Applicability of the Industry Standards for the Determination of the NPP Krško Decay Heat

Marjan Kromar
"Jožef Stefan" Institute
Reactor Physics Division
Jamova 39, 1001 Ljubljana, Slovenia
marjan.kromar@ijs.si

Vid Merljak
Nuclear Power Plant Krško
Engineering Division - Nuclear Fuel & Reactor Core
Vrbina 12, 8270 Krško, Slovenia
Vid.merljak@nek.si

ABSTRACT

The decay heat of nuclear fuel is one of the most important safety parameters that needs to be assessed to ensure safe and economically efficient storage, transportation and disposal of spent fuel. In this paper, the predictions of the four most prominent industry standards are compared with the best estimate results generated with the TRITON/NEWT sequence from the SCALE package. The following standards are considered:

- U.S. NRC RG 3.54 Rev.2, 2018
- ANSI/ANS-5.1, 2014
- DIN 25463 1/2, 2014
- ISO 10645, 2022

The spent fuel of the Krško NPP was considered for a cooling time of up to 100 years. Several sensitivity studies on fuel enrichment and burnup were performed. The U.S. NRC RG 3.54 Rev.2, 2018 and especially the DIN 25463 1/2, 2014 were quite successful in predicting the decay heat. The DIN standard never underestimated the decay heat by more than 2.5 %, while the overestimates were also modest and did not exceed 4 %. The ANSI/ANS-5.1, 2014 standard consistently overpredicted the decay heat, with deviations ranging from 7% to over 50%, particularly for lower-enriched fuel and longer cooling times. The ISO 10645, 2022 standard tended to overestimate decay heat at shorter cooling times but underestimated it at longer cooling times, with discrepancies exceeding 20% in some cases. The decay heat predictions of the standards presented in this paper show that validation of the selected standard for specific nuclear fuel and plant is required, before it can actually be applied.

Standards are a useful independent tool for rapid verification of fuel decay heat. Nowadays, when computer speed is no longer such a big problem, as e.g. the ORIGAMI code from the SCALE package can predict decay heat within seconds, provided that the needed custom one-group reaction coefficient libraries have been created in advance, the use of industry standards can be questioned. Nonetheless, an additional independent calculation tool provides an extra layer of confidence.

Keywords: *spent nuclear fuel, decay heat, sensitivity study, industry standard*

1 INTRODUCTION

The decay heat of nuclear fuel is one of the most important safety parameters that needs to be assessed to ensure safe and economically efficient spent fuel storage, transport and disposal. It is mostly sensitive to the type of fuel (UO2 or MOX), the fuel initial enrichment, burnup and cooling time. Since it takes at least one full day to measure the decay heat of a fuel assembly, it is impractical to measure the entire spent nuclear fuel inventory produced during the lifetime of one or more nuclear power plants. Therefore, calculations are required for a complete characterization of the spent fuel. In general, there are two approaches how to determine the decay heat:

- use of (semi-)empirical formulas,
- best-estimate calculation using a neutron transport code coupled with a module to solve the Bateman equations.

Although the first approach is much faster, it provides less accurate results due to the limited number of input variables and generic correlations. It is not based on the determination of the actual fuel nuclide composition by using an exact microscopic irradiation model, but relies on a predetermined correlation between the main fuel parameters (initial enrichment, burnup, cooling time, etc.) and the decay heat. The challenge for these parameterization strategies is that they have to approximate a wide range of possible fuel assembly states that may occur in practice. The approach was developed in the 1970s and 1980s, when computer resources were limited and best-estimate calculations took a long time. Nevertheless, the approach is interesting and can still be used today for fast checking of large number of cases.

In this paper, the predictions of the four most prominent industry standards are compared with the best estimate results generated with the TRITON/NEWT sequence from the SCALE package [1]. The following standards are analysed:

- U.S. NRC RG 3.54 Rev.2, 2018 [2]
- ANSI/ANS-5.1, 2014 [3]
- DIN 25463 1/2, 2014 [4]
- ISO 10645, 2022 [5]

NPP Krško spent fuel was considered for a cooling time of up to 100 years. Several sensitivity studies were carried out with regard to fuel enrichment and burnup.

