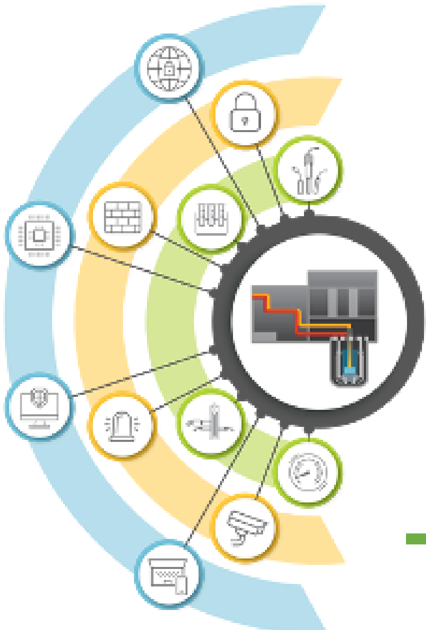

Advanced Reactor Safeguards & Security

Nuclear Material Control & Accounting for Pebble Bed Reactors

Prepared for
US Department of Energy

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Oak Ridge National Laboratory

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Nuclear Nonproliferation Division

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ACRONYMS

ANOVA	analysis of variance
ANSI	American National Standard Institute
BUMS	Burnup Measurement System
CFR	Code of Federal Regulations
COL	combined license
DA	destructive analysis
DOE	Department of Energy
Ekg	Effective Killogram
GADRAS	Gamma Detector Response and Analysis Software
HALEU	high assay low enriched uranium
HPGe	high-purity germanium
IAEA	International Atomic Energy Agency
ICA	item control area
ISO	International Organization for Standardization
KMP	key measurement point
LWR	light-water reactor
MBA	material balance area
MC&A	material control and accounting
MUF	material unaccounted for
NDA	nondestructive assay
NMMSS	Nuclear Materials Management and Safeguards System
NRC	Nuclear Regulatory Commission
ORIGAMI	ORIGEN Assembly Isotopics
ORIGEN	Oak Ridge Isotope GENeration
ORNL	Oak Ridge National Laboratory
PBMR	pebble bed modular reactor
PBR	pebble bed reactor
PEEK	PolyEtherEtherKetone
RIS	reporting identification symbol
SCALE	Standardized Computer Analyses for Licensing Evaluation
SNM	special nuclear material
TID	tamper-indicating device
TRISO	tristructural isotropic
TRITON	Transport Rigor Implemented with Time-dependent Operation for Neutronic depletion
VSOP	Very Superior Old Programs

EXECUTIVE SUMMARY

This report discusses the work done under the US Department of Energy NE-5 Advanced Reactor Safeguards and Security Program during FY 2023. It provides a summary of material control and accounting (MC&A) for pebble bed reactors (PBRs) and addresses some of the main challenges with current PBR MC&A approaches that will inform safeguards and security by design efforts. The efforts to date have focused on tristructural isotropic (TRISO) pebble fuel material accounting and control including working with partners in industry, loss and production of nuclear material as part of reactor operations, burnup modeling and measurements, uncertainty quantifications for such modeling and measurements, statistical approaches needed, and measurement methods. The unique fuel management and utilization in a PBR, where the fuel in spherical form is introduced and circulates through the reactor, poses special challenges for MC&A. This contrasts with traditional water-cooled reactors in which the fuel is contained in large assemblies and can be easily identified and counted. Even online fueled reactors, such as the CANDU reactors (none of which operate in the United States), are significantly different because the fuel is still contained in relatively large assemblies, is uniquely identified, and the number of assemblies that pass through the core on an annual basis is much fewer than the hundreds of thousands that circulate in a PBR, none of which are uniquely identified. Additionally, the nature of the TRISO fuel results in very low heavy metal loading with each pebble containing less than 10 g of uranium and on the order of less than 1 g of fissile material. This low fuel density and the robustness of the TRISO particles are major features of the TRISO fuel from a safety basis as each TRISO particle and pebble acts as a containment for the nuclear material and fission products during normal and accident conditions. This also results in very low plutonium loading per pebble during normal operations, which is on the order of 0.1 g at full burnup. A major feature of PBRs is that they will allow for significantly higher burnup, on the order of 160 GWd/THM compared to the burnup of traditional LWRs, which is on the order of 45 GWd/THM. This is achieved by monitoring the pebbles as they circulate through the reactor and allowing them to be reintroduced into the core until the desired burnup is achieved and they are removed from the reactor and enter the spent fuel storage areas.

These features and aspects of PBRs require unique approaches to material accounting and control that provides assurance that the material is accounted for at all stages of operation and controlled to prevent the theft or diversion by internal or external adversaries. Modeling approaches for PBRs provide a detailed understanding of the isotopic content of the fuel as it circulates through the reactor, but this should be validated by actual measurements of used TRISO fuel. Although used fuel is difficult to accurately measure due to the presence of fission products and the high radiation environment and other physical and operational constraints, the ability to model and measure each pebble allows for much greater accuracy than in traditional water-cooled reactors and the low heavy metal loading means that accumulating large amounts of nuclear material is very difficult for an adversary to achieve. The theft of individual pebbles is of course still a concern from a security perspective, and the MC&A must address this as part of US domestic licensing under the US Nuclear Regulatory Commission (NRC).

The MC&A of nuclear material for PBRs will differ from that used in the current fleet of light water reactors in the United States. In some ways, there are parallels to how bulk nuclear material is accounted for in fuel cycle facilities. However, the fuel pebbles are discrete objects that are easily detected, can be visually observed, and can be mechanically counted and controlled. The mass of the containers and the fuel pebbles they contain fits within the batch and piece part accounting structure commonly used. The ability to measure the burnup of individual pebbles provides a powerful tool that is not used in current LWRs. This provides validation of reactor models and thus the loss and production of nuclear material is theoretically more accurate than is possible in an LWR. This report addressed the following areas for MC&A for PBRs:

- The NRC licensing basis for PBRs

- Current industry partnerships that are helping to inform challenges and potential solutions for applying MC&A to PBRs
- Description of fuel flows in a typical PBR
- Containerization methods used for the transport and loading of fresh fuel pebbles and retirement of spent fuel pebbles
- Development of PBR reactor fuel flow models to determine the movement of fuel inside the reactor core and to calculate the burnup of individual pebbles based on their irradiation histories
- Modeling of potential gamma signatures that could be used to facilitate nondestructive measurements of irradiated fuel pebbles
- Determination of possible neutron-based nondestructive assay (NDA) systems that could measure irradiated fuel pebbles and withstand the harsh environments near the reactor systems
- Statistical approaches needed to address the unique accounting challenges of millions of mobile fuel pebbles with low densities of nuclear material
- Analysis of the features of potential inventory management systems that could be used in a PBR

Two PBR designs that are active in the US are the X-energy XE-100 helium gas cooled and the Kairos Power molten salt PBRs. Interactions with both companies have provided a context for considering MC&A during the design phase and plans for deployment. Although proprietary data and intellectual property were not included, the questions that were asked and issues addressed informed the topics and results of this report.

Recommendation 1: Continued engagement between the national laboratories and PBR designers is necessary to put into practice the results and recommendations of this report.

The licensing basis for PBRs is not entirely clear at this moment. Based on previous reports, it is not certain that the NRC will accept an MC&A program based solely on Regulatory Guide 5.29 and ANSI Standard N15.8-2009. This is because of the inability to rely on pure item counting as is done for the current licensed fleet of LWRs as well as the push to higher enrichments that will classify them as Category II facilities. There are currently no licensed Category II LWRs. Category II facilities are those that use uranium enrichments equal to or greater than 10% but less than 20%. The exceptions to the other requirements of 10 CFR Part 74 that are automatically included in the licensing process of 10 CFR Part 50 or exclusions that are granted as part of 10 CFR Part 52 may not fully apply to PBRs. In this case portions of NUREG 2159—Acceptable Standard Format and Content for the Material Control and Accounting Plan Required for Special Nuclear Material of Moderate Strategic Significance—may apply. Category II licensed facilities have tighter material and measurement controls and reporting thresholds, but these are based on fuel cycle facilities that process true bulk material in powders or solutions. Since fuel pebbles are distinct pieces, they do not fit the classification as true bulk material.

Recommendation 2: PBR designers should engage in early, pre-licensing discussions with the NRC to determine an acceptable format for the MC&A plan as part of the overall MC&A program.

The structure of the material balance areas and key measurement points for a generic PBR is straightforward. The fresh fuel is received in containers that are sealed at the fuel fabrication facility. The shipper values will likely be accepted because remeasurement would be costly. The concept of reporting in batches of fresh fuel drums with the individual pebble count listed as "pieces" within the batch are accepted MC&A reporting practices to the NRC. The same is true for the spent fuel canisters. For spent fuel containers, the dose rate at 1 m will initially be higher than for a typical LWR fuel assembly but will decay more rapidly. This is because of the higher burnups that will be achieved resulting in more fission product loading per unit volume but with lower overall actinide content of TRISO.

Recommendation 3: Designers should consider how fresh and spent fuel will be accounted for and reported when developing MC&A programs, and designers should employ containerization whenever possible in the design and in nuclear material fuel flows.

Recommendation 4: The self-protecting nature of spent TRISO fuel should be considered when developing the theft and diversion scenarios and as part of the MC&A and physical protection planning.

To adequately account for the production and loss of nuclear material inside the reactor, new models were developed to simulate the actual pathways that individual fuel pebbles take in the reactor system. The continuous on load feature combined with possible pathways through the reactor must be captured in the modeling approach. The neutron flux that a fuel pebble will experience in the reactor core will vary based on radial and axial positions as well as the aggregate fuel pebbles themselves and the operation of the reactor. This in turn affects the burnup per pass and how many passes a fuel pebble will experience based on the BUMS measurements. The design chosen to model was the PBMR-400 design. It adequately represents PBR designs, and the data is public and readily available. Based on this modeling the average and maximum total uranium loss and plutonium production were determined. Based on an initial loading of approximately 9 g total uranium at an enrichment of 9.6%, the total plutonium mass was calculated as approximately 0.1242 g per pebble, on average, after reaching a target burnup of 90 GWd/tHM.[1] The total plutonium in a spent fuel cannister holding 2,000 retired pebbles was 248.392 ± 0.26 g. The relatively low uranium and plutonium content is an inherent feature of TRISO-based fuels. The low uranium and plutonium content requires the diversion or theft of hundreds or thousands of pebbles to result in a significant amount of nuclear material. However, from a security perspective, the theft of only a small number of pebbles is a major concern because of the threat from a dirty bomb or irradiation device that can cause significant harm to workers or the public.

Recommendation 5: Models should be developed for each design to adequately represent the production and loss of nuclear material based on the specific features and operations of the reactor.

A central component of PBR operations is the Burnup Measurement System (BUMS). BUMS determines when a fuel pebble is either reinserted into the reactor or retired as spent nuclear fuel. Determining the burnup of each irradiated fuel pebble requires measuring gamma signatures that represent the burnup level reached. Measuring short-cooled nuclear fuel is very challenging because of the emanations of high-energy photons from many different fission products. Designing BUMS requires adequate considerations of the conditions where the instrument will be placed, the necessary shielding and collimation, and the geometry of the gamma measurement system. One main challenge is determining what fission products can be measured and which can provide a representation of the fuel pebble's burnup. Output from the reactor modeling was input into a GADRAS model to determine that measuring the absolute quantity of ^{137}Cs in each pebble should be possible, which is a good indicator of the burnup. Cesium-137 is a reliable burnup indicator because the fission yields of ^{137}Cs from ^{235}U and ^{239}Pu, the two primary fissioning nuclides in PBRs, are nearly identical. This assumes that the gamma instrument (e.g., a high-purity germanium detector) can be calibrated properly to determine its absolute efficiency and the geometry of the measurement system is controlled.

Recommendation 6: Work should continue to develop gamma measurement systems in collaboration with the national laboratories, vendors, and measurement equipment manufacturers.

[1] The amount of energy extracted from nuclear fuel is commonly expressed as gigawatt day per metric ton of heavy metal present in the fuel at the beginning of irradiation (GWd/tHM).

Passive neutron measurements were also considered to measure the burnup of individual pebbles. Neutron NDA techniques could replace or augment gamma measurements to determine a pebble's burnup value. Neutron measurements can potentially provide a more sensitive measurement of the burnup than the gamma measurements can because the neutron emission of an irradiated pebble is a power function of burnup, whereas the photon emission is usually a linear function of burnup. Additionally, the coincidence neutron signal of an irradiated pebble can be explored because the coincidence neutron signal is usually linear with the fissile content of the spent fuel, with the caveat that the coincidence neutron signal from a pebble's fissile content is likely to be small because of the small fissile content in a pebble; therefore, the usefulness of coincidence neutron signal for a pebble remains to be proven. One challenge of neutron measurement next to a PBR is the environment is a higher temperature than a conventional light water reactor. The commonly used neutron moderators in neutron instruments, such as high-density polyethylene, would not withstand much greater than 104 °C [1]. An alternate neutron-moderating material, PolyEtherEtherKetone (PEEK), is a semicrystalline thermoplastic with excellent strength and ductility, has good neutron-moderating power, and is suitable for continuous use at temperatures up to 260 °C [2]. This new moderating material is promising and could allow the use of passive neutron NDA techniques to be used next to a PBR as part of BUMS.

Recommendation 7: Passive neutron detectors should be explored to see if they can either perform better than gamma detectors or if they can complement gamma detectors for MC&A purposes. They may also be considered as confirmatory measurements for retired pebbles as they exit the reactor system before being placed in spent pebble storage canisters.

Accounting for the nuclear material in a PBR will include several millions of pebbles that will be received, circulate in the reactor, and retired as spent fuel over the life of the reactor. This will require careful consideration of how the loss and production of nuclear material is measured and calculated and will require statistical approaches to adequately account for Type I and Type II errors. BUMS is an integral part of PBR operations and is required to fully utilize the fuel while preventing reinserting fuel that does not produce enough power, thereby degrading reactor performance. BUMS is also needed to prevent degradation of the fuel pebble due to damage from excessive neutron flux resulting from prolonged residence times inside the reactor. The combination of BUMS measurements and the reactor core modeling provides a powerful combination of tools that can reinforce each other. During early deployment, it should be assumed that fuel pebbles will require destructive analysis, NDA, or a combination of both to verify fuel performance, and this analysis can be used to precisely quantify the actinide content and can be compared to the measured and calculated values. Sampling of fuel pebbles for destructive and NDA should be based on a statistical approach to select a representative population. These results can be used to both calibrate the BUMS values and validate the reactor code predictions for burnup and special nuclear material content. Future refinements can be made to both approaches to result in a robust measurement system and reactor burnup models. The rate of sampling can then be decreased significantly or eliminated altogether.

Recommendation 8: Designers should consider how statistical sampling of spent fuel pebbles will accommodate BUMS and the reactor core burnup models.

Recommendation 9: Additionally, a comparison of reactor models, the BUMS, and statistically based destructive analysis should be performed to validate the models and improve the BUMS performance.

Inventory management systems will be required for every PBR to facilitate the MC&A program. Currently available MC&A systems will likely need to be modified or adapted to use in a PBR. The MC&A system will be required to electronically import/ export inventory information for both operations needs and regulatory reporting. They should also have the ability to interface with the applicable reactor codes or measurement systems for updates to fissile material from reactor operations.

Recommendation 10: Future PBR owners and operators consider the MC&A software systems that are currently available to determine which one most closely meets their business and operational needs. Some modifications or adaptations may be required for PBRs. These can be performed in-house or outsourced to the software system vendor or a third-party software developer.

Recommendation 11: If it is determined that the MC&A software will be developed in-house, adequate preparation and understanding of the functional and interface requirements will be needed. Designers should plan accordingly.

There is still work to be done in modeling the reactor core and fuel handling systems due to the complexity of operations. Pebbles with different enrichments are used during the startup and run-in operations as well as graphite pebbles (nonfuel pebbles). This complicates the modeling and accounting of the nuclear material inventories. The consistency of the fuel from the fuel fabrication facility is also a consideration and will affect both the receipt verification as well as reactor operations. Non-steady-state and off-normal conditions have not yet been modeled, including reactor maintenance where the used fuel is off-loaded from the reactor, stored, and then reintroduced to restart the reactor. Development of the measurement systems and approaches still requires work. The difficulty in accurately and quickly measuring the actinide content in highly irradiated fuel must still be overcome. Measurement systems using both gamma and neutrons are being considered that can withstand the challenging radiation and thermal environments as well as space constraints. The reliability and robustness of such measurement systems must be ensured to achieve operational and economic goals.

As such, all of these issues are being addressed by industry, with support from the national laboratories, and all indications are that they will be successful and will result in a robust MC&A system that meets domestic requirements.

1. PURPOSE AND SCOPE OF WORK

1.1 PURPOSE

This report summarizes the cumulative work done under the US Department of Energy NE-5 Advanced Reactor Safeguards and Security Program up through FY 2023 on the topic of nuclear material control and accounting (MC&A) for pebble bed reactors (PBRs). The purpose of this project is to support domestic US designers of advanced reactors for compliance with NRC domestic safeguards. This includes MC&A and physical protection. It should be noted that international (IAEA) safeguards are not the focus of this report, but the discussions and analysis of MC&A for PBRs may be useful for those considerations. There are several differences between how domestic safeguards will be applied to PBRs versus the current fleet of light water reactors (LWRs) licensed to operate in the United States. This work explores these differences, provides analyses and computations, and provides insights and recommendations to the designers on how they may incorporate some features in their design and/ or future operations to meet these challenges.

1.1.1 Scope and Structure of Report

There are key differences between how MC&A is currently applied to LWRs and how it will be applied to PBRs. Title 10 of the Code of Federal Regulations (10 CFR) Part 74 defines MC&A requirements for special nuclear material (SNM). MC&A requirements are defined based on the strategic significance of the SNM. Light water reactor (LWR) fuel used in US commercial nuclear reactors is <10% enriched in the isotope ^{235}U (i.e., Category III material of low strategic significance).[2] The proposed PBR designs plan on using fuel enriched to a level of 10% to less than 20%, which results in regulations that apply to SNM of moderate strategic significance (Category II material). Even though plutonium is in LWR spent nuclear fuel,[3] LWR fuel assemblies are large, heavy, and highly radioactive, which significantly decreases the likelihood of theft. As a result, the US Nuclear Regulatory Commission (NRC) requires LWRs to meet only the sabotage design-basis threat and not the theft or diversion design-basis threat because theft or diversion is bounded by controls to mitigate sabotage. Therefore, the NRC's MC&A regulations for LWRs are less stringent and do not require the full implementation of Category I and Category II MC&A, which is required for other fuel cycle facilities. As a result, LWRs can simply rely on adequate physical protection to prevent sabotage of the facility.

For PBRs, the portability of pebbles at certain points in the process is a key difference from conventional LWRs, though the SNM content per pebble is small. The main differences stem from the movable nature of the fuel spheres (i.e., pebbles) during normal reactor operations. Pebbles are continuously inserted into and withdrawn from the reactor core from the inventory of pebbles and are moved throughout the reactor and associated systems via pneumatic (or hydraulic) pressure tubes and moved by gravity or other (e.g., mechanical) means. In LWRs the fuel bundles are fixed during the operations cycle, and the reactor must be shut down and the reactor head removed to insert, remove, or shuffle the fuel. The several hundred fuel bundles at an LWR are uniquely identified, whereas the fuel pebbles used in proposed PBR designs are not. A PBR has an inventory that comprises hundreds of thousands of fuel pebbles and, during the operational lifetime, may encounter millions of pebbles that arrive as fresh fuel and ultimately are dispositioned as spent fuel. The number of pebbles and their portability constitute the key differences in MC&A and physical protection. The concept of item control or monitoring as defined in 10 CFR Part 74 is one of the key approaches to manage this large number of pebbles.

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[2] Ten thousand grams or more of ^{235}U enriched to less than 10% is defined as Category III SNM of low strategic significance [3].
[3] The presence of a formula quantity of plutonium would typically cause SNM to be categorized as Category I strategic SNM

Even so, certain aspects of current LWR approaches apply to fresh and spent PBR fuel. However, the MC&A approaches currently in use for fuel cycle facilities[4] more closely align with anticipated PBR designs instead of LWR approaches—specifically those MC&A approaches required for facilities that handle SNM of moderate strategic significance (i.e., Category II) for the reactor vessel and recycle loops.

The low SNM content per pebble must also be balanced with the radiological aspects that result from the theft of a single spent pebble. Considering only bulk amounts or groups does not address the potential consequence of losing an individual pebble if it is used in a radiological exposure device or in a radiological dispersion device. A spent TRISO (tristructural isotropic) pebble has radiation levels equivalent to a Category 1 or 2 radiological source [5].

Collaboration with the two main PBR designers, X-energy and Kairos Power, was crucial to understanding the current state of PBR designs and provided the opportunity to identify the key challenges that they face. Since PBR designers must address these challenges in developing their domestic licensing and MC&A strategies, several areas were identified and analyzed as part of the work done under this project. Below are the focus areas:

NRC licensing and implementation of an MC&A program

Identifying how the US NRC licensing process will affect MC&A is crucial for PBR designers to understand. As such, a comprehensive review was done on the existing licensing regulations and guidance and how this might affect future compliance with MC&A requirements.

Design of fuel pebble containers

Containers will have to be designed and certified to handle both fresh fuel pebbles as well as used and spent fuel pebbles at the scale required for commercial operation of the industry. Currently, this is a challenge for HALEU (high-assay low-enriched uranium) fuel. The fresh fuel product will have to be transported from the fuel fabrication facilities and stored until the fuel is loaded into the fuel-handling system. The spent fuel containers will have to function as interim storage and are typically expected to remain on-site for the entire period of reactor operations.

Material flows

Understanding how material will be introduced into the reactor facility, the design of the material balance areas (MBAs) and the key measurement points (KMPs) will inform the domestic (NRC) licensing and potential MC&A approach.

Reactor and systems modeling

Understanding the behavior of the fuel pebbles as they circulate through the reactor is required to support the various measurement systems that will be needed for reactor operations. Chief among these is the Burnup Measurement System (BUMS), which must determine the burnup of each fuel pebble that exits the reactor and is central to the operating principles of a PBR. Variabilities such as pebble transit times, heavy metal loading, and plutonium production all affect the actinide isotopics and fission product generation for each fuel pebble. This, in turn, affects the function of the MC&A system that is required to determine the loss and production of nuclear material.

[4] Examples of fuel cycle facilities are fresh fuel fabrication and spent fuel reprocessing plants.

Destructive and Nondestructive measurement systems

Nondestructive assay (NDA) measurements of fresh, used, and spent fuel will have to be performed as part of the MC&A system. Destructive analyses (DA) will likely also be performed, especially during early operations, and may serve as a complement to NDA. As mentioned earlier, BUMS is an integral part of the reactor design and operations and determining what kind of measurement systems could be employed is a major concern for PBR designers. Additionally, other measurements of spent fuel may be advantageous to characterize “spent” fuel pebbles that have reached the end of their useful life and are placed into containers for interim or long-term storage.

Statistical approaches

Although this is a subset of MC&A, statistical approaches will play such a large role that it is treated as a separate focus area. Many fuel pebbles (approximately millions) will pass through the reactor during its lifetime and measuring, modeling, and counting them will pose challenges. Additionally, TRISO fuel is inherently low density with low uranium and plutonium content per pebble. These two features will cause uncertainties in both counting, measuring, and modeling the fuel pebbles to increase. Statistics will play a large role in determining the setpoints of BUMS to ensure both safe operations as well as effective material accounting.

Inventory management and accounting systems

Each PBR operator will have to implement an inventory management MC&A system that will support the overall MC&A program. As such, a review was performed of some of the systems currently available on the market, the functionality of such systems, and their fitness for this purpose.

These focus areas will be addressed in detail in the following sections. Current PBR MC&A approaches are still evolving as vendors continue to work on plant layouts and designs for the fuel-handling systems. This allows for the possible consideration of domestic safeguards in the designs of these facilities, and it is hoped that this report will contribute to those efforts.

