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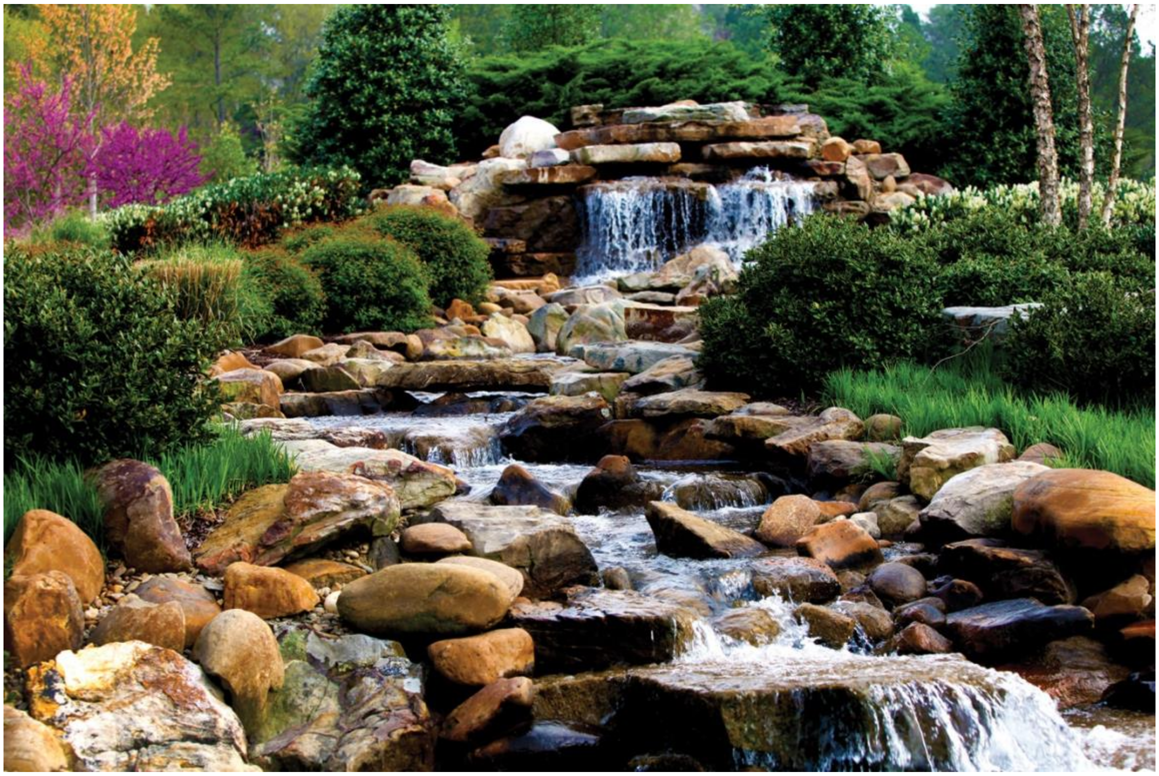
Review of Experimental Data for Validating Computer Codes Used in Shielding Calculations for Spent Fuel Storage and Transportation Systems



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Nuclear Energy and Fuel Cycle Division

REVIEW OF EXPERIMENTAL DATA FOR VALIDATING COMPUTER CODES USED IN SHIELDING CALCULATIONS FOR SPENT FUEL STORAGE AND TRANSPORTATION SYSTEMS

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ABBREVIATIONS

BWR	boiling water reactor
D-T	deuterium-tritium
ICSBEP	International Criticality Safety Benchmark Evaluation Project
OECD	Organisation for Economic Co-operation and Development
ORNL	Oak Ridge National Laboratory
NEA	Nuclear Energy Agency
PCA	Pool Critical Assembly
PWR	pressurized water reactor
RCA	radiochemical assay
REGAL	Rod-Extremity and Gadolinia Analysis
RSICC	Radiation Safety Information Computational Center
SINBAD	Shielding Integral Benchmark Archive and Database
SFCOMPO	Spent Fuel Composition database
SNF	spent nuclear fuel
TDS	thermoluminescent dosimetry
TRG	technical review group
VSC	ventilated storage cask

ABSTRACT

This report presents a review of available radiochemical assay data and shielding benchmarks applicable to spent nuclear fuel (SNF) shielding calculations. The relevant information reviewed herein includes the Spent Fuel Composition (SFCOMPO) database, the Shielding Integral Benchmark Archive and Database (SINBAD), the International Handbook of Evaluated Criticality Safety Benchmark Experiments, and published measurements of external dose rates of casks loaded with SNF. The relevant experimental data identified in this report may be used to support verification and validation of computer codes used in SNF cask/transport shielding applications, as well as development of calculation uncertainties.

