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LLNL-TR-2017627

Survey of emerging nuclear data needs for nonproliferation applications with advanced reactors

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April 2026

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This work performed under the auspices of the U.S. Department of Energy by Lawrence Livermore National Laboratory under Contract DE-AC52-07NA27344.

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Survey of emerging nuclear data needs for nonproliferation applications with advanced reactors

March 30, 2026

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This work was performed under the auspices of the U.S. Department of Energy by Lawrence Livermore National Laboratory under Contract DE-AC52-07NA27344, by Lawrence Berkeley National Laboratory under Contract DE-AC02-05CH11231, and by Argonne National Laboratory under DE-AC02-06CH357. The work at Brookhaven National Laboratory was sponsored by the Office of Nuclear Physics, Office of Science of the U.S. Department of Energy under Contract No. DE-AC02-98CH10886 with Brookhaven Science Associates, LLC.

Executive Summary

Nuclear science plays a key role in non-proliferation activities supporting advanced reactor technologies. Nuclear data underpin predictions and interpretations of nuclear material behavior and signatures in reactor fuel production, use, transport, and storage. Advanced reactors provide new challenges compared to the current fleet of thermal fission reactors.

This report consolidates reported nuclear data needs from representative workshops, conferences, and publications, identifying six themes for recommended future investments supporting non-proliferation and safeguards applications. While also identified as data needs, major fission product evaluations and (α,n) reactions were omitted as there are on-going activities producing new data under NA22/Objective O.

Each theme is summarized below with example data and association with the nonproliferation mission for advanced fuels and reactors.

Theme	Data	Mission connection
Activation/Inventory and decay data	Cross sections and products for a range of incident neutron energies; recommended compilation or "library"	Radioactive decay signatures from neutron activated structural materials, coolants, salts, reflectors, and trace impurities support monitoring and detection. In advanced reactors, harder and more variable neutron spectra can shift activation pathways and isotopic yields, changing the expected signatures. New materials and impurity chemistries can also produce activation products that introduce new signatures or dominate those traditionally relied upon.
Fission characteristics: minor actinides and emissions	Fission cross sections, notably for fast neutrons; prompt and delayed neutron production, fission product yields and gamma production	Safeguards models become more sensitive to minor actinide fission data for fast reactors and recycled fuels. Cm production affects neutron signals used for non-destructive assay. Fission products affect decay heat and gamma signatures used in verification and anomaly detection.
Neutron scatter: fuels, structural, coolant materials	Energy and angle-dependent cross sections, inelastic level data, and neutron spectrum	Scattering in fuels, coolants, moderators, and structural materials in advanced reactors strongly shapes the neutron energy spectrum and spatial distribution, which in turn drives reactivity and affects the performance of shielding and

		instruments. For safeguards applications, monitors are also sensitive to neutron energies and scatter in surrounding materials.
Light element data: Li, Be, C, F, N; including thermal models	Neutron scatter and capture cross sections, *S(α,β)* and temperature dependence across epithermal to fast energies	Light nuclides may be in the coolant, moderator, fuel matrix, salts, absorbers, and structural ceramics, and they strongly control neutron moderation, leakage, and reactivity. Non-traditional materials (graphite, salts, nitrides, carbides, hydrides) introduce light elements that dominate moderation or absorption.
Improve coverage/quality of uncertainties	Reaction and decay data covariances, including fission emissions	Uncertainties are needed to understand confidence in predictive assessments, identification of signatures, and defining detection margins for advanced reactor safeguards. Uncertainties and sensitivity studies identify priority data needs and direct future investments and data activities.
Benchmarks beyond criticality	Depletion and isotopic inventories, time-dependent neutron and gamma emissions, activation products in materials, shielding effects, and detector response functions	Nuclear benchmarks are essential for validating and improving nuclear data and software used to predict and diagnose nuclear materials. They provide well-characterized reference cases that integrate multiple data types, enabling quantification of accuracy and uncertainty.

I. Introduction and Purpose

Interest in advanced nuclear reactors has increased sharply in recent years due to a convergence of energy needs, technological progress, and policy changes. Rising electricity demand from data centers is prompting private industry to explore nuclear options that can provide dedicated capacity and long-term cost stability. Broader priorities, including energy security, firm low carbon power, and remote or resilient generation, are also accelerating development.

Industrial applications are expanding as well, including desalination, district heating, and hydrogen production. In response, new reactor concepts are being proposed and tested, enabled by advances in high performance computing, advanced manufacturing, and modern analytics, instrumentation, and controls. At the same time, licensing and regulatory reforms are shortening pathways to approval, while government investments and tax credits help lower costs and reduce risk for private investors.

The growth of advanced reactor technologies must be matched by modernized non-proliferation tools and practices to reduce the risk of unauthorized nuclear activities. Proliferation considerations should be incorporated from the outset through reactor and fuel designs that limit the production of weapons-usable material and reduce opportunities for diversion, supported by robust physical and cyber security measures and clear requirements for material handling, transport, and storage. In parallel, safeguards and monitoring will need enhanced capabilities for effective verification and surveillance across new reactor types and fuel forms.

Nuclear science plays a key role in non-proliferation activities supporting advanced reactor technologies. Nuclear data underpin predictions and interpretations of nuclear material behavior and signatures in reactor fuel production, use, transport, and storage. Cross sections, fission yields, and decay emission yields support a range of safeguards activities, including verification of predicted inventories or identification of diversion activities. Advanced reactors provide new challenges compared to the current fleet of thermal fission reactors, with:

- Different neutron spectra, affecting isotopic production
- Novel fuels and salt/metal coolants, introducing different activation products and measurement environments
- Different refueling schemes (longer cycles/higher burnup, online refueling), changing the time evolution of signatures
- Variety of reactor designs, limiting the available experience base and complicating attribution of interdicted material

Over the past five years, workshops, conferences, and publications have identified and documented nuclear data needs for advanced reactor applications. This report consolidates those findings and summarizes nuclear data priorities across the nuclear fuel cycle, with a focus on non-proliferation and safeguards applications.

