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Neutronic Characterization for a Pressurized Water Reactor Spent Fuel Assembly

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Abstract — *The transfer of nuclear spent fuel from the reactor storage pool to dry storage or for reprocessing or final disposition requires information about its isotopic composition, decay heat, and other thermomechanical properties. The spent nuclear fuel assembly of a typical advanced pressurized water reactor, AP-1000, was characterized using the Monte Carlo MCNPX code and SCALE/ORIGEN code. The simulation of operational history started from the operation of the first fresh core for an average fuel assembly with certain physical isotopic parameters until 25 GWd/tonne U discharge burnup.*

The analysis considered the calculations of the radionuclide inventories, activity, neutron emission spectrum, gamma-ray emission spectrum, and decay power after 700 effective full power days and for post different time ranges until a 1 million–year cooling period. The comparison of some results of the two codes showed small differences due to the consideration of the continuous-energy variation for neutrons in the MCNPX code and the discrete energy assumption in the SCALE/ORIGEN code.

Keywords — *Inventory, MCNPX, SCALE/ORIGEN, decay heat, neutron/photon intensity.*

Note — *Some figures may be in color only in the electronic version.*

I. INTRODUCTION

Knowledge of spent nuclear fuel (SNF) behavior during an incident is necessary to determine the magnitude of the potential release of radioactive material, the types of radioactive materials released, and the most expeditious methods to remediate the situation. The behavior of spent fuel is a key element in the fuel cycle backend during handling, transportation, wet and dry storage, and final disposal by either direct disposal in a repository or reprocessing.

The Nuclear Power Plant Krško fuel assembly was studied using the Serpent2 code at burnup up to 60 GWd/tonne U and cooling time of 100 years. This study aimed to evaluate the sensitivity of some nuclides to nuclear data libraries, such as ENDF/B-VII.0, ENDF/B-VII.1, ENDF/B-VIII.0, and JEFF-3.3, by calculating total activity, decay heat, neutron emission, and photon emission. The results presented an important difference. A comparison of nuclide activity calculations using the mentioned data libraries showed that the differences were generally less than 1%. Obvious differences were found for ^{238}Pu and ^{241}Pu. These differences were similar in all cases, about 0.5% for ^{238}Pu at a cooling time of 100 years and 0.25% for ^{241}Pu at a cooling time of 10 years. Therefore, the uncertainty of some characteristic parameters for spent fuel should be considered when selecting the data library in the simulation codes.

The two main sources of gamma radiation are fission product (FP) decay and secondary photon decay from neutron capture in fissile and nonfissile nuclides, whereas the two main sources of neutrons are spontaneous fission and (α, n) reactions in the fuel materials. These decay

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sources are important in terms of the safety and design requirements for the respective storage casks.[1]

One of the aims of the European Joint Programme on Radioactive Waste Management (EURAD) of the European Commission is joint programming of nuclear waste management in the European Union. One of these project packages is spent fuel characterization (SFC). The primary goal is to produce experimentally verified and validated procedures to determine reliable source terms for SNF. The thermal situation of SNF is most often the limiting factor in the various parts of the backend.[2]

The criticality safety of the SFC rankings was analyzed at Oak Ridge National Laboratory (ORNL) at different decay times of 5 and 30 000 years.[3] Its key features are the dominance of primary actinides ^{235}U, ^{238}U, and ^{239}Pu, along with a large number of FPs. These features are as follows:

1. At low burnup and short cooling times, actinides are responsible for 90% of all absorptions, and after 1E+5 years, they represent 87% of all absorptions.

2. At high burnup and after 5 years, actinides absorb 85% of all neutrons, but less than 79% of absorptions occur in actinides after 1E+5 years.[3]

As irradiated fuel changes over time because of alpha-decay damage and radiogenic contamination, He accumulation was investigated at both the microstructural and macrostructural levels (e.g., hardness). Properties measured on light water reactor fuel [UO2 and mixed oxide (MOX)] were supplemented by tests on alpha-doped UO2, which simulated accelerated aging processes. The hardness results obtained for 67 GWd/tonne fuel were compared to UO2 containing various portions of short-lived alpha emitters.[4]

Ion chromatography in conjunction with inductively coupled plasma mass spectroscopy (ICP-MS) was used to characterize two different spent fuel samples: UO2 and MOX. FPs were determined using isotope dilution analysis with ion chromatographic separation (Rb, Sr, Cs, Ce, Nd, Sm, Eu, and Gd). For the determination of isotopic FPs and actinides, the standard additions method was used (Y, La, Pr, ^{147}Pm, ^{237}Np, 243Am, and ^{244}Cm). Isotope Dilution-Thermal Ionization Mass Spectrometry (ID-TIMS) gamma spectrometry was used to calculate the total uranium and plutonium and to find the concentrations of Nd, Am, and Cm isotopes. Fuel burnup was calculated, and the total fuel inventory was calculated by the KORIGEN code.[3] SNF characteristics are provided by the FP and actinide buildup inside the fuel matrix along with the metallurgy of the cladding to resist fracture during fuel degradation.[4–9]