It should be noted that a relatively small subset of the identified experimental data (e.g., criticality alarm experiments) is available in a standard format established by the international community participating in experimental isotopic and shielding data evaluations. An effort of the SFCOMPO Technical Review Group (TRG) is underway to publish first isotopic evaluations of individual assay data using a standard data evaluation format. The SINBAD TRG has recently initiated benchmark evaluations and modernization of the database. Therefore, more relevant information is expected in the future that will enable users to select quality experimental data in depletion code and shielding code validations for SNF applications.

1. INTRODUCTION

The accuracy of calculated dose rates can only be assessed by comparison to benchmark measurements. The scope of this report is to identify publicly available radiochemical assay (RCA) data and shielding benchmarks that may be used to support (1) verification and validation of depletion and shielding computer codes used in spent nuclear fuel (SNF) transportation and storage applications and (2) determination of calculation uncertainties. Databases reviewed in this work include the Spent Fuel Composition (SFCOMPO) database [1], Shielding Integral Benchmark Archive and Database (SINBAD) [2], and the International Handbook of Evaluated Criticality Safety Benchmark Experiments [3]. These databases are developed and maintained by the members of the Organisation for Economic Co-operation and Development (OECD) Nuclear Energy Agency (NEA). Published measurements of external dose rates produced by loaded SNF casks [4] were also reviewed.

The SFCOMPO database, which contains publicly available experimental SNF assay data, is developed by the NEA in close collaboration with Oak Ridge National Laboratory (ORNL). The SFCOMPO 2.0 database, which was released in 2017, includes measurement data for a total of 750 fuel samples from 44 different reactors. This database can be accessed online from OECD [5]. The data in SFCOMPO have been independently reviewed for consistency with the original reports, but these data have not been formally evaluated. An effort of the SFCOMPO Technical Review Group (TRG) is underway to publish first isotopic evaluations of individual assay data using a standard data evaluation format.

SINBAD is available from the Radiation Safety Information Computational Center (RSICC) at ORNL as data package RSICC DLC-237 and from the NEA as data packages NEA-1517, NEA-1552, and NEA-1553. SINBAD contains measurement data from fission reactor, fusion reactor, and accelerator shielding experiments. Examples of measurement data include neutron reaction rates for various irradiated activation detectors, neutron spectra measured with organic liquid scintillators, and gamma radiation exposure rates measured with ionization chambers. Benchmark evaluations and calculation results are available in the database for some of the experiments. The SINBAD TRG recently initiated benchmark evaluations and modernization of the database.

The International Handbook of Evaluated Criticality Safety Benchmark Experiments is published by the International Criticality Safety Benchmark Evaluation Project (ICSBEP). The evaluated criticality safety benchmark data contain criticality benchmark specifications for critical, near-critical, or subcritical configurations, criticality alarm placement/shielding configurations, and fundamental physics measurements relevant to criticality safety applications. The criticality alarm placement/shielding configurations, which represent a small subset of these criticality safety benchmark experiments, also can be used for validation of computer codes used for spent fuel transportation and storage system applications [6].

This report is organized as follows. A summary of available RCA data for pressurized water reactor (PWR) and boiling water reactor (BWR) SNF samples is presented in Section 2. RCA data providing measurement data for nuclides important to radiation source terms and dose rate evaluations are also presented in Section 2. A summary of applicable shielding benchmarks and measurements of prototypic cask environments is presented in Section 3. Conclusions are provided in Section 4.

2. EXPERIMENTAL ASSAY DATA FOR SPENT NUCLEAR FUEL

A summary of available SNF experimental assay data widely used in depletion code validations is presented in this section. These data consist of measurements of nuclide concentrations in PWR and BWR SNF samples by post irradiation destructive RCA. Publicly available data are included in the SFCOMPO database [1]. However, proprietary data are also available for depletion code validations. The SFCOMPO database provides both the measurement data and the fuel design and irradiation data required for modeling and simulations. The data in the SFCOMPO database have been independently reviewed for consistency with the original reports but have not been formally evaluated using a standard evaluation format. Previous data evaluations have been performed as part of depletion code validations, with emphasis on validation data for burnup credit criticality safety [e.g., 7, 8, 9, 10, 11]. An effort of the SFCOMPO TRG is underway to publish the first evaluations for individual assay data using a standard data evaluation format [12]. These initial evaluations will also provide sample SCALE 6.2.3 input descriptions and depletion calculation results. As a result of this effort, data quality will be evaluated, models will be developed, depletion calculations will be performed, and recommendations will be provided to database users.

