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Decay Power Calculation for Safety Analysis of Innovative Reactor Systems

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ABSTRACT

In this work, we verified the decay heat calculation capabilities of BGCore computer code system developed recently at Ben-Gurion University. Decay power was calculated for a typical UO2 fuel in Pressurized Water Reactor environment using BGCore code and using procedure prescribed by the ANS/ANSI-2005 standard. Very good agreement between the two methods was obtained. Once BGCore calculation capabilities were verified, we calculated decay power as a function of time after shutdown for various reactors with innovative fuels, for which no standard procedure is currently available. Notable differences were observed for decay power of the advanced reactors as compared with conventional UO2 LWR. The observed differences suggest that the design of new reactors safety systems must be based on corresponding decay power curves for each individual case in order to assure the desired performance of such systems.

1 INTRODUCTION

An accurate prediction of total decay heat and its time dependence is essential for determining the heat removal requirements after reactor shutdown, during spent fuel transportation and storage, and in the reactor safety analyses for such accidents as Loss of Coolant (LOCA).

The amount of the decay heat generated by UO2 fuel after normal or emergency shutdown in the existing Light Water Reactors can be predicted with high degree of accuracy on the basis of computer simulations and extensive experimental data such as ANS standard for decay heat power in Light Water Reactors [1]. However, the amount of decay heat in future generation of nuclear reactors or in existing reactors operating with innovative fuel cycles has much greater uncertainty for the following reasons:

- Alternative to UO2 fuel compositions would have different fission product (FP) yields because they originate from different fissionable nuclides.
- Fission product yields are different in fast and thermal neutron spectra.
- Existing computer codes, capable of predicting the decay power, have at times outdated database for the FP decay constants, FP yields, and recoverable decay Q-values.
- The choice of nuclides included in the existing computer codes is based either on the nuclides' neutronic importance during irradiation or their out-of-core characteristics (radiotoxicity, decay power etc.) but never both.
- No experimental data exists to confirm the computer simulation predictions.

BGCore reactor analysis system, recently developed at Ben-Gurion University, allows calculations of in-core fuel composition and post-irradiation fuel characteristics including decay heat for all available reaction types. In this paper we present the BGcore system description, and verification of BGCore system decay heat calculation capabilities. In addition we calculate the decay power produced by various actinide bearing fuels in fast and thermal spectrum reactors using the BGCore computer code. The obtained results are compared with standard UO_2 fuel decay power curve.

2 BGCORE SYSTEM DESCRIPTION

BGCore reactor analysis system was developed for calculating in-core fuel composition and spent fuel emissions following discharge. It couples the Monte Carlo neutronic solver (MCNP [2]) with SARAF burnup and decay module. The SARAF module is independently developed at Ben-Gurion University. The module can be used in a stand-alone mode similarly to the well known ORIGEN2 [3] code. In the BGCore system, the SARAF module receives the relevant data from MCNP, executes the depletion time step, and passes back the updated fuel composition for the next MCNP time step.

The BGCore system is written entirely in the MATLAB programming environment. This greatly simplifies the source code, making it more transparent, efficient, and less error prone. In addition, the MATLAB data can be stored in a standard, fast and easy access, platform independent binary format which is also easy to visualize.

Similar to the ORIGEN2 code, SARAF uses matrix exponential method to solve a set of first order differential equations representing the evolution of each nuclide concentration with time. However, no asymptotic approximations are used. That is, the main calculation matrix always includes all isotopes available from the SARAF library. The execution time for the depletion step is notably larger than that of the ORIGEN code for the similar input. However, it is still negligible compared to the Monte Carlo step.

The depletion calculations are executed for each burnable region with the calculated real neutron flux values in two stages known as Predictor-Corrector (P-C) algorithm. The P-C procedure is required to reduce the error introduced by the fact that the depletion calculation is performed using the beginning of timestep values of the flux and cross sections, which are, in reality, may change significantly during the timestep.

The BGCore approach for generation of one group cross-sections takes advantage of the fact that dividing the neutron flux tally into multiple energy bins has practically no effect on the MCNP execution time. The following calculation procedure is therefore adopted. A fine group spectrum (currently 50000 lethargy points) is tallied at each burnable region by MCNP and passed on to the SARAF module. The one-group cross-sections are then calculated in a separate subroutine using pre-generated multi-group cross-section set and the fine group neutron spectrum obtained from MCNP.