According to IAEA publications, spent fuel is discharged from research reactors and transferred to wet spent fuel pools. It is characterized using the WIMS and MCNPX and other codes for the safety and safeguards requirements. The isotopic inventory, radionuclides activity, decay heat, and gamma intensities can be calculated and analyzed. Technically, after a certain period of wet cooling of the spent fuel, it will be stored in casks at dry storage until the final disposal or transport to the country of origin.[1] In addition, scientific groups in the academic institutions can study the performance of nuclear fuel in all types of nuclear reactors, especially for research reactors and power reactors such as VVER-1000, VVER-1200, and pressurized water reactors (PWRs), for the purpose of capacity building in the field of nuclear science and SFC. The characterization of spent fuels can support the handling and safe management of the backend of the nuclear fuel cycle.

In this paper, neutronic characterization of an average spent fuel assembly from the AP-1000 advanced PWR was performed. The characterization was performed using two computer codes: MCNPX and SCALE/ORIGEN. The initial characteristics of the fresh nuclear fuel was considered to study the fuel irradiation cycle from the beginning until the end of its operational cycle and discharge of the reactor core. After reactor shutdown, SNF is transferred to wet storage for decay heat removal and cooling for about 10 years, and after that, it can be reprocessed or stored in dry storage. The detailed nuclide inventories, activities, decay heat, neutron emission rates, and photon emission rates were calculated at different cooling times of up to 1 million years.

II. SNF CHARACTERIZATION

Spent nuclear fuel characterization investigates issues related to the properties of SNF in the backend of the nuclear fuel cycle. Decay heat, nuclide inventory, burnup, and mechanical properties are the primary focal points of SFC. These are important for the safety and operational aspect of the backend fuel cycle. It includes cooling, transportation, intermediate storage, encapsulation, reprocessing, and final disposal.

The main source terms of SFC are neutron emission rates, photon (gamma-ray) emission spectra, decay heat rate, and inventory of specific nuclides, long-lived FPs, fissile nuclides (^{235}U and ^{239}Pu), and minor actinides (^{241,243}Am and ^{242,244}Cm).

The inventory of fissile nuclides is needed for safeguards control, reactivity calculations to prevent

recriticality, and the design of interim storage and final repository facility cask/canister loading. Also, the inventory of activation products and long-lived FPs is important to study the impact on the biosphere.

Various compositions of simulated SNFs were used as lanthanide (Ln) (La, Nd, Ce, Gd, and Eu)–doped UO2, and SIMFUEL pellets doped with metallic particles (Ln-PGM-Mo-UO2 ceramics) to be tested under conditions relevant in the backend of the fuel cycle (dry and wet).

Morphological and chemical/spectroscopic analyses of the fuel behavior were performed by means of conventional techniques (i.e., ICP-MS, Scanning Electron Microscopy-Energy Dispersive X-ray analysis (SEM-EDX), Environmental Scanning Electron Microscope (ESEM), Transmission Electron Microscopy (TEM), X-Ray Diffraction (XRD), and Raman spectroscopy). Cutting-edge synchrotron methods at the Rossendorf beamlines (high-resolution diffraction, X-ray absorption spectroscopy, and X-ray emission spectroscopy) were also applied. The results provided degradation performance and mechanisms at a multiscale level ranging from the atomic scale to the macroscopic scale.[3]

The high-purity germanium (HPGe) detector enabled high-resolution measurements, which could be used to identify the presence of different radionuclides in the sample. The measurement of gamma rays using the HPGe detector for SNF can provide information on the activity of different radionuclides, such as ^{137}Cs, ^{154}Eu, and ^{134}Cs. These radionuclides are linked to some safety parameters, such as gamma dose rate and burnup. In particular, ^{137}Cs is linked to decay heat.[2]

Some of FPs, such as ^{148}Nd, are used for burnup determination. The complete list of source terms is hard to measure directly, particularly during reactor operation. Therefore, they are estimated based on a combination of calculations and results of nondestructive analysis measurements to verify the calculations. This required the calculation of a complex inventory of nuclides with strongly varying characteristics. The selection of the data library in the simulation codes can affect the uncertainty of spent fuel properties.[3,10,11]

There are several approaches to the management of SNF and radioactive waste. One of these is nuclear transmutation, which is designed to reduce the level of accumulated radioactivity. Transmutation means exposing long-lived radioactive nuclei to neutrons, which are ultimately converted into stable nuclei. Transmutation of radioactive waste is considered by the world scientific community to be an integral part of the future nuclear fuel cycle. It allows for the transformation of long-lived radioactive nuclides into a stable state or for nuclides with a shorter half-life, reducing the amount and hazard of waste to be finally disposed of and easing the requirements for long-term storage.[12]

III. DESCRIPTION OF AP1000 FUEL ASSEMBLY

The advanced PWR AP1000 fuel assembly is configured with 264 fuel rods, 24 guide tubes arranged in a matrix of a 17 × 17 square lattice array with an active fuel length of 4.3 m, and an enrichment of 3.4% UO2. The fuel rod pellets are slightly dished to better accommodate thermal expansion and fuel swelling, increase the void volume for FP release, and accommodate the differential thermal expansion between the clad and the fuel as the pellet density increases in response to irradiation. It is contained in cold-worked and stress-relieved ZIRLO tubing (advanced zirconium-based alloy), which is plugged and seal-welded at its ends to encapsulate the fuel, as shown in Figs. 1 and 2.