RCA data for 302 PWR SNF samples and 249 BWR SNF samples are currently available in the SFCOMPO database. The PWR SNF samples, obtained from 15 PWRs representative of old and modern reactor designs, have a maximum evaluated burnup of 75 GWd/MTU. The BWR SNF samples, obtained from 12 BWRs primarily representative of old reactor designs, have a maximum evaluated burnup of 77.6 GWd/MTU. Table 1 presents the main characteristics of measured PWR fuel samples, including reactor names, fuel assembly designs, the range of the initial enrichments of the measured fuel samples, and the range of estimated sample burnup. The same information for the BWR is presented in Table 2. However, not all RCAs provide measurement data for nuclides important to radiation source term and shielding calculations.

The most important radionuclides in SNF with respect to radiation source terms have been identified in previous work based on cask shielding calculations [13, 14, 15]. The cooling times analyzed in those studies are 1 to 40 years, 100 years, and 10,000 years. The list of nuclides includes radioactive fission products, transuranic nuclides, and ^{60}Co. Principal gamma emitters in SNF include ^{144}Ce ($T_{1/2}$=284.89 days)/^{144}Pr ($T_{1/2}$=17.29 min), ^{106}Ru ($T_{1/2}$=1.02 years)/^{106}Rh ($T_{1/2}$=2.18 h), ^{134}Cs ($T_{1/2}$=2.0652 years), ^{154}Eu ($T_{1/2}$=8.593 years), and ^{137}Cs ($T_{1/2}$=30.1 years)/^{137m}Ba ($T_{1/2}$=2.6 min). ^{90}Sr ($T_{1/2}$=28.78 years)/^{90}Y ($T_{1/2}$=64 h) contributes bremsstrahlung radiation. Principal neutron emitters in the SNF currently in dry storage include ^{244}Cm ($T_{1/2}$=18.1 years) and ^{242}Cm ($T_{1/2}$=0.45 years). Other transuranic nuclides important for SNF with long cooling times (100 years or longer) include ^{246}Cm, ^{241}Am, ^{238}Pu, ^{239}Pu, ^{240}Pu, and ^{242}Pu. Available RCA data for nuclides important to dose rate are marked with the character "×" in Table 3. This table shows that limited RCA data measurements are available for fission product gamma emitters than are for the transuranic nuclides. Notably, BWR ^{90}Sr measurement data only exist for old fuel assembly types (6×6 and 7×7) from two BWRs.

Proprietary RCA data for fuel samples from the BWR reactors Leibstadt-3 [16] and Limerick [17] have been evaluated at ORNL for burnup credit criticality safety validations [10]. The evaluated Leibstadt-3 UO_2 fuel samples, which have an initial ^{235}U of 3.9% and an evaluated burnup range of 56–63 GWd/MTU, were obtained from a 10×10 (SVEA-96) fuel assembly. The evaluated Limerick fuel samples were obtained from UO_2 (3.95% ^{235}U) and UO_2 – Gd_2O_3 (3.6% ^{235}U – 5.0% Gd_2O_3) rods of a 9×9 (GE-11) fuel assembly and have an evaluated burnup range of 37–65 GWd/MTU.

Most recent experimental program at the time of this writing, called the Rod-Extremity and Gadolinia Analysis (REGAL) program [18], is an international program with the scope to provide high quality nuclide inventory data to fill in existing gaps in the database of nuclide inventories for irradiated PWR UO_2 and UO_2-Gd_2O_3 fuel rods. These data are primarily intended for code validations for burnup credit criticality safety. The UO_2 sample analyzed so far for the REGAL program [18] was obtained from a UO_2 rod with initial enrichment of 4.25% and rod average burnup of 50 GWd/MTU. The analyzed UO_2-Gd_2O_3 sample was obtained from a fuel rod with an average burnup value of 12 GWd/tHM. The ^{235}U enrichment of the UO_2-Gd_2O_3 fuel rod is 2%, and the Gd_2O_3 percentage in the UO_2-Gd_2O_3 fuel is 10%.