2 CALCULATION METHODS AND CODES

2.1 SCALE

The SCALE Code System [1] is a widely used modelling and simulation suite for nuclear safety analysis and design that is developed, maintained, tested, and managed by the Reactor and Nuclear Systems Division (RNSD) of Oak Ridge National Laboratory (ORNL). SCALE provides a comprehensive, verified and validated, user-friendly toolset for criticality safety, reactor and lattice physics, radiation shielding, spent fuel and radioactive source term characterization, and sensitivity and uncertainty analysis. SCALE version 6.2.4 with the TRITON/NEWT module and the internal 56-energy group library v7-56, based on the ENDF/B-VII.1 evaluated nuclear data files [6], was used for this assessment.

2.2 Standards

In general, the four standard methods, listed in the introduction, rely on the determination of the decay heat by evaluating the following components:

- Contribution of the fission products from nuclear fission,
- Contribution of isotopes resulting from neutron capture in fission products,
- Contribution of the actinides.

RG 3.54 also takes into account the contribution of activated structural materials.

The methods were implemented in a STDSNF tool [7], which is available at the Nuclear Energy Agency, in the working group of the Working Party for Nuclear Criticality and Safety, called ‘‘SG12: Decay heat of spent nuclear fuels’’, provided by Studsvik company. This tool was used to determine the decay heat of evaluated cases. It should be noted that RG 3.54 Rev.2 (2018) has been implemented, while Rev.3 (2022) has been already issued, but has not yet been incorporated into the STDSNF.

The main features of the methods are summarized in Table 1, while the limitations and ranges of applicability are listed in Table 2.

Table 1: Main characteristics of standards

Method	Fission products	Capture in F. Pr.	Actinides
RG 3.54 Rev.2, 2018	9-term exponential fit ANSI/ANS-5.1, 2014	Explicit: $^{133\text{-}134}$Cs Correction for the rest	semi-empirical for 7 actinides
ANSI/ANS-5.1, 2014	23-term exponential fit	t≤10^4s: G-factor t>10^4s Explicit:$^{133\text{-}134}$Cs Correction for the rest	Explicit: ^{239}U, ^{239}Np Correction for the rest
DIN 25463 1/2, 2014	24-term exponential fit	Depletion chains	Depletion chains
ISO 10645, 2022	ANSI/ANS-5.1, 2014	Explicit: $^{133\text{-}134}$Cs Correction for the rest	Explicit: ^{239}U, ^{239}Np Correction for the rest

Table 2: Limits and ranges of applicability

Method	Reactor	Fuel	Decay time [sec]	Burnup [MWd/kgU]	Enrichment [wt%]
RG 3.54 Rev.2, 2018	PWR BWR	UOX	1÷110 years	10÷65 10÷55	2.0÷5.0 (^{235}U)
ANSI/ANS-5.1, 2014	LWR	UOX	<10^{10} (~316y)	No limit	No limit

DIN 25463 1/2, 2014	PWR	UOX MOX	<2.10^{9} (~62 y)	<80.0 <60.0	3.0÷5.0 (^{235}U) 1.8÷7.5 (Pu^{fiss})
ISO 10645, 2022	LWR	UOX	<10^{9} (~31 y)	<62.0	<5.0

Readers should refer to the given references for the details of the standard calculations. In order to apply these standards, a certain number of information needs to be provided, such as assembly burnup, mass, initial enrichment, cooling time and power history. For the ANSI/ANS and ISO standards, fission fractions of the ^{235}U, ^{238}U, ^{239}Pu and ^{241}Pu fissions in the system as a function of burnup steps, have to be provided. In this analysis, fission fractions calculated from the reference SCALE calculation were applied. If these values are not available, the default values in the RG 3.54 can be used.

3 RESULTS AND DISCUSSION

3.1 Reference case

NPP Krško 16×16 spent fuel (Figure 1) with no burnable absorbers was considered. A fuel assembly with an initial enrichment of 5 % was irradiated with a specific power of 40 kW/kgU up to a burnup of 60 MWd/kgU. Three identical cycles of 20 MWd/kgU, with a downtime of 30 days were modelled. A cool-down time of 0.1 years to 100 years was assumed. It should be noted that RG 3.54 is applicable for cooling times larger than 1 year and ISO 2022 applies to times up to ~31 years.

