last NPP Krško cycles. Comparison of both codes is presented in Fig. 1(a), where decay heat per kg of initial fuel uranium is plotted for the burnups up to 60000 MWd/tU. Agreement between the two codes is reasonably good. Differences are within 2-3 %. Closer examination shows some disturbances at burnup points where burnup interval is increased i.e. from 1000 MWd/tU to 2500 MWd/tU (at 5000 MWd/tU) and from 2500 MWd/tU to 5000 MWd/tU (at 30000 MWd/tU). The reason behind it is a continuous variation of the fission nuclide inventory and neutron energy spectrum during the specific burnup step. The effect is present in both codes, since they use basically the same matrix exponential method. To evaluate possible discrepancies, burnup interval was shortened to 1000 MWd/tU up to burnup of 20000 MWd/tU and 2000 MWd/tU for higher burnups. Results are plotted in Fig. 1 and are denoted as “Serp_fine”. A nonlinearity at 20000 MWd/tU can still be seen but the curve is much smoother than before. Relative differences due to finer burnup mesh calculations near the final burnups are around 1 %. We can make a conclusion, that burnup calculations should be performed at most with 1000 MWd/tU burnup interval increment, if a very high calculation accuracy is required.

Usually we are more interested in the decay heat during the fuel cooling period. For this purpose fuel cooling was modelled at the end of the 60000 MWd/tU burnup interval. Results are presented in Fig. 1(b). Agreement between the two codes’ results is again very good; differences are less than 1 % for the cooling period of up to 5 years. Both figures are indicating that the majority of discrepancies between codes could be contributed mainly to the differences in the short lived isotopes.



(a)



(b)

FIGURE 1. Origen and Serpent comparison during the fuel (a) burnout and (b) cooling

Effect of the finer burnup mesh during the fission process is again visible and is even stronger than the differences between both codes. However, computational requirements for the Serpent code are substantial. It takes almost an hour of the wall clock time to calculate one burnup point on the cluster node with 2.93 GHz processor having 12 cores and 24 threads. Since at this stage we are not interested in the maximal absolute accuracy but only relative comparison, it was decided to use coarser time mesh in the further evaluation.

SENSITIVITY STUDY

Decay heat is produced by the radioactive decay of unstable nuclides. Isotopic inventory of such unstable nuclides is constantly changing during the reactor operation and is a function of several parameters such as neutron energy spectrum, fission nuclides inventory present etc. Reactor operating conditions (temperatures of the fuel and moderator, water density, neutron absorbers etc.) directly influence neutron spectrum. It is prudent to evaluate effect of these parameters for the development of optimal calculation strategy to achieve desired accuracy. In addition, in the decay heat evaluations usually some averaging of material properties or conditions is applied. It is not self-evident that such averaging process would yield also an averaged decay heat. To verify such assumptions a sensitivity analysis is needed to confirm such approach or to identify crucial nonlinear parameters.

Enrichment

The effect of the fuel enrichment is shown in Fig. 2. Two cases (2.1 % and 3.525 %) are compared to the reference 4.95 % case. It can be estimated that a decrease of 1 % in enrichment can induce increase of up to 7 % in the decay heat. However, the effect is not linear over the cooling period.

As already mentioned, in practice it is a usual procedure to calculate larger fuel areas with some average properties. But strictly speaking, if the fuel assembly is constituted of 2 geometrically equal regions, the one with 2.1 % enrichment and the other with 4.95 %, the average decay heat would not be equal to the decay heat of the fuel with an average 3.525 % enrichment. Differences between the average

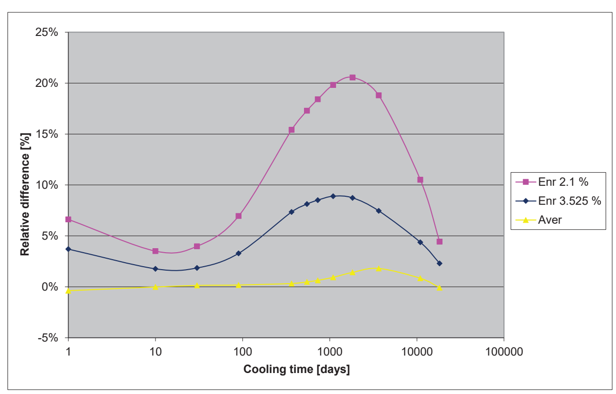


FIGURE 2. Effect of the fuel enrichment on the decay heat

decay heat of the 2.1 % and 4.95 % fuel region and the decay heat of the averaged 3.525 % region are plotted as "Aver" in Fig. 2. Differences of up to 2 % are visible in the 1000-10000 days region.