Cobalt-60 is an activation product primarily produced by neutron capture reactions of the cobalt impurity existing in fuel hardware and non-fuel hardware materials such as steel and nickel-based alloys. Calculating ^{60}Co activation sources in SNF assembly hardware materials requires determining the average neutron flux and spectrum in the hardware regions outside the fueled region of the reactor. The accuracy of ^{60}Co predictions is affected by the uncertainty associated with the cobalt impurity amount in fuel assembly structural materials. Because the cobalt impurity amount is typically unknown, the historic analysis approach has been to calculate ^{60}Co activation sources using a bounding cobalt impurity concentration in hardware materials and the average neutron flux and spectrum from the active fuel region. The neutron flux in a hardware region is then factored in by applying flux scaling factors [19] to this activation source.

Table 1. PWR reactor, fuel assembly design, sample initial enrichment, and estimated sample burnup

Reactor (country)	Assembly lattice	Number of samples	UO_2 sample enrichment (% ^{235}U)	UO_2 – Gd_2O_3 sample enrichment (wt% ^{235}U–wt% Gd_2O_3)	Sample burnup range (GWd/MTU)
Calvert Cliffs-1 [a] (US)	14×14	33	2.453; 2.72; 3.038	–	12.92–47.05
Genkai-1 (JPN)	14×14	2	3.415	–	38.1–38.7
GKN II [b] (GER)	18×18	1	3.8	–	54.1
Gösgen (SWTZ)	15×15	9	3.5; 4.1	–	21.76–59.7
H. B. Robinson-2 (US)	15×15	7	2.561	–	16.0–30.92
Mihama-3 (JPN)	15×15	9	3.203; 3.208; 3.21	–	6.9–34.2
Obrigheim (GER)	14×14	33	2.83; 3.00; 3.13	–	15.6–37.5
Ohi-1 (JPN)	17×17	1	3.2	–	52.434
Ohi-2 (JPN)	17×17	5	1.6874	1.6874–6.0	21.465–38.496
Takahama-3 (JPN)	17×17	16	2.63; 4.11	2.63–6.0	7.4–47.3
TMI-1 (US)	15×15	24	4.01; 4.66	–	22.8–55.7
Trino Vercellese (ITLY)	15×15	49	2.719; 3.13; 3.897	–	7.2–27.8
Turkey Point-3 (US)	15×15	18	2.556	–	20.0–31.6
Vandellos II (SPN)	17×17	17	4.5	–	43.5–75.0
Yankee-1 (US)	18×18-B; 18×18-A	78	3.4	–	6.3–43.2

[a]Additional data for two Calvert Cliffs-1 SNF samples measured at ORNL are documented in a paper by J. Hu et al. [20]. The initial enrichment and burnup of these samples are 3.038% and ~43.5 GWd/MTU, respectively.
[b]Under the name of Neckarwestheim-2 in the SFCOMPO database, reactor name abbreviated as GKN II [21, 22].

Table 2. BWR reactor, fuel assembly design, sample initial enrichment, and estimated sample burnup

Reactor (country)	Assembly lattice	Number of samples	UO_2 sample enrichment (wt% ^{235}U)	UO_2 – Gd_2O_3 sample enrichment (wt% ^{235}U–wt% Gd_2O_3)	Sample burnup range (GWd/MTU)
Cooper-1 (US)	7×7 (GE-3B)	17	2.94	–	17.84–34.45
Dodewaard-1 (NETH)	6×6	5	4.92	–	55.49
Forsmark-3 (SWDN)	10×10 (SVEA-100)	2	3.97	–	60.7
Fukushima-Daiichi-3 (JPN)	8×8-1	36	1.45; 1.87; 3.01	3.01– N/A[a]	4.24–33.6
Fukushima-Daini-1 (JPN)	9×9-9	13	2.1; 4.9	3.0–5.0	27.89–68.42
Fukushima-Daini-2 (JPN)	8×8-2; 8×8-4	44	3.4; 4.5	3.4–4.5	7.19–59.1
Garigliano-1 (ITLY)	8×8; 9×9	26	1.6; 2.1; 2.41	–	5.58–12.83
Gundremmingen-1 (GER)	6×6GUN	18	2.53	–	15.22–30.12
JPDR-1 (JPN)	6×6	30	2.6	–	2.185–7.04
Monticello-1 (US)	8×8	30	1.45; 1.87; 2.14; 2.87	2.87–1.5	32.7–58.7
Quad Cities-1 (US)	8×8	18	3; 3.8	–	52.5–77.6
Tsuruga-1 (JPN)	7×7	10	1.44	–	8.64–27.74

[a]Not available.