II. Recommendations and Priorities for Nuclear Data Investments

Nuclear data needs were assessed across fuel production, reactor use, and storage or recycling. Input from both data producers and end users identified a broad set of requirements, with substantial overlap among stakeholder groups. Common themes are:

- Activation/inventory and decay data
- Fission characteristics: minor actinides and emissions
- Neutron scatter: fuels, structural, coolant materials
- Light element data: Li, Be, C, F, N; including thermal scattering models
- Improve coverage/quality of uncertainties
- Benchmarks beyond criticality

While also identified as data needs, major fission product evaluations and (α,n) reactions were omitted as there are on-going activities producing new data under NA22/Objective O.

The six themes identified above support the safeguards and nonproliferation mission by enabling predictive capabilities, signature identification, and monitoring/detection activities. Advanced reactors present added challenges because novel materials and neutron spectra push modeling and measurement beyond well-validated nuclear data regimes. In the following sections, additional details on nuclear data needs across the fuel cycle are summarized.

A. Fuel production and processing

Nuclear safeguards for conventional fuel fabrication rely on Material Control and Accounting (MC&A) and physical protection, which are often integrated. Nuclear measurement techniques, using passive or active methods, help verify material presence and detect loss or diversion. For advanced reactor fuels, the underlying purpose is the same, but implementation may need to adapt to higher enrichment levels, novel material forms, and different process layouts. Light Water Reactor (LWR) fuel fabrication processes have been in use for over 70 years, evolving with improvements in enrichment, fuel design (e.g. burnable poisons), and performance. Advanced reactors stress fuel production throughput, capabilities, and regulatory limits.

Common themes for nuclear data needs in fuel fabrication include:

- (α,n) data on light elements for non-destructive assays
- Thick target yields for Li, Be, B, C, O, F, Al and compounds with these elements
- Reactions in fuel fabrication, notably alloys
- Predictive x-ray, γ-ray, and neutron emission from processing and fresh fuel forms
- Nuclear data uncertainties
- Benchmarks for non-LWR fuels

High Assay Low Enriched Uranium (HALEU)

HALEU, or high assay low enriched uranium (typically 5 to 19.75 percent ^{235}U), enables several advanced reactor designs. The higher enrichment enables smaller reactor cores and more compact designs with longer operating cycles and higher burnup. HALEU also expands the design space, including concepts that rely on fast spectra and high temperature operation. HALEU is a diverse array of forms, including oxide pellets, kernels, pebbles, molten salt, or metallic U-Zr alloys.

Nuclear data benchmarks for HALEU have been identified as a key need to cover anticipated fuel forms, moderators, and configurations. New collaborative activities between DOE and the NRC, for example the DOE/NRC Collaboration for Criticality Safety Support for Commercial-Scale HALEU Fuel Cycles and Transportation (DNCSH), are beginning to address this gap through new criticality measurements targeting representative fuels and reactions involving structural materials and priority isotopes such as ^{35}Cl.

Verifying and measuring uranium enrichment is a core activity in nonproliferation safeguards. Nuclear data underpin the interpretation of results from non-destructive assay (gamma and neutron emissions), active interrogation (induced emissions), and destructive analysis (chemical sampling and mass spectrometry). Non-destructive methods are generally preferred because they reduce cost, measurement time, and operational burden. HALEU increases the challenge: enrichment values approaching 20 percent drive tighter accuracy requirements, while higher enrichment also tightens criticality safety constraints that can limit measurement geometries and reduce available sample sizes. Robust and reliable nuclear data uncertainties are critical toward defining margins and confidence. Other key data needs include high-precision gamma and x-ray emission probabilities for ^{234}U, ^{235}U, plus improved approaches for inferring enrichment from complex or attenuated spectra associated with HALEU material forms. Photon emission data for ^{131m}Xe, ^{133m}Xe, and ^{135}Xe are also important to support measurements and modeling for new HALEU fuel forms and processing steps, particularly where fission gas signatures may be used for monitoring or validation.

Representative references:

A. Barto (US NRC), D. Algama (DOE NE), *HALEU Transportation and fuel cycle licensing and the DNCSH Project*, in Workshop for Applied Nuclear Data Activities (WANDA 2025), Feb. 10–Feb. 13, 2025.

N. Thomson, T. Cutler (LANL), *Deimos and other HALEU Critical Experiments at NCERC*, in Workshop for Applied Nuclear Data Activities (WANDA 2025), Feb. 10–Feb. 13, 2025.

R. Pulido, D. Ames (SNL), *Applied Nuclear Data Supporting HALEU Fuel Transport: DNCSH Critical Benchmark Experiments at SPRF/CX,* in Workshop for Applied Nuclear Data Activities (WANDA 2025), Feb. 10–Feb. 13, 2025.

M. Croce (LANL), *Data Needs for Nuclear Material Accounting and Safeguards in the HALEU Fuel Cycle*, in Workshop for Applied Nuclear Data Activities (WANDA 2025), Feb. 10–Feb. 13, 2025.

Tri-structural ISOtropic (TRISO) particle fuel

Unlike traditional light water reactors (LWRs), advanced reactors that use TRISO-based fuels do not benefit from decades of operating experience with this fuel type. To better understand the limitations of TRISO fabrication and existing studies, several areas require additional analysis and further study.

Carbon is a major constituent of TRISO particles. In a modeling study using high temperature reactors (HTTR and HTR-10) for various cross-section data evaluations, it was found that there were significant variations in the estimated k-effective. These discrepancies were associated with inadequacies in the graphite/carbon evaluations in ENDF/B-VII.1 to ENDF/B-VIII.0. These differences can get larger in higher burnup scenarios.

A vast majority of advanced reactors proposed to operate on TRISO use similar compositions with slight variations in size. The study using the HTTR and HTR-10 highlighted how sensitive modeling of these systems can be to nuclear data especially due to the carbon and uranium data. For ENDF/B-VII.0, both systems show significant positive reactivity biases (i.e. about 1300-1400 pcm), largely due to underpredicted thermal carbon capture. The increase in the thermal carbon absorption cross-section in ENDF/B-VII.1 reduces k_{eff} by several hundred pcm, bringing predictions closer to benchmark values. However, subsequent updates in ^{235}U and ^{238}U cross-sections in ENDF/B-VII.0 reintroduce a ~300 pcm increase relative to VII.1, highlighting that TRISO modeling accuracy depends not only on moderator data but also on fissile isotope evaluations.