1.2 INDUSTRY PARTNERSHIPS AND NATIONAL LABORATORIES

The industry partnerships described in this report provide important examples of leading PBR designs, which are useful in considering possible MC&A approaches and offer detailed design information. The results from this report do not contain any designer’s proprietary information and can therefore be applied to any PBR design, within the given general reactor design characteristics and can be used by any vendors or designers considering MC&A approaches for PBRs.

1.2.1 Collaboration with X-energy

In October 2020, X-energy was selected as one of two recipients of the DOE’s Advanced Reactor Demonstration Program. The cooperative agreement between the DOE and X-energy was signed on March 1, 2021. As part of that agreement, Oak Ridge National Laboratory (ORNL) and Sandia National Laboratories are partnering with X-energy on domestic safeguards. The appropriate nondisclosure agreements were put in place in September 2022. The work packages, supplier agreements, and funding are also in place to support the work.

Technical consultations have taken place since 2021 for burnup measurements, reactor modeling, and spent fuel plutonium declarations and included a visit by an ORNL team to X-energy headquarters in Rockville, Maryland, in July 2021 and a visit by X-energy staff to ORNL in September 2021.

Additionally, the ORNL team has done work under the DOE-NE Advanced Reactor Development Program (ARDP), which provided an additional opportunity for collaboration with X-energy. These discussions informed the work and planning reflected in this report.

Discussions have also started about the MC&A program. The functional requirements for an MC&A program are being developed, and the final specification will be used to identify a qualified supplier or developer. Continued discussions are needed about measurement and statistical methods used in accounting for plutonium production and uranium depletion in the spent fuel.

The results presented in this report were informed by discussions with X-energy, including detailed design reviews. However, no proprietary information or intellectual property is included in this report.

1.2.2 Collaboration with Kairos Power

In FY 2021, the project team met with Kairos Power on August 3–5, 2021, at the company's Alameda, California, headquarters and discussed MC&A approaches for PBRs. The objective of these meetings was twofold: one was for DOE's Advanced Reactor Safeguards and Security (ARSS project team to better understand Kairos Power's commercial fluoride salt-cooled high-temperature reactor and Hermes test reactor design and current needs at Kairos Power. Second, the laboratory experts provided additional background on work done to date at the national laboratories under this and other projects related to PBRs. The goal was to assist the Kairos Power design team in understanding and providing contacts/resources that could be of assistance.

In November 2021, Kairos Power personnel visited ORNL. Although some topics related to MC&A were discussed, the main purpose of the visit was to discuss hot-cell design approaches that would be applicable to both the test reactor, Hermes, and the power reactor design of Kairos Power. In May 2022, ORNL and Sandia National Laboratories personnel visited Kairos Power's Albuquerque, New Mexico, facility, where the test reactor and fuel manufacturing were discussed. Additional work that involves MC&A is anticipated under an Advanced Reactor Demonstration Program award. The final approvals and funding are expected for this work in FY 2024.

The results presented in this report were informed by discussions with Kairos Power, including detailed design reviews. However, no proprietary information or intellectual property is included in this report.

1.2.3 Collaboration with National Laboratories, the NRC, and Universities

The project team coordinated with the following national laboratories: Sandia National Laboratories, Brookhaven National Laboratory, Argonne National Laboratory, and Idaho National Laboratory.

Throughout the project, multiple briefings and consultations were held with the NRC to discuss the scope of work and to solicit the NRC's advice. The briefings focused on the predicted plutonium content and the uncertainty in the spent fuel based on modeling work presented in May 2022 at the Advanced Reactor Systems meetings in Albuquerque and is reflected in this report. Attendees also discussed accounting options for handling the different uranium enrichments that each vendor will use during reactor startup.

Argonne National Laboratory, Brookhaven National Laboratory, and ORNL partnered with Virginia Commonwealth University under Funding Opportunity Announcement DE-FOA-0002516 for *CFA-22-26931: Creation of a Pebble Database for Material Control and Accountancy in Pebble Bed Reactors, CT-4: Advanced and Small Modular Reactor Materials Accountancy and Physical Protection.* This work was started in FY2023.

2. FOCUS AREAS

2.1 U.S. NRC REGULATORY FRAMEWORK RELEVANT TO PBR MC&A

PBR designers and operators must meet US NRC requirements for MC&A. While a facility's MC&A plan may not be required to be submitted as part of the licensing application, a PBR applicant is nonetheless required to establish, implement, and maintain an MC&A program that secures, documents, and protects the SNM at the facility.[5]

2.1.1 License Applications

The applicant that possesses SNM must ensure the control and accounting of licensed materials. Since the MC&A program is a system of material control and accounting measures to prevent, deter, and detect unauthorized removal or misuse of SNM, the applicant's program should be developed and implemented before SNM receipt and must be maintained while any SNM is present on-site.

2.1.1.1 Production and Utilization Facilities

The regulations in 10 CFR 70.22(b) require that each application for a license to possess and use SNM in a quantity exceeding 1 effective kilogram (Ekg) contain a full description of the program for control and accounting of the SNM. Also required is a full description of how compliance with the applicable requirements in 10 CFR 74.31, 74.33, 74.41, and 74.51 will be accomplished. In addition, the provisions of 10 CFR 70.32(c) require a license authorizing the use of SNM to include, and be subject to, a condition requiring the licensee to maintain and follow a program for controlling and accounting for SNM, a measurement control program, and other material control procedures that include corresponding record management requirements. However, the requirements in 10 CFR 70.22(b) and 70.32(c) contain exclusions for licensees governed by 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," including nonpower reactors (i.e., research and test reactors), for uses of SNM as sealed sources, and operations involved in waste disposal. The same exclusions are contained in the MC&A requirements in 10 CFR 74.31, 74.33, 74.41, and 74.51. Therefore, a PBR applicant under 10 CFR Part 50 is not required to submit a Fundamental Nuclear Material Control plan or the facility's MC&A plan to the US NRC along with the licensing application. Although the plan itself is not required with the licensing application, a licensee is required to establish, implement, and maintain an MC&A program that secures, documents, and protects the material at the facility.

2.1.1.2 Combined License

The proposed rule of 10 CFR Part 53, "Risk-Informed, Technology-Inclusive Regulatory Framework for Commercial Nuclear Plants," is not final (as of the writing of this report) for use by applicants for licensing future commercial nuclear plants, including non-LWRs and LWRs. Consequently, a PBR applicant can apply for a combined license (COL). A COL is a combined construction permit and operating license with conditions for a nuclear power facility under Subpart C of 10 CFR Part 52. If an application is submitted and accepted under 10 CFR Part 52 (COL), the exclusions in 10 CFR Parts 70 and 74 described above are not included, even though, for the purposes of the requirement, the applicants are of the same facility type. In that case, the applicant must submit a ***request for exemption*** from 10 CFR 70.22(b), 70.32(c), 74.31, 74.33, 74.41, and 74.51. The US NRC reviews exemption requests and may

[5] Special nuclear material is defined in 10 CFR 74.4 as "Plutonium, uranium-233, uranium enriched in the isotope ^{233}U or in the isotope ^{235}U, and any other material which the Commission, pursuant to the provisions of section 51 of the Atomic Energy Act of 1954, as amended, determines to be special nuclear material, but does not include source material; or any material artificially enriched by any of the foregoing, but does not include source material."

allow those determined not to lessen the reasonable assurance of public health and safety and the common defense.

COL Exemption Requests

The subject exemption requests will allow the applicant to have similar exceptions for the COL under 10 CFR Part 52. Therefore, the same regulations applied to nuclear reactors licensed under 10 CFR Part 50 (i.e., the regulations under Part 74 Subpart B) will apply to the PBR's SNM MC&A Program. Note that the US NRC will determine that (1) these requested exemptions are consistent with the Atomic Energy Act and are authorized by law; (2) the exemptions will not present an undue risk to public health and safety; (3) these exemptions are consistent with common defense and security; and (4) special circumstances may exist so that the application of the regulations is not necessary to achieve the underlying purpose of the rule. If the US NRC finds that the applicant has satisfied the exemption criteria in 10 CFR 50.12, then considers these exemption requests also to satisfy the exemption criteria in 10 CFR 52.7, 70.17(a), and 74.7; therefore, the exemptions from 10 CFR 70.22(b), 70.32(c), 74.31, 74.41 and 74.51 are justified.

However, it is not entirely clear if the NRC will allow a PBR licensee to be exempt from the reporting requirements traditionally provided to the current operating fleet of LWRs. For reasons explained later in this report, accountancy for PBRs will not be based on item counting of large fuel assemblies. This is because PBR fuel spheres cannot be counted as items as they exist in a form that has some features of bulk material.

2.1.2 Development of an MC&A Program for PBRs

Even if the applicant is exempted from submitting the facility's MC&A plan itself with the application, the applicant for a PBR facility should present information about the MC&A program to the US NRC. An adequate application submittal would describe the applicant's MC&A program elements that will meet certain applicable requirements of 10 CFR Part 74, "Material Control and Accounting of Special Nuclear Material." Applicants that plan to possess, transfer, or receive SNM in a quantity of 1 g or more, will be subject to the general reporting and recordkeeping of 10 CFR Part 74, Subpart B (excluding 10 CFR 74.17), "General Reporting and Recordkeeping Requirements."

The following requirements of Subpart B apply to PBR applicants:

- 10 CFR 74.11, "Reports of loss or theft or attempted theft or unauthorized production of special nuclear material," requires the applicant to notify the US Nuclear Regulatory Commission (NRC) Operations Center in the event of any lost, stolen, or unlawfully diverted SNM, including attempts, within 1 hour of discovery.

- 10 CFR 74.13, "Material status reports," requires the applicant to prepare material balance reports concerning SNM that the licensee has received, produced, possessed, transferred, consumed, disposed of, or lost.

- 10 CFR 74.15, "Nuclear material transaction reports," requires the applicant who transfers or receives SNM in certain quantities or adjusts its SNM inventory to submit a nuclear material transaction report.

- 10 CFR 74.19, "Recordkeeping," requires the applicant to maintain and retain records of the receipt, inventory, acquisition, transfer, and disposal of all SNM. This section also requires applicants to establish, maintain, and follow written MC&A procedures that are sufficient to enable the applicant to account for the SNM in its possession under license. This section also requires the applicant possessing certain quantities to take a physical inventory of all SNM in its possession at intervals not to exceed 12 months.

2.1.2.1 Reports of Loss or Theft or Attempted Theft of SNM

The regulations in 10 CFR 74.11 require the licensee to notify the NRC Operations Center in the event of any lost, stolen, or unlawfully diverted SNM, including attempts, within 1 hour of discovery.

The applicant should describe actions that will be taken if a loss, theft, or diversion of SNM is discovered or suspected. The applicant describes how indicators of a possible loss, theft, or diversion of SNM, whether arising from errors or deliberate actions, will be investigated, and resolved. The applicant should have well-defined procedures for promptly investigating and resolving indications of possible missing SNM and procedures for promptly determining whether an actual loss of SNM has occurred. Resolving a loss indicator means that the licensee has determined that loss, including possible diversion or theft, has not occurred, and is not occurring. Any investigation of an indication of a loss or theft should provide, whenever possible, (1) an estimate of the quantity of SNM involved, (2) the material type or physical form of the material, (3) the type of unauthorized activity or event detected, (4) the time frame within which the loss or activity could have occurred, (5) the most probable cause(s), and (6) recommendations for precluding reoccurrence.

For indications that a loss or theft may have occurred, the resolution process should include (1) thoroughly checking the accountability records and source information, (2) locating the source of the problem, (3) isolating the exact reason for the problem within the area, (4) determining the amounts of SNM involved, and (5) determining that the indication is or is not resolved. If an investigation of an indicator results in a conclusion that the indication is true, such a conclusion must be reported to the NRC within 1 hour of its determination in accordance with 10 CFR 74.11. Procedures should identify all documentation requirements associated with the methods for the reporting, investigation, and resolution of missing SNM indicators. The applicant should identify facility positions responsible for implementing these notification and reporting requirements.

For PBRs, this will be more involved than for LWRs. This is because the number of pebbles are orders of magnitude greater than fuel assemblies, are much smaller, and are not individually identified. In an LWR, the individual fuel pellets are contained within welded fuel rods, which are manufactured in a fixed fuel assembly. Fuel assemblies are normally kept intact and therefore, there should be very few instances in which individual fuel rods, or pellets, are not part of a fuel assembly. Since PBR fuel is in the form of many small pieces and has some of the qualities of "bulk material"[6] pure item counting will not be possible. Material accountancy will be in the form of batch/ piece count and weight measurements with the incoming fresh fuel pebbles contained in drums (or other types of containers) and then loaded into the reactor and the spent fuel being loaded into canisters and then sealed. Therefore, the opportunity for a fuel pebble to be "lost" is much greater than for an LWR.

For Category II facilities (i.e., SNM of moderate strategic significance, such as HALEU), 10 CFR Part 74.43 requires that SNM items are stored and handled, or subsequently measured, in a manner such that unauthorized removal of 200 g or more of plutonium or ^{233}U or 300 g or more of ^{235}U, as one or more whole items and/or as SNM removed from containers, will be detected. As was stated previously, currently this does not pertain to LWRs. However, it is not clear if this limit would apply to a PBR. Additionally, ANSI N15.8-2009, "Methods of Nuclear Material Control—Material Control Systems—Special Nuclear Material Control and Accounting Systems for Nuclear Power Plants," suggests that an

[6] In bulk nuclear material, individual items cannot be distinguished.

item control area (ICA) within the owner-controlled area for which the SNM (fuel assemblies, fuel components, or non-fuel SNM) is maintained in such a way that, at any time, an item count and related SNM quantities can be obtained from the records for the SNM located within the area [3]. ICAs have defined physical boundaries; these generally comprise fresh and irradiated fuel storage areas, including independent spent fuel storage installations, reactor vessels, spent fuel pools, and non-fuel SNM. The regulatory intent of an item control program would have applicability to both the fresh and spent fuel areas of the PBR.

2.1.2.2 Material Status Reports

The regulations in 10 CFR 74.13 require the licensee to prepare material balance reports and physical inventory listings concerning SNM that the licensee has received, produced, possessed, transferred, consumed, disposed of, or lost. US Department of Energy (DOE)/NRC Form 742, "Material Balance Report," and DOE/NRC Form 742C, "Physical Inventory Listing," is the means for submitting reports of material balance and physical inventory listing data to the Nuclear Materials Management and Safeguards System (NMMSS), which is the national database used for tracking certain nuclear material. DOE/NRC Form 742 is used to report a summary of activity for a specified material within a material balance reporting period, as specified in 10 CFR 74.13. The report conveys beginning and ending inventory balances, activities such as shipment and receipts involving other facilities, decay, transmutation, and production calculations. DOE/NRC Form 742C is used to report a facility's physical inventory listing as of a specified date.

The applicant should generally describe how material status reports are prepared and submitted to NMMSS. Reports must be submitted for each reporting identification symbol (RIS), which can only be obtained after the NRC license is issued. Once the license is issued, the licensee should contact the NRC's Office of Nuclear Material Safety and Safeguards, Division of Fuel Management, to request an RIS. Processing the request for a RIS will require the NRC license number, the address where the material will be used and stored, the business address of the licensee, and the name and telephone number of a contact person.

The applicant should have well-defined procedures for preparing and submitting reports in a computer-readable format in accordance with the detailed instructions contained in NUREG/BR-0007, "Instructions for the Preparation and Distribution of Material Status Reports (DOE/NRC Forms 742 and 742C)," and in NMMSS Report D-24, "Personal Computer Data Input for Nuclear Regulatory Commission Licensees." The procedures should ensure that reports are made and filed within the required time periods, as defined in 10 CFR 74.13. If it possesses US government-owned material, the applicant should also have procedures in place to ensure that it will meet the DOE reporting requirements for all receipts, transfers, and inventories of US government-owned, loaned, or leased material, as specified in NUREG/BR-0007 as well.

If the PBR facility has materials that are nationally tracked sources, the applicant should have procedures in place to ensure reporting to the National Source Tracking System, which is a secure, user-friendly web-based database designed to track Category I and II radioactive sources regulated by the NRC and the agreement states. Applicants with less than a critical mass and plutonium sources (less than 16 Ci) or plutonium/beryllium sources should report them to the National Source Tracking System.[7]

If the applicant is subject to the requirements in 10 CFR Part 75, "Safeguards on Nuclear Material—Implementation of Safeguards Agreements between the United States and the International Atomic

[7] Information on NSTS about the National Source Tracking System can be found on the NRC public website at: http://www.nrc.gov/security/byproduct/ismp/nsts.html.

Energy Agency," it should describe how it will submit the required material status reports in accordance with 10 CFR 75.35, "Material status reports."

For PBRs, material status reports will be based on the containers of fresh fuel that enter the facility and are stored until required to be loaded into the reactor. Once loaded, the fuel becomes part of the reactor system until it exits into spent fuel containers. Since the reactor can operate for some time without fresh fuel being loaded and spent fuel removed, physical inventories can be performed with the fuel pebbles in the reactor being considered "material in process" and accounted for based on the total sum of fuel pebbles and corresponding mass that was loaded into the reactor. The accounting for loss and production will not occur until the spent fuel pebbles exit the reactor system into spent fuel storage and the amount can either be calculated, measured, or both.

2.1.2.3 Nuclear Material Transaction Reports

The regulations in 10 CFR 74.15 require the licensee who transfers or receives SNM in certain quantities or who adjusts its inventory of SNM to submit that information to NMMSS. DOE/NRC Form 741, "Nuclear Material Transaction Report," is how licensees submit transaction data to NMMSS. DOE/NRC Form 741 is used to report physical transfers of nuclear materials between facilities and to report exchanges of foreign obligations on material between facilities even when no physical transfer occurs. The form is also used to report on-site transactions, such as inventory corrections that otherwise increase or decrease foreign obligation balances or nuclear material categories within a facility.

The applicant should generally describe how it will track licensed materials from "receipt to disposal" to ensure accurate accounting records and that possession limits listed on the license are not exceeded. The applicant should describe how it prepares nuclear material transaction reports and submits them to NMMSS. The applicant should have well-defined procedures for preparing and submitting reports in a computer-readable format in accordance with the detailed instructions contained in NUREG/BR-0006, "Instructions for Completing Nuclear Material Transaction Reports (DOE/NRC Forms 741 and 740M)," and in NMMSS Report D-24. If it possesses US government owned material, the applicant should also have procedures in place to ensure that it will meet the DOE reporting requirements for all receipts, transfers, and inventories of US government owned, loaned, or leased material, as specified in NUREG/BR-0006 as well. If the applicant will be subject to the requirements in 10 CFR Part 75, it should describe how it will submit the required inventory change reports in accordance with 10 CFR 75.34, "Inventory change reports."

For PBRs, material transactions will be based on the fresh fuel received and any nonconforming fuel returned to the fuel manufacturer. Also, if fresh or spent fuel is sent off-site for analysis that will also constitute a transaction. Since most PBR designs plan to store spent fuel on-site indefinitely, it will remain inside the MBA of the site, depending on the structure of the MBA for each reactor design. If spent fuel is transferred to another MBA or off-site, transaction reports will have to be generated.

2.1.2.4 Recordkeeping

Receipt, inventory, acquisition, transfer, and disposal

The regulation in 10 CFR 74.19(a) states that licensees that are not subject to 10 CFR 74.31, 10 CFR 74.33, 10 CFR 74.41, or 10 CFR 74.51 are subject to the recordkeeping requirements in 10 CFR 74.19(a)(1)–(4), which require a licensee to maintain records of receipt, inventory, acquisition, transfer, and disposal of all SNM in its possession. Each record relating to MC&A that is required by this regulation or by license condition is to be maintained and retained in accordance with the appropriate regulation or license condition. If a retention period is not specified, the licensee is required to retain the record until the commission terminates the license.

The applicant should generally describe the recordkeeping system used to maintain records of receipt, use, transfer, and disposal (as waste) of all licensed material. Table 1 lists each type of record and how long the record must be maintained. Other records, such as transfer records, could be linked to radioactive material inventory records. Receipt records should also document cases the licensee found excessive radiation levels or radioactive contamination on packages or containers of material received and describe the action taken.

Table 1. Records maintenance

Type of Record	How Long Record Must Be Maintained
Receipt, acquisition, or physical inventory	For as long as the material is possessed until 3 years after transfer or disposal
Transfer	For 3 years after the transfer
Disposal	Until the NRC terminates the license

Receipt, transfer, and disposal records typically contain the following information:

- Radionuclide, quantity, and date of measurement of SNM

- For each sealed source, manufacturer, model number, location, and, if needed for identification, serial number and, as appropriate, manufacturer and model number of the device containing the sealed source

- Date of the transfer and name and license number of the recipient, and description of the affected radioactive material (e.g., radionuclide, quantity, manufacturer's name and model number, serial number)

- For licensed materials disposed of as waste, the radionuclide, quantity, date of disposal, and method of disposal (e.g., decay, sewer)

Written material, control, and accounting procedures

The regulation in 10 CFR 74.19(b) states that each licensee authorized to possess SNM in a quantity exceeding 1 Ekg shall establish, maintain, and follow written MC&A procedures that are sufficient to enable the licensee to account for the SNM in its possession under license. The applicant should indicate that procedures will be established, maintained, and followed to account for SNM and describe the written procedures established to ensure all applicable MC&A requirements are met. The applicant should provide some specific examples addressing, at a minimum, the following: organization, records and reporting, notification of events, receiving and shipping, internal transfers, physical inventory, element and isotopic calculation method, and identification of SNM and non-SNM items to preclude loss. The applicant should indicate that provisions are made for the written approval of procedure revisions.

Physical inventories

The regulation in 10 CFR 74.19(c) states that each licensee not subject to 10 CFR 74.31, 10 CFR 74.33, 10 CFR 74.41, or 10 CFR 74.51 and authorized to possess SNM in a quantity greater than 350 g of contained ^{235}U, ^{233}U, or plutonium, or any combination thereof, shall make a physical inventory of all SNM in its possession under license at intervals not to exceed 12 months. The applicant should have well-defined procedures for the planning, conducting, assessing, and reporting the physical inventories. The applicant should generally describe how it performs physical inventories of its SNM. The inventory description should address the regulatory requirement for conducting a physical inventory at intervals not to exceed 12 months and for maintaining and retaining associated inventory records, although the results of the physical inventories need not be reported to the NRC. Concerning the physical inventory requirement, the applicant should define the term *physical inventory*, identify the overall responsibility for the implementation of physical inventories, and address other inventory topics such as conduct, coverage, inventory procedures, inventory methods for fuel types, fuel components, fuel inside the reactor, fuel outside the reactor, storage of fuel and non-SNM fuel, inventory reconciliation, and documentation. The applicant is required to submit reports regarding the physical inventory in accordance with the requirements in 10 CFR 74.13. The applicant should describe how it maintains and retains inventory records in accordance with 10 CFR 74.19.

See previous discussion for PBR physical inventory taking and recordkeeping.

Records access and storage

The regulation in 10 CFR 74.19(d) requires the licensee to ensure that the recordkeeping system can produce clear and legible copies of records after storage for the period specified by the regulations. The section also states that the licensee shall maintain adequate protection against tampering and loss of records. The applicant should describe how it stores records and how it controls its access to records to meet the requirement in 10 CFR 74.19(d). The applicant should define the term *MC&A records* and provide examples of various types of records such as SNM receipt, acquisition, internal transfer, measurement and calculation, reconstitution, inventory, shipment, and disposal. The applicant should identify the organization responsible for maintaining records for the SNM in the facility's possession. The applicant should indicate that adequate controls against tampering with and losing records will be maintained and that periodic review and assessment of records will be documented. In terms of the record retention requirements, the applicant should indicate that SNM records and reports will be retained as required.