Fuel temperature

The effect of fuel temperatures is shown in Fig. 3. Two cases (800 K and 1000 K) are compared to the reference 900 K case. Relative differences of both cases are less than 0.5 %. Average decay heat is very close to the reference case supporting averaging approach.

However, in the reality fuel temperature is not constant over fuel pellet but is highest in the centre and decreases towards the cladding due to heat transfer to the moderator. If we assume constant power across the pellet, with constant thermal characteristics, we get a parabolic temperature profile. A 10 region annular case was considered assuming a parabolic temperature profile with a pellet average of 900 K. Obtained results are plotted in Fig. 3 ("T profile"). Effect is relatively small and is estimated to be less than 0.1 % in the decay heat.

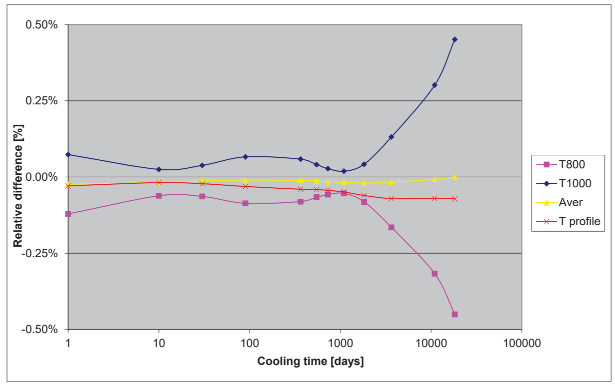


FIGURE 3. Effect of the fuel temperature on the decay heat

Moderator temperature

Direct influence of the moderator temperature on the decay heat is negligible. However, reactor is operating in the pressure region, where relatively small moderator temperature changes cause significant differences in the moderator density producing significant differences in neutron moderation and energy spectrum. The effect is shown in Fig. 4. Temperature was varied for ±20 K (water density goes from 0.74972 g/cm3 to 0.65642 g/cm3). This is approximately NPP Krško inlet–outlet range at 100 % reactor power. Therefore, upper fuel regions with near core outlet temperatures have up to 5 % higher decay heat than bottom regions. Average decay heat is up to 0.5 % higher than decay heat at average temperature.

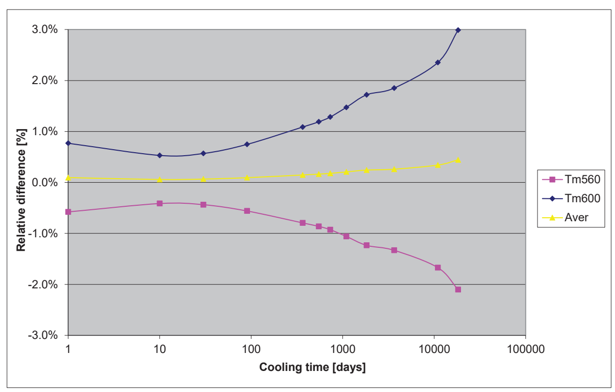


FIGURE 4. Effect of the moderator temperature on the decay heat

Soluble boron

The effect of soluble boron present in the moderator is shown in Fig. 5, where ±500 ppm variations are analyzed. 500 ppm increase in the boron concentration can cause up to 1.5 % higher fuel decay heat. Average decay heat is almost the same as the decay heat at average boron concentration supporting averaging approach.

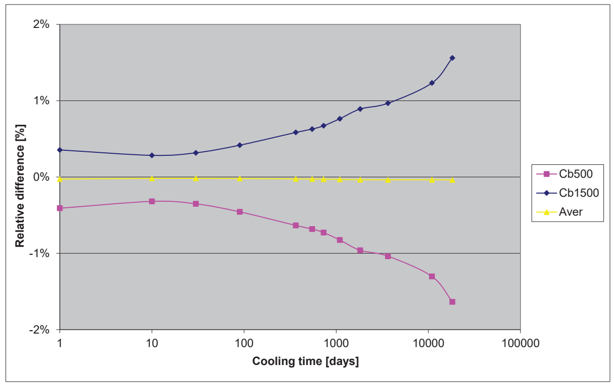


FIGURE 5. Effect of soluble boron on the decay heat

Average power

Cases with 70 % nominal power and 130 % nominal power are presented in Fig. 6. Decay heat at shutdown is almost proportional to the specific power. 1 % increase in reactor power results in almost 1 % higher decay heat. That means that the main decay heat contributors experience some saturation concentration, which is proportional to the specific power. Differences are gradually decreasing over time and are after 10 years cooling approximately 30 times smaller (less than 1 %). Average decay heat is up to 1.5 % lower than the decay heat at average specific power.