This procedure reduces dramatically the computation time of MCNP. The reduction in execution time by factor of 3 to 10 between the standard and multi-group approaches was observed depending on the nature and complexity of the problem. Additional advantage of the method is the fact that the one group cross-sections are calculated for all available isotopes and, in principle, for all available reaction types without any increase in computation time. In the conventional coupling method, on the other hand, the calculated cross-section data is limited only to the most neutronically important nuclides and reactions, while the rest of the data is taken from a standard reactor-type dependant ORIGEN library.

Considerable effort was made to ensure that the multi-group approach gives one group cross-section values identical to those obtained with the conventional direct reaction rate tally approach. It

was found that for the nuclides with complex resonance structure, which present in the fuel at high concentrations, increasing the number of energy groups does not reduce the error in one group cross section below ~1% especially at low fuel temperatures. The error originates almost exclusively from the unresolved resonances energy region. This is due to the probabilistic treatment of unresolved resonances in MCNP, which correctly predicts the average value of reaction rates but in principle cannot provide the fine structure of the neutron flux. As a result, the self-shielding effect in the unresolved resonances energy region is not accounted for correctly. The error introduced is not statistical in nature but systematic since shielded cross-section is always smaller than the infinite dilution one.

In order to overcome that inaccuracy, the multi-group approach was extended by introducing the background cross-section (σ_0) tabulation into the calculation scheme. A series of multi-group cross-section sets is generated for selected isotopes with significant resonance cross-sections for several values of σ_0. The background cross section is automatically calculated by the code for each calculation case using a simplified formulation described in Reference 4 and then used to extract (through linear interpolation) the appropriate multi-group cross-section set for a specific resonance isotope which is further used to obtain the “shielded” one group cross-section value. Introduction of such extension reduces the difference between directly tallied and collapsed from multi-group cross-sections to well below 1% while still taking advantage of the fast MCNP execution.

A number of benchmark cases of BGCore against well established and verified, state of the art computer codes for thermal and fast spectrum lattices were performed. An excellent agreement in prediction of most important parameters was observed [5].

The SARAF data library required for its execution is based on JEFF3.1 evaluated data files. These include fast and thermal fission product yields for over 30 fissionable nuclides, decay constants, atomic masses, decay reactions’ branching ratios and recoverable energy per decay for decay heat calculations.

Currently, about 1700 isotopes are tracked in the SARAF code. The following guidelines were used in choosing these nuclides:

- All the nuclides that have evaluated cross-sections in JEFF-3.1 file with their respective decay chains.
- All nuclides with available fission yield data including their decay chains

The fact that all of the isotopes, and not just the most neutronically important, are tracked throughout all depletion steps, allows calculations of post-irradiation fuel characteristics such as, activity, radiotoxicity, and decay heat.

3 VERIFICATION OF DECAY HEAT CALCULATION CAPABILITIES OF BGCORE SYSTEM

In this section, we verified decay heat calculations capabilities of BGCore system against ANS decay heat standard. The ANS standard is applicable to LWR fuels containing U-235 as the primary fissile material and U-238 as the major fertile material. In the ANS standard, fission product decay heat power is provided for thermal neutron spectrum fissions of U-235, Pu-239, and Pu-241 and for fast fissions of U-238. In addition, the contribution of U-239 and Np-239 to the total decay power is taken into account.

In order to test decay heat estimation capabilities of BGCore system, we calculated the decay power produced by 4.2% enriched UO2 fuel in a typical PWR core using the BGCore computer code.

The obtained results were compared (Fig. 1) with those of the most recent edition of ANS Standard UO_2 fuel decay power curve.

Figure 1 shows very good agreement between BGCore and ANS standard curve. The ANS standard curve is based on conservative calculations and experimental data. Therefore, the ANS standard data slightly overestimates the decay heat during first few hours. However, the difference between ANS standard data and that predicted by the BGCore does not exceed 5% (Fig. 2).

The obtained results indicate that the BGCore system can be used with confidence for prediction of decay heat of innovative reactor systems.