2.1.3 Additional Information

Guidance that may be useful to the applicant developing the MC&A program is contained in Regulatory Guide 5.29, "Special Nuclear Material Control and Accounting Systems for Nuclear Power Plants," Revision 2 [6], which provides American National Standard Institute (ANSI) publication N15.8-2009 as an acceptable approach for complying with the NRC's MC&A requirements in Subpart B of 10 CFR Part 74. Although ANSI N15.8-2009 is intended to develop MC&A programs for LWRs that use low-enriched uranium oxide fuel, aspects of this guidance may be useful in developing the MC&A program for PBRs.

Additionally, NUREG 2159 [7] may contain useful information about Category II fuel cycle facilities. If a PBR uses Category II fresh fuel, the NRC may require that portions of other requirements for fuel cycle facilities in the 10 CFR may apply. Table 2 shows that a facility that has 10 kg or more of uranium enriched to greater than 10% but less than 20% will be considered a Category II facility.

Table 2. Facility category is based on quantities of special nuclear material present

URANIUM-235			
CATEGORY	**I**	**II**	**III**
ENRICHMENT (E % U-235)	SNM of Strategic Significance (Q grams of U-235)	SNM of Moderate Strategic Significance (Q grams of U-235)	SNM of Low Strategic Significance (Q grams of U-235)
$0.711 < E < 10$	_____	_____	$Q \leq 10{,}000$
$10 < E < 20$	_____	$Q \geq 10{,}000$	$1{,}000 < Q < 10{,}000$
$E \geq 20$	$Q \geq 5{,}000$	$1{,}000 < Q < 5{,}000$	$15 < Q \leq 1{,}000$

Category II licensed facilities have tighter material and measurement controls and reporting thresholds, but these are based on fuel cycle facilities that process true bulk material in powders or solutions. Since fuel pebbles are distinct pieces, they do not fit the classification as true bulk material. As described later in this report, they can be controlled and reported as batches that exist in containers with the "pieces" being part of the container and itemized in the MC&A system and reported to the NRC. An ORNL report completed in 2020 developed a Model MC&A Plan for PBRs [8]. It has useful information about the use of NUREG 2159 and how it may apply to the licensing for PBRs.

As previously noted, applicants who intend to possess certain amounts and types of SNM not in sealed sources may be subject to additional MC&A requirements in 10 CFR Part 74 other than those in Subpart B. A license to possess SNM of low strategic significance (Category III) or SNM of moderate strategic significance (Category II), not in sealed sources, may be subject to requirements in 10 CFR Part 74, Subparts C and D, respectively. Applicants for licenses to possess such material should contact the NRC for further guidance.

2.2 IMPACTS OF PEBBLE FUEL FLOWS ON MC&A

Understanding how material will be introduced into the reactor facility, the designation of the MBAs, and the configuration of key measurement points (KMPs) will inform the MC&A approach.

The loss and production of fissile material during the operations of a nuclear reactor result from using the nuclear fuel to produce power and the corresponding transmutation of actinides. In a U and Pu system, U is consumed, and Pu is produced. MC&A systems must track the depletion (loss) of uranium and the production (gain) of plutonium to account for fissile and fissionable material. This tracking is normally done by computer modeling to determine the depletion and production based on reactor parameters and operations. For reactors with fixed fuel in their cores, such as LWRs, these models are well developed, and the uncertainties are relatively low because the fuel enrichment and precise location in the core are known. Reactor parameters such as power level, burnup, core configuration, fuel loading, and control-rod movement are well understood. Typically, the loss and production are recorded when the used (i.e., burned) fuel is removed periodically during refueling from the reactor core and placed into wet storage.

However, for PBRs, the fuel consists of hundreds of thousands of pebbles that pass through the reactor core via flow loops multiple times before they are removed after having reached the desired burnup. From an MC&A standpoint, there are multiple differences between PBRs and fixed fuel reactors:

- It is not straightforward to determine when to record loss and production because each fuel pebble is circulated within the reactor several times.
- Because the fuel is not fixed, each pebble's pathway through the reactor and its position at any given time cannot be known.
- The number of passes and the length of time each pebble resides in the reactor is not directly known and can be estimated only via system pebble flow characteristics and actual burnup measurements.
- There will be a range of values for determining when a pebble should be discharged, and this range influences overall loss and production in the system. Therefore, several uncertainties exist when attempting to measure loss and production during normal operations and directly affect material accounting.
- The pebbles consist of varying enrichment during startup and equilibrium operations. Additionally, there may be non-fuel, graphite "moderator" pebbles used throughout operation.

2.2.1 Pebble Fuel

Fuel pebbles are discrete objects that can be counted. In a PBR, the fuel will not change chemical or physical properties unless the fuel pebbles are physically damaged. The fuel pebbles will, however, undergo significant isotopic changes, as in any power reactor that consumes and produces fissile and fertile isotopes. PBR fuel is more portable and concealable than traditional LWR fuel assemblies, but each pebble only contains a relatively small amount of SNM, so a large number of pebbles would need to be taken in an abrupt theft scenario to equal the quantity of nuclear material in an LWR fuel assembly. Protracted theft scenarios would require many attempts to obtain a significant amount of nuclear material.

As an example, one 60 mm TRISO fuel pebble has a mass of total mass of approximately 200 g. Assuming an enrichment of 9.6% ^{235}U, the total quantity of SNM contained in each pebble will be between 7 and 9 g of low-enriched uranium, or just less than 1 g ^{235}U before irradiation. After an irradiation of approximately 85 GWd/tHM upon discharge,[8] the pebbles would contain less than 0.12 g of plutonium and less than 8.2 g of residual uranium at 3.8% ^{235}U. Other PBR designs are expected to utilize fuel with uranium enrichments from natural up to less than 20%, and some designs plan to combine pebble enrichments across that range.

The limit for promptly detecting and reporting the loss of Category II material is 200 g plutonium or ^{233}U or 300 g ^{235}U. For the TRISO fuel example, each individual pebble is well below the material quantity thresholds. This corresponds to approximately 350 fresh fuel pebbles and 1,600 spent fuel pebbles. That amount of fuel pebbles is approximately what would be contained in a fresh fuel container or a spent fuel canister.

A PBR may use fuel of varying enrichments, as well as graphite pebbles. Visually, graphite pebbles cannot be distinguished from fuel pebbles, and pebbles of differing enrichments are indistinguishable.

[8] In this scenario, a pebble is discharged if its burnup is greater than 80 GWD/THM, with a maximum expected burnup of 90 GWD/THM.

The MC&A system must identify an approach that distinguishes different levels of enrichments and precludes non-SNM items from being mistaken for SNM items (Section 7.4 of ANSI N15.8-2009 [3]).

2.2.2 Material Flow in a PBR Facility

The fuel pebbles would be received at the facility in containers with a declared quantity of nuclear material that would be received as an item. It is unknown if the number of pebbles would be identified in the shipper's documentation as "pieces" contained within the container. The nuclear MC&A and pebble count may be stated on a batch basis, and a batch would comprise one or more containers. The fresh fuel receipts and storage area may be an item control area (ICA). An ICA is defined as an identifiable physical area for the storage and control of SNM items [9]. Control of items moving into or out of an ICA is by item identity and SNM quantity as determined by a previous measurement.

Containers of pebbles would be stored to await transfer of a set number of pebbles into the feed mechanism (e.g., a feed hopper) of the reactor. After the reactor is loaded with pebbles, the reactor itself could become a defined containment boundary. Per ANSI 15.8-2009, once an LWR reactor vessel is loaded with fuel and closed, it is considered an item for accountancy purposes. This could be analogous to a PBR because at any moment the number of fuel pebbles and associated mass can be stated. However, unlike a traditional LWR, the PBR itself will have a continual flow of nuclear material into and out of it. Owing to differences in timing of additions and removals of nuclear material (i.e., fuel pebble feed and withdrawal), the inventory of the reactor will fluctuate. The PBR is more analogous to a *unit process* as defined in 10 CFR Part 74.4 instead of an item, and it behaves similarly to the dynamic inventory in an enrichment facility. Although the PBR is dynamic in that pebbles are moving through the reactor and the primary fuel loop, the inventory of pebbles will stay in the reactor for an extended period. During the initial fuel loading, the inventory in the reactor can be established from the number of pebbles required to fill the reactor. After a PBR has achieved an equilibrium operating level, the contained fissile mass and average burnup could be calculated. Until equilibrium is reached, the average burnup could be estimated by the location of the control rods (assuming operation of the reactor at a constant power level).

After residing in the reactor core for some time interval, pebbles will exit the core for either recycling or removal. Pebbles can exit the primary fuel loop of the reactor in one of three ways: as damaged fuel, as spent fuel, or as used fuel (see Figure 1). Upon exiting the core, an integrity check will be performed on the pebble to determine if it remains physically intact. Damaged pebbles and pebble fragments will be transferred to a separate area outside of the primary fuel loop. These damaged pebbles could be loaded into a container that could be sealed and counted as an item. After the integrity check, each intact pebble will be measured individually to determine if the burnup exceeds a threshold. If the pebble exceeds the burnup threshold, it will be sent to a collection area and ultimately placed in spent fuel containers and most likely permanently sealed (e.g., welded). After the nuclear material content of each container is determined, the container could be counted as an item with a quantity of nuclear material and a pebble count. The average burnup of the spent fuel in a container, along with a corroborating burnup measurement on the container, could be used to determine the nuclear material content.

Pebbles may also be removed from the core during reactor maintenance and stored as used fuel. The term "used fuel" is differentiated from "spent fuel" in that used fuel has some degree of burnup but could be reinserted for additional power operations. Spent fuel has a degree of burnup that is no longer suitable for reinsertion. Used fuel may be temporarily containerized for possible item counting and then returned to the primary fuel loop for recycle later.

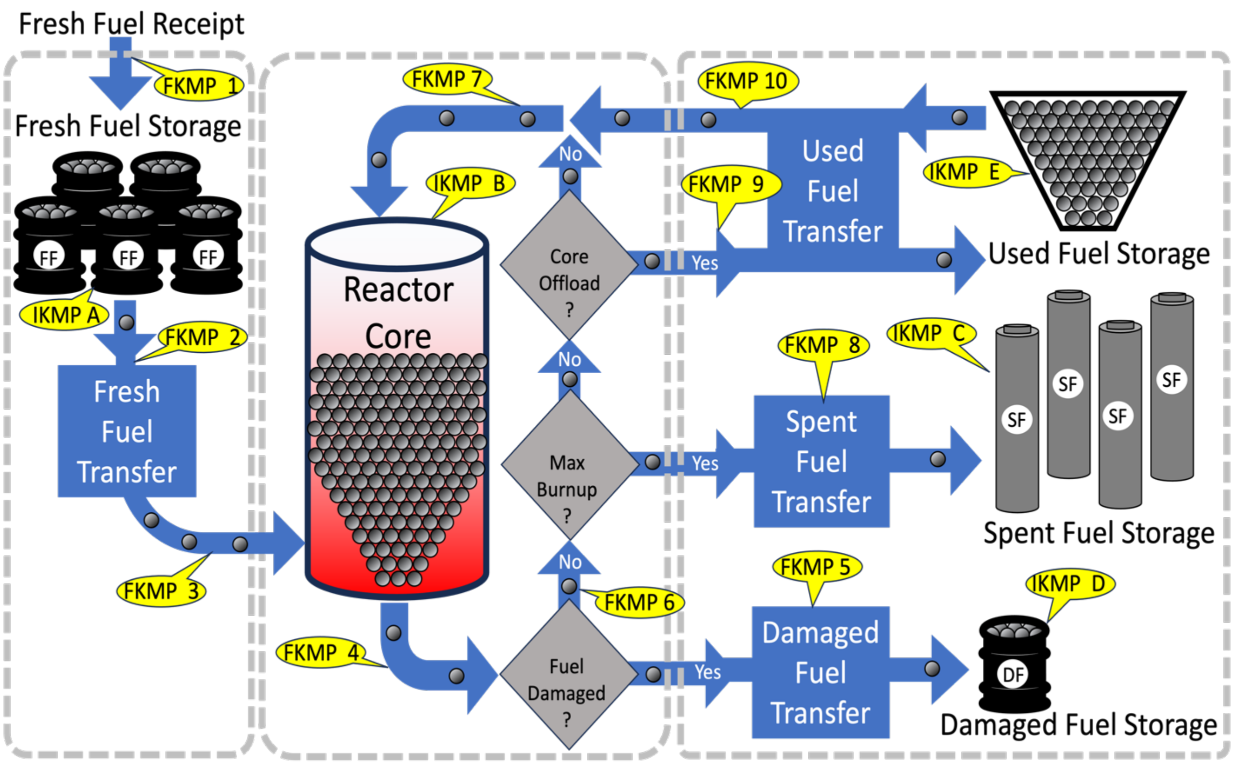

Figure 1. Material flow in a PBR facility. The grey dashed boundaries represent one possible subdivision of the facility's MBA or sub-MBAs/ICAs.

A KMP is a physical location where nuclear material appears in such a form that it may be measured to determine material flow or inventory [10]. Many possible inventory KMPs (IKMPs) and flow KMPs (FKMPs) could be identified in a PBR facility, including the following:

- IKMP A: Fresh fuel storage area (items)
- IKMP-B: Reactor system
- IKMP C: Irradiated (intact) spent fuel storage area (items)
- IKMP D: Irradiated (damaged) fuel and waste storage area (items)
- IKMP E: Irradiated intact used fuel storage area (items)
- FKMP 1: Fresh fuel receipt (items)
- FKMP 2: Fresh fuel transferred to reactor pebble feed system
- FKMP 3: Fresh fuel insertion into reactor core
- *FKMP 4: Irradiated fuel removal from reactor core
- FKMP 5: Irradiated damaged fuel and waste transferred to damaged pebble storage area
- *FKMP 6: Irradiated intact fuel transfer to BUMS
- *FKMP 7: Irradiated intact fuel reinsertion to reactor core
- FKMP 8: Irradiated intact fuel removed from reactor system and transferred to spent fuel storage
- FKMP 9: Irradiated intact fuel removed from reactor system for temporary core off-loading
- FKMP 10: Irradiated intact fuel returned to the reactor system after temporary core off-loading

(* FKMP marked with an asterisk are for recording internal flows and would not be required for material accountancy.)

Note, the cooling time may be needed before transfer to containerized storage. The used, spent, and damaged pebbles storage areas could constitute ICAs. The nuclear material quantity in a spent fuel container can be determined from the aggregate values from the pebbles exiting the reactor and being placed in a container or by weighing the filled container and then determining the burnup. This can be done by using the values from the BUMS measurements used to discharge the pebbles, by calculating the average burnup of the pebbles in the container, by other NDA measurements (e.g., total neutron and total or specific isotope gamma measurements), or by some combination of these methods.

2.3 FRESH FUEL CONTAINERS: INVENTORY AND RECEIPT VERIFICATION

Fuel pebble containers must be designed and certified to handle both fresh and used fuel pebbles at the industrial scale required for commercial reactor operations. Container design for HALEU fuel pebbles is a current challenge because modification of the container size requires recertification. The fresh fuel product must be transported from the fuel fabrication facility and stored until the fuel is loaded into the fuel-handling system. The spent fuel containers will likely need to function as interim storage and are typically expected to remain on-site for multiple decades over the operational lifetime of the plant.

Pebbles of various enrichments and pebbles containing only graphite cannot be visually distinguished from one another. The MC&A system will need to identify an approach that distinguishes different levels of enrichments and precludes non-SNM items being mistaken for SNM items (Section 7.4 of ANSI N15.8-2009 [11]).

2.3.1 Fresh Fuel Containers

A primary fresh fuel container candidate is the Versa-Pac (VP55), a specially configured 55-gal package for shipment of uranium oxides, uranium metal, and other uranium compounds including TRISO fuel. The VP55 would contain approximately 350 pebbles with a total heavy metal (uranium) weight of 3 kg and a fissile content of <400 g ^{235}U. Key design characteristics are shown in Figure 2.

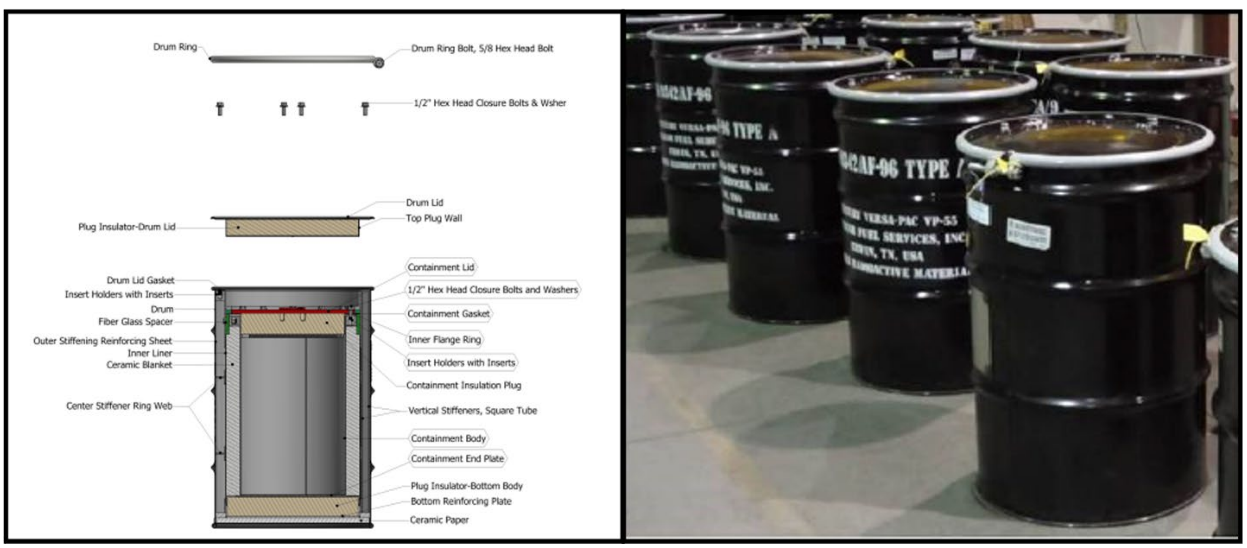

Figure 2. (*Left*) Cross sectional diagram of VP55 container. (*Right*) Photo of VP55 containers.

2.3.2 Fresh Fuel MC&A Approach

The MC&A approach for fresh fuel is not expected to be challenging with numerous existing acceptable parallels within other types of facilities. The general rule of thumb is in the case of closed containers of SNM pieces, provisions should be made to assure that the removal of contents would be observable. Based on diluteness of the SNM in the pebble, 10 CFR Part 74 required detection thresholds for Category

II approach an entire VP55 container, which would have to be compromised. Currently, all LWRs are considered Category III facilities because their fresh fuel is below 10% enrichment. However, if a PBR uses Category II nuclear material (greater than 10% but less than 20% enrichment), then some of the limits for Category II may apply.

Although MC&A plans for nuclear reactors may not be subject to the other portions of 10 CFR Part 74 (see previous discussion on licensing), if Category II fresh fuel is used some aspects of the other portions may apply. As such, the NRC may look at how the following areas are addressed in the MC&A plan and operating procedures.

- Current Knowledge of Items—10 CFR 74.43(b)(5)(i) and 10 CFR 74.43(c)(1)
- Tamper-Safing—10 CFR 74.43(c)(3)
- Receipt, Shipment, and Transfer of SNM—10 CFR 74.43(c)(2)
- Item Storage and Handling—10 CFR 74.43(b)(5)(ii)
- Item Control Program—10 CFR 74.43(b)(5)(ii)

Particular to VP55s used for PBRs there will be containers of differing enrichments and non-SNM (graphite) pebbles. It should be clear in handling and inventory procedures how these are identified, stored, and managed for operations and inventory.

Receipt verification is also expected to be straightforward with several potential options shown from easiest to more difficult:

- Simply verifying the container and tamper-indicating device's serial number
- Opening the container and counting the pebbles
- Opening the container, counting the pebbles, and measuring a sample
- Opening the container and individually measuring each pebble

The first, least rigorous option is defined as "unopened receipts" in 10 CFR Part 74.

> *Unopened receipts* means receipts not opened by the licensee, including receipts of sealed sources, and receipts opened only for sampling and subsequently maintained under tamper-safing.

If there is high confidence in the transfer process, then an acceptable MC&A approach would be the first option, which is to confirm the container's and TID's serial number and inspect the tamper-safing upon receipt and for inventory. This may be coupled with a weight measurement or quick enrichment measurement to confirm differing enrichments and non-SNM-bearing containers.

The term *high confidence* means that no credible diversion strategies during transfer and storage processes were identified. This term also assumes that the processes implemented by the fuel fabricator and reactor operator for transfers and receipts have a low probability of discrepancies, which would affect MC&A, operations, and safety. Although fuel fabrication is not covered in this report, the MC&A plan at the fuel fabrication facility will affect the receipt of shipments at the reactor.

If gross weight and the container/tamper-indicating device's serial number verification are used for verification, then the numbers of pebbles and the SNM content within those pebbles would be placed into the inventory based on the shipper's values and maintained until the container is opened for the fuel pebbles to be fed into the reactor.

Even for cases with low confidence in the transfer process but with fresh fuel having a short residence time in the receipt area, this approach may be acceptable. This is because opening the container for fuel loading shortly after receipt provides another potentially equally acceptable opportunity for verifying the SNM received within acceptable timelines for loss detection.

In cases with lower confidence either from credible diversion scenarios or performance violations in the transfer and storage processes, additional rigor in receipt and inventory checks may be warranted. These efforts could range from random sampling of containers and application of independent measurements up to 100% verification. Owing to the labor-intensive nature of these checks, this is not an ideal approach.

2.4 SPENT FUEL CONTAINERS

Undamaged pebbles are discharged from the primary reactor loop once their burnup has been determined to exceed the burnup threshold. Pebbles are then transferred directly to a spent fuel canister. PBR reactor designers are considering spent fuel containers of various designs. Generally, these spent fuel containers are tall narrow stainless-steel cylinders that hold hundreds to thousands of spent pebbles. The geometry of these vessels is driven by several factors, including passive heat dissipation rate, criticality safety, and external dose rate. Figure 3 shows several sizes of spent fuel container based on the TLK container design with an inner radius of 27.95 cm and a 1.6 cm thick SS314 stainless steel wall [12]. The height of the container is based on the pebble capacity. Modeling results for discharged pebble isotopics and NDA measurements are included in Section 2.5.

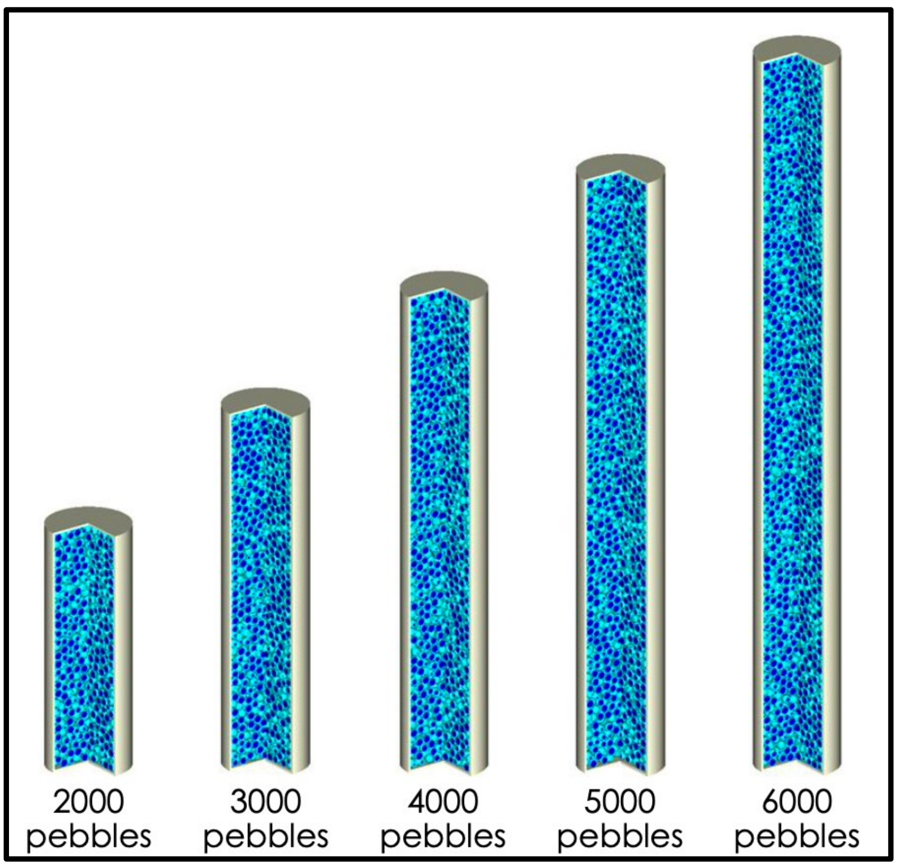

Figure 3. Spent fuel pebble canister geometry of various capacities based on the TLK canister design [12].