For TRISO fuel, this is amplified by the double heterogenous structure of the coated fuel particles embedded in graphite matrices and surrounded by graphite reflectors. The introduction of multiple graphite evaluations in ENDF/B-VIII.0 (perfect crystal, 10% porosity and 30% porosity) reveals that graphite porosity strongly affects system reactivity when applied throughout all graphite regions. While reflector-only porosity changes k_{eff} by less than ~150 pcm, assuming 10-30% porosity in all graphite structures increases eigenvalues by 300-600 pcm at room temperature. This occurs because porosity effectively reduces the graphite density, thereby lowering moderation power and absorption per unit space, hardening the neutron spectrum and increasing fast fission contributions in ^{238}U. In TRISO modeling, this directly alters the thermal utilization factor and resonance escape probability, particularly within the compact matrix region where neutron slowing down lengths are short and spatial self-shielding is significant. Moreover, the temperature dependence of porosity effects indicates coupling between thermal scattering laws and density dependent moderation. Since TRISO particles rely on precise spectral shaping within the fuel compact, inaccurate graphite density or carbon capture data can misrepresent intra-particle self-shielding and kernel power distribution.

TRISO modeling in High Temperature Reactors (HTR) is particularly sensitive to thermal carbon scattering and capture, uranium resonance and fission cross section and accurate representation of graphite porosity and density. Neglecting these effects can introduce reactivity biases on the order of several hundred pcm, comparable to design margins. With advanced pebble bed reactors anticipated to operate at high temperatures and longer burnups, these concerns with the evaluations of carbon could have significant impacts on the quality and accuracy of simulated results for TRISO based reactor systems.

Representative reference:

M. L. Zerkle, J. L. Wormald (NNL), *Update on ENDF/B-VIII.1 TSLs for Moderator and Fuel Materials*, in Workshop for Applied Nuclear Data Activities (WANDA 2025), Feb. 10–Feb. 13, 2025.

Mixed OXide (MOX) fuel

Mixed oxide (MOX) fuel is composed of plutonium and uranium oxides. The PuO_2 is typically recovered from spent nuclear fuel, while the UO_2 is depleted, natural, or low-enriched. MOX enables recycling of material from spent fuel and can reduce the volume of material requiring long-term storage. Although MOX is most commonly used in light-water reactors, it is also considered for fast-spectrum advanced reactors, particularly sodium-cooled fast reactors (SFRs) where there are reactors in service, and in some designs using lead or lead-bismuth (LFR) or gas (GFR) coolants.

MOX fuels raise proliferation concerns because they require the production, separation, and handling of plutonium. Fuel fabrication involves bulk powder processing, which can complicate material accountancy due to holdup, scrap generation, and measurement uncertainties. For safeguards-relevant material characterization, non-destructive assay relies on isotopic gamma and neutron signatures from ^{238}Pu through ^{242}Pu and ^{241}Am, along with (α,n) neutron production with oxygen in oxide matrices. Nuclear data covariances and correlations are also needed to support defensible uncertainty budgets for accountancy and verification measurements.

Representative references:

IAEA, “Safeguards Techniques and Equipment: 2011 Edition” (International Nuclear Verification Series, No. 1)

IAEA Department of Safeguards, “Nuclear Material Accounting Handbook”

IAEA Nuclear Energy Series, “International Safeguards in the Design of Fuel Fabrication Plants”, No. NF-T-4.7 (2017).

Accident Tolerant Fuel (ATF)

Accident tolerant fuels (ATF) are new nuclear fuel and cladding concepts designed to improve off-normal and accident conditions relative to light water reactor (LWR) materials while maintaining reactor performance. For example, high density fuel forms such as nitrides (UN) and silicates (U_3Si_2) can withstand higher temperatures and retain fission products, notably gases. Alternative cladding materials, such as FeCrAl and silicon carbide ceramics reduce high-temperature oxidation and hydrogen generation compared to Zr alloys.

Validated nuclear data are needed for new ATF materials, supported by both differential measurements and integral benchmarks. For nitride fuels, this includes improved neutron scattering, (n,p), and capture data for ^{14}N and ^{15}N, along with new UN integral benchmarks to validate calculations. For alternative claddings and coatings, priorities include neutron reaction cross sections, thermal scattering laws $S(\alpha,\beta)$ and activation data, including reaction cross sections, product yields, and decay data. Cr-doped fuels and Cr-coated Zr cladding are anticipated in the near term, making Cr data a priority. For new fuel compositions, updated fission yields for relevant neutron spectra and fuel chemistries are needed to enable predictive depletion and inventory calculations, as well as decay heat and radionuclide source term estimates.

Representative references:

A. Simpson (UK NNL), *Nuclear Data for Advanced Fuel Cycles*, in Workshop for Applied Nuclear Data Activities (WANDA 2021), Jan. 25–Feb. 3, 2021.

L. Kyriazidis (US NRC), *Nuclear Data in the Regulatory Framework*, in Workshop for Applied Nuclear Data Activities (WANDA 2025), Feb. 10–Feb. 13, 2025.

B. Reactor applications and monitoring

Predictive capabilities for modeling and simulation are crucial to the design, licensing, and operation. Safety assessments and accident analyses may require unusual physics regimes to be examined. For nonproliferation activities, these same modeling capabilities are needed to determine signatures of off-normal operations or inconsistencies with safeguards reporting. Additional data, e.g. decay processes that produce neutrinos, may also be needed.

Common themes for nuclear data needs in reactor operations and monitoring include:

- Activation and decay data; coupled reaction and decay database
- (α,n) data on light elements
- Neutron induced reactions on actinides, notably minor actinides, ^{235}U capture in fast regime, ^{238}U neutron scatter, U fission nu-bar
- Fission product yield data, including branching ratios and uncertainties; neutron capture reactions on fission products; uncertainties and correlations on delayed neutron fractions and decay constants
- Thermal scatter law on light elements and compounds (e.g. Be, BeO, Be_2C, MgO, SiC, reactor-grade C, FLiBe, Zr_3Si_2, YH_2), including angular distributions and uncertainties
- Benchmarks for reaction rates, depletion/burnup, and minor actinides
- Uncertainties in reaction, decay, and fission data, including angular distributions, thermal scatter laws, and temperature dependence

Small Modular Reactor (SMR) and microreactor

Small modular reactors (SMRs) are typically defined as producing up to about 300 MWe and are designed for modular, factory-based fabrication with shipment and assembly on site. Multiple modules can be deployed together to scale total output. SMR fuel can be LEU or HALEU, with HALEU more common in advanced, non light-water designs. Microreactors are smaller systems, generally up to about 20 MWe, designed for high transportability and rapid deployment. They often use HALEU to enable compact cores and long operating life between refueling.