Each fuel assembly has a reconstitutable top nozzle and a debris filter bottom nozzle (DFBN) to minimize the potential of fuel damage due to debris in the reactor coolant. The AP1000 fuel design also includes a protective grid adjacent to the DFBN for enhanced debris resistance.[11]

IV. MATHEMATICAL MODEL OF AVERAGE FUEL ASSEMBLY

The Monte Carlo N-Particle computer code (MCNP) and the SCALE/ORIGEN code were used to simulate the spent fuel assembly for inventory, activity, and the neutron and gamma spectrum, along with the decay heat.

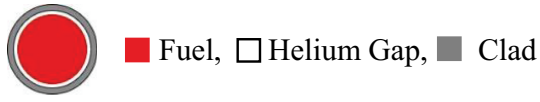

Fig. 1. Simulated fuel rod.

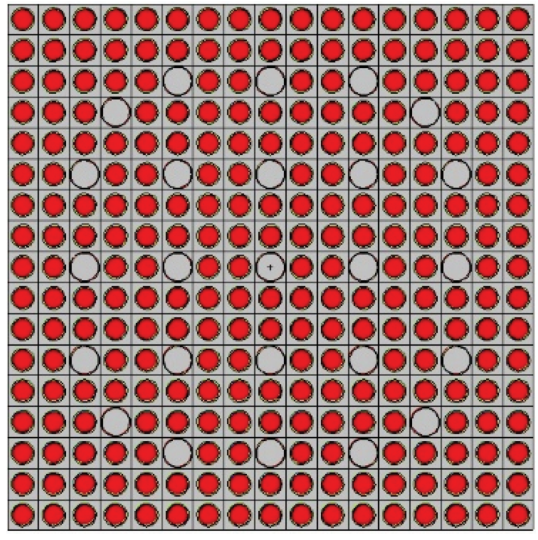

Fig. 2. AP1000 simulated clean fuel assembly.

IV.A. MCNP Description

MCNP is a general-purpose, continuous-energy, generalized-geometry, time-dependent code that tracks many particle types over broad ranges of energies, like neutrons, which are tracked from birth until they are lost due to fission, capture, or leakage from the system.

The MCNP code provides numerous flexible tallies to calculate different physical parameters, such as surface current, volume flux, fission heating, pulse height tally for energy or charge deposition, mesh tallies, and radiography tallies.

IV.B. Fuel Depletion Calculation

MCNPX depletion is a linked process involving steady-state flux and nuclide depletion calculations in CINDER90.[15] MCNPX runs a steady-state calculation to determine the system eigenvalues. Then CINDER90 takes those generated values and performs the depletion calculation to generate new number densities for the next time step. It starts a loop with these new number densities and generates another set of fluxes and reaction rates. The process repeats itself until the specified final time step.

CINDER90 utilizes decay- and energy-integrated reaction rate probabilities along with fission yield information to calculate the temporal nuclide buildup and depletion values. The depletion equation for a specified isotope is

$$\frac{\mathrm{dN}_m}{\mathrm{dt}} = -N_m(t)\beta_m + \overline{Y}_m + \sum_{k\neq m} N_k(t)\gamma_{k\to m} \ , \qquad (1)$$

where

β_m = total transmutation probability of isotope m

$\gamma_{k\to m}$ = probability of isotope k transmuting, by either decay or absorption, into isotope m

$\overline{Y}_m$ = production rate

$N_m(t)$ = time-dependent atom density of isotope m

$N_k(t)$ = the time-dependent atom density of isotope k.

Equation (1) is a nonlinear equation, as the probabilities of transmutation depend on the time-integrated flux, which depends on time-dependent number densities. To linearize the equation, the probabilities of transmutation are set to be constant over the time step. The simplified one-group depletion equation without decay is

$$\frac{dN(r,t)}{dt} = -\Phi(r,t) * \sigma(r) * N(r,t) \ . \qquad (2)$$

The corresponding solution for nuclide density is

$$N(r,t) = N_0(r)e^{-\sigma(r)\int_{t_1}^{t_2}\Phi(r,t)dt} \ . \qquad (3)$$

Hence, nuclide concentration changes based on the time-integrated flux. Unfortunately, the time-dependent flux also is dependent on the nuclide density, which makes Eq. (3) nonlinear. To linearize the equation, time-dependent flux is assumed constant throughout the burn step in MCNPX:

$$N(r,t) = N_0(r)e^{-\sigma(r)\Phi(r)_{t,average}} \ . \qquad (4)$$

Using the Cross-Section Averaging for Depletion Acceleration (CSADA) technique, the average flux behavior is approximated. This technique involves the multistep process, predictor step, and corrector step,[12,13] as shown in Fig. 3.