The dose rate at a 1 m distance with no shielding will depend on the aggregate burnup of the pebbles. It will also depend on whether we assume that the contents are homogeneously distributed or distributed nonuniformly, such as preferentially toward the bottom or the outer wall. Figure 4 shows the dose rate contours of five canisters of various capacities with homogeneously distributed pebbles. Figure 5 shows dose rates at 1 m over time from canisters containing PBMR-400 pebbles at a burnup of 90 GWd/tHM. When compared to the dose rates from LWR assemblies to this high burnup example, the pebble cannister starts out with an order of magnitude higher gamma dose but decays more quickly. This is because the burnup of the pebbles is much higher than in a LWR, which results in a higher concentration of fission products, but the amount of fuel in the containers is less than a single LWR fuel assemble. For example, 6000 pebbles in canister has approximately one-tenth the fuel compared to the single LWR fuel assembly.

The current threshold for self-protection of spent fuel is 100 rad/h at 1 m. However, future NRC Rulemaking is expected to raise this threshold to 5,000 rad/h at 1 m. In this case the spent fuel cannisters would be self-protecting (above 5,000 rad/h) for approximately 3 - 4 months depending on the number of pebbles in the container. The 100 rad/h threshold will continue to be exceeded for 40 – 50 years. This information will be important when developing the potential diversion and theft scenarios as part of developing the MC&A and physical protection features of the facility.

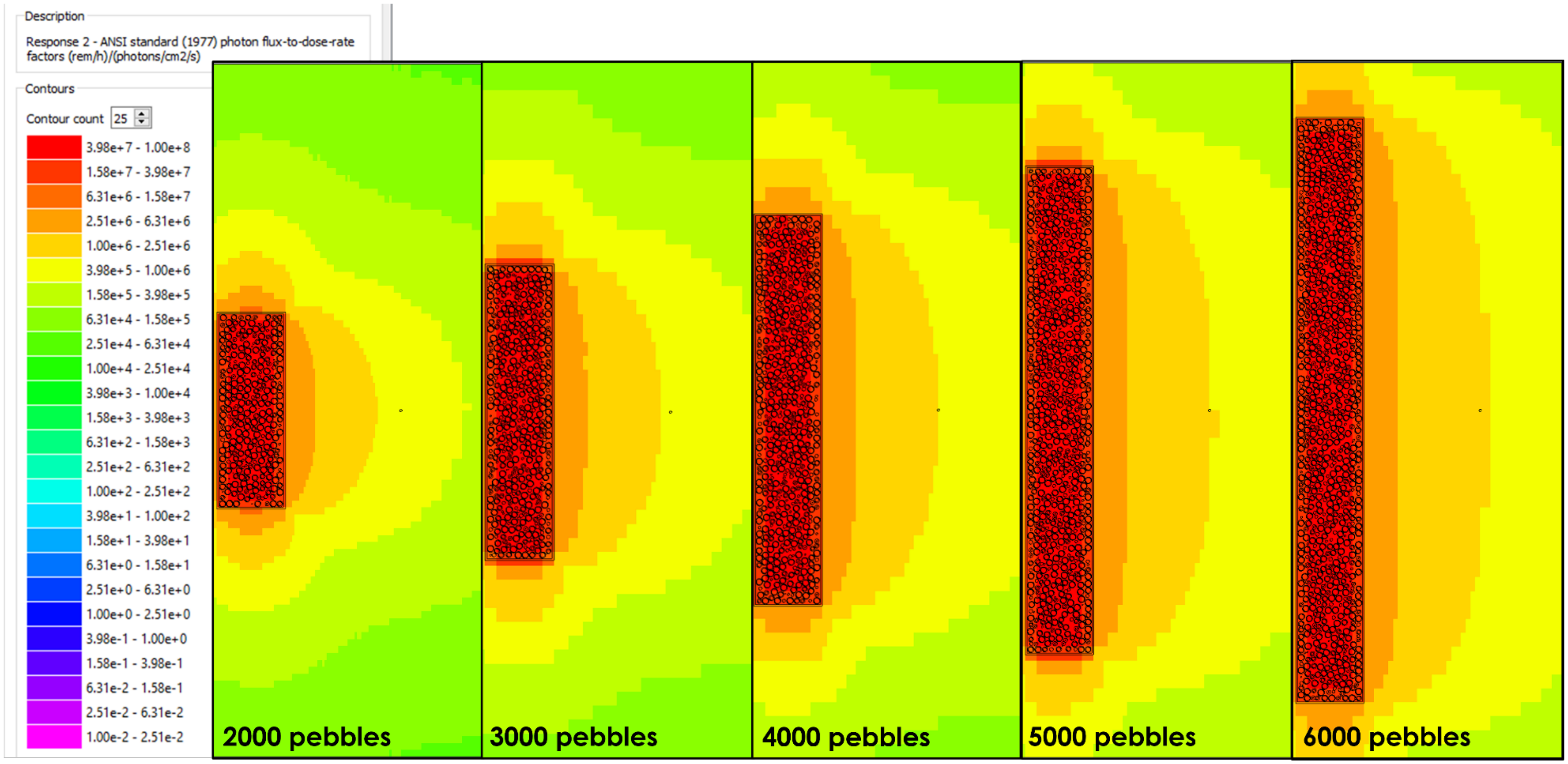

Figure 4. Gamma dose contours for spent fuel cannisters of various pebble capacities. Pebbles are PBMR-400 at a burnup of 90 GWd/tHM.

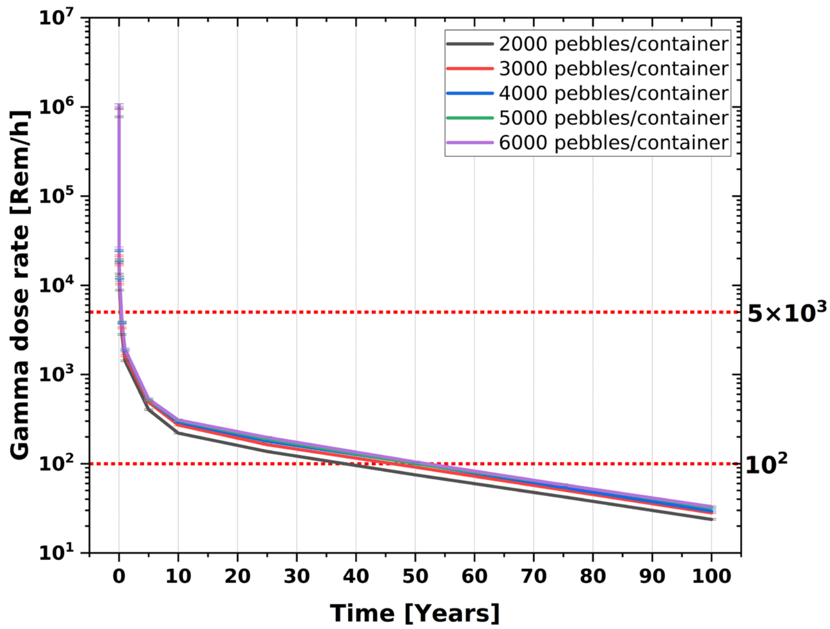

Figure 5. Gamma dose rates at 1 m from the surface as a function of time for containers with various pebble capacities. Pebbles are PBMR-400 at a burnup of 90 GWd/tHM.

2.5 REACTOR AND SYSTEMS MODELING

Understanding the behavior of the fuel pebbles as they circulate through the reactor is needed to support the various measurement systems for reactor operations. Chief among these is BUMS, which must determine the burnup of each fuel pebble that exits the reactor and is central to the operating principles of a PBR. Variabilities such as pebble transit times, heavy metal loading, and plutonium production all affect the actinide isotopics and fission product generation for each fuel pebble. This, in turn, affects the key function of the MC&A program that is required to determine loss and production of nuclear material.

2.5.1 Loss and Production of Nuclear Material in PBRs

Different areas of the core experience different neutron spectra and fluxes and yield different rates of plutonium production in pebbles. This variability will have a direct effect on MC&A and the uncertainties around measuring the burnup of each pebble, determining the range of burnup values, estimating the plutonium content, aggregating these values over thousands of pebbles, and characterizing the spent fuel cannisters. The reference design for this work is the PBMR-400 reactor because of the available design and operating information. This work assumed a core that achieved steady-state (i.e., equilibrium) operations and did not assume start-up or shutdown/ refueling operations. Nonequilibrium conditions should be addressed in future work. The following questions will be addressed here:

- What is the total uranium and plutonium content for individual spent fuel pebbles, and what are the associated uncertainties as well as in the aggregate, as for a spent fuel container?
- What is the effect of different pebble flow paths in the reactor core, and what is their effect on burnup measurements?

- Based on analyses of statistical approaches and sampling schemes, what are the pebble discharge values? Specifically, when can it be determined that a pebble has reached its maximum burnup and is removed from the core? What is the range of values, and what aggregate effect this has on overall loss and production amounts? Which nondestructive measurements can be used to determine pebble burnup and their accuracy? Since it is desired to measure each pebble every time it exits the reactor, the nondestructive measurement time is limited.

- What are the measurement uncertainties and their effect on overall loss and production and on MC&A?

The modeling effort focused on determining the uranium and plutonium content of spent fuel pebbles based on variations in their flow paths through the reactor core and different residency times. The goal was to determine the total uranium and plutonium values in the spent fuel containers, the associated uncertainties, and the standard deviation that can be used in determining overall loss and production of fissile and fertile material. To this end, the team employed several software tools:

- As part of the Nuclear Regulatory Commission's code suite assessment of non-LWRs, 3D full-core models in SCALE (Standardized Computer Analyses for Licensing Evaluation) and Two-Dimensional Runoff Inundation Toolkit for Operational Needs (TRITON) were developed to understand spectral zones and power distributions within the core.

- A new capability was developed for estimating pebble inventory in SCALE and ORIGAMI (ORIGEN Assembly Isotopics), and it enabled defining the flow path of a pebble and assessing the effect on isotopics.

- Oak Ridge Isotope GENeration (ORIGEN) was used to reproduce the pebble power history required to define the transit history in each channel and pass.

2.5.2 Modeling of Pebble Discharge Burnup and Isotopic Inventory

To achieve higher fuel utilization, lower excess reactivity, and flatter power distribution, PBR designs adopted the multi-pass fuel-reloading scheme. In this scheme, each spherical fuel element, or pebble, circulates multiple times through the core until the desired discharge burnup is attained. Typically, the pebbles enter the core through the charging chute at the top and are randomly located at the top of the core, descend along its trajectory through the core, and then exit through the discharge chute at the bottom. Each pebble can accumulate a unique amount of burnup with each pass because the neutron flux varies at different radial and axial locations within the reactor, and the speed and pathway of a pebble also vary. Therefore, the burnup of each retired pebble may vary even though it experiences the same number of passes through the core, and distributions in the burnup, isotopic inventory and decay heat of the retired pebbles are expected. Note that this simulation is for equilibrium core conditions where it is assumed the flux in each region will be constant at any time [14].

An equilibrium core was used to develop the models where the reactor was assumed to have achieved a steady state in its operations history after the startup of the reactor. Additionally, the same initial enrichment level for every pebble was assumed.

2.5.3 Introduction of PBMR-400

The pebble burnup and isotopic inventory uncertainty quantification in this study was carried out for the PBMR-400 reactor [15-18] (Figure 6) because data such as power profiles, temperature distributions, and equilibrium core composition are publicly available for this reactor. The PBMR-400 was developed by a

South African company from 1994 to 2009, and its core is an annular cylindrical core with an inner radius of 1 m and an outer radius of 1.85 m. There are two fixed graphite reflectors: one is in the center, and the other is just outside the core. The core is loaded with about 452,000 pebbles, and the fresh fuel pebble enrichment is 9.6%. Moreover, in the multi-pass scheme adopted in the PBMR-400, a pebble passes the core six times on average before achieving a burnup of 90 GWd/tHM and then being retired as spent fuel.

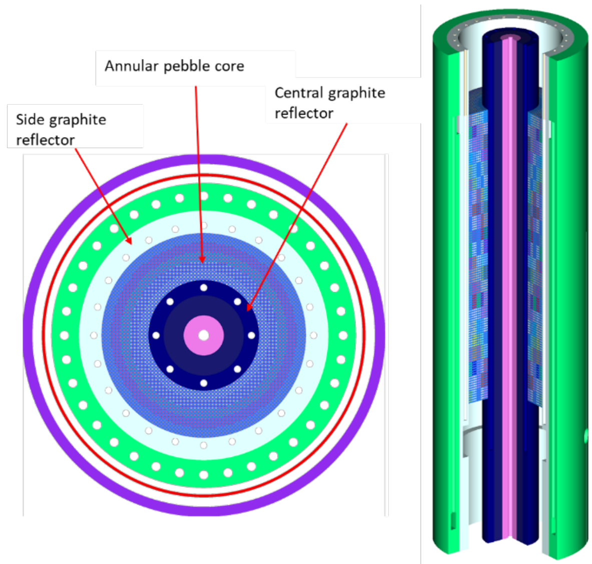

Figure 6. PBMR-400 core model [18].

2.5.4 SCALE/ORIGAMI-Sequence

ORIGAMI in the SCALE code [19] was used to perform the pebble burnup simulation because it was recently developed to rapidly model the depletion of flowing pebbles. Pebble depletion is carried out as a series of axial (i.e., vertical) segments called *transit zones*. The radial characteristics in each axial transit zone can also be accounted for using the radial power shape, radial pebble population distribution, and radial zone library. Multiple passes can be simulated, and each pass is defined as a *transit history* that includes the pebble power, irradiation time, cooling time, and a series of sequential transit zones. The fractional irradiation time and the axial power factor are part of the inputs for each transit zone. An example of depletion modeling in ORIGAMI for a PBR with three radial channels and five transit zones in each radial channel is depicted in Figure 7. Generally, several radial channels in a transit history can be included, for example, to obtain the average core-wise fuel composition or to select one radial channel in each transit history, as applied in this work, to evaluate the uncertainty for a pebble flowing through different channels.

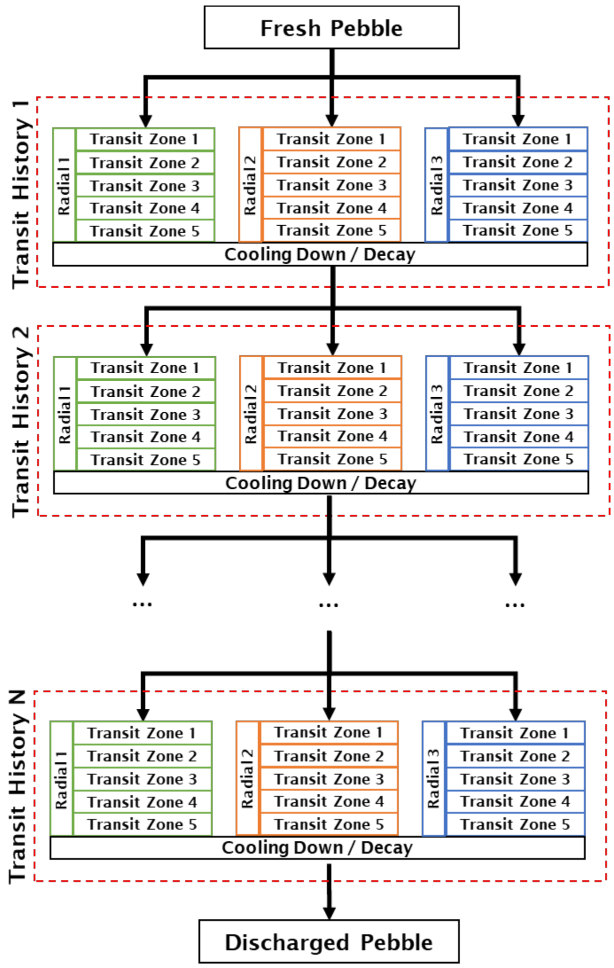

Figure 7. Example of depletion flowchart in ORIGAMI.

2.5.4.1 Methodology

The publicly available data from PBMR-400 were used to develop the ORIGAMI cross section and model. The power and temperature profiles were used to define the transit zone and history, whereas the equilibrium core composition was used to generate the libraries for ORIGAMI. Transit time, which depends on flow speed, was also used to simulate the pebble depletion accurately. The pebble flow can be modeled as a 1D laminar flow in the axial direction, and the axial velocity depends on the radial position. For example, a pebble near the graphite reflector walls tends to move slower than other pebbles because of the friction between the pebble and the walls. The transit time of pebbles that flowed adjacent to the reflector walls was 54% longer than those that flowed in other locations [17, 20].

Previous work on the Very Superior Old Programs (VSOP) modeling of the PBMR-400 equilibrium core divided the core into five radial channels (Figure 8) [17, 20]. This approach was also considered in this work [21]. A channel has a unique neutron spectrum according to its temperature profiles and proximity to the graphite reflectors. To obtain an accurate cross section for each radial channel, a single pebble with the reflective depletion model cannot be used. Owing to the continuous circulation of the pebble in the core, the compositions of the neighboring pebbles cannot be represented by a single pebble depletion. Moreover, the neighboring pebbles will determine the neutron spectrum in the pebble. Consequently, the channel-dependent spectrum cannot be represented correctly by using a single pebble depletion model. In this work, the library for ORIGAMI was produced using a series of TRITON axial slices of the core. This

2D core model was chosen because the variations in neutron energy spectra are larger in the radial direction than in the axial direction. The neutron spectra are softer in the two channels next to the reflectors than in others, as depicted in Figure 9.. In this model, a depletable pebble is placed radially within an array of pebbles at an assumed constant burnup level/equilibrium core composition. The depletable pebble is initially fresh fuel, located in each radial channel to capture the variation of the neutron spectrum, and the cross sections are tabulated from the depletable pebble as a function of radial position, burnup, fuel, and reflector temperatures. Three temperatures that cover the operating temperature range are considered and 28 burnup steps that cover 0–100 GWd/tHM are available.

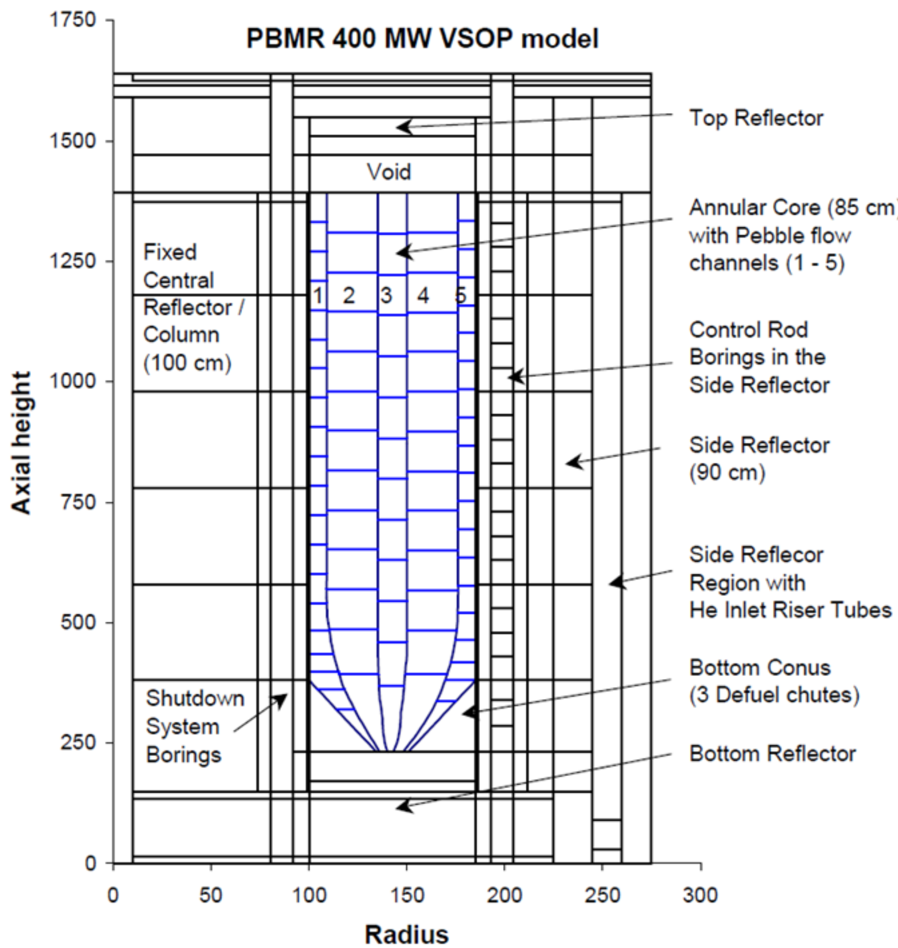

Figure 8. Radial flow channels [16].

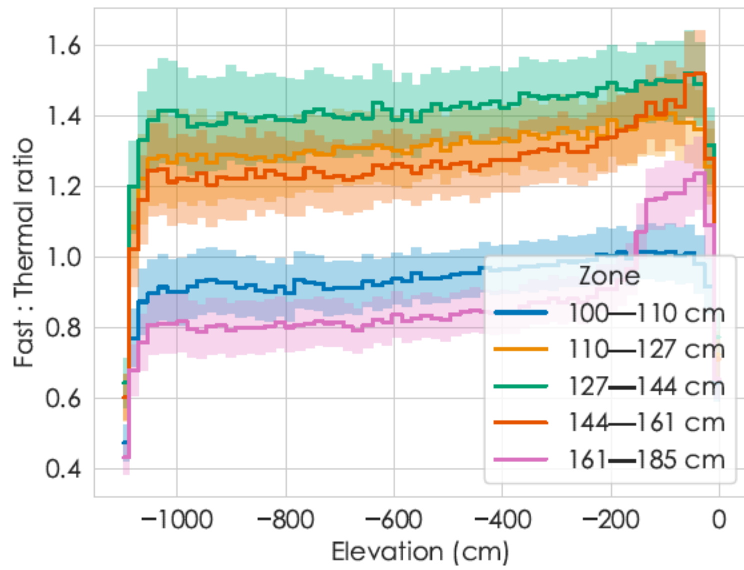

Figure 9. Axial and radial variations in neutron-energy spectra elevation at 0 cm, which represents the top of the core [18].

Another set of information required to construct the ORIGAMI model is the pebble power history in each channel and pass. ORIGEN [19] was used to reproduce the pebble power history required to define the transit history in each channel and pass, using the data given by Reitsma [17] and the channel-wise flux profile [16]. The results of the accumulated burnup distribution are shown in Figure 10, and it can be converted to pebble power per channel and per pass. The pebbles generated the highest average power in the first pass and gradually decreased with the number of passes.