Nuclear needs supporting dose and shielding requirements are needed, given the close proximity of people to these smaller reactors. These data also support nonproliferation safeguards monitoring activities and include prompt neutrons and gammas from fission, fission product gamma-rays, neutron capture reactions, material activation and decay, and neutron and gamma attenuation in materials. Thermal scatter data for advanced moderators, such as YH_x for compact, high temperature microreactors and space reactors, and reflectors are also needed. Advanced SMR designs include sodium or lead fast reactors SFR, LFR), He-cooled high temperature gas reactors (HTGR), and molten salt reactors (MSR).

Heat pipe reactors rely on passive heat transport to remove heat from the reactor core and deliver it to the power conversion system. They use sealed metal tubes partially filled with a working fluid, commonly sodium or potassium. For these designs, accurate neutron elastic and inelastic scattering data are needed for sodium and potassium, along with data for structural materials (for example iron) and reflector materials (such as Al_2O_3, and BeO). Thermal scattering laws for key light moderators and reflectors, including beryllium and BeO, are also important. In addition, high-quality uranium nuclear data are required, particularly ^{235}U neutron capture in the fast energy range.

Representative references:

B. Reardon (X-energy), *Summary of Nuclear Data Needs for X-Energy Reactor Designs,* in Workshop for Applied Nuclear Data Activities (WANDA 2021), Jan. 25–Feb. 3, 2021.

F. Bostelmann, G. Ilas, C. Celik, A. M. Holcomb, W. A. Wieselquist, *Nuclear Data Assessment for Advanced Reactors*, NUREG/CR-7289, ORNL/TM-2021/2002 (2022).

High Temperature Gas-Cooled Reactor (HTGR) and Very High Temperature Reactor

High Temperature Gas-Cooled Reactors (HTGRs) and Very High Temperature Reactors (VHTRs) rely heavily on precise nuclear data to validate safety margins, optimize fuel cycle economics, and model the unique physics of graphite-moderated, high-temperature systems. Unlike Light Water Reactors (LWRs), which operate in a lower temperature regime with water moderation, HTGRs require specific data sets that account for spectral shifts, complex fuel geometries, and extreme thermal operating conditions.

One of the most critical data needs involves the thermal scattering laws (TSL) for reactor-grade graphite. As the primary moderator and structural component in the core, graphite's crystalline structure significantly influences neutron thermalization. Current data often relies on historical measurements or theoretical models that may not fully capture the phonon density of states for modern, nuclear-grade graphite, particularly at the elevated temperatures and varying porosity levels found in HTGRs. Inaccuracies in graphite TSL data contribute to uncertainties in the temperature coefficient of reactivity, which is a fundamental safety parameter. Furthermore, data are needed to understand how irradiation damage alters the crystalline structure and subsequently changes the scattering cross-sections over the reactor's lifetime.

The fuel form used in these reactors, typically TriStructural ISOtropic (TRISO) particles, necessitates highly accurate cross-section data for uranium and plutonium isotopes in the resonance region. Because HTGRs target high burnup levels, the isotopic vector of the fuel changes significantly over time, leading to a build-up of plutonium and minor actinides. Consequently, there is a strong need for improved capture and fission cross-sections for ^{239}Pu, ^{240}Pu, and ^{241}Pu, as well as minor actinides like Am and Cm. The "double heterogeneity" of TRISO fuel—where self-shielding occurs both within the particle and within the fuel compact—requires precise resonance parameters to correctly calculate the Doppler feedback effect, which ensures reactor stability during temperature transients.

High operating temperatures, often exceeding 750°C and potentially reaching 1000°C in VHTR designs, accentuate the importance of Doppler broadening data. Nuclear data libraries typically focus on the operating ranges of LWRs; therefore, uncertainties increase when extrapolating cross-sections to the very high temperatures experienced during HTGR normal operation or accident scenarios. Reducing uncertainties in the capture resonances of ^{238}U at these elevated temperatures is particularly important for accurately predicting the negative reactivity feedback that shuts the reactor down during a power excursion.

Beyond the fuel and moderator, there are significant data needs regarding fission products and structural materials. HTGR safety cases often rely on the retention of fission products within the fuel particle, but understanding the source term requires accurate production cross-sections for metallic fission products like ^{110m}Ag, ^{137}Cs, and ^{90}Sr. ^{110m}Ag is of particular interest due to its tendency to migrate through silicon carbide layers, creating maintenance challenges. Additionally, cross-sections for structural materials such as high-temperature nickel alloys, iron, and chromium are vital. Specifically, (n,α) and (n,p) reaction cross-sections are needed to predict helium and hydrogen production in these metals, which causes embrittlement and limits the lifespan of reactor components.

Finally, the passive safety measures inherent in most HTGR designs require accurate decay heat data. While decay heat standards are well-established, the specific nuclide inventory generated by the unique neutron spectrum and high burnup of an HTGR requires validation. Reduced uncertainties in decay heat summation calculations are necessary to demonstrate that the reactor can cool itself via natural circulation and conduction without fuel damage during a loss-of-forced-cooling accident. Along with this, covariance data—which quantifies the uncertainties and correlations between different nuclear data sets—is increasingly requested by regulators to provide a rigorous statistical basis for the safety margins claimed in reactor designs.

Representative references:

F. Bostelmann, G. Ilas, C. Celik, A. M. Holcomb, and W. A. Wieselquist, "Nuclear Data Assessment for Advanced Reactors," NUREG/CR-7289, Oak Ridge National Laboratory; U.S. Nuclear Regulatory Commission (2022).

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J. M. Brown, E. O'Brien, T. Bredeweg, J. Jo Ressler, A. Couture, M. Loughlin, P. Romano, L. Bernstein, T. Cutler, J. Ortensi, A. Daskalakis, D. A. Brown, and R. Vogt, Proceedings for the Workshop on Applied Nuclear Data Activities 2025, ORNL/TM-2025/3966, Oak Ridge National Laboratory, Oak Ridge, TN, (2025).