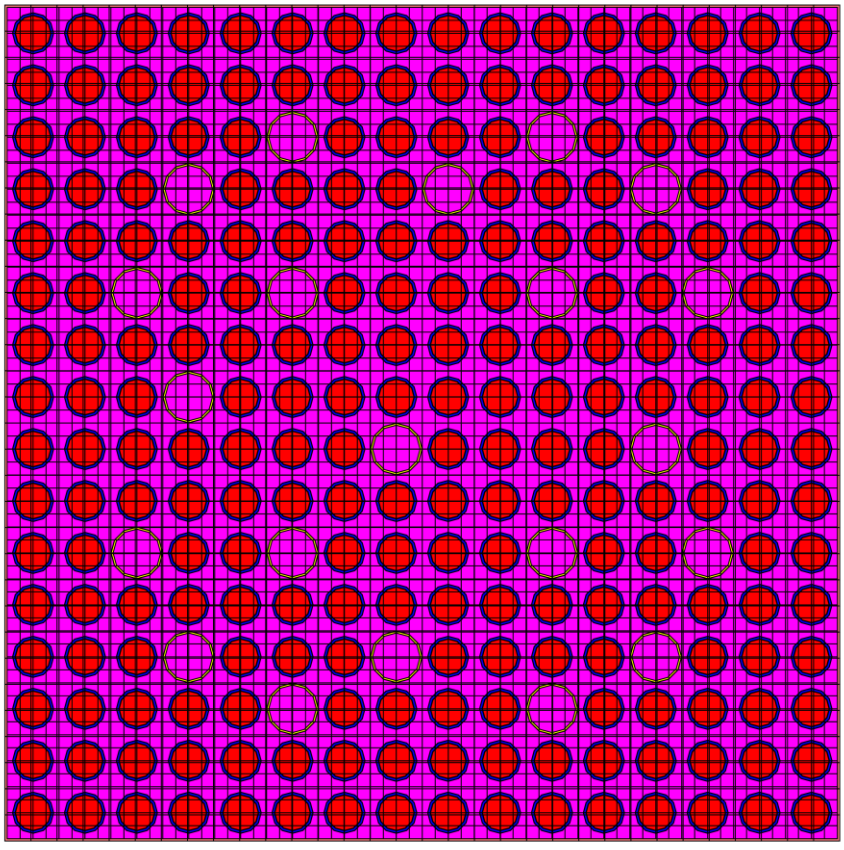

Figure 1: NPP Krško fuel

Geometric and material properties of the fuel assembly are given in Table 3. The values are close to the average nominal conditions at full power (Hot Full Power, HFP), taking into account thermal expansion.

Table 3: Geometric and material properties of the reference case

Pellet outer radius	0.412060 cm
Clad inner radius	0.418542 cm
Clad outer radius	0.475789 cm
Radius of assumed Inconel grid layer	0.478344 cm
Fuel enrichment	0.05 (weight fraction of ^{235}U in the U)
Fuel density/temperature	10.24341 g/cm^3 / 850 K
Moderator density/temperature	0.70500 g/cm^3 / 583.07 K
Rod pitch	1.236692 cm
Assembly pitch	19.896272 cm

The comparison of the results with the SCALE values is shown in Figure 2. The cooling time vector is the same in all calculations. The ANSI/ANS standard overestimates the decay heat by 7 % - 17 % in the cooling period. RG 3.54 and ISO standard are within ± 6 %. Please note that the ISO standard is only valid for cooling times of up to 31 years. The DIN standard provides surprisingly good results. The differences are within 3 %. The results are consistent with the observations reported in [8].