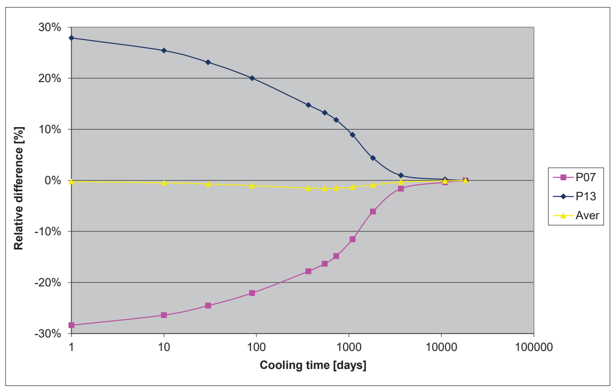


FIGURE 6. Effect of specific power on the decay heat

Burnable absorbers

Introduction of burnable absorbers causes spectrum hardening and therefore changes in fuel isotopic composition and consequently decay heat. The effect is presented in Fig. 7, where cases with 64 IFBA rods and 116 IFBA rods per assembly are plotted. The effect is relatively small, since the majority of boron in the pellet coating disappears after a few thousands MWd/tU. The fuel with 116 IFBA rods is producing only up to 0.5 % more decay heat than the fuel with no IFBA rods.

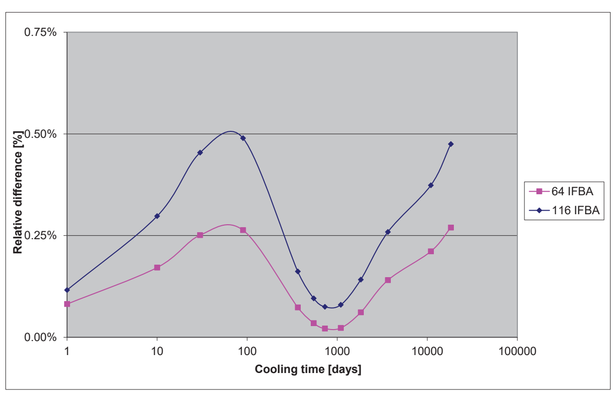


FIGURE 7. Effect of IFBA rods on the decay heat

Burnup

Decay heat versus fuel burnup has already been presented in Fig. 1. However, as in the specific power case a difference in the isotopic inventory is significant during the fuel burnout. Differences in the decay heat for the 40000 MWd/tU and 50000 MWd/tU cases are presented in Fig. 8. In contrast to the specific power case, differences during fuel cooling are increasing over time. Averaging process over this ±10000 MWd/tU burnup interval is producing up to 2 % error in the decay heat.

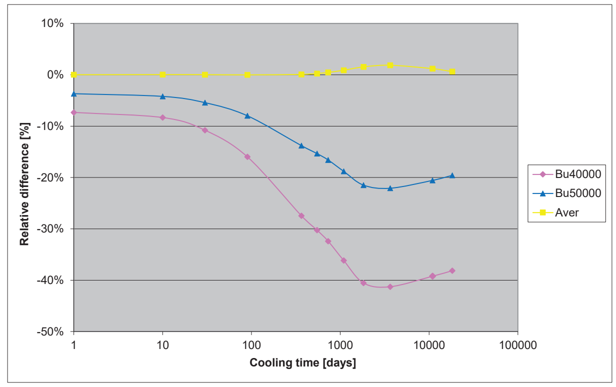


FIGURE 8. Effect of burnup on the decay heat

CONCLUSION

Analysis of the NPP Krško nuclear fuel decay heat with the Serpent code has been performed. Comparison with Origen code has shown very good agreement. It was demonstrated, that the burnup interval not larger than 1000 MWd/tU is needed, if the numerical accuracy less than 1 % is desired.

Since the fuel decay heat is dependent on the fuel isotopic composition and consequently also on neutron flux spectrum, the effect of several operational parameters was examined. Influence of the following parameters on the production of decay heat was analyzed:

1. Fuel enrichment,
2. Fuel temperature, profile,
3. Moderator temperature,
4. Soluble boron concentration,
5. Average power,
6. Burnable absorbers,
7. Burnup.

Realistic operating ranges were considered. In addition, it was demonstrated, especially in the analysis of the enrichment, specific power, moderator temperature (density) and burnup, that the decay heat of the fuel at average conditions is observably different than the average decay heat at specific conditions.

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