Molten Salt Reactor (MSR)

There are two main types of Molten Salt Reactors (MSRs): salt-cooled reactors that use solid fuel with a molten-salt coolant, and liquid-fueled reactors in which the fissile material is dissolved directly in the molten salt. In solid-fueled designs, uranium fuel in TRISO particles (formed into pebbles or prismatic blocks) is common. In both cases, the molten salt, typically fluoride-based for thermal-spectrum concepts or chloride-based for fast-spectrum concepts, transports heat to a heat exchanger.

Thermal-spectrum MSR designs typically use graphite as the neutron moderator, and improved graphite nuclear data are needed, consistent with the needs identified for High

Temperature Reactors. Key gaps include graphite reaction data (for example neutron capture) and thermal scattering law data. Data are also needed for reactions on graphite impurities, including (n,γ), (n,p), and (n,α), which can produce activation and fission products such as ^{152}Eu, ^{60}Co, and ^{137}Cs.

Thermal systems generally use fluoride salts, for example LiF-BeF$_2$-UF$_4$ or NaF-RbF-UF$_4$ for liquid-fuel/coolant mixtures, and FLiBe as a coolant. Related nuclear data needs include improved reaction data for ^{19}F, particularly neutron inelastic scattering, where evaluated libraries differ. Thermal scattering law data are also needed for relevant fluoride compounds, including temperature dependence and associated uncertainties. In addition, data are needed for reactions on light elements, notably ^{7}Li neutron capture and ^{6}Li reactions that produce ^{3}H.

Fast-spectrum MSR concepts typically use molten chloride salts, for example UCl$_3$-NaCl or PuCl$_3$-NaCl, often combined with NaCl-KCl-MgCl$_2$ based mixtures to tailor melting temperature and support chemistry control. These reactors are generally liquid-fueled, with the actinide chlorides dissolved directly in the circulating salt. Some reactors use a reflector, such as MgO, where improved neutron data for ^{24}Mg are needed. Other key nuclear data needs include neutron reaction data for Cl and Na, particularly radiative capture and (n,p) channels. Improved neutron scattering data, including differential (angular) distributions, are also required, along with well-characterized uncertainties across all relevant reactions.

Molten fuel-salt systems pose significant challenges for nonproliferation safeguards. Unlike solid-fueled reactors, the nuclear material is not sealed in discrete fuel assemblies, it is mobile, distributed throughout the plant, and can change chemical form and isotopic composition during operation. As a result, improved predictive and measurement capabilities that couple nuclear behavior, salt chemistry, and materials properties are needed to support accurate material accountancy and inventory estimates. The operating environment, including high radiation fields, elevated temperatures, and corrosive salts, further complicates instrumentation, sampling, and verification. Improved nuclear data are needed for short-lived fission products, as well as for the production of gaseous species (for example ^{135m}Xe) that can be used for monitoring and diagnostics. Data are also needed for neutron-producing reactions, such as (α,n) on light isotopes, which are important for non-destructive assay and related analyses.

Representative references:

F. Bostelmann, G. Ilas, C. Celik, A. M. Holcomb, W. A. Wieselquist, *Nuclear Data Assessment for Advanced Reactors*, NUREG/CR-7289, ORNL/TM-2021/2002 (2022).

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N. Satvat (Kairos Power), *Nuclear data for Kairos Power's Fluoride-salt cooled High Temperature Reactor (KP-FHR),* in Workshop for Applied Nuclear Data Activities (WANDA 2021), Jan. 25–Feb. 3, 2021.

M. P. Dion (ORNL), *Nuclear Data for Nondestructive Assay and Advanced Reactor Applications,* in in Workshop for Applied Nuclear Data Activities (WANDA 2021), Jan. 25–Feb. 3, 2021.

T. Drzewiecki (US NRC), *Nuclear data uncertainties and its impact on design margins and advanced reactor licensing*, in Workshop for Applied Nuclear Data Activities (WANDA 2022), Feb. 28–Mar. 4, 2022.

C. Campbell (Kairos Power*) Bridging Gaps: Nuclear data needs for optimized deployment of KP-FHR reactors,* in Workshop for Applied Nuclear Data Activities (WANDA 2025), Feb. 10–Feb. 13, 2025.

Sodium Fast Reactor (SFR) and Liquid Metal-Cooled Fast Reactor (LFR)

Liquid Metal-Cooled Fast Reactors (LMFRs), including Sodium-cooled Fast Reactors (SFRs) and Lead- or Lead-Bismuth-cooled Fast Reactors (LFRs), demand a significantly different set of nuclear data compared to thermal systems. Because these reactors operate with a high-energy neutron spectrum, the focus shifts from thermal scattering laws to accurate cross-sections in the fast energy range, typically from the keV region up to several MeV. The most pressing need usually centers on the inelastic scattering cross-sections of structural materials and coolants, which dominate the neutron energy loss mechanisms in the absence of a traditional moderator like water or graphite.

For SFRs, the precise characterization of sodium nuclear data is paramount, particularly regarding the sodium void reactivity coefficient. This safety parameter is heavily influenced by the resonance structure of ^{23}Na. Inaccuracies in the capture and elastic scattering cross-sections of sodium, especially around the major resonance peaks, can lead to significant uncertainties in predicting whether the reactor reactivity will increase or decrease if the coolant boils or is lost. Additionally, knowing the precise inelastic scattering cross-sections for sodium is crucial for determining the neutron spectrum hardness, which in turn dictates the breeding ratio and the efficiency of actinide burning.

In the case of LFRs, the data needs center on the scattering properties of ^{204}Pb, ^{206}Pb, ^{207}Pb, ^{208}Pb, as well as ^{209}Bi. Lead has very low absorption but acts as a spectral shifter through inelastic scattering. Uncertainties in Pb inelastic scattering data propagate into errors in critical mass calculations and control rod worth predictions. For Lead-Bismuth-cooled systems specifically, there is a distinct radiological safety need for accurate capture cross-sections of ^{209}Bi leading to the formation of ^{210}Bi, which subsequently decays into the highly radiotoxic and volatile ^{210}Po. Reducing the uncertainty in this production chain is essential for designing shielding and maintenance protocols.