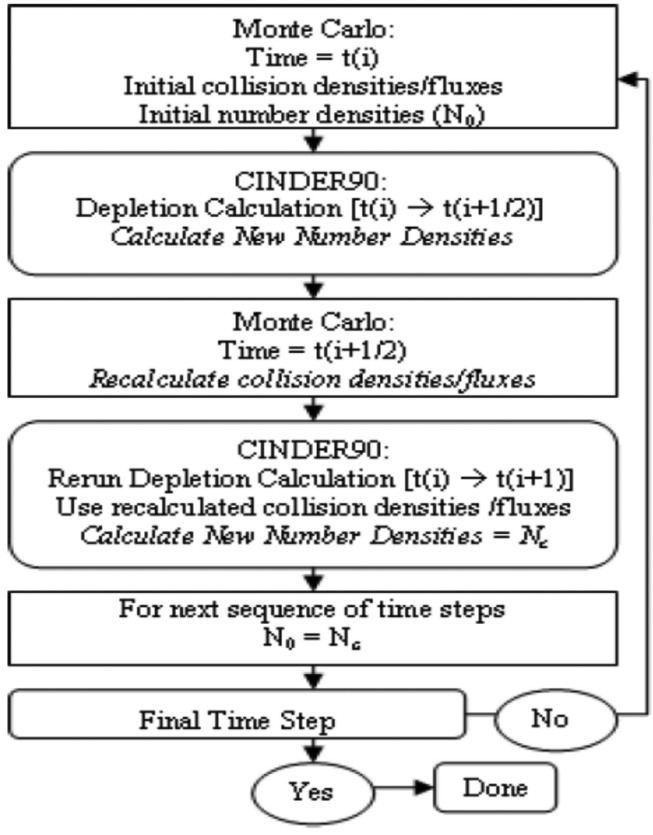

Fig. 3. MCNPX and CINDER90 predictor-corrector method.

IV.C. SCALE/ORIGEN Code Description

The SCALE 6.1 package was developed by ORNL for nuclear design and safety analyses. It provides various codes that address criticality safety, reactor physics, radiation shielding, radioactive source term characterization, and sensitivity and uncertainty analyses.[16]

ORIGEN (Oak Ridge Isotope GENeration) is the code within the SCALE 6.1 package for point depletion, buildup, and decay calculations.[17] This allows for several results to be obtained, including isotopic activities, compositions, and decay photon spectra as a function of time. It includes libraries for decay chains, including decay radiation, FP yields, activation products, actinide production, and photon emission data. In addition, transport codes within SCALE can be used to model a user-defined system, and the COUPLE code can be applied to calculate problem-dependent neutron spectrum–weighted cross sections that are representative of conditions within any given reactor or fuel assembly and convert these cross sections into a library that can be used by ORIGEN.

The matrix exponential method is used to solve a large system of coupled, linear, first-order ordinary differential equations with constant coefficients that describe the evolution of various isotopes over time.[14]

IV.D. Model Validation

Criticality calculations were carried out for an assembly with a maximum enrichment of 4.5% of UO2 in the first cycle using the ENDF/B-VII.0 cross-section data library and fuel temperatures of 900 and 600 K for the moderator temperatures. MCNP6.1 runs criticality with 10 000 neutron histories. The model results were compared with the reference design in the Design Control Document of the AP1000 issued by Westinghouse and with other published results using the SCALE and WIMS9/PARCS/TRACE computer codes.[18,19]

V. RESULTS AND DISCUSSION

The source term of the average AP1000 fuel assembly (3.4%) was calculated using the MCNPX and ORIGEN codes at an average burnup of 27 GWd/tonne U with an irradiation period 700 days and a cooling time from discharge time up to 1 million years.

In Table I, the inventories of some important radionuclides are evaluated. One can notice that the values are nearly the same, with small differences between some radionuclides due to the fact that MCNPX is linked with CINDER90, which contains 63-group cross-section data while ORIGEN uses a 1-group collapsed cross section; also MCNPX uses continuous-energy nuclear and atomic data libraries and ORIGEN uses a discrete energy data library.

In addition, nuclide transformations starting with ^{235}U lead to the formation of ^{237}Np and ^{238}Pu. The main uranium isotope, ^{238}U, is the source of formation of all isotopes of plutonium, beginning with ^{239}Pu. The last significant isotopes of plutonium are ^{241}Pu and ^{242}Pu, with highly radioactive elements, such as ^{137}Cs and ^{90}Sr, which are relatively short lived (30 years).