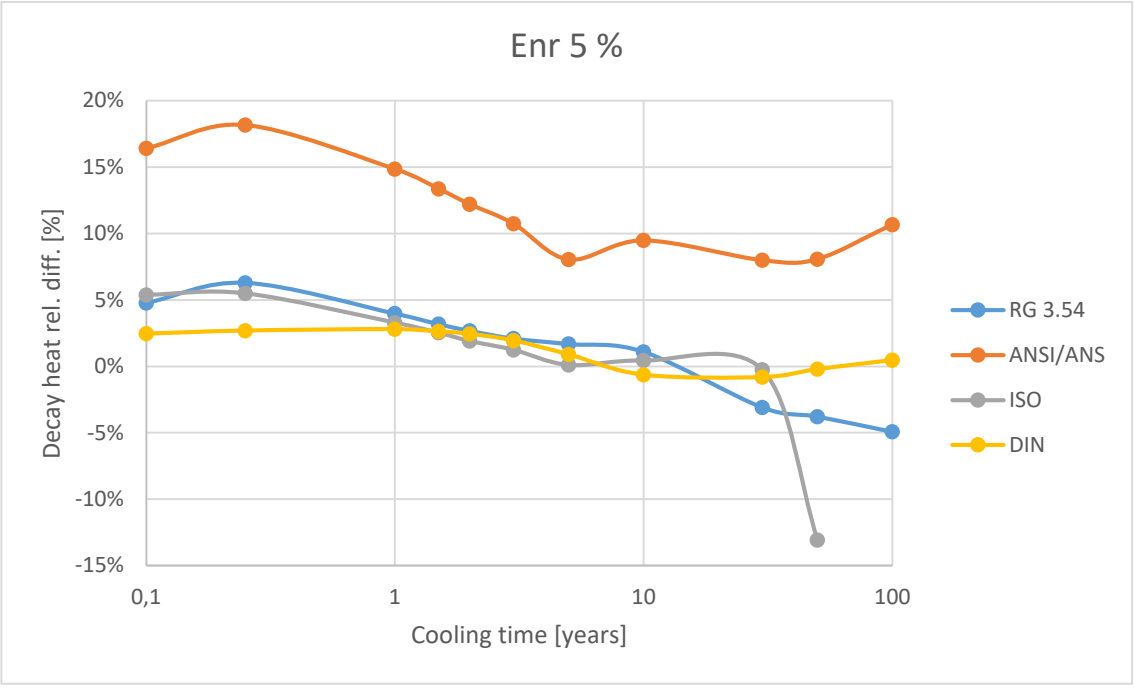

Figure 2: Decay heat comparison for the reference 5 % case

3.2 Sensitivity study - enrichment

Additional enrichments of 2 % to 4 % were considered to test the generality of the prediction of the standards. The results are shown in Figures 3 to 5. Almost the same trend can be observed as in the 5 % reference case. The ANSI/ANS standard overpredicts the decay power by several tens of percent. The ISO standard overpredicts for shorter cooling times and underpredicts for longer times. The RG 3.54 and the DIN standards provide quite good results. Overall, the results of RG 3.54 are within -5 % to + 9 %, while the DIN standard is excellent with results within -2.5 % to 3.5 %.