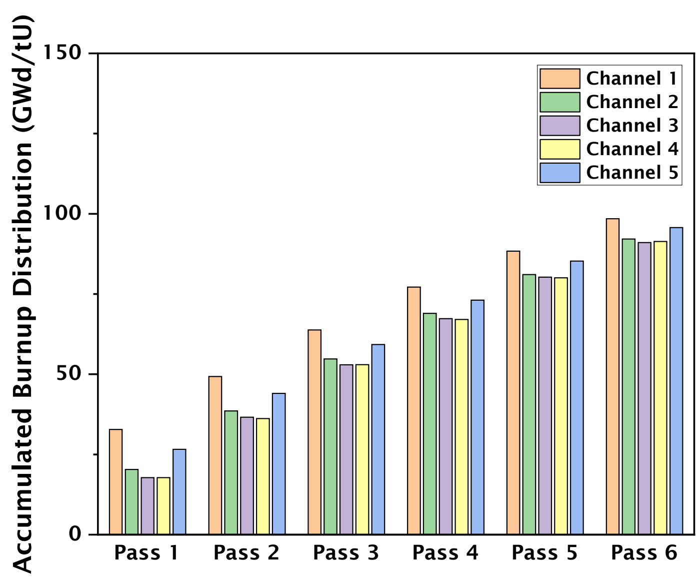

Figure 10. Accumulated burnup distribution per channel and pass.

The ORIGAMI depletion was calculated by using a series of transit histories. A single radial flow channel with 22 transit zones was modeled in each transit history to evaluate the uncertainty of the radial flow channel. The transit history was randomly selected by considering the probability that a pebble flows in a specific radial channel, which was assumed to be equal to the channel's volume fraction. A pebble was assumed to stay within one radial channel for each pass it took. Moreover, the uncertainties of the pebble transit time (±2 days) and pebble power (±2%) were considered at each pass. The average transit time was about 152.4 days and was calculated considering the target burnup value of 90 GWd/tHM. After each pass, a non-irradiation (cooling) time of 4.5 days is considered based on the expected operational practice. This non-irradiation period is due to the pebbles reaching the bottom of the core where they are outside of the neutron flux and therefore not actively fissioning. In this study, a total of 20,000 depletion cases were sampled by using these approaches (Figure 11), and the results are discussed in the next section.

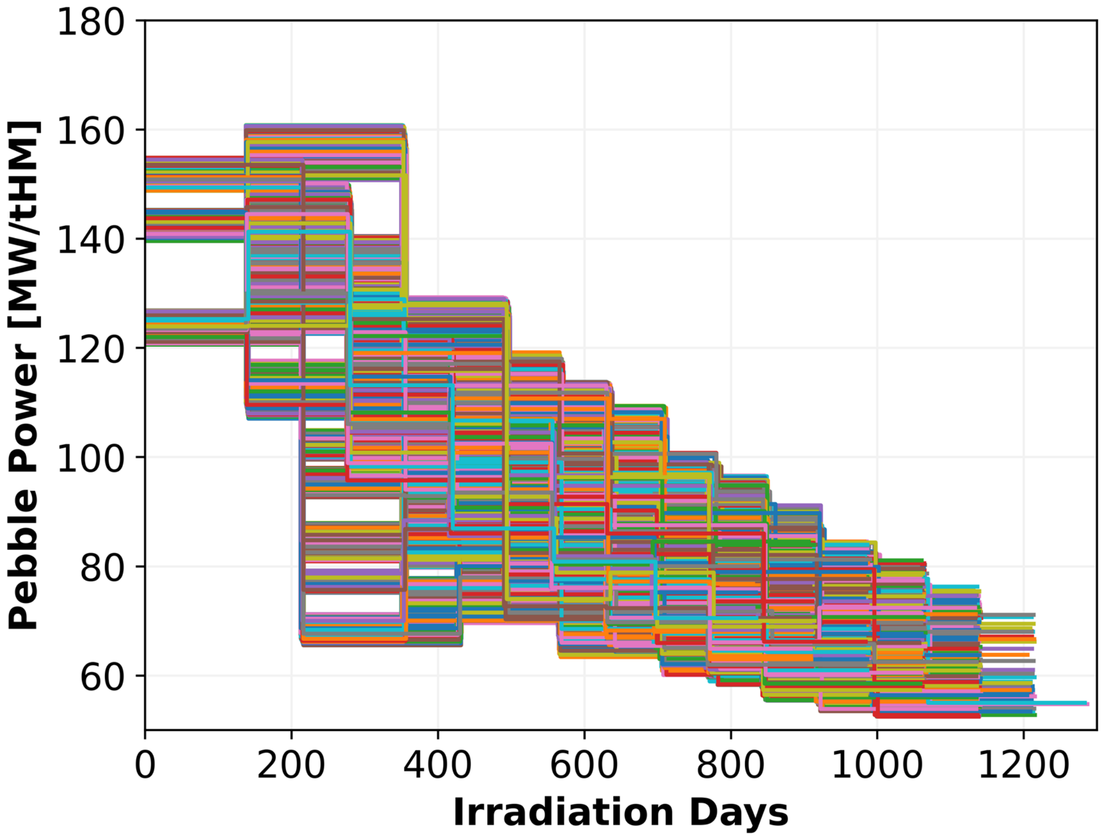

Figure 11. Average pebble power in each pass from the simulation.

2.5.4.2 Results

Using the sampled cases, the distributions of the discharge burnup and the isotopic compositions of the pebbles were evaluated. The fraction of retired pebbles after each pass through the core is shown in Figure 12. The pebbles that reached the burnup limit in a pass were not counted in the next pass. The results show that at the end of pass four, a small percentage (0.1%) of the pebbles were retired because they achieved the burnup limit. After pass seven, all the pebbles (100%) reached their target burnup. The fraction of pebbles that reached the target burnup after each pass and the average burnup of retired pebbles are shown in Figure 12 and Figure 13. The average discharge burnup value of the retired pebbles is about 90.13 GWd/tHM, which is close to the burnup limit.

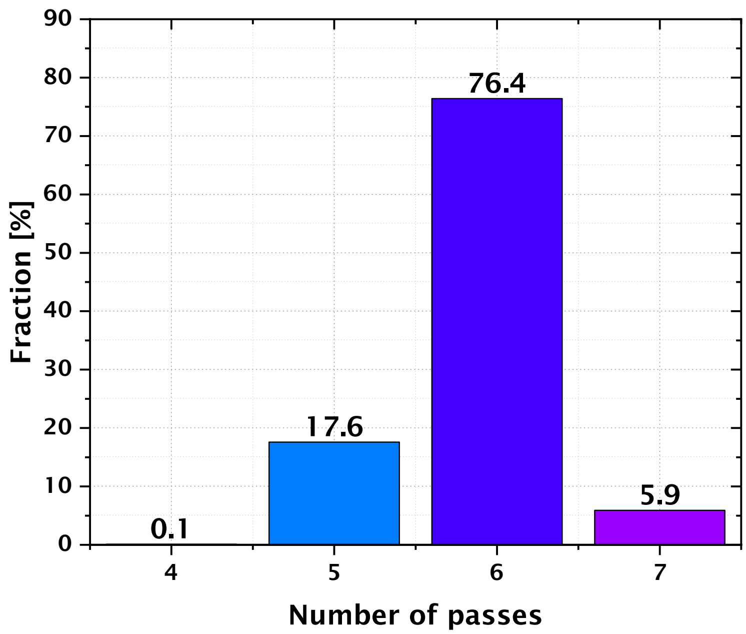

Figure 12. Fraction of permanently discharged (retired) pebbles from each pass.

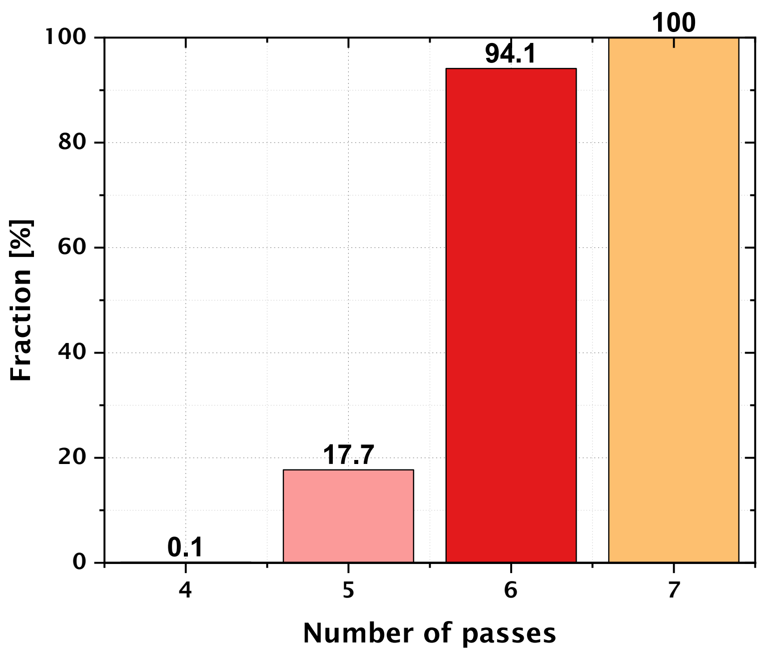

Figure 13. Fraction of pebbles that reached the target burnup after each pass.

Table 3 summarizes uranium and plutonium isotopic mass distributions for the retired pebbles and all burned pebbles after six passes in comparison to previous work done. The retired pebbles include the pebbles that reached the burnup limit after passes 4, 5, and 6. This table illustrates that a few milligrams of ^{234}U were present per pebble in the fresh fuel, and the amount gradually decreased in the next pass by capturing neutrons to produce ^{235}U. Uranium-235 was consumed continuously because it was the main fissile isotope. However, some of the neutron reactions with ^{235}U produced ^{236}U. The neutron capture reaction of ^{236}U produced ^{237}U, which decayed into ^{237}Np. Neptiunium-237 may capture a neutron and produce ^{238}Np, which rapidly decays into ^{238}Pu. Meanwhile, the neutron capture reactions of ^{238}U form ^{239}U, which quickly decays into ^{239}Np and then into ^{239}Pu.

To check these inventory results, the team compared them with the reported results of previous work done for the discharged pebbles at the end of pass 6 and used the PEBBED code [22]. Good agreement was observed in general between this work and reported results [21] with most of the discrepancies within the standard deviation [22].

Table 3. Comparison of average isotopic mass (mg) per pebble with previous work

Nuclide	Initial	Retired pebbles	All pebbles after pass 6 (this work)	All pebbles after pass 6 (previous work [22])
^{234}U	7.690	4.8 ± 0.1	4.8 ± 0.1	N/A
^{235}U	864.00	141.8 ± 19.6	145.5 ± 21.9	185 ± 11
^{236}U	3.974	116.1 ± 2.2	115.6 ± 2.5	N/A
^{238}U	8,124.34	7,755.8 ± 21.1	7,753.1 ± 13.6	7,690 ± 50
Total U	9,000	8,018.4 ± 28.9	8,019.1 ± 25.9	7,875 ± 51.2
^{238}Pu	0	2.3 ± 0.3	2.3 ± 0.3	2.8 ± 0.2
^{239}Pu	0	52.2 ± 6.0	53.6 ± 5.3	57 ± 9
^{240}Pu	0	35.5 ± 1.3	35.7 ± 0.9	30 ± 3
^{241}Pu	0	19.3 ± 1.5	19.7 ± 1.1	28 ± 4
^{242}Pu	0	14.6 ± 1.6	19.4 ± 1.5	20 ± 2
Total Pu	0	123.9 ± 6.5	125.9 ± 4.6	137.8 ± 10.5

One interesting observation is shown in the ^{239}Pu distribution, as illustrated in Figure 14. Besides being produced, ^{239}Pu is also consumed because it has a high thermal neutron cross section. Most of the neutron interaction with ^{239}Pu results in a fission reaction; however, some of the interactions produce ^{240}Pu, which can capture neutrons to produce ^{241}Pu. Plutonium-241 is another fissile isotope, and similarly, some of its neutron absorptions lead to the formation of ^{242}Pu. It is also noticed that 20% of the pebbles have lower ^{239}Pu mass than average (represented by the "Outliers" in Figure 14), and they are the pebbles flowing next to the graphite reflectors that have higher thermal spectrum that increase the consumption of ^{239}Pu. Table 4 shows the average and maximum mass values of ^{239}Pu.

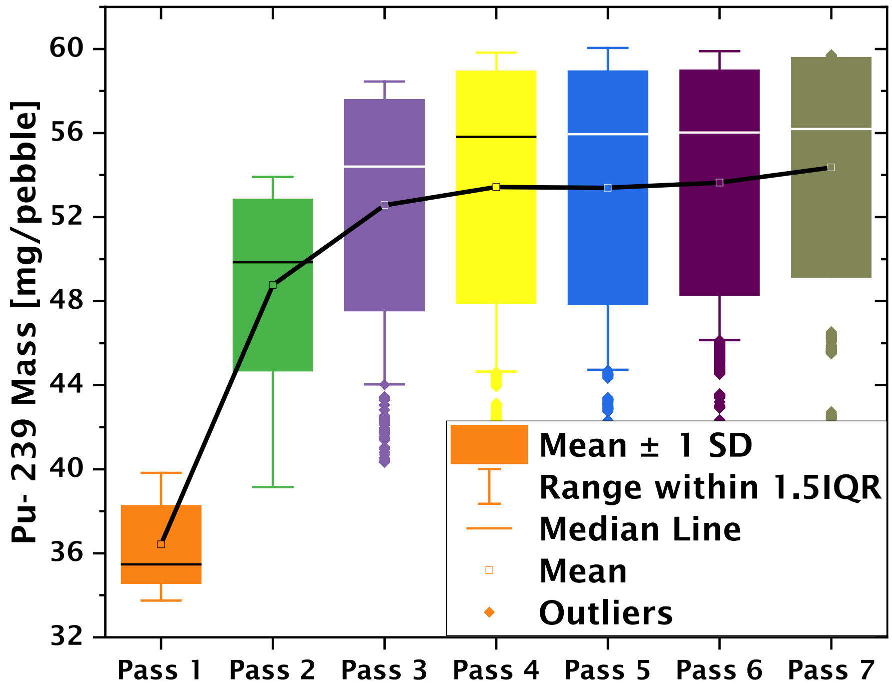

Figure 14. Statistical distribution of ^{239}Pu mass. Note: IQR = interquartile range.

Table 4. Average and maximum values of ^{239}Pu mass

Pass	Average ^{239}Pu mass (mg/pebble)	Maximum ^{239}Pu mass (mg/pebble)
1	36.4 ± 1.8	39.8
2	48.8 ± 4.1	53.9
3	52.6 ± 5.0	58.5
4	53.4 ± 5.5	59.8
5	53.4 ± 5.5	60.0
6	53.6 ± 5.3	59.9
7	54.4 ± 5.2	59.7

2.5.5 Spent fuel canisters

Retired fuel pebbles are assumed to be sent to canisters where they will be accumulated in the reactor system until full. Once full, they will likely be decoupled from the reactor system and are then assumed to be sealed and then become part of the spent fuel storage area. Although the design of these spent fuel canisters is not yet known, based on discussions with the reactor designers a canister that contained 2,000 spent fuel pebbles was modeled. A random selection of 2,000 retired pebbles was used to simulate the condition in a spent pebble canister. Of these pebbles, 83.6% had undergone six to seven passes. The average burnup of these 2,000 pebbles was 90.079 ± 3.448 GWd/tHM, where the burnup uncertainty is due to uncertainties in pebble transit time and power. Figure 15 shows the fraction of permanently discharged pebbles after each pass. Note, the fraction of the randomly selected pebbles is the same as previously shown for the entire population of fuel pebbles in the reactor. The total plutonium content of a permanently discharged pebble was approximately 124.2 mg.

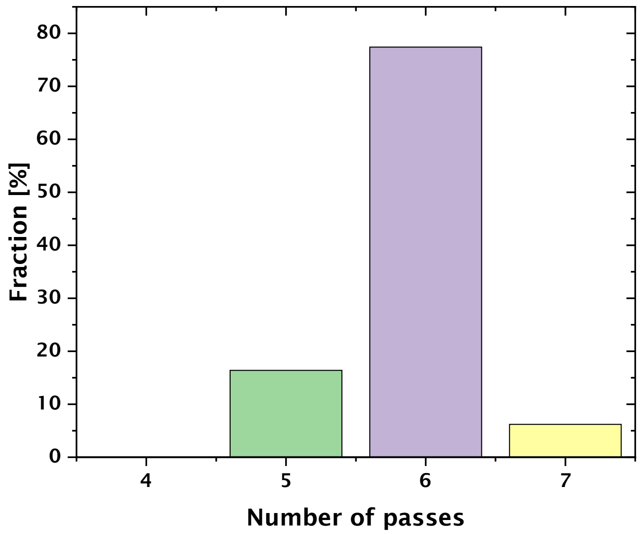

Figure 15. Fraction of retired pebbles after each pass for a randomly selected sample of 2,000 pebbles.

Assuming a canister contains 2,000 spent fuel pebbles and a random selection of spent pebbles, the team estimated that the total plutonium would be 248 g based on the assumptions used in this work (e.g., 9.6% enrichment and 90 GWd/tHM target burnup). Uncertainty in total plutonium mass in a canister was small because of error cancelations when the pebbles were randomly selected. Table 5 shows the total mass of uranium and plutonium and the main isotopes per canister. The values are rounded to the nearest gram.

Table 5. Average isotopic mass (g) in the canister

Nuclide	Average mass per pebble	Total mass per canister (unrounded)	Total mass per canister (rounded)
^{234}U	0.0048	9.65 ± 0.006	10
^{235}U	0.1423	284.636 ± 0.818	285
^{236}U	0.1160	232.022 ± 0.097	232
^{238}U	7.7555	15,510.92 ± 0.922	15,511
Total	8.0186	16,037.23 ± 1.266	16,037
^{238}Pu	0.0023	4.68 ± 0.013	5
^{239}Pu	0.0525	104.92 ± 0.263	105
^{240}Pu	0.0355	71.02 ± 0.056	71
^{241}Pu	0.0193	38.656 ± 0.067	39
^{242}Pu	0.0146	29.114 ± 0.071	29
Total	0.1242	248.392 ± 0.286	248

This analysis used the neutronic environment of the PBMR-400 equilibrium core to investigate the possible variation in the burnup and nuclide inventory of a single pebble. The assumed individual pebble characteristics (e.g., flow speed and operational history) were sampled from distributions that were assumed to be consistent with the equilibrium core. Based on these assumptions, the results show that some pebbles have some probability of achieving the burnup limit earlier than pass six, but on average, the pebbles required six passes to achieve the target discharge burnup. Moreover, an important isotope for nuclear material accounting, ^{239}Pu, had a maximum mass of about 52 mg per pebble at discharge, and the total plutonium of a canister with 2,000 randomly selected spent pebbles was estimated to be 248.4 ± 0.286 g based on assumptions used in this work.

2.6 NONDESTRUCTIVE MEASUREMENT SYSTEMS

BUMS will be an integral part of PBR operations and will determine when a pebble is to be permanently discharged based on its burnup history. Several non-destructive measurement systems exist that could provide measurements of fission products to infer the isotopic content of the irradiated pebbles.

2.6.1 Gamma Measurements

Work was performed to determine potential NDA measurement systems and methods for measuring fuel-pebble burnup and isotopic content. The results of the burnup modeling presented in the previous section were used to develop synthetic spectra using GADRAS to simulate gamma detector response. Measurement systems were investigated to determine the most effective approach and to consider system robustness, tolerance to high-radiation fields, and acceptable accuracies and uncertainties.

BUMS is used to measure a pebble when it exits the PBR core to determine whether the pebble should be recirculated in the core or sent to a spent fuel canister. The measured burnup value of a pebble is used together with a reactor core-modeling code to estimate the uranium and plutonium isotopic quantities in a spent pebble.

To provide some background information to assist in the design of such a measurement system, the expected gamma spectra of two representative irradiated pebbles were simulated and analyzed. The pebbles represent pebbles that have had three or six passes through the core, referred to as Pebble 3 and Pebble 6, respectively.

The synthetic gamma spectra from these two pebbles were generated using two simple GADRAS models. Figure 16 shows the 1D model used by GADRAS to generate synthetic spectra in this work. The model consists of two concentric spheres, with the inner sphere having a density of 2.12 g/cm^3 and representing the mixture of TRISO particles and graphite matrix, and the outer sphere having a density of 1.74 g/cm^3 and representing the graphite shell [16]. The mixture in the inner sphere included a homogenous blend of irradiated uranium dioxide (UO_2) fuel and graphite. According to the core simulations on PBMR-400, Pebble 3 had a burnup value of 52.65 GWd/tHM, and Pebble 6 had a burnup value of 88.41 GWd/tHM. The GADRAS model included 322 nuclides for Pebble 3 and 332 nuclides for Pebble 6 (the additional burnup of Pebble 6 produces more nuclides). The cooling time was assumed to be 100 h after the pebble's final exit from the core when the gamma measurements were simulated (based on the non-irradiation period when each pebble is at the bottom of the core outside the neutron flux). GADRAS treats most types of gamma source terms in materials comprehensively, including neutron-induced (e.g., [n, γ] reaction) photons, electron-induced (e.g., bremsstrahlung radiation) photons, and decay photons [23]. The detector response function provided by GADRAS for a coaxial high-purity germanium (HPGe) gamma detector with 110% efficiency was selected for this work. The distance between the detector and the pebbles was assumed be 100 cm, and the distance from the detector to the floor was also assumed to be 100 cm. For simplicity, no attenuator or collimator was used in this part of the simulation.

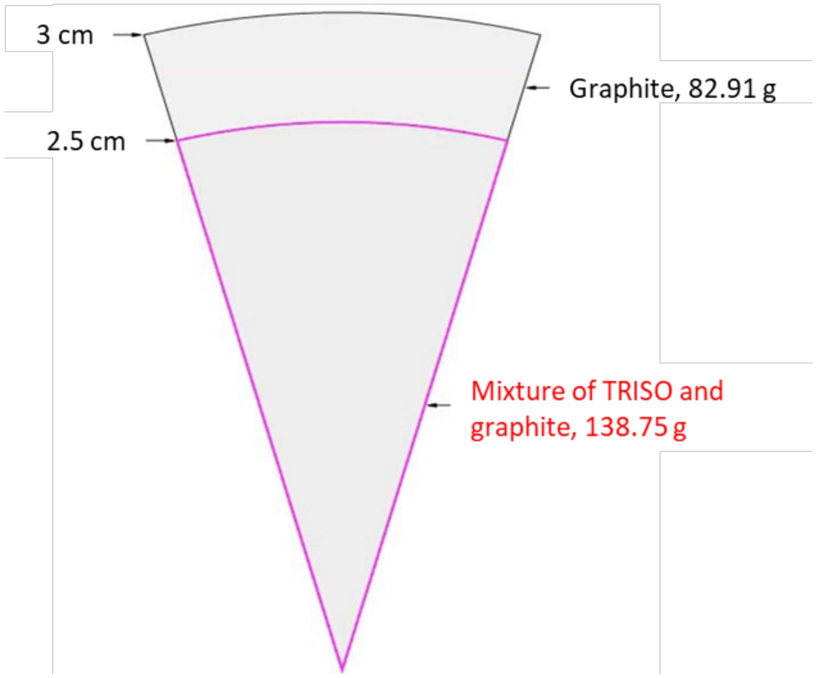

Figure 16. Pebble model used by GADRAS to generate synthetic gamma spectra.

Figure 17 compares the synthetic gamma spectra generated by the GADRAS models between Pebbles 3 and 6. Both spectra are similar except that (1) the backgrounds differ at some energy ranges and (2) the magnitudes of some gamma peaks differ. Figure 18 provides a close-up of Figure 17 for the energy range of 600–800 keV. Cesium-134 and cesium-137 are two commonly used burnup indicators that can be measured by gamma spectroscopy [24, 25]. The activity of ^{137}Cs in the irradiated fuel is linearly proportionate to the burnup upon discharge from the reactor. If the cooling time is known, the 30-year half-life of ^{137}Cs can be accounted for, and the discharge activity can be inferred based on the measured activity and the decay correction. The relative uncertainty in the burnup is the same as the relative uncertainty in the decay corrected activity, which is the square root of the number of net counts under the peak divided by the net counts under the peak. Figure 19 provides a close-up of Figure 18 to show the details of the 661.6 keV gamma line from ^{137}Cs and the neighboring 668 keV line from ^{132}I, which has a half-life of 2.3 h. The two main gamma lines from ^{134}Cs decay, 604.7 and 795.8 keV, are clearly shown and labeled in Figure 19, as is the prominent 661.6 keV line of ^{137}Cs. The other three gamma lines are labeled in this figure for information only, including the 765.8 keV line that has the largest peak area in these two spectra.