TABLE I

Inventory of AP-1000 Fuel Assembly After 700 Days of Irradiation at Discharge Time Using the MCNPX and ORIGEN Codes

Nuclide	MCNPX (g)	ORIGEN (g)	Percentage Error (%) $\left(\frac{\text{MCNPX}-\text{ORIGEN}}{\text{ORIGEN}}\right) \times 100$
^{238}U	5.70E + 05	5.69E + 05	0%
^{235}U	7.49E + 03	7.90E + 03	−5%
^{239}Pu	3.67E + 03	3.14E + 03	17%
^{236}U	2.19E + 03	2.25E + 03	−3%
^{240}Pu	9.05E + 02	1.01E + 03	−10%
^{241}Pu	5.13E + 02	6.00E + 02	−15%
^{237}Np	1.68E + 02	1.84E + 02	−9%
^{242}Pu	1.08E + 02	1.63E + 02	−34%
^{136}Xe	1.03E + 03	1.06E + 03	−3%
^{134}Xe	6.93E + 02	7.09E + 02	−2%
^{137}Cs	5.48E + 02	5.68E + 02	−3%
^{144}Nd	3.66E + 02	3.76E + 02	−3%
^{90}Sr	2.67E + 02	2.66E + 02	0%

V.A. Radionuclide Inventory

After 10 years, the inventory was calculated using ORIGEN-ARP to investigate the spent fuel behavior before transferring from wet storage to interim storage, as shown in Table II.

Comparing Tables I and II, we found that the concentration of the radionuclides was slightly reduced as the cooling time increased due to nuclide transformation with different decay times. Also, we can notice that there was no formation of americium because it is represented by two isotopes: ^{241}Am and ^{243}Am. The ^{241}Am isotope was formed in the reactor by the decay of ^{241}Pu (half-life of 14.4 years). The same process occurred when spent fuel was placed outside the reactor in a storage facility. Furthermore, the ^{241}Am isotope in the reactor was ultimately converted to ^{238}Pu. The contribution of ^{241}Am and ^{238}Pu to the overall radiation hazard must be considered during the long-term storage of spent fuel.

After 100 years of storage, the radiotoxicity of ^{241}Am accounts for more than 90% of the total radiotoxicity of actinides in the storage facility. Two dangerous nuclides that accumulate in large amounts, namely, ^{90}Sr and ^{137}Cs, with a half-life of about 30 years, decay completely in about 100 years and do not require transmutation. The rest of the long-lived FPs from spent fuel decay in less time and do not require transmutation.

V.B. Neutron Emission Rate

The distribution of neutrons, including total (alpha-n) and spontaneous (fission/delayed), at discrete decay times from 1 day to 1.E6 years calculated using the ORIGEN-ARP module are shown in Fig. 4. The intensity is per second per unit energy, which will be an input for the recriticality evaluation of the dry cask.

V.C. Photon Emission Rate

The photon distribution at decay times from 1 day to 1 million years calculated using the ORIGEN-ARP module are shown in Fig. 5. We can notice that the intensity is per second per megaelectron-volt, which will be used later as input for shielding calculations of the proper storage cask.

V.D. Activities of Actinides and FPs

The main contributions of the long-lived radionuclides to the total activity of the AP1000 fuel assembly along a 1 million-year decay time were calculated using the ORIGEN-ARP module and are shown in Fig. 6. The resulting secondary activity due to ^{90}Sr and ^{137}Cs was higher than that of ^{99}Tc and ^{129}I.

From the radiation safety point of view, Figs. 4, 5, and 6 are very important as input data for evaluating the

TABLE II

Inventory of AP-1000 Fuel Assembly After 10 Years of Decay Time Using the ORIGEN Code

Isotope	Mass (g)	Isotope	Mass (g)	Isotope	Mass (g)
^{238}U	5.69E + 05	^{143}Nd	4.22E + 02	^{104}Ru	2.35E + 02
^{235}U	7.90E + 03	^{99}Tc	3.81E + 02	^{89}Y	2.22E + 02
^{239}Pu	3.19E + 03	^{98}Mo	3.81E + 02	^{131}Xe	2.16E + 02
^{236}U	2.26E + 03	^{96}Zr	3.80E + 02	^{90}Sr	2.08E + 02
^{136}Xe	1.06E + 03	^{97}Mo	3.79E + 02	^{130}Te	1.72E + 02
^{240}Pu	1.01E + 03	^{241}Pu	3.70E + 02	^{148}Nd	1.70E + 02
^{134}Xe	7.09E + 02	^{95}Mo	3.67E + 02	^{135}Cs	1.68E + 02
^{138}Ba	6.07E + 02	^{101}Ru	3.64E + 02	^{88}Sr	1.66E + 02
^{140}Ce	5.84E + 02	^{94}Zr	3.64E + 02	^{105}Pd	1.63E + 02
^{144}Nd	5.81E + 02	^{102}Ru	3.53E + 02	^{106}Pd	1.42E + 02
^{139}La	5.72E + 02	^{93}Zr	3.41E + 02	^{150}Sm	1.37E + 02
^{133}Cs	5.54E + 02	^{145}Nd	3.30E + 02	^{137}Ba	1.32E + 02
^{141}Pr	5.27E + 02	^{146}Nd	3.13E + 02	^{147}Sm	1.31E + 02
^{142}Ce	5.26E + 02	^{92}Zr	3.12E + 02	^{87}Rb	1.20E + 02
^{132}Xe	4.81E + 02	^{91}Zr	2.90E + 02	^{86}Kr	8.87E + 01
^{137}Cs	4.51E + 02	^{103}Rh	2.37E + 02	^{107}Pd	8.50E + 01