Structural materials play a much more active role in the neutronics of fast reactors than in thermal reactors. Because the core is tightly packed and lacks a moderator, the steel cladding and wrapper materials (containing ^{56}Fe, ^{52}Cr, and ^{58}Ni) constitute a significant portion of the scattering medium. Consequently, inelastic scattering cross-sections on ^{56}Fe are with low uncertainties, including covariances, are needed for the fast reactor community. Errors here directly affect the neutron leakage and the overall neutron economy. Furthermore, because the neutron flux is highly energetic, threshold reactions such as (n,p) and (n,α) in steel components become prevalent. Accurate data for these gas-producing reactions is vital for predicting material swelling, embrittlement, and the ultimate lifetime of the fuel assemblies.

Fuel cycle physics in fast reactors also necessitates improved data for actinides, specifically regarding inelastic scattering and fission cross-sections in the unresolved resonance and fast energy regions. ^{238}U inelastic scattering is a major contributor to spectral softening; uncertainties here can skew predictions of the Doppler feedback coefficient, a key safety mechanism. For transmutation missions, where fast reactors are used to burn nuclear waste, better data is required for the fission and capture cross-sections of minor actinides like ^{237}Np, ^{241}Am, and ^{243}Am in the MeV range. If the capture-to-fission ratio is not well known, it is difficult to validate the transmutation efficiency of the core design.

Finally, the kinetics of fast reactors rely heavily on accurate delayed neutron data. Since fast reactors utilize plutonium or minor actinide-bearing fuels which have lower delayed neutron fractions (β_{eff}) than uranium, the margin for error is smaller. Precise data on delayed neutron yields and the decay constants of precursor groups for high-energy fission of ^{239}Pu, ^{240}Pu, and ^{238}U are critical for transient analysis and designing reliable control systems. This ensures that the reactor can be safely controlled despite the shorter prompt neutron lifetime characteristic of fast systems.

Representative references:

F. Bostelmann, G. Ilas, C. Celik, A. M. Holcomb, and W. A. Wieselquist, “Nuclear Data Assessment for Advanced Reactors,” NUREG/CR-7289, Oak Ridge National Laboratory; U.S. Nuclear Regulatory Commission (2022).

F. Bostelmann, G. Ilas, and W. A. Wieselquist, “Key Nuclear Data Impacting Reactivity in Advanced Reactors,” ORNL/TM-2020/1557, Oak Ridge National Laboratory, Oak Ridge, TN, (2020).

F. Bostelmann, A. M. Holcomb, J. B. Clarity, W. J. Marshall, V. Sobes, and B. T. Rearden, Nuclear Data Performance Assessment for Advanced Reactors, ORNL/TM-2018/1033, Oak Ridge National Laboratory, Oak Ridge, TN, (2019).

J. M. Brown, E. O’Brien, T. Bredeweg, J. Jo Ressler, A. Couture, M. Loughlin, P. Romano, L. Bernstein, T. Cutler, J. Ortensi, A. Daskalakis, D. A. Brown, and R. Vogt, Proceedings for the Workshop on Applied Nuclear Data Activities 2025, ORNL/TM-2025/3966, Oak Ridge National Laboratory, Oak Ridge, TN, (2025).

Thorium-based reactors

Nuclear reactors operating on the thorium fuel cycle generate energy primarily through the fission of ^{233}U, in contrast to conventional reactors which rely on ^{235}U. In thorium cycle-based reactors, ^{233}U is produced from ^{232}Th via neutron capture to ^{233}Th, followed by its beta decay to ^{233}Pa which subsequently decays to ^{233}U. The thorium cycle offers several notable advantages: thorium is more abundant in nature than uranium, and thorium-based reactors tend to produce lower quantities of long-lived radioactive waste, thereby presenting potential benefits for sustainable fuel utilization and waste management. Several reactor designs have been proposed or are under development to exploit the thorium fuel cycle.

Accurate reactor physics modeling and reliable performance predictions for thorium-cycle reactors depend on accurate nuclear data. Of particular importance are the neutron capture cross sections of ^{232}Th and ^{233}Pa, as well as the fission cross section of ^{233}U. Fission yields for ^{233}U remain significantly less studied than those of ^{235}U, resulting in larger uncertainties that can affect reactor design, safety analyses, and fuel cycle predictions. Consequently, improved measurements of both fission yields and (α,n) yields for ^{233}U are essential. In addition, for non-destructive assay of ^{233}U-bearing materials, precise data on gamma-ray energies, emission probabilities, half-lives, decay branching ratios, and mass attenuation coefficients are required to ensure accurate material characterization.

For fast-spectrum thorium reactors, further challenges arise due to discrepancies between experimental measurements and the values reported in current nuclear data libraries for the ^{232}Th fission cross section, particularly at neutron energies exceeding 1 MeV. Addressing these discrepancies is critical to achieving reliable reactor performance predictions and to supporting the safe, efficient operation of thorium-based fast reactors.

Representative references:

H. Unesaki, K. Kobayashi, and S. Shiroya, "Assessment of ^{232}Th nuclear data through analysis of thorium-loaded critical experiments in thermal-neutron systems using the Kyoto University Critical Assembly," J. Nucl. Sci. Technol., 38, 6, pp. 370–378, 2021.

M. P. Dion, M. Croce, S. Croft, M. S. Smith, R. Venkataraman, and L. Worrall (ORNL), *Nuclear Data for Nondestructive Assay and Advanced Reactor Applications*, in Workshop for Applied Nuclear Data Activities (WANDA 2021), Jan. 25–Feb. 3, 2021.

Accelerator-driven systems

Accelerator-Driven Subcritical Reactors (ADSRs), more commonly referred to as Accelerator-Driven Systems (ADS), represent an advanced nuclear energy concept in which a subcritical reactor core is coupled to a high-energy particle accelerator. In this configuration, the fission chain reaction cannot sustain itself and instead depends on an externally supplied neutron source generated by the accelerator. This external source provides a high degree of operational control and enhances inherent safety characteristics. ADS concepts have been proposed for electricity generation while simultaneously enabling the transmutation of long-lived nuclear waste or the utilization of thorium-based fuel cycles.