Table 3. Summary of RCA data for nuclides important to radiation source terms and dose rate

Nuclide	^{144}Ce	^{106}Ru	^{134}Cs	^{154}Eu	^{90}Sr	^{137}Cs	^{242}Cm	^{244}Cm	^{238}Pu	^{241}Am	^{246}Cm	^{240}Pu	^{239}Pu	^{242}Pu
Half-life (years)	0.78	1.02	2.065	8.59	28.9	30.07	0.446	18.1	87.81	433	4.73×10^{3}	6.56×10^{3}	2.41×10^{4}	3.75×10^{5}
PWR														
Calvert Cliffs-1				×	×	×			×	×		×	×	×
Genkai-1		×	×	×		×		×	×	×		×	×	×
GKN II	×			×		×	×	×	×	×		×	×	×
Göesgen-1	×	×	×	×	×	×	×	×	×	×	×	×	×	×
H. B. Robinson-2	×		×	×	×	×	×	×	×	×		×	×	×
Mihamma-3		×	×	×		×	×	×	×	×		×	×	×
Obrigheim-1			×	×		×	×	×	×	×		×	×	×
Ohi-1	×	×	×	×		×	×	×	×	×	×	×	×	×
Ohi-2	×	×	×	×		×	×	×	×	×	×	×	×	×
Takahama-3	×	×	×	×		×	×	×	×	×	×	×	×	×
TMI-1			×	×		×	×	×	×	×		×	×	×
Trino Vercellese	×	×	×	×		×	×	×	×	×		×	×	×
Turkey Point-3									×			×	×	×
Vandellos II	×	×	×	×		×		×	×	×	×	×	×	×
Yankee-1					×	×			×	×		×	×	×
BWR														
Cooper-1					×	×			×	×		×	×	×
Dodewaard-1	×	×	×	×	×	×	×	×	×	×	×	×	×	×
Forsmark-3	×							×	×	×		×	×	×
Fukushima-Daiichi-3												×	×	×
Fukushima-Daini-1			×	×		×			×			×	×	×
Fukushima-Daini-2	×	×	×	×		×	×	×	×	×	×	×	×	×
Garigliano-1	×	×	×			×	×	×	×	×		×	×	×
Gundremmingen-1			×	×		×	×	×	×	×		×	×	×
JPDR-1	×	×	×	×		×	×	×	×	×		×	×	×
Monticello-1							×					×	×	×
Quad Cities-1			×			×		×	×	×		×	×	×
Tsuruga-1									×			×	×	×

3. SHIELDING CODE BENCHMARKS

Radiation shielding for transportation packages is typically provided by thick layers of steel, lead, and neutron absorber materials such as polymers and/or borated materials. Concrete is the primary shielding material used in dry storage casks. The shielding materials for transfer casks used in storage systems may include carbon steel, lead, concrete, water, and borated materials. The external cask dose rate consists of gamma and neutron radiation components. It is therefore of interest to validate shielding code results against benchmark experiments designed to test neutron and gamma transport through typical shielding materials.

A dry storage cask typically requires maintaining an air flow around the fuel canister to control the cladding temperature of the stored fuel. The air flow is facilitated by a set of inlet and outlets vents or openings through the shielding layers. To minimize the impact of vents on dose rates produced by a dry storage cask, these vents are typically shaped as a labyrinth. Such configuration prevents a direct streaming path from the radiation source. It is therefore important to benchmark the computer codes against the measurements of radiation streaming through shielding labyrinths.

At large distances from a dry storage installation, the skyshine and groundshine components become a dominant part of the far-field dose rate from the installation. It is therefore important to benchmark radiation transport codes against measurements of skyshine and groundshine radiation.

Table 4 presents a summary of applicable fission reactor benchmark experiments documented in the SINBAD 2018 release [2]. Table 5 presents a summary of applicable shielding benchmarks included in the International Handbook of Evaluated Criticality Safety Benchmark Experiments [3, 6]. These selected experiments have tested radiation attenuation in shielding materials (e.g., Fe, steel, Pb, water, elements found in concrete and soil) and radiation scattering phenomena (e.g., radiation streaming through ducts with labyrinth-like configurations, skyshine and groundshine) relevant to radiation transport calculations for SNF transportation packages/dry storage casks/transfer casks. The typical radiation sources in these selected experiments are thermal-neutron reactors, ^{252}Cf sources, and ^{60}Co sources.