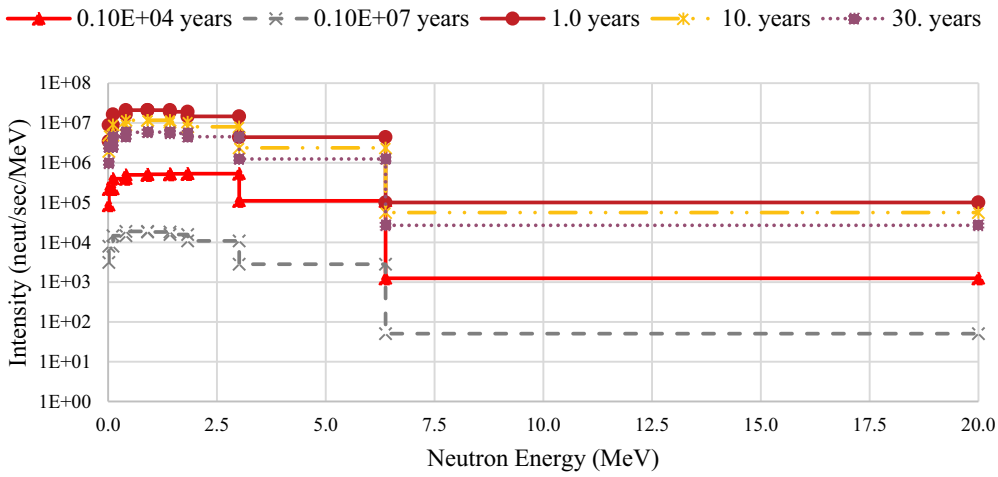

Fig. 4. Neutron intensities at different neutron energies and decay times.

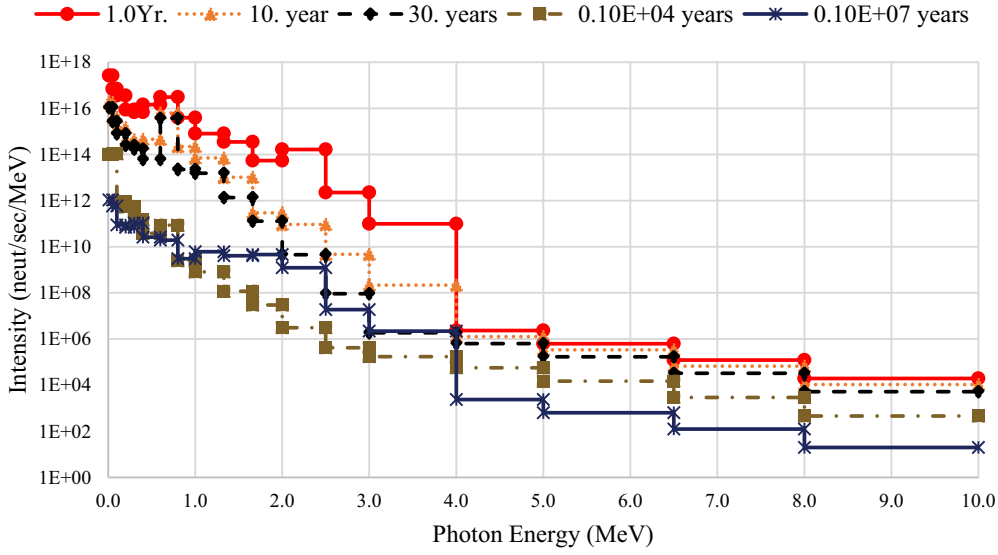

Fig. 5. Photon intensities at different photon energies and decay times.

dose rate around the storage cask. The neutron/photon intensities and distributions are also important for choosing the geometry and the shielding materials used in the construction of dry storage.

The evaluation of activity values is important in relation to the radiological considerations for wet/dry storage, personal protection, and environmental safety.

V.E. Decay Power

The decay power for the selected radionuclides for different decay times until 1 million years were calculated using the ORIGEN-ARP module and are shown in Fig. 7.

The decay power calculation is crucial during the storage time to maintain and prepare the proper cooling conditions. We can notice that the total decay power after 1 year of cooling is around 300.0 W, after 100.0 years of cooling, it is around 100.0 W, and after 1000.0 years of cooling, it is around 30.0 W. This shows the main actinides that contribute to the decay power along the decay time.