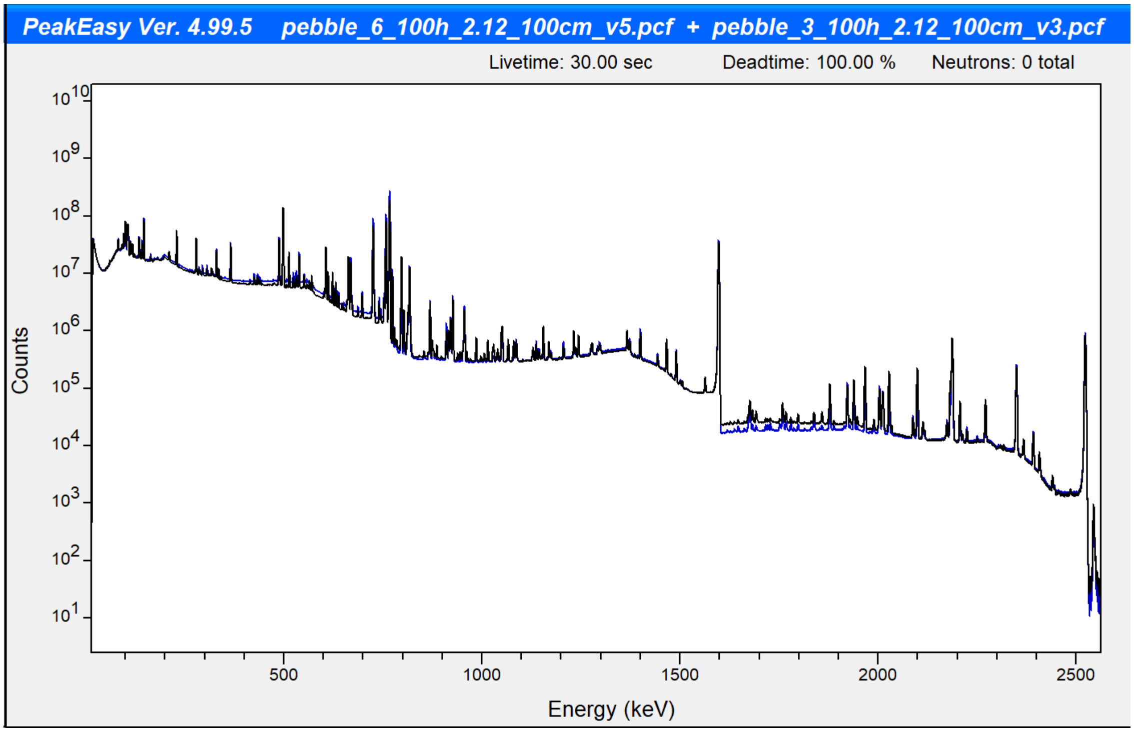

Figure 17. Synthetic HPGe gamma spectra generated by GADRAS for Pebble 3 (blue) and Pebble 6 (black).

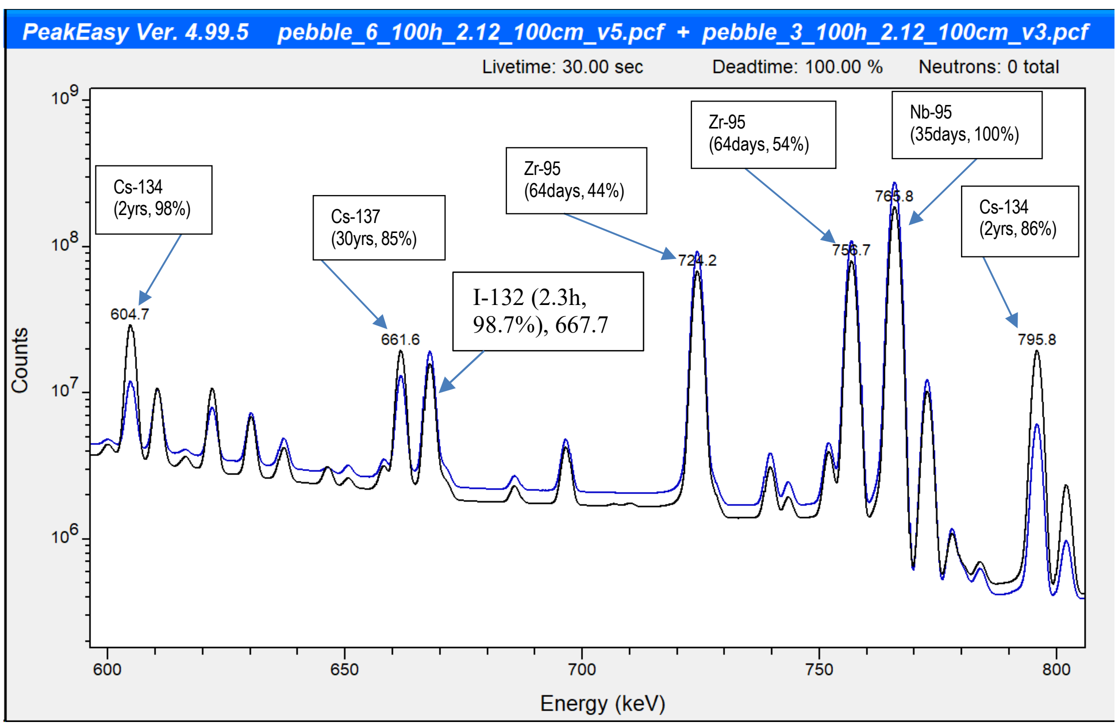

Figure 18. Close-up of the synthetic HPGe gamma spectra generated by GADRAS for Pebble 3 (blue) and Pebble 6 (black) in the energy range of 600–800 keV with the prominent gamma lines labeled. The labels include the primary emitter, its half-life, and the branching ratio (in percentage) of the respective gamma line.

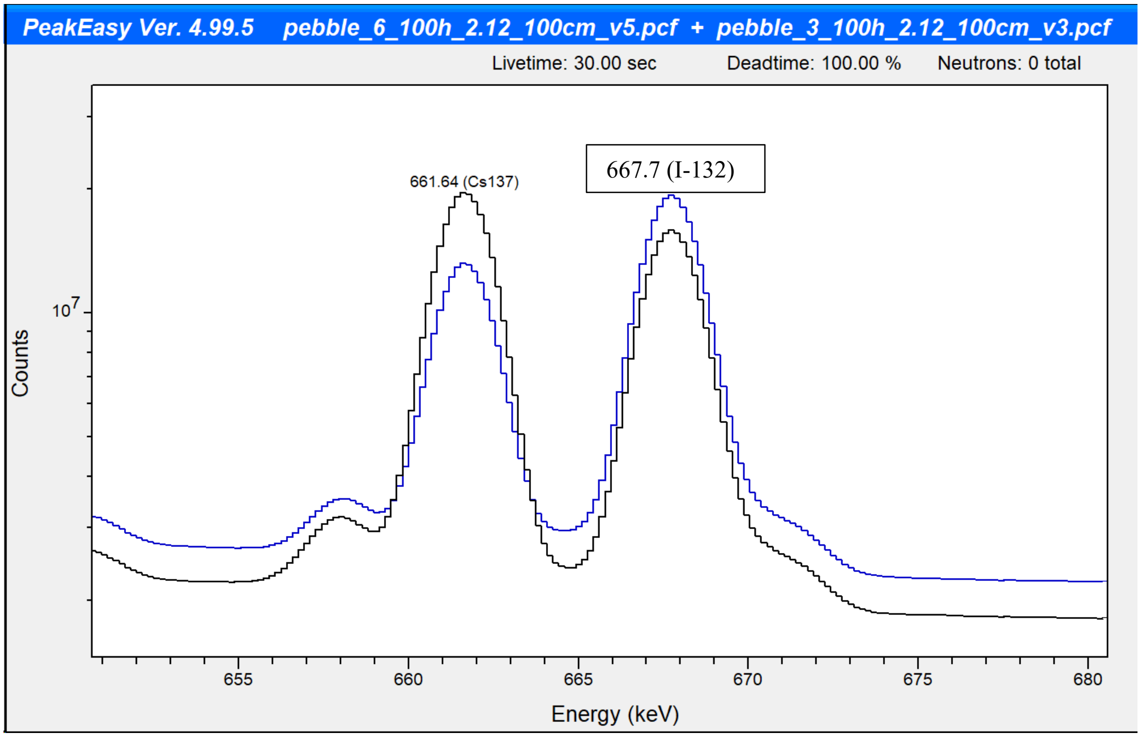

Figure 19. Close-up of the synthetic HPGe gamma spectra generated by GADRAS for Pebble 3 (blue) and Pebble 6 (black) in the energy range of 650–680 keV, showing the 661.6 keV line from ^{137}Cs and the 668 keV line from ^{132}I.

Europium-154 is another commonly used burnup indicator, but all the major gamma lines from europium-154 are either severely interfered or shadowed by neighboring gamma lines from other nuclides in these two spectra. That can be attributed to the relatively longer half-life of ^{154}Eu (8.6 years). For example, the 1,274.4 keV line, the most intense gamma line of ^{154}Eu, was severely interfered by the 1,278 keV line from ^{156}Eu, which has a half-life of only 15 days. Such interference makes it nearly impossible to use the peak areas of the subject gamma lines to infer the mass of the source nuclides.

A careful examination that weighs the concentrations of the nuclides from the burnup calculations with the 100 h decay time, half-lives, and branching ratios confirmed that the 604.7 and 795.8 keV gamma lines were predominately emitted by ^{134}Cs and were free of any significant interference from other nuclides. The same confirmation was made for the 661.6 keV line of ^{137}Cs. This means that the measured peak areas of 604.7 and 795.8 keV can be used to infer the mass of ^{134}Cs in the pebble being interrogated; likewise, the measured peak area of the 661.6 keV gamma line can be used to infer the mass of ^{137}Cs. Although the ^{134}Cs mass generally trends positively with burnup, it is also affected by the following two factors: First, the ^{134}Cs production depends on the neutron-energy spectra in which the fuel has been irradiated [24]. Second, the ^{134}Cs accumulation is affected by irradiation power because it takes longer to reach the same burnup under lower irradiation power, and some of the ^{134}Cs has already decayed away before the final discharge because of ^{134}Cs's relatively short half-life of 2 years [24]. These two factors are expected to have larger impacts on PBRs than LWRs because PBRs usually have hundreds of thousands of fuel elements (i.e., pebbles) in their cores, and the fuel pebbles travel through the core somewhat randomly several times. For example, the neutron energy spectra experienced by a pebble can vary significantly among different radial and axial zones. Moreover, the power profile of a pebble can also vary significantly because of the different flow paths taken through the core.

On the other hand, ^{137}Cs is a reliable burnup indicator because the fission yields of ^{137}Cs from ^{235}U and ^{239}Pu, the two primary fissioning nuclides in PBRs, are nearly identical. There are usually two ways to use the measured gamma-peak areas to determine the mass of the source nuclides and then burnup: first, using the absolute gamma-peak areas and, second, using the gamma-peak area ratios. Given that the absolute efficiency of an emitted photon being detected is usually unknown because the absolute efficiency depends on the self-attenuation in the source, transport from source to detector, detector efficiency, and so on, using peak-area ratios can circumvent this issue because only relative efficiency is needed among two subject gamma lines. For example, the relative efficiency in the energy range of 600–800 keV can be determined by comparing the measured peak areas of 604.7 and 795.8 keV and their respective branching ratios, given that both gamma lines were from the same source nuclide. However, because of the complications of ^{134}Cs production discussed earlier, the commonly used ^{134}Cs to ^{137}Cs gamma line ratio for burnup measurement cannot be applied to PBRs unless the dependencies of ^{134}Cs on neutron-energy spectra and power profile are accounted for. Using the absolute peak areas of the 661.6 keV gamma line of ^{137}Cs appears to be a more practical way to measure burnup for PBRs if the absolute efficiency is quantified by using calibrations. Such calibrations can be done by using a known ^{137}Cs source in a similar source-material mix and geometry and measurement configuration. The peak areas of the 661.6 keV gamma line of these two synthetic spectra of Pebbles 3 and 6 were also extracted by GADRAS. The peak-area ratio of Pebble 6 to Pebble 3 for the 661.6 keV gamma line was 1.72, which is close to the burnup ratio of Pebble 6 to Pebble 3 (1.68). The geometry of the BUMS system will be known and consistent, facilitating calibrations using the material and geometry.

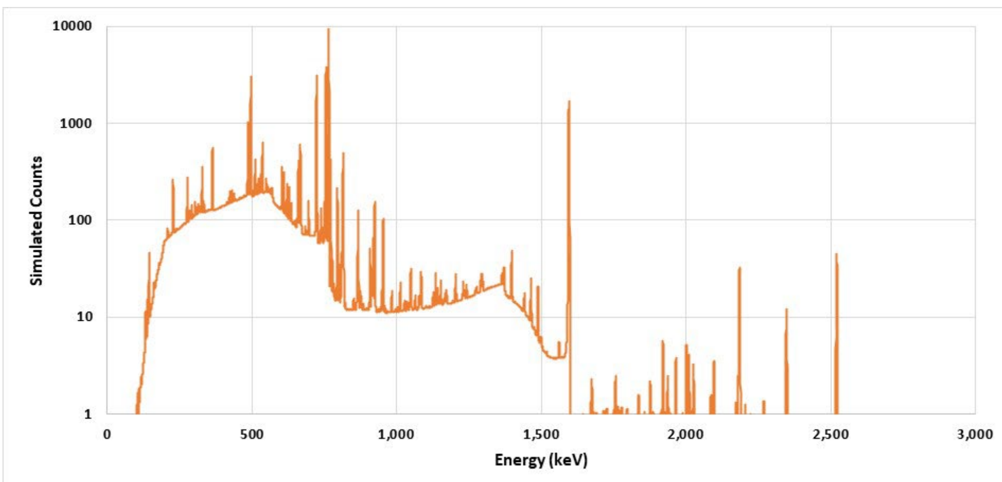

Figure 20. Synthetic HPGe gamma spectra for Pebble 3 with hypothesized attenuation and collimation applied.

As shown in the spectra above, the gamma counts for 30 s were high (e.g., tens to hundreds of millions counts per second in the 600–800 keV range), and the dead time was nearly 100%. The dose rate in the detector at 100 cm away from Pebble 6 was estimated to be 33.5 R/h based on a simple Monte Carlo N-Particle Transport calculation without attenuation or collimation applied, which was four orders of magnitude higher than the upper dose rate limit on a typical HPGe detector (~2.5 mR/h). This means that attenuation and collimation are needed to bring the count rates and dose rates down to acceptable levels. Because it is beyond the scope of this project to design actual attenuators and collimators for pebble measurements, a simple study was performed by simulating the use of a 1 in. thick stainless-steel attenuator and a 0.5 mm diameter collimator. Figure 20 shows an updated synthetic spectrum of Pebble 3 with the use of such attenuator and collimator. Compared to what is shown in Figure 18, the count rates in the 600–800 keV range were reduced by 4 orders of magnitude, and the count rates in the low-energy range (<200 keV) were reduced even further. The dose rate was reduced in similar orders of magnitude.

The total count rate is now reduced to ~30,000 counts per second, which would yield a dead time of 20%–30%. In this spectrum, a measurement precision of 2.5% in 30 s for the 661.6 keV peak can be expected.

In summary, based on the simulations of the expected gamma spectra of Pebbles 3 and 6 from a typical HPGe detector, the prominent gamma lines from ^{134}Cs and ^{137}Cs can be measured at a cooling time of 100 h. The simulated peak areas of the 661.6 keV gamma line were proportional to the burnups of Pebbles 3 and 6. Although the peak-area ratios of ^{134}Cs to ^{137}Cs have been routinely used to determine burnup values in LWRs, ^{134}Cs accumulation in a PBR is more complicated because of its dependence on neutron-energy spectra and power levels, which are different because of PBR's unique core and refueling characteristics. Using the absolute peak area of the 661.6 keV line of ^{137}Cs seems to be a more practical approach to determine a pebble's burnup, but calibrations will be required to determine the efficiency of a 661.6 keV photon being measured in the detector. Additionally, using a 1 in. thick stainless-steel attenuator and a 0.5 mm diameter collimator brought the count rates of Pebble 3 down to acceptable levels.

2.6.2 Neutron Measurements

Passive neutron measurements were also considered to measure the burnup of individual pebbles. Neutron NDA techniques could replace or augment gamma measurements to determine a pebble's burnup value. Neutron measurements can potentially provide a more sensitive measurement of the burnup than the gamma measurements can because the neutron emission of an irradiated pebble is a power function of burnup, whereas the photon emission is usually a linear function of burnup. Additionally, the coincidence neutron signal of an irradiated pebble can be explored because the coincidence neutron signal is usually linear with the fissile content of the spent fuel, with the caveat that the coincidence neutron signal from a pebble's fissile content is likely to be small due to the small fissile content in a pebble; therefore, the usefulness of coincidence neutron signal for a pebble remains to be proven. Figure 21 depicts a model of a typical passive neutron coincidence counter. The neutrons are measured by a series of helium-3 (^{3}He) tubes that are embedded in neutron-moderating materials such as polyethylene and graphite. Significant modifications of the detector geometry are expected to customize this detector for a pebble measurement. For example, the detector can be split into two half-cylinders to allow a pebble-containing pipe to pass through the middle of the detector. High-Z materials such as tungsten are needed to reduce the gamma-flux exposure on the ^{3}He tubes to reduce pileups.

Table 6 shows the neutron-emission rates from Pebble 6 determined at a cooling time of 100 h. Spontaneous fission accounted for 90.5% of the neutron emissions, and Pebble 6 emitted ~10,000 neutrons per second (n/s), which are expected to yield sufficient counting statistics from a passive neutron coincidence counter measurement.

One challenge of neutron measurement next to a PBR is the higher temperature environments than a conventional LWR because a PBR operates at a much higher temperature. The commonly used neutron moderator, such as high-density Polyethylene, in a neutron instrument would not withstand much greater than 104 °C temperatures [1]. An alternate neutron-moderating material, PolyEtherEtherKetone (PEEK), is a semicrystalline thermoplastic with excellent strength and ductility, has good neutron moderating power, and is suitable for continuous-use at temperatures up to 260 °C [2]. This new moderating material is promising and could allow the use of passive neutron NDA techniques to be used next to a PBR as part of burnup measurement system.

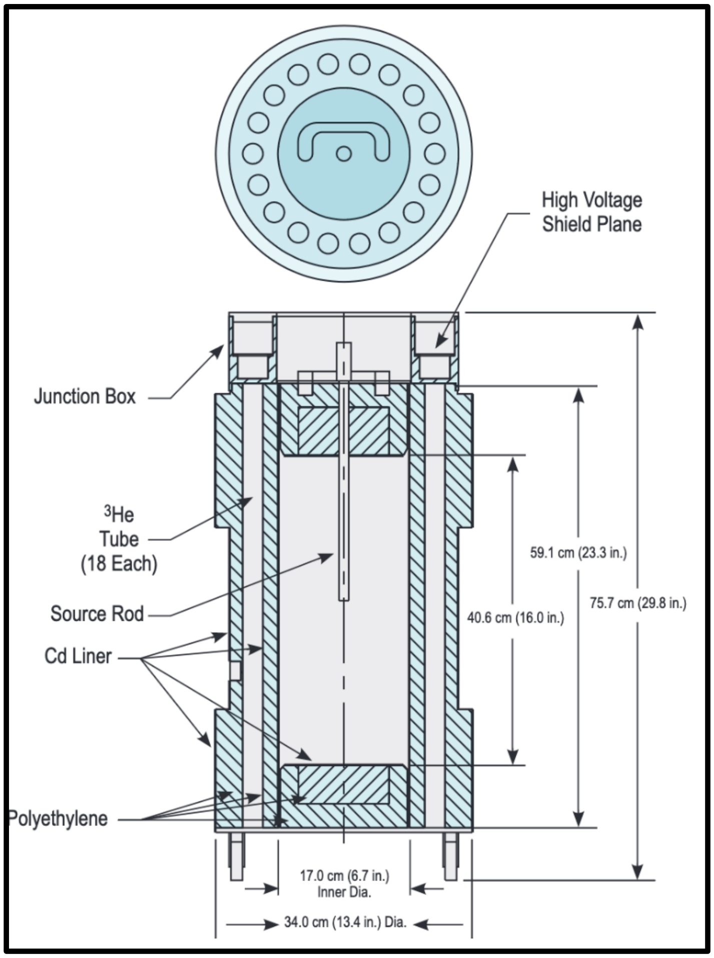

Figure 21**. An example of a PNCC: JCC-31 a high-level neutron coincidence counter[26]..**

Table 6. Neutron-emission rates from Pebble 6

Burnup	**(α, *n*)**	**Spontaneous fission**	**Total**
(GWd/tU)	(n/s)	(n/s)	(n/s)
90	958.59	9,094.5	10,053.0

2.7 STATISTICAL APPROACHES

Although this is a subset of MC&A, since statistical approaches will play such a large role it is treated as a separate focus area. Many fuel pebbles (approximately millions) will pass through the reactor during its lifetime and measuring, modeling, and counting them will pose challenges. Additionally, TRISO fuel is inherently low density with low uranium and plutonium content per pebble. These two features will cause uncertainties in counting, measuring, and modeling the fuel pebbles to add up. Statistics will play a large role in determining the setpoints of BUMS to ensure both safe operations as well as effective material accounting.

2.7.1 Rounding

PBRs have many discrete objects (i.e., pebbles) containing small quantities of nuclear material that will require special considerations designed into the MC&A system to handle rounding for inventory and transaction reporting. This should not be construed as an issue for MC&A or physical protection from a theft/diversion perspective because all work to date has focused on balancing around numbers of pebbles. It is simply an accounting and reporting nuance due to large numbers of small items that must be addressed for periodic MC&A reporting. The MC&A system should have the capability and approach to adjust the inventory for rounding.

The reason this capability will be needed is that this becomes an issue in reporting NMMSS because of reporting units (i.e., number of decimal places) and different methods of data grouping used for inventory and transaction reporting.

2.7.1.1 Rounding Guidance for Reporting

Although many facility MC&A systems may—and many do—carry additional decimal places for transaction and inventory reporting purposes to NMMSS, reporting units for enriched uranium and plutonium are constrained to the nearest gram. Entries for natural uranium are rounded to the nearest kilogram.

For both the NRC and DOE, the guidance for rounding is

- quantities equal to or greater than 0.5 of the reporting unit are rounded up to the next whole reporting unit, and quantities less than 0.5 are rounded down; and

- transfers of multiple discrete items of 0.5 of a reporting unit or less but of the same material type (e.g., U, Pu) are typically summed to a total weight of that material type before applying rounding criteria. (This guidance is very applicable for handling the plutonium content which is very small on a per-pebble basis.)

The following sections provide an overview of the types of differences that might be observed, depending on choices in grouping, rounding, and summing of nuclear material weights for MC&A reporting.

2.7.1.2 Uranium Rounding Example

Assuming a 15,000 TRISO particle loading in a 15 wt.% enriched pebble and a reactor vessel containing 452,000 pebbles, gram quantities are shown in Table 7.

Table 7. Unrounded pebbles and reactor uranium values

	Unrounded Uranium in Pebble (g)	Unrounded Uranium in Reactor (g)
^{234}U	0.00769	3,475
^{235}U	0.86393	390,498
^{236}U	0.00397	1,796
^{238}U	8.12372	3,671,922
Total U	8.99932	4,067,692

2.7.1.3 Rounding Based on Two Example Groupings: Pebble and VP55 Fresh Fuel Container

Multiple size groups could be used to receive and transfer fuel into the reactor. For the example shown in Table 8, a single pebble and a VP55 (fresh fuel container ~350 pebbles) are compared to illustrate the impact of rounding and summing by these two potential groupings. Column (B) shows the actual uranium content in the reactor vessel. Rounding and summing the total reactor mass at the pebble level are shown in column (C), and the difference from actual is shown in column (C–B). Rounding and summing at the VP55 level are shown in column (F), and column (F–B) shows rounding and summing differences.