Table 4. Summary of applicable fission reactor integral shielding experiments available in the 2018 SINBAD release

Experiment name	Shielding materials	Radiation source and facility	Measurement/phenomena tested
Experiments testing radiation attenuation in individual shielding materials			
YAYOI iron	Fe	YAYOI reactor, University of Tokyo, JPN	Transmission of neutron spectra through iron slabs with a thickness up to 20 cm
Karlsruhe iron sphere [a]	Fe	^{252}Cf source, Karlsruhe Nuclear Research Center, GER	Neutron leakage spectra from a set of iron spheres of diameters 15, 20, 25, 30, 35, and 40 cm, with a ^{252}Cf source in the center
CSEWG SB2 [a]	Fe, Al, Cu, Zn, Ti, Ni, Si, Ca, P, Na, Ba, Cl, S; stainless steel	The Tower Shielding Reactor, ORNL, US	Secondary gamma-ray spectra from thermal-neutron capture in materials important to reactor shielding
CSEWG SB3 [a]	Same as above	Same as above	Secondary gamma-ray spectra from fast-neutron capture in different shielding materials
Iron broomstick [a]	Fe	Same as above	Spectra of uncollided fission neutrons transmitted through thick samples of iron
Oxygen broomstick [a]	O	Same as above	Spectra of uncollided fission neutrons transmitted through thick samples of oxygen
Nitrogen broomstick [a]	N	Same as above	Spectra of uncollided fission neutrons transmitted through thick samples of nitrogen
Stainless steel broomstick [a]	Stainless steel	Same as above	Spectra of uncollided fission neutrons transmitted through thick samples of stainless steel
Neutron transport through iron and stainless steel [a]	Fe, stainless steel	Same as above	Spectra of uncollided fission neutrons transmitted through thick samples of iron/stainless steel
Ispra iron (EURACOS)	Fe	Fission plates placed at the end of the thermal column of the 250 kW TRIGA MARK II reactor of the University of Pavia, ITLY	Neutron flux and spectra measured up to 130 cm in iron
Neutron leakage from water spheres	H_2O	^{252}Cf source, NIST, US	Fission reaction rates and neutron leakage from water surrounding a ^{252}Cf source
NAÏADE 1 light water [a]	H_2O	Fission plates irradiated by a beam of purely thermal neutrons coming from the graphite reflector of the ZOE heavy water reactor, NAÏADE facility, Fontenay aux Roses, FRAN	Fission neutron transport in light water for penetration up to 50 cm for the fast and up to 150 cm for the thermal neutrons
NAÏADE 1 iron [a]	Fe	Same as above	Fission neutron transport in iron for penetration up to 80 cm
NAÏADE 1 concrete	Concrete	Same as above	Fission neutron transport in concrete for penetration up to 100 cm for fast neutron measurements and up to approximately 120 cm for thermal neutrons

[a]An input file for shielding calculations is not currently available in SINBAD for this experiment.

Table 4. Summary of applicable fission reactor integral shielding experiments available in the 2018 SINBAD release (continued)