VI. CONCLUSION

The neutronic characterization of SNF for an average assembly from the AP1000 was simulated and modeled using the MCNPX and SCALE/ORIGEN-ARP

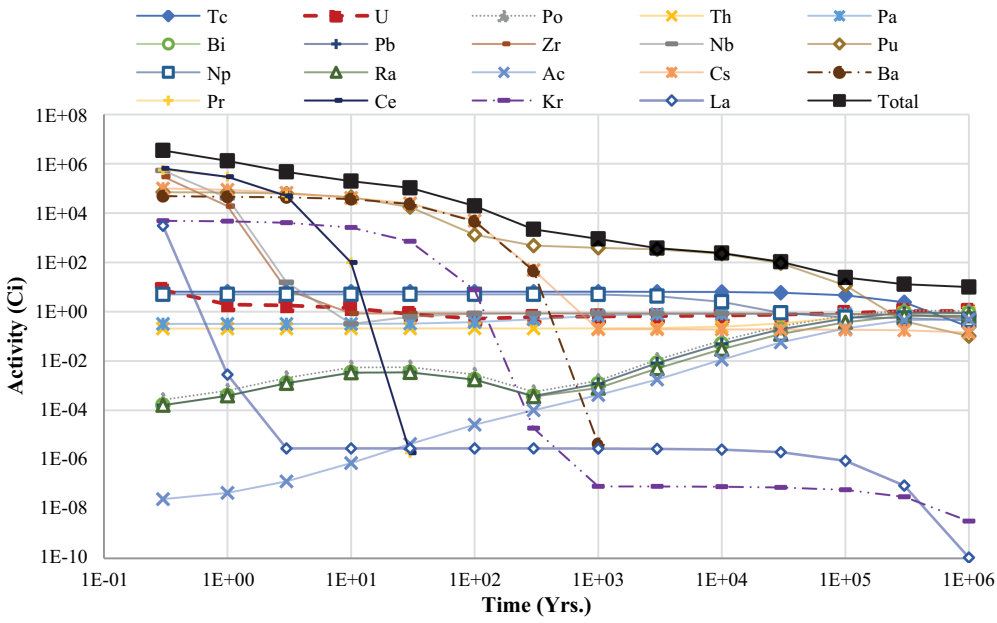

Fig. 6. Total and selected activities for selected radionuclides at decay times.

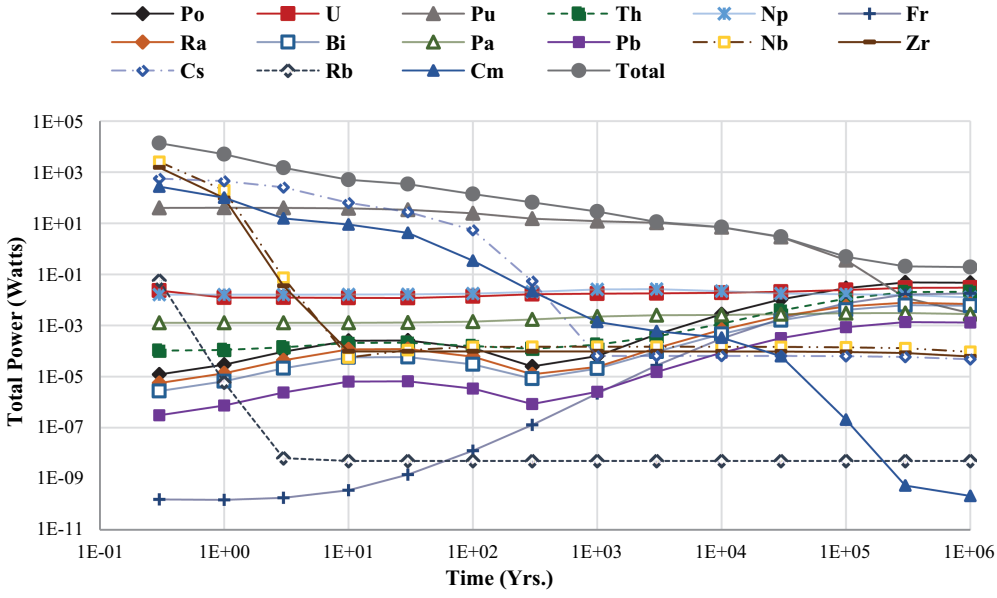

Fig. 7. Contribution of radionuclides in decay power at decay times.

codes. The depletion process started from the first loading as a clean and fresh fuel assembly and continued until its average discharge burnup. The characterization after 10 years of cooling time results for the radionuclide inventory for ^{239}Pu, ^{235}U, and ^{238}U were 3.19, 7.49, and 569.7 kg, respectively. The total activity was 1.9E+05 Ci, the total decay heat was 502 W, the total neutron emission was 3.6E+07 neutrons/s, and the total gamma intensity was 2.56E+15 photons/s. These results are necessary for dry cask design and radiation protection management during SNF handling, transport, and final storage or reprocessing.