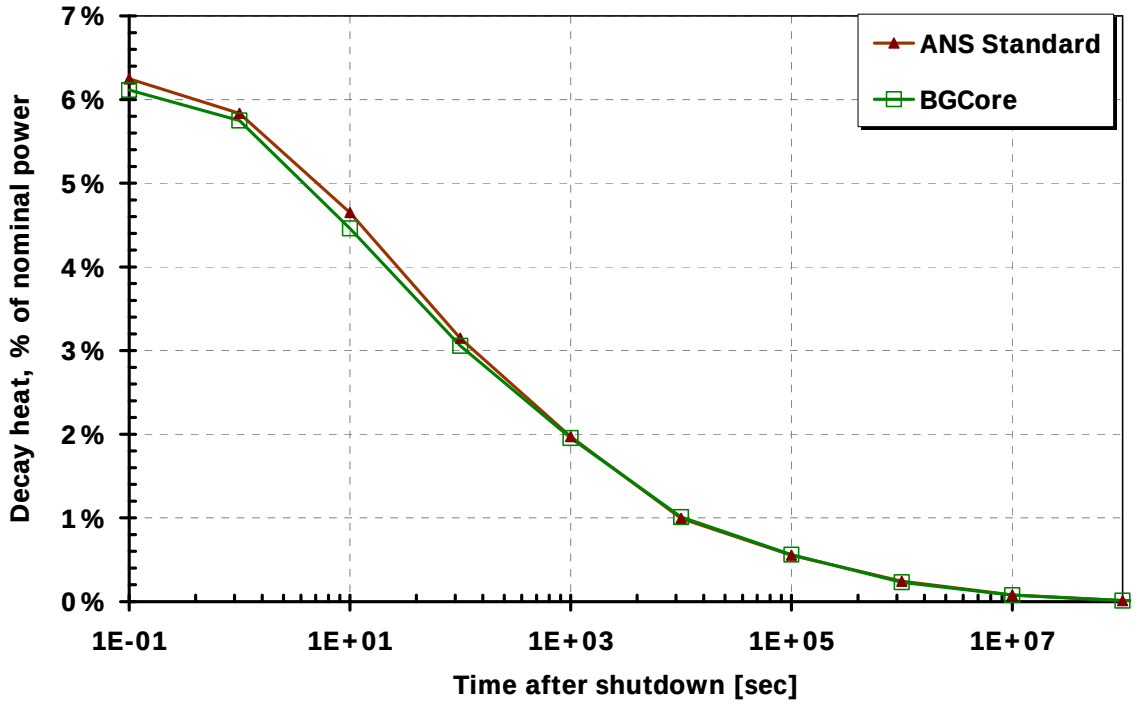

Figure 1: Decay heat comparison: BGCore vs. ANS standard curve, absolute values

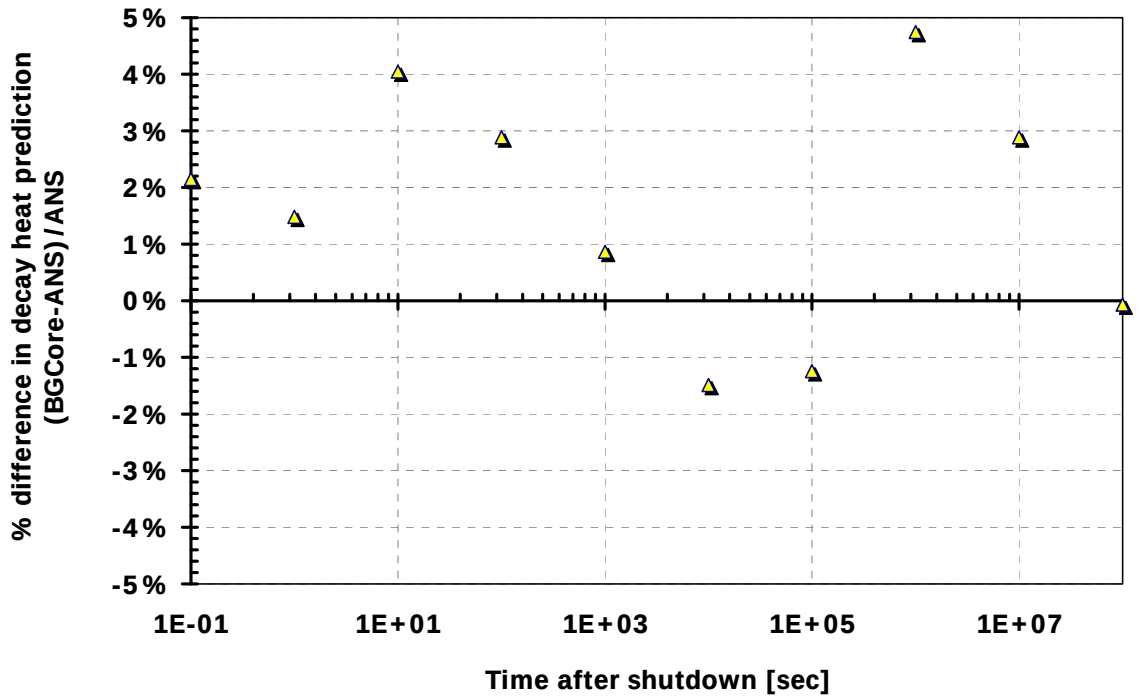

Figure 2: Decay heat comparison: BGCore vs. ANS standard curve, % difference

4 DECAY POWER CALCULATION OF INNOVATIVE REACTOR SYSTEMS

In this section, we performed decay heat calculations for different advanced fuels in both thermal and fast neutron spectrum cores. Brief description of considered cases is presented in Table 1. The fast spectrum reactor concepts used for this analysis were adopted from Reference 6. Lead cooled and sodium cooled fast reactors operating as actinide burners with zero conversion ratio (CR) or operating in a self-sustainable fuel cycle (CR=1) were analysed.