Table 8. Rounding at various groupings and differences

	(A) Mass per pebble (g)	(B) Mass in reactor unrounded (g)	(C) Mass in reactor rounded at the pebble level (g)	(D) Mass per VP55 drum unrounded (g)	(E) Mass per VP55 Drum rounded at drum level (g)	(F) Rounding at drum level ×1,292 drums (g)	(C-B) Reactor mass accounting difference if rounded at pebble level (g)	(F-B) VP55 drum mass accounting difference if rounded at drum level (g)
^{234}U	0.00769	3,475	—	2.6912	3	3,876	(3,475)	401
^{235}U	0.86393	390,498	452,000	302.3771	302	390,184	61,502	(314)
^{236}U	0.00397	1,796	—	1.3909	1	1,292	(1,796)	(504)
^{238}U	8.12372	3,671,922	3,616,000	2,843.3024	2,843	3,673,156	(55,922)	1,234
Total U	8.99932	4,067,692	4,068,000	—	—	4,068,508	308	816

Although the differences are much smaller when rounding at the container level, any difference will result in a discrepancy to be resolved with NMMSS during the reconciliation process. This is due to how values are grouped and summed for reporting. Periodic rounding adjustments by the process outlined in NRC regulations will be necessary to balance licensee records with NMMSS.

2.7.1.4 Spent Fuel Containers and Plutonium Content

The difference becomes much more pronounced when the plutonium content in the spent fuel is considered because of the small amount in each pebble (about 0.1242 g, see Table 9). Using the PBMR-400 and assuming a burnup of 90 GWd/tHM and a spent fuel container of 2,000 pebbles, the differences are shown in Table 10. Individually each pebble will round to zero. The total plutonium content in the 2,000-pebble group rounded at the pebble level is zero; however, the actual content in the container is 248 g. Therefore, declaring plutonium production and rounding at the spent fuel container level is a more reasonable and accurate approach.

Table 9. Plutonium rounding spent fuel

	Unrounded plutonium in a pebble (g)	Rounded at the pebble level and summed (g)	Rounded at the spent fuel container (g)
^{238}Pu	0.0023	0	5
^{239}Pu	0.0525	0	105
^{240}Pu	0.0355	0	71
^{241}Pu	0.0193	0	39
^{242}Pu	0.0146	0	29
Total Pu	0.1242	0	248

2.7.1.5 Rounding Summary

Approaches to handle rounding adjustments are common in MC&A systems, especially for fuel cycle facilities. Although not prevalent in LWRs, PBRs will behave more like fuel cycle facilities with respect to rounding. The basis and approach for rounding adjustments should be covered in the facility's MC&A plan, which should describe how rounding adjustments are captured within the MC&A system and subsequently reported to NMMSS.

2.7.2 Statistical Approaches for PBR Operations and MC&A

To account for the SNM in spent and used fuel storage, the operator will use BUMS measurements, the reactor codes, or a combination of the two. To ensure the accuracy and validity of these methods, a statistical sample of pebbles entering these two zones will be selected for more extensive NDA or DA to get a better representation of the isotopes and masses. These results will be used to calibrate the BUMS burnup measurements and validate the reactor code predictions for burnup and SNM content.

A pebble entering either the spent or used fuel storage can be treated as a manufactured product. Therefore, the statistical techniques employed here should more closely resemble those in manufacturing and process control than traditional MC&A approaches. Thus, the statistical sampling of pebbles for NDA or DA should be chosen in a manner analogous to choosing products to be analyzed in a manufacturing process. An added advantage to this approach is the ability to monitor reactor performance for both operations and MC&A that is not possible in other reactor designs.

Reactor code performance in predicting SNM content for LWRs required special measurement campaigns or fuel reprocessing, but for PBRs the comparison of BUMS values with reactor code-predicted values accomplishes this in real time. The transit time and path of the pebble fuel through the reactor determine the plutonium production and uranium depletion. So, the SNM distribution will serve as an indicator of performance. Analyzing the predicted and measured values affords opportunities to adjust operating parameters, fuel design, and other properties to optimize reactor performance and fuel use. As for MC&A, this approach provides the requisite information to validate declared values and evaluate if the reactor is operating as expected or not.

2.7.2.1 Statistical Approaches for Burnup Measurements and Calculations

Two statistical approaches for the measurement systems (i.e., BUMS) and reactor codes are expected to be used in PBRs. At this point, the relationship between the reactor codes and BUMS with respect to MC&A declared values is still being evaluated. Specifically, it has not been determined if BUMS will be used to both determine the burnup that is used to decide when to discharge a pebble and provide the values used in the SNM declarations for burnup/production or if BUMS will strictly support the burnup discharge decision, and the reactor codes provide the values for the SNM declared values. Either approach may be adequate for the purposes of MC&A because of the diluteness of the SNM in the fuel and the fact that MC&A's main goal is the detection of theft or diversion that is based on the removal of a whole pebble. The total plutonium in a spent fuel pebble is so small (i.e., ~0.12 g) that slight variations in this amount have no practical impact on how the MC&A physical protection programs are designed to accomplish that goal.

As a reference, LWRs provide a declared value of SNM in the irradiated nuclear fuel based solely on reactor codes. The regulatory guidance for this practice is provided in ANSI N15.8-2009 [3] Section 9 on SNM calculations for power reactors.

> **9 SNM Calculations**
>
> **9.1 Element and Isotopic Computations**
>
> Methods of computation shall be established and utilized for determining the total element and isotopic composition of SNM in irradiated nuclear fuel assemblies and fuel components. The computed values are the basis for shipment documents, as required in 10 CFR 74.15, and material status reports, as required in 10 CFR 74.13.
>
> **9.2 Analysis of Results**
>
> Refinement of the element and isotopic computations used in determining the SNM content of irradiated fuel should be considered as new technologies evolve. For reprocessed fuel, this may include a collection and comparison of reprocessing plant measurement data with computed data for fuel assemblies.

However, PBRs will employ an NDA measurement (BUMS) that is part of the fuel-handling system, which can be used in addition to the reactor code. Therefore, there are two possible "methods of computation" that could be used in determining the SNM content in the irradiated pebbles. One approach would be to use reactor codes like LWRs. The other approach would be to use the BUMS measurement. The method, or combination of methods, that will be the most accurate predictor of the SNM content in the irradiated pebbles remains to be determined.

Additionally, integration of these two approaches provides a unique opportunity to monitor reactor performance. For LWRs, knowledge about reactor code performance in predicting irradiated SNM content historically was only achieved from special measurement campaigns or from fuel reprocessing (see Section 9.2 ANSI N15.8-2009). Conversely, through statistical comparison of BUMS with the reactor code-predicted values, PBRs can achieve this in real time. The resulting SNM distribution is also an indicator of reactor performance because factors such as transit time and path of the pebble through the reactor determine the plutonium distribution and uranium depletion. By analyzing the predicted and measured values, this information can provide opportunities to adjust operating parameters, fuel design, and other characteristics to optimize performance and fuel utilization.

From a process control and MC&A perspective, there are three statistical decisions to be considered for PBRs.

1. Burnup measurement discharge decision
2. Burnup measurement versus reactor code comparison

3. Analysis of variance between reactors in a modular multiunit deployment

Statistical Model for Burnup Discharge Decision

A burnup measurement uncertainty will be associated with the decision on when to discharge a pebble. PBRs have a maximum allowed burnup for the fuel plus a burnup threshold that serves as the decision point for discharge. One goal of BUMS is to determine if the current burnup has reached the discharge threshold. The relationship between the discharge threshold and maximum burnup is a buffer to make sure that a pebble is not returned to the reactor such that if it took the highest energy path it might exceed the maximum allowable burnup. The applicable uncertainties are for both the BUMS and the reactor models, which provide an estimate of the highest energy path taken by a pebble. The decision point for discharge is a balance between Type I and Type II[9] statistical errors:

- Type I error—Discharging a pebble when it should have been returned to the reactor, resulting in underutilized fuel.

- Type II error—Returning a pebble to the reactor when it should have been discharged, resulting in a pebble exceeding the maximum desired burnup and creating possible safety concerns and/or less-than-desirable operational performance.

In some situations, such as burial of nuclear waste, the statistical approach used is that the measured value plus the measurement uncertainty at 2σ must be less than the maximum allowed limit. Using this approach, a decision threshold would factor in the maximum allowed (or never exceed burnup) minus the maximum likely burnup from an additional pass if placed back in the reactor. The measured burnup plus 2σ would then be less than or equal to that limit (Figure 22).

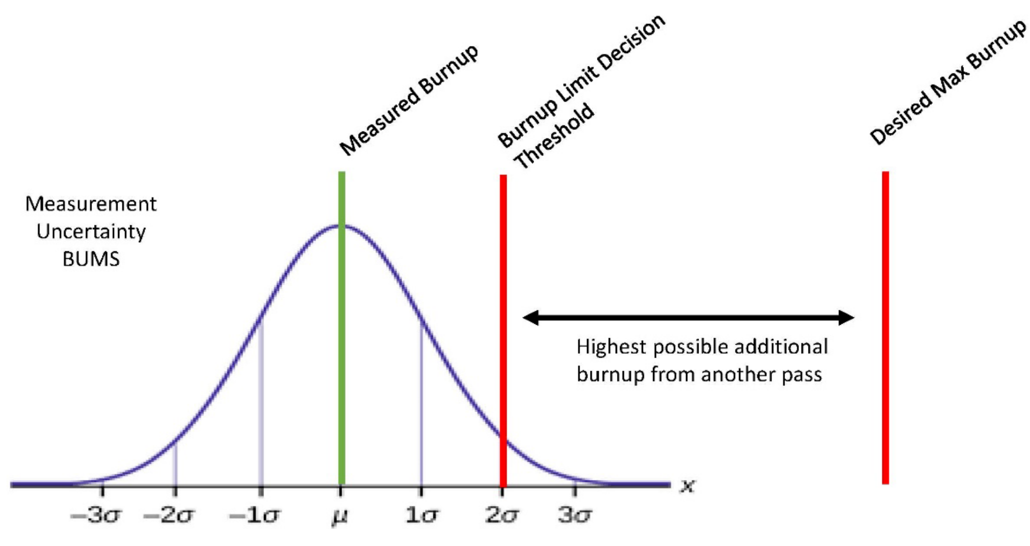

Figure 22. Two sigma limit illustration.

[9] A Type I error (false positive) occurs if an investigator rejects a null hypothesis that is actually true in the population; a Type II error (false negative) occurs if the investigator fails to reject a null hypothesis that is actually false in the population.

The issue with the 2σ limit approach is that statistically there will be pebbles that will inadvertently be placed back in the reactor that should have been discharged. (This would be a Type II error, assuming it was below the target threshold when it really is not.)

In some processes, such as the 6σ approach in quality or other approaches used in criticality, an additional buffer is added to allow for the measurement method uncertainty and reduce the probability of Type II errors. By setting the rejection threshold at a 4σ to even 6σ buffer, the probability of a Type II error can be minimized to whatever extent is desirable. However, this is at the expense of potentially underutilizing the fuel.

Figure 23 provides an illustration of what this would look like statistically. Although the burnup limit detection threshold and desired max burnup remain the same, the average measured burnup target is reduced by additional standard deviations to create a larger buffer, reducing the probability of Type II statistical errors.

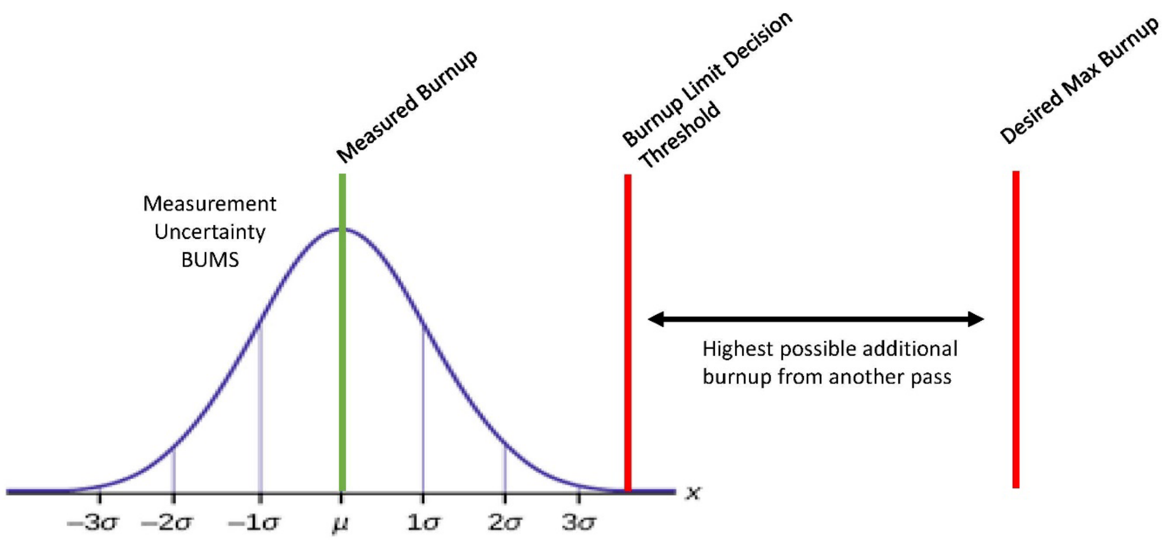

Figure 23. Setting limit to reduce Type II errors.

Once the BUMS measurement uncertainty has been established and the reactor codes are validated, a distribution for the possible burnup after an additional pass of the pebble can be used to establish the burnup limit decision threshold precisely in agreement with the desired balance of statistical errors.

Here we construct an example using the burnup data from the model in Section 2.5. Since we lack true values for burnup, we will treat the simulated values as "true" values. We consider the maximum allowable burnup to be 100 GWd/tHM for safety reasons as in [14, 27]. In the modeling data, the largest difference in burnup for a pebble that was below100 GWd/tHM, and after an additional pass was above 100 GWd/tHM, was 16.771 GWd/tHM. For example, the burnup for a pebble after pass five was at 83.246 and after pass six was at 100.017 GWd/tHM. So, we will consider the highest possible additional burnup from another pass to be 16.771 GWd/tHM. *(Note: In a real system, the maximum additional burnup will not be a fixed number but will instead be a distribution dependent on many factors including the current contents of the pebble, the probability of the pebble going down a given path, and the transit time of the pebble. With enough operating history and validated reactor codes this distribution can be accurately estimated, and the associated probabilities would be used in the calculations of the statistical errors and the burnup threshold.)*

Given the 16.771 GWd/tHM maximum additional burnup, if there was no measurement error, we could set 83.229 GWd/tHM as the decision point of measured pebbles and have no chance of exceeding the maximum allowable burnup of 100 GWd/tHM. However, all measurements have uncertainty. Since the BUMS measurement uncertainty is unknown, we will suppose the BUMS measurement follows a normal distribution with mean μ equal to the true value and a standard deviation σ = 2.5%.

We will consider the decision to discharge a pebble correct if the true value of burnup for that pebble exceeds the burnup threshold, and we will consider the decision incorrect in the case where it does not. So, a Type I error occurs when a pebble is discharged when it should not have been (i.e., the measured value of burnup for a pebble is greater than the decision point for the measured burnup but the true value of burnup is below the burnup threshold). Type II error occurs when a pebble is returned to the reactor when it should not have been (i.e., the measured value of burnup for a pebble is less than the decision point for the measured burnup but the true value of burnup is greater than the burnup threshold). We will represent the true burnup value with x, T is the burnup threshold, the measured value is m, and the L is the decision point for the measured burnup. So, we can formulate the Type I Error (α) and Type II Error (β) as follows:

$$\text{Type I Error: } \alpha = P(m \geq L | x < T)$$

$$\text{Type II Error: } \beta = P(m < L | x \geq T)$$

Both errors will depend on the underlying true burnup value of the pebble being measured. The further away the true burnup is from the threshold, the lower the errors will be. Without having the joint distribution for x and m, it is not possible to directly compute these errors. Instead, we can compute these probabilities for values for x, and at values of L or, we can use the results of the model study as an empirical distribution for x to approximate the errors. We do both here.

Based on the model results, we set T = 83.229 and consider L = 83.229, L = 81.1990, L =7 9.2657, L = 76.2267, and L = 72.3730, where the values of L are chosen corresponding to 83.229 being 0, 1, 2, 4, and 6σ from the respective L based on the assumption σ is 2.5%. We can then compute the associated errors for a pebble at a given true burnup. For example, if we set the threshold at 83.229 GWd/tU and the limit for the measured values at 79.2657 GWd/tHM, we can find the Type I error associated with a pebble that has a true burnup value less than 83.229 GWd/tHM such as 83.1703 GWd/tHM.

Let $Y \sim N$ (83.1703, 0.025% × 83.1703), then

$$\text{Type I Error: } P(m \geq 79.2657 | x = 83.1703) = P(y \geq 79.2657) = 0.9698.$$

Similarly, we can find the Type II error associated with a pebble that has a true burnup beyond the threshold for example 85.6222 GWd/tHM. Let $Y \sim N$ (85.6222, 0.025% × 85.6222), then

$$\text{Type II Error: } P(m < 79.2657 | x = 85.6222) = P(y < 79.2657) = 0.00149.$$

We can perform these calculations for every possible true burnup value. Here we include a plot based on the five choices of L for the associated Type I errors (Figure 24).

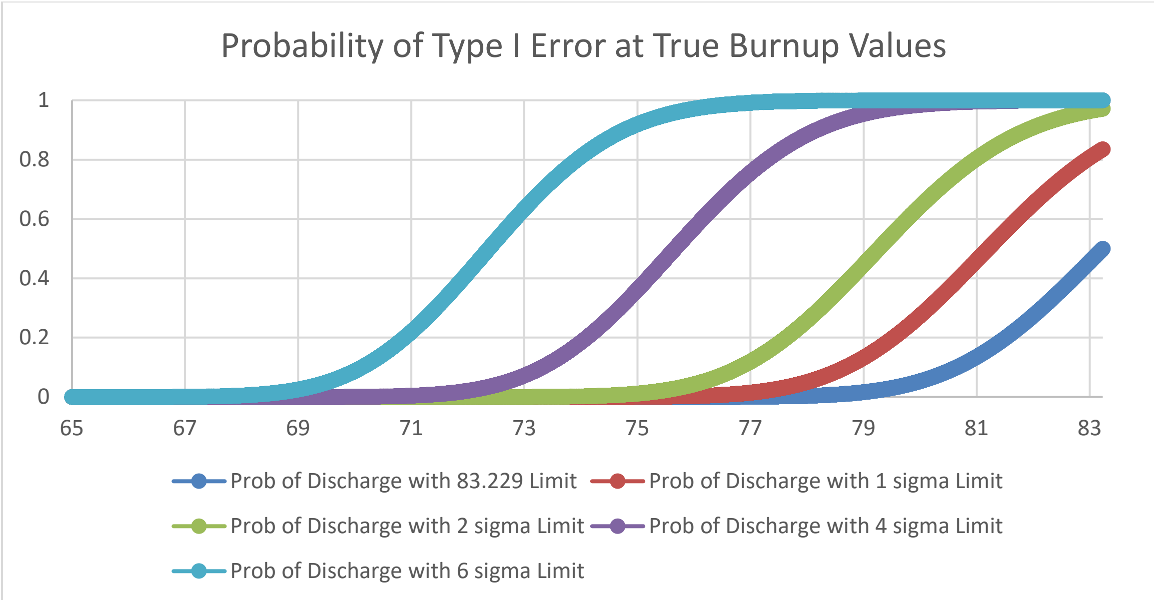

Figure 24. Probability of Type I error at true burnup values.

As expected, the probability of Type I error decreases the further the true burnup value is from the threshold. Moreover, we notice that as we decrease the decision point *L* (i.e., increase safety margins) for the measured values, we see that the probability of Type I error at each true burnup value increases. Both agree with our intuition because a true value of burnup being considerably lower than the threshold would make it unlikely that it would be discharged and discharging a pebble at a lower measured value increases the likelihood that it was discharged prematurely.

Here we include a plot based on the five choices of *L* for the associated Type II errors (Figure 25). Note, the probability of Type II error at the 4σ and 6σ limits are both near zero. Therefore, the probability curves overlap in the plot.

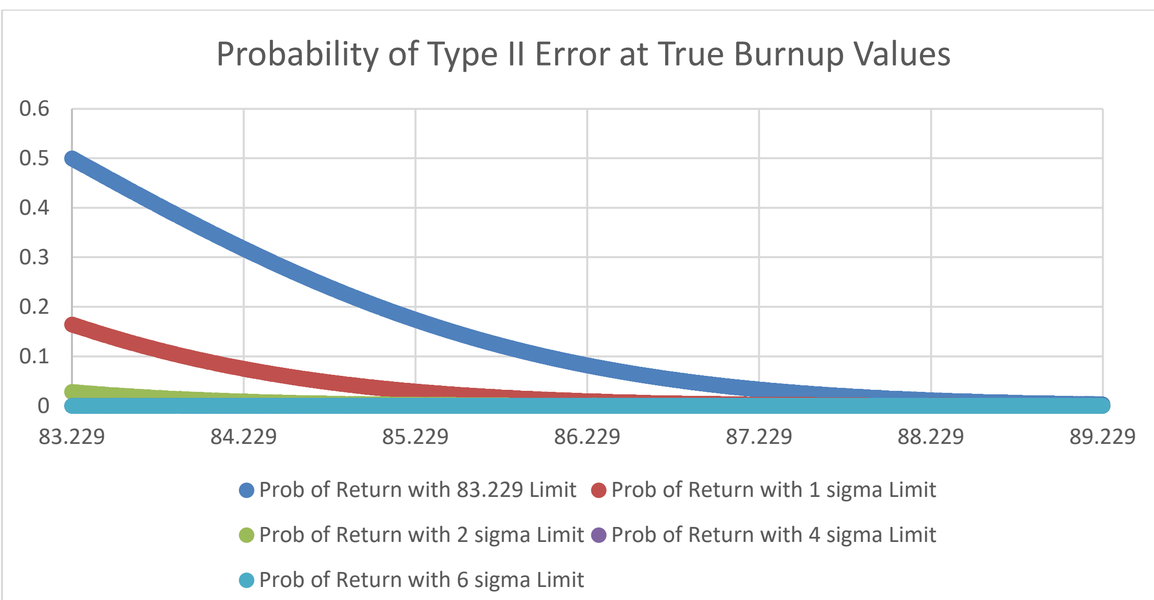

Figure 25. Probability of Type II error at true burnup values.

Once again, the results are consistent with what we expect. The Type II error will be lower if we set the decision point for measured burnup lower and the Type II error will be lower as the true value increases away from the threshold. The results of both this plot and the Type I error plot demonstrate that choosing the decision point for the measured burnup is a balance of Type I and Type II error.

This is analogous to a manufacturing process—the tighter the tolerances (i.e., increased safety margins), the more units would not meet the specification and be rejected (Type I error when a non–fully burned pebble is withdrawn from the reactor). The lesser the tolerances, the greater the probability that a nonconforming unit would pass inspection and be shipped as product (Type II error when a pebble is reinserted into the reactor that results on an overburn situation).

Recall that if we want to know the total Type I or Type II of the system we would use:

$$\text{Type I Error: } \alpha = P(m \geq L | x < T)$$

$$\text{Type II Error: } \beta = P(m < L | x \geq T)$$

As stated above it is not possible to compute these for a given L and T without having the joint distribution. We can, however, estimate the values by using the results from the model as an empirical distribution of pebble burnup and using the previously mentioned normal distributions as a source of measurement error at each of the values in the distribution.