One of the principal advantages of ADS technology is its flexibility in fuel and coolant selection. Because the system remains subcritical and relies on an external neutron source, it can accommodate fuels and coolants that would present reduced intrinsic safety margins in conventional critical reactors, such as fuels with high minor actinide (MA) content. The presence of the accelerator may allow simplification or replacement of certain traditional safety systems and provides a broader operational range to compensate for reactivity loss during burn-up. In particular, ADS has been identified as a potentially unique solution for dedicated minor actinide (and plutonium) transmutation systems.

Neutron production in ADS is typically achieved through spallation reactions induced by high-energy protons incident on a heavy target. Optimal neutron yields occur for proton energies in the range of approximately 800 MeV to 1 GeV. The neutron production rate peaks once the incident proton "punches through” the target nucleus and initiates a hadronic cascade process (Russell, 1990). Current and planned facilities illustrate a range of accelerator energies: MYRRHA employs 600 MeV protons, CiADS utilizes 500 MeV protons, and the proposed JAEA-ADS design considered proton energies up to 1.5 GeV (Yee-Rendon, 2022). Consequently, reliable high-energy proton reaction data are required for the accelerator driver, along with accurate data for the secondary high-energy neutrons produced in the spallation process.

Existing evaluated nuclear data libraries such as ENDF provide neutron and proton cross sections up to 150 MeV, near the pion production threshold. In the energy range 20-150 MeV, the available extensions are largely model-based evaluations developed in the late 1990s and require modernization. These high-energy evaluations lack associated covariance data, limiting their utility for uncertainty quantification. Updated high-energy cross sections and detailed emitted-particle distributions are particularly important for spallation target materials. Depending on the ADS concept, these materials may include refractory metals such as tungsten or liquid-metal systems that serve both as coolant and spallation target, as in the MYRRHA design, which employs lead–bismuth eutectic.

Accurate inventory calculations for transuranic (TRU) elements and fission products (FP) during burn-up present a substantial challenge. For many isotopes of interest, neutron capture and fission cross sections remain unmeasured or poorly constrained. Non-destructive assay (NDA) requirements depend strongly on the specific fuel composition, and for many TRU isotopes essential decay data are incomplete, including gamma-ray energies and emission probabilities, half-lives, decay branching ratios, and mass attenuation coefficients.

In the specific case of MYRRHA, which is intended both as an isotope production facility and as a test bed for fusion materials research, improved nuclear data and reduced uncertainties are required across a broad range of reactions (Baeten, 2019). Priority needs include neutron capture and scatter reactions on ^{239}Pu, ^{235}U, ^{238}U, ^{209}Bi, and ^{56}Fe in the resonance and fast energy regions. Fission data, such as (n,f) cross sections, prompt fission neutron spectra (PFNS), and neutron multiplicities are also needed for ^{238}Pu through ^{242}Pu.

Additional data needs include proton-induced reactions relevant for shielding and target design, such as p + ^{nat}C (e.g., ISOL targets), p + Al, and p + W (fusion applications); transmutation pathways involving ^{206}Tl, lead isotopes, and bismuth, particularly in relation to ^{210}Po production; decay heat data; and gas and other volatile production yields.

Representative references:

Baeten, P. (2019). Nuclear *data needs for MYRRHA and MINERVA*. OECD/NEA, June 6, 2019.

Gonzalez, E. M. (2004). *European Physics Society Nuclear Physics Board*, Valencia, May 1, 2004.

Yee-Rendon, B (2022). Proceedings of LINAC2022. DOI: 10.18429/JACoW-LINAC2022-TU2AA01

Russell, G. J. (1990). *1CANS-XI International Collaboration on Advanced Neutron Sources*, KEK, Tsukuba, October 22–26, 1990.

M.B. Chadwick, et al. (2001), "Nuclear data for accelerator-driven systems", Progress in Nuclear Energy, Volume 38, Issues 1–2, Pages 179-219. https://doi.org/10.1016/S0149-1970(00)00102-5.

C. Spent fuel, transmutation, and reprocessing

Future advanced-reactor spent fuel streams can differ substantially from current LWR used fuel. While some designs discharge conventional spent fuel assemblies, others produce particulate fuels (for example TRISO), metallic fuels, or separated waste streams (as in some MSR concepts). In addition, fast-spectrum operation and higher burnup can shift fission product and actinide inventories, which can challenge existing monitoring and diagnostic technologies. Transmutation and reprocessing can reduce long-lived radiotoxicity and repository heat loads and improve fuel utilization. However, significant challenges remain in technical maturity, policy and regulatory frameworks, and nonproliferation safeguards.

Common themes for nuclear data needs for spent fuel include:

- (α,n) data on light elements
- Activation and decay data
- Neutron induced reactions on actinides, notably minor actinides
- Fission product yield data, including branching ratios and uncertainties; neutron capture reactions on fission products; uncertainties and correlations on delayed neutron fractions and decay constants
- Predictive x-ray, γ-ray, and neutron emission from spent fuel forms
- Benchmarks for minor actinides in fast neutron spectra, depletion/burnup, isotopic inventories and decay heat
- Uncertainties in reaction, decay, and fission data

Spent fuel storage

Storing spent fuel is often the simplest, most mature, and lowest near-term risk option. In contrast, “burning” (transmutation) or reprocessing typically requires substantial new infrastructure, higher cost, and added safeguards complexity. Spent fuel also provides a strong intrinsic radiation barrier; it is extremely difficult to access, handle, or steal without heavy shielding, remote handling equipment, and specialized facilities. As a result, intact spent fuel is “self-protecting” to a degree; it cannot be safely manipulated with simple tools, and any attempt to move or process it is more likely to be detectable and resource intensive.

Nonproliferation safeguards face two coupled challenges: accurately predicting the fuel’s isotopic inventory and reliably interpreting measured radiation signatures. In many advanced reactor concepts, fast neutron spectra, higher burnup, multi-recycle operation, and in some cases online refueling shift actinide compositions and fission product inventories, pushing existing predictive tools beyond their validated range. Addressing this requires high-quality fast-energy neutron reaction data, especially capture and fission cross sections (with uncertainties) for key actinides such as Np, Pu, and Am, along with energy-dependent fission product yield data. Robust non-destructive assay and material attribution also depend on accurate neutron source terms from spontaneous fission and (α,n), reactions, decay schemes, and gamma emission data. These needs become even more acute for online monitoring or circulating-fuel systems, where material may reside in salts, off-gas streams, filters, or process hold-up rather than in discrete fuel assemblies.