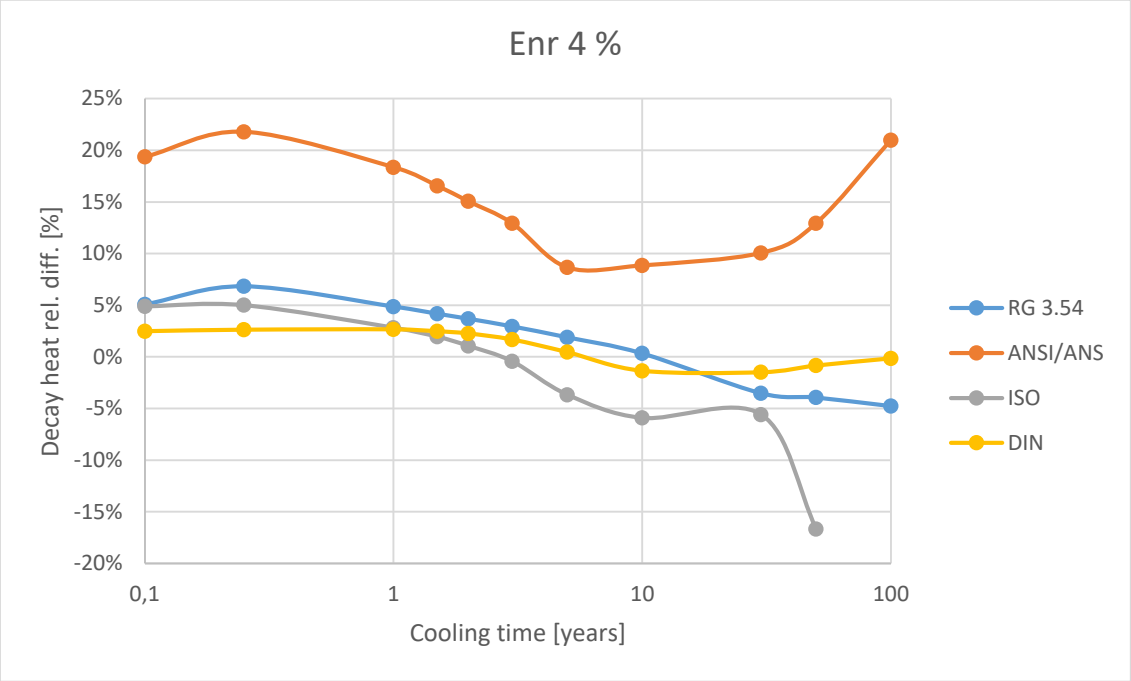

Figure 3: Decay heat comparison for the 4 % case

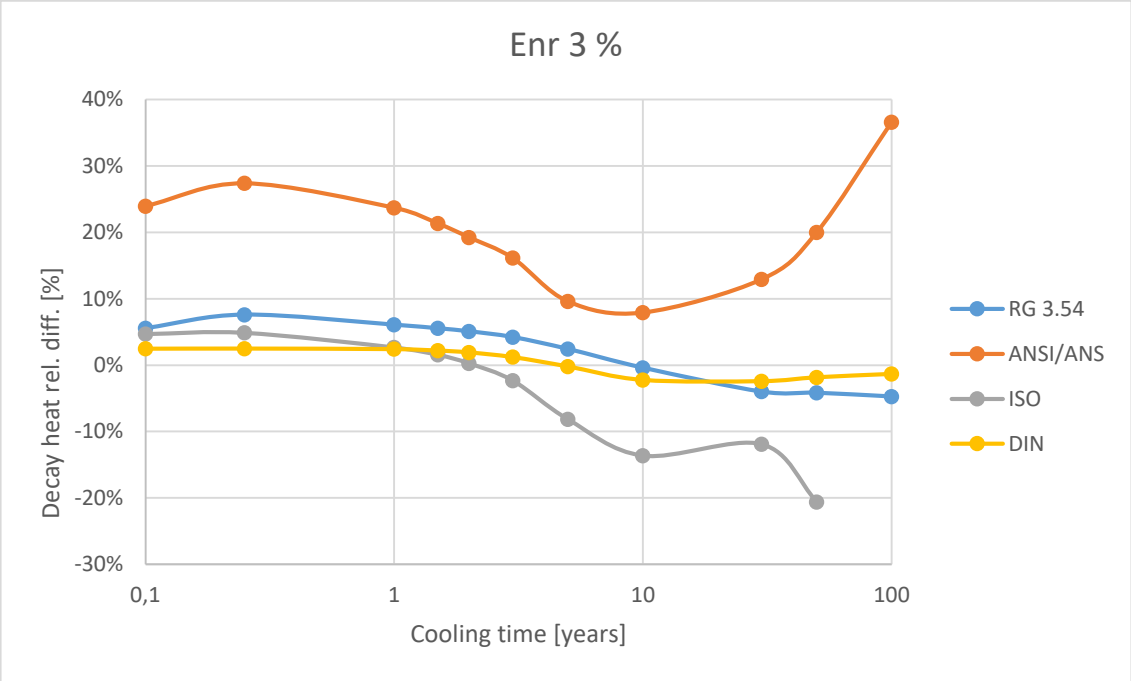

Figure 4: Decay heat comparison for the 3 % case

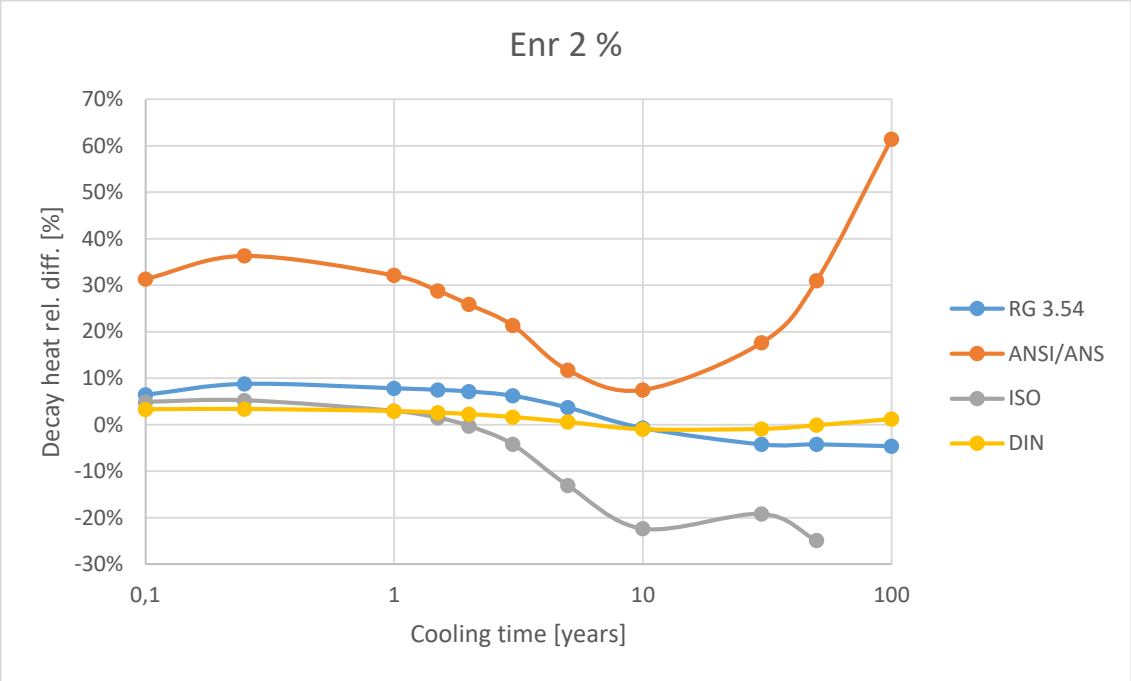

Figure 5: Decay heat comparison for the 2 % case

3.3 Sensitivity study - burnup

Two additional burnup levels at 54 MWd/kgU (-10 % of the reference value) and 48 MWd/kgU (-20 %) were analysed for the reference 5 % case. The results are shown in Figures 6 and 7. The ANSI/ANS standard overestimates the decay heat by up to 23 %. The ISO standard overpredicts for shorter cooling times and underpredicts for longer times. The RG 3.54 and DIN standard give again reasonable good results. Overall, the results of RG 3.54 are within -2.5 % to + 8 %, while the DIN standard is again excellent with results within -0.5 % to 3 %.