Table 1: List of calculated cases

Case designation	Fuel	Reactor Type
MOX	$(Pu\text{-}U)O_2$, 7.5 vol% RG† Pu + Nat. U	PWR
TOX	$(Pu\text{-}Th)O_2$, 9.2 vol% RG Pu + Th	PWR
LFR1	Metallic Zr-U-TRU, 16% TRU enrichment	Lead Cooled Fast Reactor, CR‡ =1
LFR0	Metallic Zr-TRU, 30% TRU enrichment	Lead Cooled Fast Reactor, CR = 0
SFR1	Metallic Zr-U-TRU, 16% TRU enrichment	Sodium Cooled Fast Reactor, CR =1

†RG – Reactor Grade
‡CR – Conversion Ratio

The results of calculations were compared to those of a 4.2% enriched UO2 fuel in a standard PWR. Fig. 2 and 3 present the results of comparison for thermal and fast spectrum cores respectively.

The figures plot deviation of the decay power to nominal power ratio relative to UO2 LWR case according to the following formula:

$$\frac{\left[P/P_o\right]_{Case} - \left[P/P_o\right]_{UO_2}}{\left[P/P_o\right]_{UO_2}} \times 100\%$$

Main observations from the presented results may be summarized as follows:

1. Thermal spectrum cases:
    a. At 1 sec. after reactor shutdown the decay heat of the MOX fuel is lower by 2.3% and of the TOX fuel by 5.3% than that of the typical UO2 fuel. At this point, the power production of the reactor core is still largely determined by delayed neutron fissions.
    b. The decay heat generated by MOX and TOX fuels stays lower but becomes equal to that of UO2 fuel after ~$4x10^3$ sec and ~$4x10^4$ sec respectively.
    c. The licensing authorities typically require to provide the core cooling capabilities for at least 72 hours after emergency shutdown, which correspond to about $2.6x10^5$ seconds. Following a few thousand seconds after shutdown, the absolute value of the decay heat falls below 1% of the nominal core power rating. However, most of the decay energy still remains to be removed. Therefore, even slightly higher decay power of the advanced fuels as compared with standard UO2 case between 10^4 and $2.6x10^5$ seconds may have significant impact on the design of the emergency core cooling systems.
    d. After the cooling period of 10^5 sec, Figure 3 shows particularly significant increase in decay heat power as compared with that of the UO2 fuel. For example, at $1x10^6$ sec (~11.6 days) after reactor shutdown the difference reaches 22% for MOX fuel and 60% for TOX fuel.

2. Fast spectrum cases:
    a. At 1 sec after shutdown the decay heat values of the LFR0, LFR1, and SFR1 cases are lower than that of UO2 by 12.3%, 2.8% and 1.8% respectively.
    b. The decay heat becomes equal to that of UO2 fuel after about 40 seconds for LFR1 and SFR1 cases and after ~$6x10^4$ sec for LFR0 case.
    c. For all fast spectrum cases considerable increase in decay heat as compared to that of UO2 fuel can be observed for cooling times greater than 10^5 sec.