Experiment name	Shielding materials	Radiation source and facility	Measurement/phenomena tested
Experiments testing radiation attenuation in individual shielding materials			
University of Illinois iron sphere	Fe	(1) ^{252}Cf source, and (2) deuterium-tritium (D-T) fusion neutron source provided by a neutron generator, University of Illinois, US	Fast neutron leakage spectra from a spherical shell of iron to test the validity and accuracy of the neutron cross section data
Winfrith water (ASPIS)	H_2O	^{252}Cf source, AEE Winfrith, UK	Fast neutron spectra above 1 MeV and detector reaction rates up to 50 cm in water
Winfrith iron (ASPIS)	Fe	Fission converter plates driven by a thermal flux from the extended graphite reflector of the NESTOR reactor, ASPIS shielding facility, AEE Winfrith, UK	Neutron spectra and detector reaction rates at different depths in a bulk iron shield about 1 m thick
Winfrith iron 88 (ASPIS)	Steel	Same as above	Neutron transport for penetrations up to 67 cm in steel
Janus phase I	Stainless steel	Same as above	Testing the prediction of neutron penetration through stainless steel where the incident spectrum was typical of that emerging from a fast reactor
Reactor / mock-up reactor shielding configurations			
VENUS-3	Mock-up of the pressure vessel internals representative of a 3-loop Westinghouse reactor	VENUS-3 zero power core, VENUS Critical Facility, CEN/SCK Mol, BELG	Power distribution for validating the analytical methods needed to predict the azimuthal variation of the fluence in the pressure vessel
H.B. Robinson-2 pressure vessel dosimetry	H.B. Robinson-2 shielding materials	H.B. Robinson-2 reactor, US	In- and ex-vessel neutron dosimetry measurements for verifying neutron transport calculations
NESDIP-2 (ASPIS) [a]	H_2O, stainless steel, and carbon steel	Fission converter plates driven by a thermal flux from the extended graphite reflector of the NESTOR reactor, ASPIS shielding facility, AEE Winfrith, UK	Neutron transport in a shield simulating the radial shield of a PWR, including the cavity region and the backing shield
NESDIP-3 (ASPIS) [a]	Water and stainless steel	Same as above	Same as above
ASPIS neutron/gamma-ray transport through water/steel arrays [a]	Fe and H_2O	Same as above	Neutron activation and gamma-ray dose rate in an experimental configuration comprising a shield of iron and water
Winfrith water/iron (ASPIS-PCA REPLICA)	Model of the Oak Ridge Pool Critical Assembly (PCA) 12/13 configuration	Same as above	Neutron transport in a water/iron shield reproducing the ex-core radial geometry of a PWR

[a]An input file for shielding calculations is not available in SINBAD for this experiment.

Table 4. Summary of applicable fission reactor integral shielding experiments available in the 2018 SINBAD release (continued)

Experiment name	Shielding materials	Radiation source and facility	Measurement/phenomena tested
Experiments involving neutron streaming through ducts			
Streaming through ducts	Steel and concrete	Research reactor, Institute of Nuclear Techniques (NTI), the Technical University of Budapest, HUNG	Fast and thermal neutron reaction rates in straight and bent steel-walled cylindrical ducts in concrete
Skyshine experiments			
Baikal-1 skyshine	Air and soil	The research RA reactor, Semipalatinsk Nuclear Test Site, KAZ	Spatial energy distributions of neutrons and photons scattered in the air near the ground-air interface up to 1500 m from the reactor axis
Photon skyshine	Air and soil	^{60}Co photon sources, Kansas State University Nuclear Engineering Shielding Facility, US	Skyshine from ^{60}Co photon sources measured at distances in air up to 700 m

Table 5. Summary of criticality alarm experiments relevant to shielding

ICSBEP evaluation identifier	Shielding materials	Radiation source and facility	Measured quantity
Experiment involving individual shielding materials			
ALARM-CF-PB-SHIELD	Pb	^{252}Cf source, Institute of Physics and Power Engineering, RUS	Neutron and photon leakage spectra from ^{252}Cf at centers of lead spheres of various diameters
ALARM-CF-FE-SHIELD	Fe	Same as above	Neutron and photon leakage spectra from ^{252}Cf at centers of iron spheres of various diameters
ALARM-CF-Air-SHIELD	Air	Same as above	^{252}Cf neutron and photon spectra in air
ALARM-TRAN-Air-SHIELD	Air	SILENE reactor facility, Valduc, FRA	Neutron activation and thermoluminescent dosimetry (TDS) in air
ALARM-TRAN-CH_2-SHIELD	CH_2	Same as above	Neutron activation and TDS in polyethylene
ALARM-TRAN-PB-SHIELD	Pb	Same as above	Neutron activation and TDS in lead
Skyshine and groundshine			
ALARM-REAC-AIR-SKY [a]	Air and soil	The research RA reactor, Semipalatinsk Nuclear Test Site, KAZ	Spatial energy distributions of neutrons and photons scattered in the air near the ground-air interface
Radiation streaming experiments			
ALARM-CF-AIR-LAB	Concrete	^{252}Cf source, Institute of High Energy Physics, RUS	Neutron fields in concrete labyrinth with additional plates of polyethylene and borated concrete
ALARM-CF-CH2-LAB	Concrete	^{252}Cf source filtered by 30.5 cm diameter polyethylene sphere, Institute of High Energy Physics, RUS	Same as above

[a]Baikal-1 skyshine experiment in SINBAD.