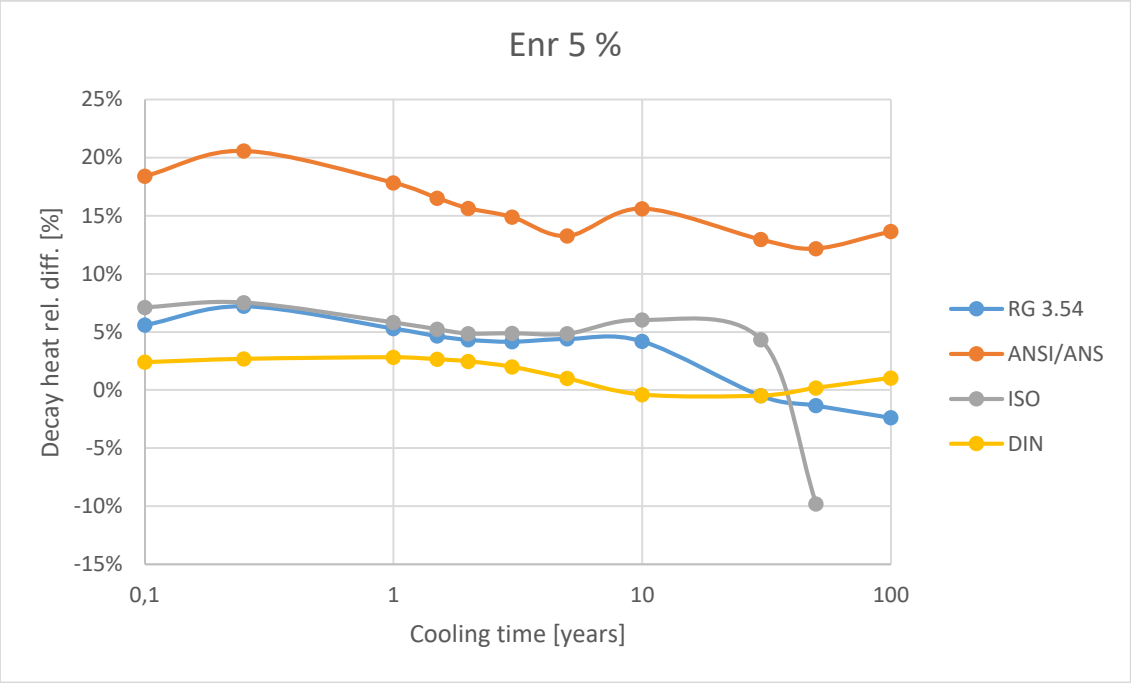

Figure 6: Decay heat comparison for the 54 MWd/kgU case

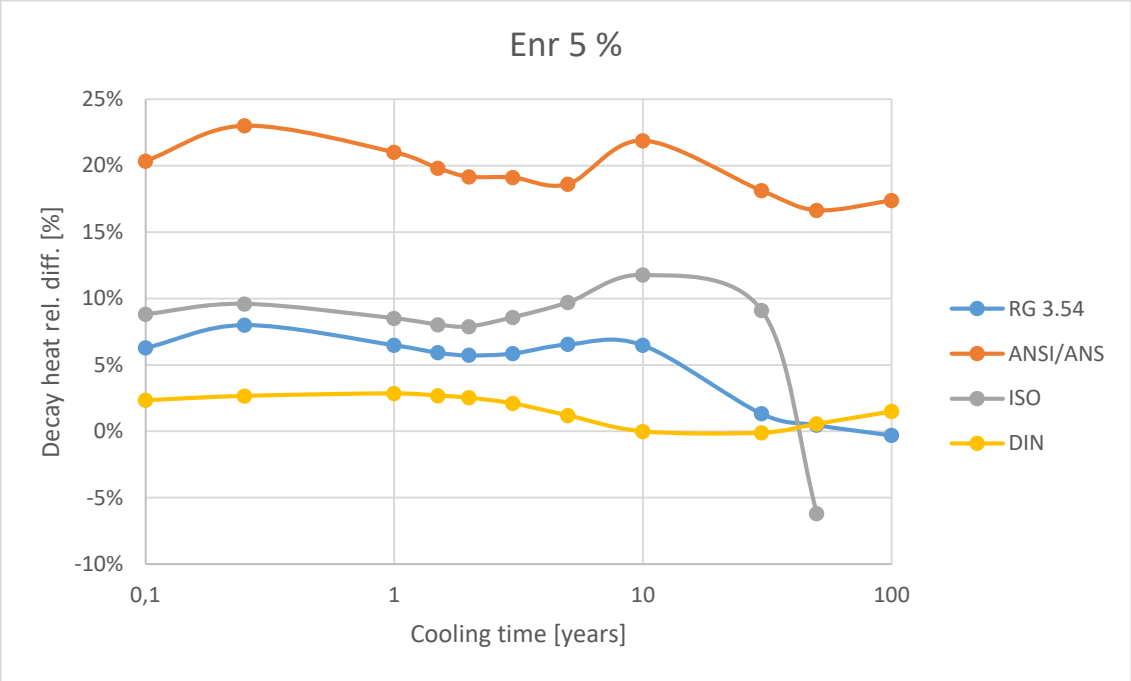

Figure 7: Decay heat comparison for the 48 MWd/kgU case

4 CONCLUSION

The decay heat predictions of the four best-known industry standards were compared with the best estimate results generated with the TRITON/NEWT sequence from the SCALE package. The following standards were analysed:

- U.S. NRC RG 3.54 Rev.2, 2018
- ANSI/ANS-5.1, 2014
- DIN 25463 1/2, 2014
- ISO 10645, 2022

NPP Krško spent fuel was considered for a cooling time of up to 100 years. The ANSI/ANS-5.1, 2014 standard overpredicted the decay heat in all cases. The overprediction is pronounced for lower enriched fuel and especially for longer cooling times, where it can exceed the best estimate of decay heat by more than 50 %. The ISO 10645, 2022 overestimates the decay heat at lower cooling times and underpredicts at longer cooling times even for the more than 20 %. The standard has quite severe limitation, that it is only valid for a cooling time of 30 years. U.S. NRC RG 3.54 Rev.2, 2018 and DIN 25463 1/2, 2014 were quite successful in predicting decay heat. The RG 3.54 underpredicted by no more than 5 %, while the maximum overprediction did not exceed 9 %. The most accurate standard in our analysis was the DIN standard. It never underpredicted the decay heat by more than 2.5 %, while the overestimations were also modest and did not exceed 4 %. It should be mentioned that no additional conservatism was applied to the DIN results. The standard namely specifies the expected uncertainties. With these uncertainties, it could be expected that the results would always be above the best estimate predicted values, but without being too conservative. However, the spread of the decay heat predictions and the occasional underestimation show that validation of the respective standard for a specific fuel and plant is necessary to obtain meaningful results.

Standards are a useful independent tool for rapid verification of fuel decay heat. Nowadays, computer speed is not such a big problem anymore, as e.g. the ORIGAMI code from the SCALE package can predict the decay heat within seconds, provided that the needed custom one-group

reaction coefficient libraries have been created in advance. However, an additional independent calculation tool provides an extra layer of the confidence.

ACKNOWLEDGEMENTS

The authors are grateful to the Studsvik Scandpower, Inc., and especially Teo Simeonov, for providing STDSNF code through the Nuclear Energy Agency international cooperation in the working group of the Working Party for Nuclear Criticality and Safety, called ‘‘SG12: Decay heat of spent nuclear fuels’’.

Some of the underlying research was performed within the Euratom Research and Training Program under Grant Agreement No. 847593 (EURAD project) and IAEA coordinated research project T13018.

REFERENCES

[1] W. A. Wieselquist et al., SCALE Code System, ORNL/TM-2005/39, Version 6.2.4, Oak Ridge National Laboratory, Oak Ridge, Tennessee, April 2020.

[2] U.S. NRC, Regulatory Guide 3.54, Revision 2, Spent Fuel Heat Generation in an Independent Spent Fuel Storage Installation, Washington, DC., 2018.

[3] American Nuclear Society Standards Committee Working Group ANS-5.1, Decay Heat Power in Light Water Reactors, Technical Report ANSI/ANS-5.1-2014, American Nuclear Society, 2014.

[4] DIN Standards Committee Materials Testing, Calculation of the decay power in nuclear fuels of light water reactors - Part 1: Uranium oxide nuclear fuel for pressurized water reactors, English translation of DIN 25463-1:2014-02, DIN Standards Committee Materials Testing, Germany, 2014.

[5] Technical Committee ISO/TC 85, Subcommittee SC 6, Nuclear Energy - Light Water Reactors - Decay Heat Power in Non-Recycled Nuclear Fuels, Technical Report ISO 10645:2022, International Organization for Standardization (ISO), 2022.

[6] M. B. Chadwick, et al., ENDF/B-VII.1 Nuclear Data for Science and Technology: Cross Sections, Covariances, Fission Product Yields and Decay Data, *Nuclear Data Sheets*, 112.12, pp. 2887–2996, 2011.

[7] T. Simeonov, STDSNF Code Input - Standards for Decay Heat Power, Studsvik Scandpower, Inc., SSP-22/446 Rev. 0, October 17, 2022.

[8] Rochman, D. A., Kromar, M., et al., On the estimation of nuclide inventory and decay heat: a review from the EURAD European project, *EPJ Nuclear Sci. Technol.*, 9, https://doi.org/10.1051/epjn/2022055, 2023.