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Disclosure Statement

No potential conflict of interest was reported by the author(s).

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References

1. D. CALIC, "Marjan Kromar, Spent Fuel Characterization Analysis Using Various Nuclear Data Libraries," *Nucl. Eng. Technol.*, **54**, *9*, 3260 (Sep. 2022); https://doi.org/10.1016/j.net.2022.04.009.
2. A. SJOLAND et al., "Spent Nuclear Fuel Management, Characterization, and Dissolution Behavior: Progress and Achievement from SFC and DisCo," *EPJ Nucl. Sci. Technol.*, **9**, 13 (2023); https://doi.org/10.1051/epjn/2022029.
3. B. L. BROADHEAD et al., "Investigation of Nuclide Importance to Functional Requirements Related to Transport and Long- Term Storage of LWR Spent Fuel," ORNL/TM-12742, Oak Ridge National Laboratory (1995).
4. V. RONDINELLA et al., "Spent Nuclear Fuel Characterization in View of Long-Term Storage," *Proc. 15th Int. High-Level Radioactive Waste Management Conf.*, JRC94471, pp. 479–485, American Nuclear Society (2015).
5. J. M. BARRERO MORENO et al., "Characterization of Spent Nuclear Fuels by Ion chromatography–Inductively Coupled Plasma Mass Spectrometry," *J. Anal. At. Spectrom.*, **11**, *10*, 929 (1996); https://doi.org/10.1039/JA9961100929.
6. "High Burn-Up Uranium Oxide and Mixed Uranium-Plutonium Oxide Water Reactor Fuel on Spent Fuel Management," IAEA Nuclear Energy Series No. NF-T-3.8, International Atomic Energy Agency (2011).
7. J. LEPPÄNEN et al., "The Serpent Monte Carlo Code: Status, Development and Applications in 2013," *Ann. Nucl. Energy*, **82**, *2015*, 142 (2015); https://doi.org/10.1016/j.anucene.2014.08.024.
8. K.-S. KIM and W. A. WIESELQUIST, "Neutronic Characteristics of ENDF/BVIII.0 Compared to ENDF/B-VII.1 for Light-Water Reactor Analysis," *J. Nucl. Eng.*, **2**, *4*, 318 (2021); https://doi.org/10.3390/jne2040026.
9. D. ROCHMAN et al., "Nuclear Data Uncertainties for Typical LWR Fuel Assemblies and a Simple Reactor Core," *Nucl. Data Sheets*, **139**, 1 (2017); https://doi.org/10.1016/j.nds.2017.01.001.
10. F. STUREK and L. AGRENIUS, "Measurements of Decay Heat in Spent Nuclear Fuel at the Swedish Interim Storage Facility, CLAB," SKB Report R-05-62, 1402, SKB (2006).
11. D. ROCHMAN et al., "Best Estimate Plus Uncertainty Analysis for the ^{244}Cm Prediction in Spent Fuel Characterization," presented at the ANS Best Estimate Plus Uncertainty Int. Conf. (BEPU 2018), Real College, Lucca, Italy, May 13, 2018.
12. G. V. TIKHOMIROV and A. S. GERASIMOV, "The Main Problems of Transmutation of Actinides and Long Lived Fission Products from NPP Spent Fuel, ICNRP Volga 2020," *J. Phys. Conf. Ser.*, **1689**, 012032 (2020); https://doi.org/10.1088/1742-6596/1689/1/012032.
13. "Rev. 16-Tier 2, Chapter 4 Reactor, Section 4.3, Nuclear Design, ML071580897," Westinghouse AP1000 Design Control Document, Westinghouse (2017).
14. D. B. PELOWITZ, J. T. GOORLEY, M. R. JAMES et al. "MCNP6.1 User's Manual Code," Code Version 6.1., license number #150863, LA-CP-13-00634, Los Alamos National Laboratory, May 2013.
15. J. S. HENDRICKS et al., "MCNPX 2.6.0 Extensions," LA-UR-08-2216, Los Alamos National Laboratory (Apr. 11, 2008).
16. "SCALE: A Comprehensive Modeling and Simulation Suite for Nuclear Safety Analysis and Design," ORNL/TM-2005/39, Version 6.1, Oak Ridge National Laboratory (2011).
17. "ORIGEN 2.0: Isotope Generation and Depletion Code—Matrix Exponential Method," ORNL-RSIC-CCC 371, Oak Ridge National Laboratory (1980).
18. R. FAYEZ MAHMOUD et al., "Modeling and Validation of an Advanced Pressurized Water Reactor Using Monte Carlo Technique," *Int. J. Sci. Technol.*, **5**, *9* Sep. 2017.
19. M. A. ELSAWI and A. S. BIN HRAIZ, "Benchmarking of the WIMS9/PARCS/TRACE Code System for Neutronic Calculations of the Westinghouse AP1000TM Reactor," *Nucl. Eng. Des.*, **293**, 249 (2015); https://doi.org/10.1016/j.nucengdes.2015.08.008.