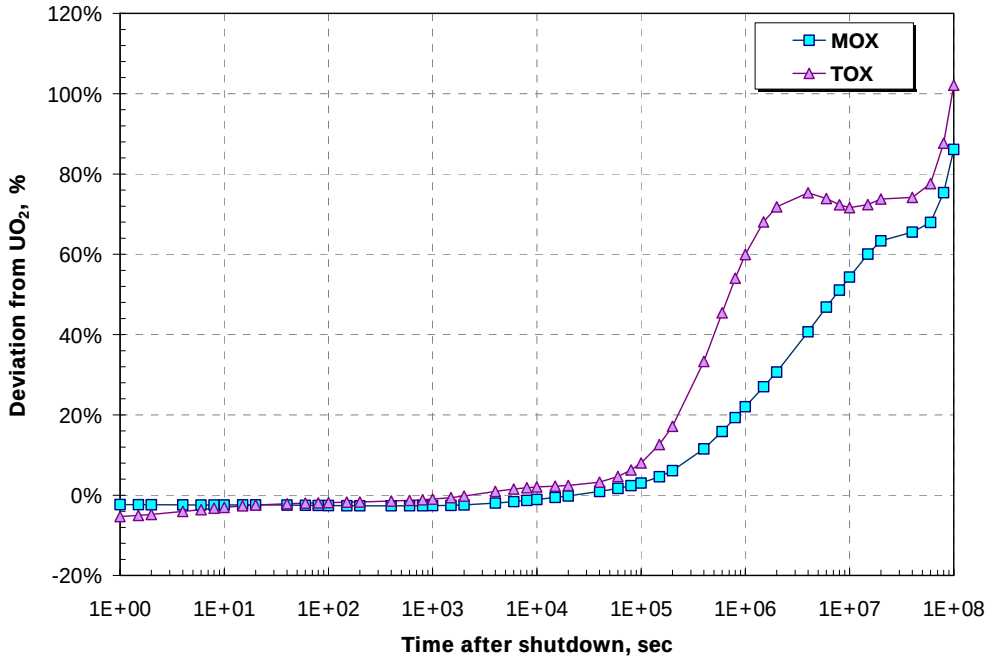

Figure 3: Decay heat of advanced fuels in thermal spectrum

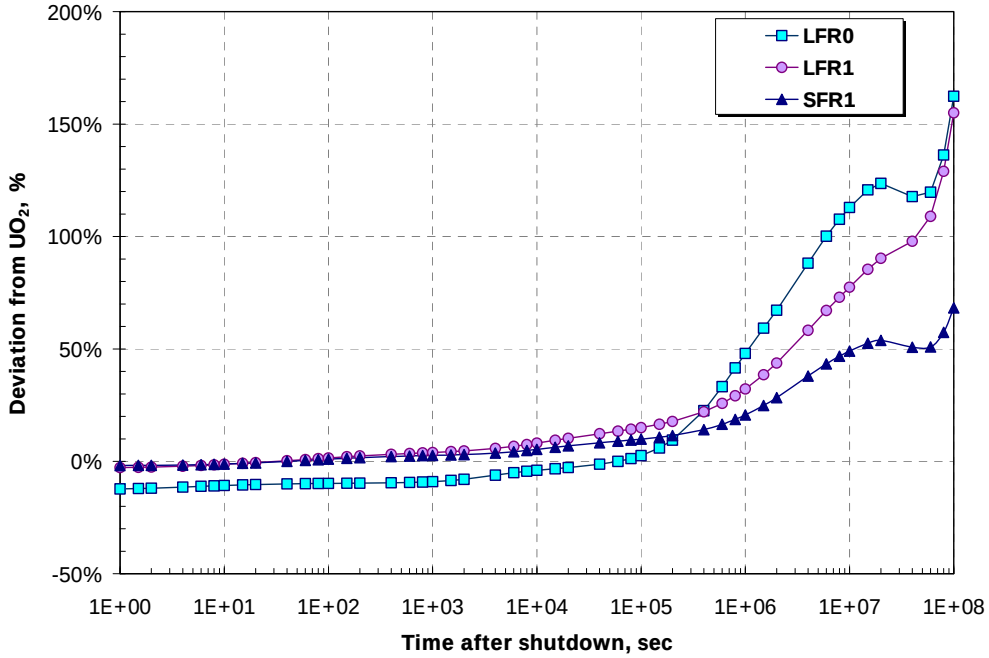

Figure 4: Decay heat of advanced fuels in fast spectrum

5 SUMMARY AND CONCLUSIONS

Dissipation of the decay heat during the normal operation and in case of an accident is one of the major safety concerns of any reactor.

BGCore reactor analysis system, recently developed at Ben-Gurion University, couples the Monte Carlo neutronic solver with a burnup module in a unique way that allows accurate tracking of hundreds of nuclides during irradiation and following shutdown as well as calculation of post-irradiation fuel characteristics including decay heat power.

In this study we tested decay heat calculation capabilities of BGCore system against the most recent edition of ANS Standard UO2 fuel decay power curve. Very good agreement between BGCore and ANS standard was obtained.

In addition, we used BGCore system for calculation of the decay heat generated by various advanced fuel in Light Water Reactors, Lead Cooled Fast Reactor, and Sodium Fast Breeder Reactor. The results of calculations were compared with those of typical PWR UO2 fuel. It was observed that decay heat produced by advanced fuels is somewhat lower but close to that of UO2 for short cooling times. However, decay heat of advanced fuels is significantly larger for cooling times greater than 10^4 seconds, which may impact the safety systems design.

Relying on the standard LWR UO2 decay heat curve for the safety system designs of new reactors with innovative fuels would most definitely result in large uncertainty in the performance of such systems.

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