Benchmark data are needed to validate fission yield predictions and isotopic inventories for advanced-reactor neutron spectra and burnup periods. Shielding and measurement benchmarks for gamma dose rates and neutron fields are also essential, since alternative fuel forms and geometries can complicate radiation transport and signature interpretation.

Representative references:

W. Wieselquist (ORNL), *Reactor Physics Nuclear Data Needs,* in Workshop for Applied Nuclear Data Activities (WANDA 2020), Mar. 3–Mar. 5, 2020.

K. Koop Hogue (ORNL), *Multi-Physics Code Prediction of Nuclear Material Quantities for Advanced Reactor Safeguards,* in Workshop for Applied Nuclear Data Activities (WANDA 2025), Feb. 10–Feb. 13, 2025.

Transmutation

Nuclear waste transmutation aims to convert long-lived radioisotopes in spent nuclear fuel into shorter-lived or stable isotopes, facilitating safer and more manageable disposal. For instance, spent fuel from a once-through fuel cycle contains isotopes such as ^{239}Pu which has a half-life exceeding 24,000 years. The transmutation process typically involves the chemical separation of plutonium and minor actinides such as neptunium, americium, and curium from the spent fuel. These separated actinides are subsequently irradiated in fast-spectrum reactors or accelerator-driven systems where these isotopes are depleted.

Despite its promise, transmutation introduces significant safeguards and security challenges. The separation process, both chemical and pyroprocessing, create potential pathways for diversion of nuclear materials. Monitoring and inspection of transmutation facilities are complicated by the high radioactivity and chemical hazards of the processed materials, placing stringent demands on safeguards instrumentation and inspection. Additionally, the presence of separated actinides can hinder non-destructive assay (NDA) techniques traditionally employed to characterize samples, complicating efforts to verify nuclear material inventories.

Representative reference:

E. Buhmann and G. Kirchner, "Proliferation relevance and safeguards implications of partitioning and transmutation nuclear fuel cycles," Science & Global Security, vol. 26, pp. 91–108, 2018.

Reprocessing

Nuclear reprocessing chemically treats spent nuclear fuel to recover reusable materials and reduce the volume of waste requiring high level disposal. It can separate and recycle actinides, such as uranium and plutonium, for fabrication into new reactor fuels, including mixed oxide (MOX) fuel. However, processes that separate plutonium introduce proliferation and security concerns, requiring stringent safeguards, physical protection, and international oversight.

Nonproliferation safeguards activities in reprocessing facilities require accurate nuclear data for uranium (^{234}U to ^{238}U) and plutonium (^{238}Pu to ^{242}Pu) isotopics, along with minor actinides (Np, Am, Cm) and key fission products, to support reliable nuclear material accountancy. Radioactive decay data, including half-lives and branching information, are needed to generate time-dependent inventories and to predict how measured signatures change with cooling time. Data on gamma and neutron emissions from radioactive decay, including neutrons from spontaneous fission and (α,n) reactions, are also needed to inform shielding design and to select and interpret detectors for process monitoring. Finally, fission product yield data are required to estimate the nuclides that dominate decay heat, dose rates, chemical behavior, in addition to some monitoring signatures.

Representative references:

IAEA, “Safeguards Techniques and Equipment: 2011 Edition” (International Nuclear Verification Series, No. 1)

IAEA Department of Safeguards, “Nuclear Material Accounting Handbook”

III. Conclusions

This report identifies and prioritizes nuclear data gaps that limit nonproliferation and safeguards capabilities for advanced reactor designs, drawing on needs articulated in recent workshops, conferences, and publications. It synthesizes these gaps into six cross-cutting research themes for targeted investment, emphasizing areas of highest-impact on predictive modeling and measurements where novel materials, high temperatures, and fast or mixed spectra drive new activation pathways, different isotopic yields, and altered signature evolution.

1. **Activation/inventory and decay data** for structural materials, coolants, salts, reflectors, and impurities, to support signature prediction and monitoring in environments with nontraditional materials and spectra.
2. **Fission characteristics for minor actinides**, including fast-energy fission cross sections, prompt and delayed neutron production, fission product yields, and

gamma production, reflecting increased sensitivity to minor actinide content in fast reactors and recycled fuels.

3. **Neutron scattering data** for fuels, structural materials, and coolants, including energy- and angle-dependent information and inelastic level data, to improve spectrum prediction that drives both reactor physics and safeguards-relevant instrument response.
4. **Light-element nuclear data** (Li, Be, C, F, N), including thermal scattering laws and temperature dependence, because these nuclides often dominate moderation, absorption, and leakage in advanced materials, salts, ceramics, and moderators. Carbon is notable for both fuel (e.g. TRISO) and moderator (e.g. graphite) needs, and efforts to incorporate and understand data have been undertaken by the Cross Section Evaluation Working Group (CSEWG, a collaboration across national laboratories, universities, and industries)[1].
5. **Improved uncertainty coverage and quality**, emphasizing covariances for reaction, decay, and fission emission data, enabling sensitivity studies, uncertainty-aware safeguards assessments, and regulator-relevant confidence statements.
6. **Benchmarks beyond criticality**, prioritizing integral data for depletion and inventories, time-dependent neutron and gamma emissions, activation products, shielding, and detector response, to validate end-to-end predictive capability used in nonproliferation and safeguards applications.

A key outcome is that these priorities map naturally onto shared mission needs and capabilities across multiple offices, creating strong opportunities for coordinated investment and execution. The six themes provide a structured basis for collaboration with partners across NA-20, NA-80, NA-30, DOE Office of Science (SC), Nuclear Energy (NE), and NA-11. Coordinated programs can align differential measurements, evaluations and covariance development, library improvements, and integral benchmark campaigns with the most safeguards-relevant use cases, reducing duplication while accelerating delivery of fit-for-purpose data. Collectively, these investments will strengthen the technical foundations for monitoring, verification, and attribution in advanced reactor fuel cycles, and improve confidence in both predictive assessments and the interpretation of measured signatures across emerging reactor deployments.

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[1] https://www.osti.gov/biblio/2998877