3.1 PROTOTYPIC ENVIRONMENTS

Dose rate measurements around a cask loaded with SNF assemblies may be used to validate the overall calculational procedure that includes both source term and shielding calculations [4]. Examples of prototypic measurements performed during the period 1984 to 1990 are described in EPRI TR-104329 [4] for five casks loaded with PWR fuel. The cask designs are CASTOR-V/21, MC-10, TN-24P, and VSC-17. The fuel initial enrichment varied from 1.9 to 3.2%, and fuel assembly average burnup varied from 24 to 36 GWd/MTU. The Westinghouse MC-10 cask [23] is a representative overpack for transportation packages, with a fuel basket for 14 PWR fuel assemblies, a containment vessel made of forged steel, and neutron shielding on the periphery. The Ventilated Storage Cask (VSC)-17, which has a capacity of 17 fuel assemblies, is a typical dry storage concrete cask with inlet and outlet vents for cooling. Source term and shielding calculations were performed at ORNL [4] and compared against the measured dose rates. The calculated neutron dose rates differed from the measured dose rates by less than 30%, and the calculated gamma dose rate over predicted dose rate by approximately 60%. The discrepancy between calculated and measured gamma dose rate values was attributed to uncertainties associated with the cobalt impurity amount in fuel hardware materials. Dose rate measurements for the CASTOR-V/21 cask were again obtained in 2001 [24]. More recent gamma dose rate measurements were obtained at ORNL for the NAC International Light Weight Truck Cask containing 25 high burnup fuel rods. The calculated gamma dose rate values for axial locations away from cask top and bottom regions were in good agreement (within 20%) with the measurements, and a larger discrepancy was obtained for measurement locations at the top and bottom of the cask [25].

4. CONCLUSIONS

This report identifies RCA data and shielding benchmarks that may be used to support verification and validation of computer codes used in SNF cask/transport shielding applications, as well as determination of calculation uncertainties, based on a review of publicly available information. The reviewed relevant information includes the SFCOMPO database for isotopic validations, the shielding benchmarks available in SINBAD and the International Handbook of Evaluated Criticality Safety Benchmark Experiments, and published measurements of external dose rates produced by loaded casks.

RCA data for 302 PWR SNF samples and 249 BWR SNF samples are currently available in the SFCOMPO database. The PWR SNF samples, obtained from 15 PWRs representative of old and modern reactor designs, have a maximum evaluated burnup of 75 GWd/MTU. The range of initial ^{235}U enrichment in the measured PWR UO_2 fuel rods is 1.6874% to 4.66%. PWR RCA data also include isotopic measurements for fuel samples from two UO_2-Gd_2O_3 rods with a 6% gadolinia content and initial ^{235}U enrichments of 1.6874% and 2.63%. The BWR SNF samples, obtained from 12 BWRs primarily representative of old reactor designs, have a maximum evaluated burnup of 77.6 GWd/MTU. The range of initial ^{235}U enrichment in the measured BWR UO_2 fuel rods is 1.44% to 4.92%. For the measured BWR UO_2-Gd_2O_3 rods, the range of ^{235}U enrichment is 2.87% to 3.4% and the range of gadolinia content is 1.5% to 5%. However, not all of these RCAs provide measurement data for nuclides important for radiation source term and shielding. Notably, BWR ^{90}Sr measurement data only exist for old fuel assembly types (6×6 and 7×7) from two BWRs.

The shielding benchmark measurements relevant to radiation transport calculations for SNF transportation packages/dry storage casks/transfer casks have tested radiation attenuation in shielding materials (25 benchmarks), including Fe, steel, Pb, water, elements found in concrete and soil; radiation streaming through ducts with labyrinth-like configurations (3 benchmarks); and radiation scattering in the air and the ground (i.e., skyshine and groundshine) (2 benchmarks). The typical radiation sources in these selected experiments are thermal-neutron reactors, ^{252}Cf sources, and ^{60}Co sources. Dose rate measurements around a cask loaded with SNF assemblies may be used to validate the overall calculational procedure that includes both source term and shielding calculations.

It should be noted that only a relatively small subset of the identified experimental data (e.g., criticality alarm experiments) is available in a standard format established by the international community participating in experimental isotopic and shielding data evaluations. An effort of the SFCOMPO TRG is underway to publish first isotopic evaluations of individual assay data using a standard data evaluation format. The SINBAD TRG has recently initiated benchmark evaluations and modernization of the database. Therefore, more relevant information will be available in the future that will enable users to select quality experimental data in depletion code and shielding code validations for SNF applications.

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