First, we can estimate the total Type I error by using the set of all burnup values less than 83.229 GWd/tHM found in the modeling results as the true burnup values and using the discharge probabilities we calculated for each of those individual points and multiplying this by the probability that the pebble was not discharged before reaching that point as the proportion of discharged pebbles for that given true burnup value. We can then take the sum of all these proportions and divide it by the total number of pebbles from the model that had burnup values less than 83.229 GWd/tHM to produce a proportion of discharged pebbles that should not have been (i.e., the estimated Type I error). We include here a summary of these estimators in Table 10.Table .

Table 10. Total number of measured pebbles with true burnup less than 83.229 GWd/tHM, *n* = 47752.

Decision point for measured burnup (GWd/tHM)	Average number of pebbles discharged prematurely	Proportion of pebbles discharged prematurely, Type I error estimate
83.229	516.6673	0.01082
81.1990	1,481.6804	0.03103
79.2657	2,945.2306	0.0617
75.6627	6,780.3562	0.1420
72.3730	8,958.1836	0.1876

Again, we notice that decreasing the decision point for measured burnup (i.e., decreasing uncertainty) increases the probability of Type I error. Since most of the 47,752 pebbles are not discharged early in these scenarios, these are reliable estimates.

However, we can use Monte Carlo methods to produce another estimate that considers the retirement of pebbles. To do this, we first list the burnup values from the first pass and then randomly select a value from the associated normal distribution for the measurement of a pebble with that value. If the true value is less than the burnup threshold, then we increase the tally of measured pebbles with a true burnup below the threshold. If the measured value is greater than the decision point for measured burnup, we increase the tally of discharged pebbles and remove the pebble from further counts. If both of those conditions are met, then we increase the tally of prematurely discharged pebbles. If the true value is greater than the burnup threshold, then we increase the tally of measured pebbles with a true burnup above the threshold. If the measured value is less than the decision point for measured burnup and the true burnup value is greater than the burnup threshold, then we increase the tally of pebbles that should have been discharged but were not. If the measured burnup is less than the threshold, then after counting we replace it in the list with the value from the second pass, then third, fourth, and so on until all seven passes are considered and all pebbles have been retired. We repeat this process 10,000 times to produce estimators based on the law of large numbers. We do this for each of the values of L considered before and at L = 85 GWd/tHM to meet the average burnup of 90 GWd/tHM. The results of this Monte Carlo study are found in Table 11.. Table

Table 11. Measured burnup versus Type I and Type II mean and SD.

Decision point for measured burnup (GWd/tU)	Average burnup of discharged pebble (GWd/tU)	Type I error estimate mean	Type I error estimate SD	Type II error estimate mean	Type II error estimate SD
85.0000	90.1867	0.00280	0.000234688	0.14245	0.002353471
83.2290	88.7127	0.01079	0.000409156	0.05836	0.002017674
81.1990	87.1999	0.03101	0.000559823	0.01286	0.001155108
79.2657	85.5135	0.06165	0.000655822	0.00178	0.000501737
75.6627	81.2152	0.14202	0.000714746	1.01E-05	5.59E-05
72.3730	78.3929	0.18890	0.000363925	0*	0*

*0 indicates below machine precision.

The estimates in Table 11 agree with the estimates in Table 10. The results also agree with the fact that decreasing one of the two errors will increase the other. We also see that, as expected, decreasing the decision point for measured burnup will decrease the average burnup of discharged pebbles while decreasing the Type II error and increasing the Type I error.

In summary, the burnup discharge decision is a function of the BUMS measurement uncertainty, the potential maximum energy path through the reactor, and the maximum allowable burnup. The decision point for measured burnup is set by factoring in a margin for the BUMS measurement uncertainty. The size of that margin is a balance between Type I and Type II statistical errors.

Burnup Measurement versus Reactor Code Comparison

Statistically comparing and integrating the two independent values from the BUMS measurements for SNM content with the predicted reactor code values will support MC&A, disposition of spent fuel, and reactor performance/operations. It also provides a pathway for continuous process improvement of key aspects of this technology.

The goal is to statistically compare the predicted pebble SNM content from the reactor code with the measured SNM content to identify any significant differences between the two and to investigate and identify the underlying cause(s) of the differences. Once the cause is identified, changes can be implemented as needed to adjust the process and eliminate the cause. In some cases, this could be updates to the reactor code and/or underlying assumptions about the pebble flow paths in the reactor. Adjustments to the underlying assumptions supporting the BUMS calibration could also be made.

For illustration purposes using a normal distribution, Figure 26 shows the differences in the statistical distributions that could be expected from these two approaches. Although the distributions overlap, they are statistically different from each other.

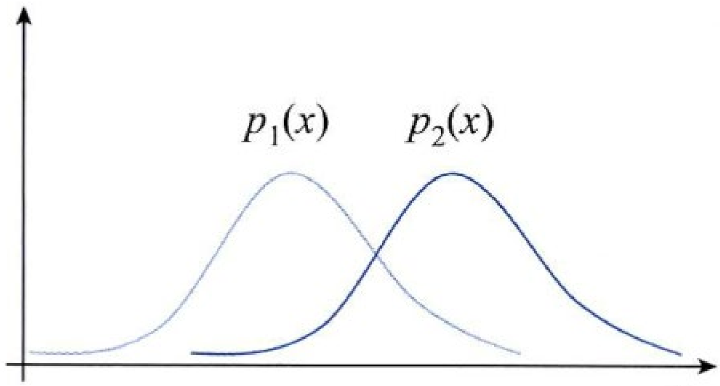

Figure 26. Two different but overlapping distributions.

The desired end state from this approach is shown in

Figure 27, where the two approaches for spent fuel SNM content produce essentially the same results with only minor systematic differences.

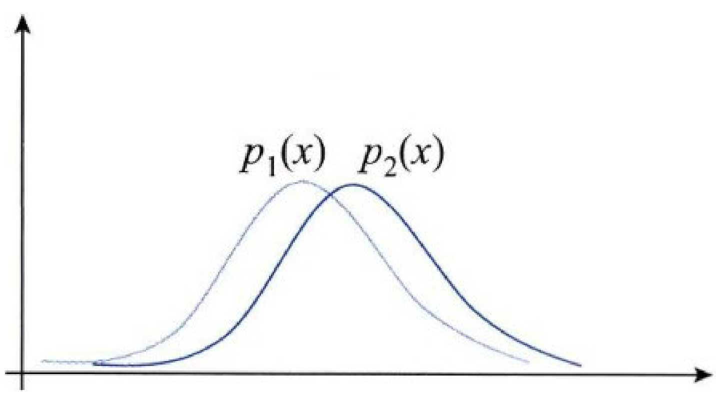

Figure 27. Similar overlapping distributions.

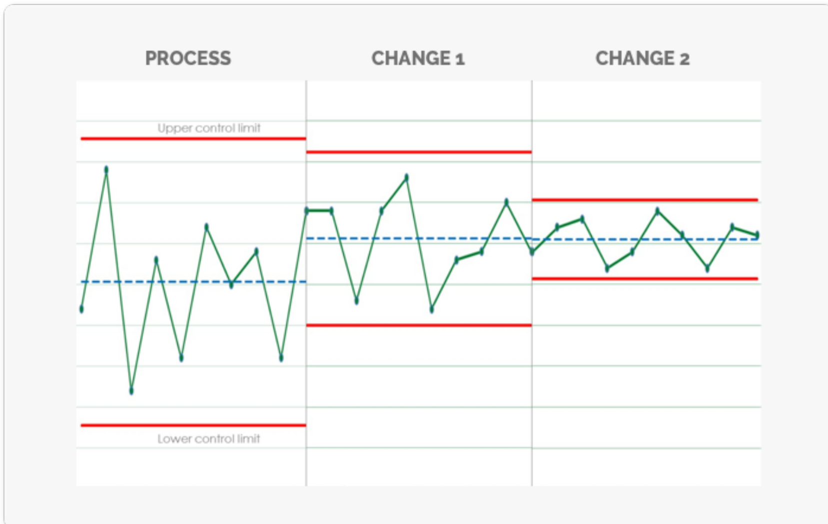

Figure 28. Continuous improvement to reduce process variability.

The goal of this approach is to implement incremental process improvements as needed to reduce process variability to desired levels as determined by key parties (Figure 28).). Lastly, this approach also provides independent or redundant quality checks on the process to monitor performance. Through this statistical process control approach, changes in the process can be identified, and corrective actions can be taken as needed.

Statistical Sampling Plan for Selection of Fuel Pebbles for Additional DA or NDA

First, the implementation of this approach could simply be based on comparison of the BUMS measurement with the reactor code for a 100% comparison without a sampling plan. However, because the BUMS measurement is not currently a direct measurement of the SNM content, it may be desirable, at least initially, to sample pebbles and perform more extensive NDA or DA, which would serve to validate the calculated and measured burnup and other values. Typically, these additional measurements are more time-consuming or expensive, and therefore, the pebbles selected for analysis would be a subset of the pebble population.

The International Organization for Standardization (ISO) offers two series of standards for sampling schemes based on acceptance quality limits: (1) ISO 2859 Sampling Procedures for Inspection by Attributes and (2) ISO 3951 Sampling Procedures for Inspection by Variables. In these standards, the acceptance quality limit is defined as the "quality level that is the worst tolerable process average when a continuing series of lots is submitted for acceptance sampling." Both standards are designed to "ensure that lots of acceptable quality have a high probability of acceptance and that the probability of not accepting inferior lots is as high as practicable." To this end, the standards suggest when to switch between normal inspection, tightened inspection, and reduced inspection [28, 29][28, 29].

The major difference between the two series of standards is how a sample is determined to be acceptable. In ISO 2859, acceptance sampling is by attribute: "inspection whereby either the item is classified simply as conforming or nonconforming with respect to a specified requirement or set of specified requirements, or the number of nonconformities in the item is counted." Whereas, in ISO 3951, acceptance sampling is by variables: "acceptance sampling inspection in which the acceptability of the process is determined statistically from measurements on specified quality characteristics of each item in a sample from a lot." Furthermore, ISO 3951 requires the measured variable to be distributed according to a normal distribution or a small deviation from normal [28, 29][28, 29].

In this analysis, we assume that the sampling plan will be used to select pebbles that will be measured to detect ^{137}Cs activity, which is correlated to burnup values. In this case, ISO 2859 would not be applicable because it is based on acceptance by attribute. Therefore, we should consider ISO 3951, which is based on acceptance sampling by variable. It requires the measured variable to be distributed according to a normal or near normal distribution. However, the results from the pebble burnup simulation indicate that the ^{137}Cs activity levels from a random sample of pebbles are unlikely to follow a normal distribution. The results of work done by ORNL on the pebble burnup simulation showed a linear estimation of burnup with ^{137}Cs (Figure 29), which can be used to create the plot shown in Figure 30.. The distribution of ^{137}Cs activity levels in the plot clearly does not follow a normal distribution (bell curve). Hence, the ^{137}Cs activity levels from a random sample of pebbles is unlikely to follow a normal distribution.

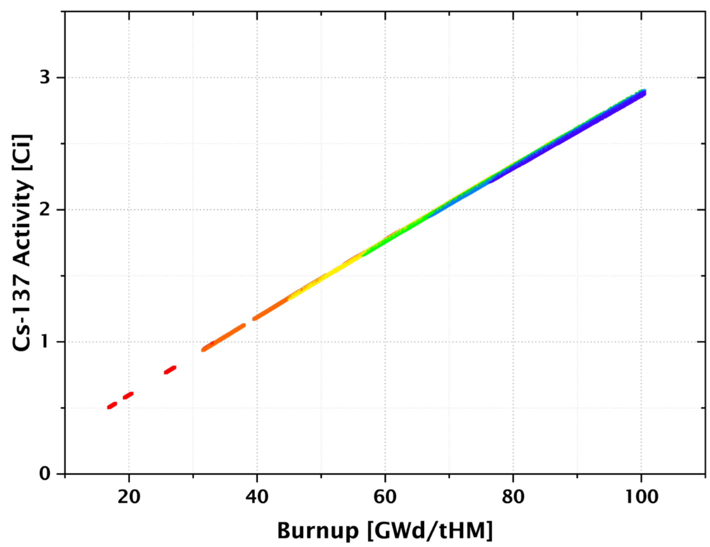

Figure 29. ^{137}Cs activity as function of burnup.

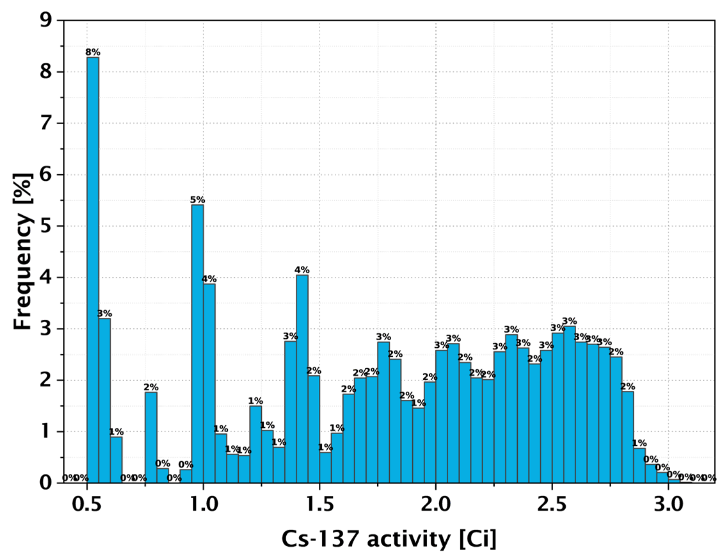

Figure 30. Distribution of measurement for ^{137}Cs.

One approach to consider is taking random sets of samples, computing the means of those sets, and treating those means as the individuals. With a sufficiently large sample, the central limit theorem should apply, and the mean would follow a normal distribution. However, there are two issues with this approach. First, in this case, it is not clear that the mean of the ^{137}Cs activity is necessarily what we need to monitor as opposed to the distribution of the samples. Second, ISO 3951 is based heavily on Shewhart-type control charts. Studies, such as by Huberts et al. [30][30] show that these control charts are sensitive to the normality condition. So, it is not advisable to implement ISO 3951 as is.

A solution is to modify the recommended approach in ISO 3951 in such a way that a distribution-free control chart could be used in place of the charts that assume normality. Distribution-free control charts can be used effectively when the underlying distribution of a process is unknown or complex. One such scheme is the location variable and scale chart introduced by Mukherjee, A., P. Qiu, and M. Marozzi [31].

Code Data Validation—Goodness of Fit Tests

A nonparametric goodness of fit test must be performed to determine if the ^{137}Cs activity samples taken with BUMS are from the same unspecified distribution as those predicted by the reactor code model. The null hypothesis of these tests is that the data comes from the same distribution. The two common tests are the two-sample Anderson–Darling test and the Kolmogorov–Smirnov test. The two-sample Anderson–Darling test is a more powerful test than the Kolmogorov–Smirnov test. This means given a significance level α, the Anderson–Darling test is less likely than the Kolmogorov–Smirnov test to result in Type II error (i.e., fail to reject a null hypothesis when the null hypothesis is false) [32]. Therefore, implementing the two-sample Anderson–Darling test to compare the burnup measured by BUMS with the reactor code is recommended. This same test can be used to validate the reactor code predictions for the SNM content when compared to the values that will be found from the pebbles sampled from the sampling plan.

Tests Analysis of Variance between Reactor Performance

The discussion so far has addressed monitoring the performance of the reactor codes and BUMS for a single reactor. Rather than single reactors, PBRs are modular and are expected to be deployed as multiple units at a site. Variations in performance between reactor units and systems within those units should be expected. Analysis of variance, or ANOVA, is an approach to look at and monitor this variability, make an assessment about its significance, and initiate corrective action if necessary.

Implementing ANOVA would demonstrate that PBRs can dependably attain the same level of reliable performance from reactor to reactor, which ultimately supports widespread deployment of PBRs to address long-term energy and climate goals.

ANOVA is an exceptionally large topic and would be difficult to cover in full detail in this paper. Jaech [33][33] provides additional information and a few simple examples that are likely applicable. There are also other publications that can be consulted as well. Future collaboration with vendors and academia to further explore application of ANOVA to a multi-reactor deployment in the future is recommended.

2.8 INVENTORY MANAGEMENT AND ACCOUNTING SYSTEMS

Each PBR operator will have to implement an inventory management system that will support the overall MC&A program. As such, a review was performed of some of the systems currently available on the market, the functionality of such systems, and their fitness for purpose.

Work was conducted to explore MC&A systems requirements and concepts that will be applicable for PBRs. A review was conducted of several current inventory management and MC&A systems that are in use at NRC-licensed commercial reactors and within DOE facilities. Many capabilities of these current systems are directly applicable to and would support PBRs without modification; however, certain aspects of the PBRs would necessitate some customization of any of the systems studied. Implementation of any system should be informed by the organizational structure anticipated within the utility used to support MC&A and associated regulatory reporting functions. It would be practical to include consultations with current utilities during the design process so that design choices for implementation could be optimized in ways that would best fit current structures.

An inventory management or MC&A reporting system used in PBRs will need to meet the following requirements:

- Fuel life cycle tracking from receipt, to operation, to spent fuel management (audit trail and/or continuity of knowledge)

- Ability to electronically import/export inventory information for both operational needs and regulatory reporting into and out of the MC&A system.

- Ability to handle all regulatory reporting per applicable NRC Manuals

    - U.S. NRC, Instructions for the Completing Nuclear Material Transaction Reports (DOE/NRC Forms 741 and 740M), U.S. Nuclear Regulatory Commission, NUREG/BR-0006, Rev. 9, 2020. [34]
    - U.S. NRC, Instructions for the Preparation and Distribution of Material Status Reports (DOE/NRC Forms 742 and 742C), U.S. Nuclear Regulatory Commission, NUREG/BR-0007, Rev. 8, 2019. [35]
    - U.S. NRC, D-24 Personal Computer Data Input for NRC Licensees - Directory of Electronic File Formats for Data Submission, August 2016. [36]

- Ability to interface with applicable reactor codes or measurement systems for updates to fissile material from reactor operations.

2.8.1 Current MC&A Systems

For the market overview, key differences between these systems are the extent to which they approach and integrate inventory management and regulatory reporting. The Westinghouse and Standish Technologies systems cover both aspects, inventory management and reporting, with specialized functionality to interface with reactor codes and handle spent fuel management. The US DOE system is highlighted and used to illustrate certain concepts central to any MC&A. The two systems supported by the NMMSS and the Nuclear Assurance Corporation are focused more on regulatory reporting requirements.

The TRACWORKS® fuel data management system is a product of Westinghouse and is one of the two main applications currently in use commercially in the United States. It is also deployed internationally. TRACWORKS facilitates life cycle tracking, data management, and reporting for all fuel assemblies or bundles and components for both pressurized water and boiling water reactor units. It is an example of a system that covers inventory management from both an operational and MC&A perspective along with supporting reporting to NMMSS.

SNMtrac is a product of Standish Technologies and is the second of the main two products in use commercially in the United States. Like the Westinghouse product, it covers inventory management from both an operational and MC&A perspective. SNMtrac facilitates reporting, inventory management, operations, MC&A, and spent fuel management. It also supports reporting to NMMSS.

LANMAS is the standard MC&A system used by DOE. The direct application of LANMAS to PBRs would likely be more difficult than deploying either of the two products discussed because it was custom designed for DOE facilities and processes. At the time it was developed, DOE was no longer operating production reactors, so a lot of the reactor-specific requirements are not covered within the application.

Safeguards Management Software (SAMS) and Nuclear Accounting and Compliance (NAC) reporter™ both only cover reporting, and their use would require additional software to address inventory management or interfaces with other systems. SAMS is free of charge from NMMSS. NAC Reporter™ is a similar product and used to be the prime contractor for NMMSS.

2.8.2 MC&A Concepts

There are several key concepts that are used in most MC&A systems. Insight into these concepts is provided along with implementation methods and, where applicable, additional discussion about how the concepts relate specifically to PBRs.

2.8.2.1 Account structure

The basic device of nuclear material accounting, like financial accounting, is the account. In nuclear material accounting, several levels of accounts are used to document the nuclear material inventory, which allows material to be quickly located and to localize any accounting errors or nuclear materials.

The highest level of accounting is the account assigned to each reporting organization by DOE or the NRC, or the site's reporting identification symbol (RIS). RISs are unique combinations of three or four letters assigned to each nuclear facility or sub-facility. A single site or organization may have one or more RIS, which are used for identification when the organization reports inventory data to NMMSS.

The highest level of accounting with a RIS is the MBA, which represents an area that is both a subsidiary account of materials at a facility and a geographical area with defined boundaries that is used to identify the location and quantity of nuclear materials in the facility. Each RIS must have at least one MBA. Subdivisions of an MBA, as recognized by site accounting practices, are sometimes referred to as *sub-MBAs*. Many organizations use sub-MBAs to divide large MBAs to make it easier to identify the source of accounting problems or just to track material. Sub-MBAs can be used to represent a single location, a group of locations, or a specific type of material.

Consultation with utilities that currently operate nuclear reactors is recommended for setting up the account structure because it will likely be very similar to approaches already in use for LWRs, and implementations should logically be consistent with existing MC&A plans.

2.8.2.2 Locations

The term *location* refers to the physical location of the nuclear material, a process line, or any identified logical grouping within a facility. Many accounting systems have location structures that support multiple levels of a location description (e.g., building, wing, room, shelf, position). The decision about how many levels to use is a function of organization preference in implementing inventory and material control. Most modern accounting systems are designed with the flexibility to accommodate an organization's preference with authorized locations populated in the accounting system as part of the system administration functions.

Although the physical location of the material is separate from the account in which the material is maintained, the two concepts are interrelated. Materials may be present in one or more locations (buildings, rooms, etc.) within an MBA or a single location may contain more than one MBA. Either option may be suitable to achieve MC&A goals; however, material attractiveness and physical protection requirements can influence which approach to use. For example, MBAs typically must be fully contained within certain types of security areas used for highly attractive materials.

2.8.2.3 Accounting period

To manage the nuclear material inventory, the amount of nuclear material present in the RIS must be reported periodically. These reports are generated by selecting a specific period of time and determining the amount of material on-hand at the end of the period. This time is referred to as the *accounting period*.

For domestic reporting, applications typically use an accounting period of 1 calendar month. These accounting periods start at midnight on the first day of the calendar month and end at 23:59:59 on the last day of that month. Some accounting systems also support the use of alternative accounting periods as required for inventory governed by International Atomic Energy Agency (IAEA) requirements. For these accounts, the accounting period can be set to run from one physical inventory to the next.

The accounting systems establish the amount of material present at the beginning of the accounting period (beginning inventory), identify all inventory changes, and calculate the amount of material present at the end of the accounting period (ending inventory).

2.8.2.4 Transactions

The process of accounting for nuclear materials begins when some activity is conducted that affects the nuclear material. This activity is sometimes referred to as an *inventory change* or a *transaction*. The primary objective of the transaction process is to create an audit trail that supports the continuity of knowledge for the nuclear material. The accounting principle of double entry is used, which means for every transaction there is a from and to side or in financial terms, a debit and credit.

There are two types of transactions—reportable and nonreportable. For the purposes of this report, *reportable transactions* are transactions that are generally reported to an off-site organization. In most cases, this organization is the US government, through NMMSS; however, many sites also report data to the IAEA. Generally, reportable transactions are changes in account (either on- or off-site), material type, and material weight. They also include any additions to or removals from inventory.

Nonreportable transactions make inventory changes that affect parameters that are not usually reported to an outside organization or that do not affect the nuclear material quantity. Nonreportable transactions are recorded for internal control and accounting of nuclear material again to maintain continuity of knowledge by developing an audit trail or history of every event that has occurred for a given item or container.

2.8.2.5 Physical inventory

There are multiple options for implementation of physical inventory ranging from prelists, barcodes, and other approaches. This approach and design should be informed by current approaches in use at LWRs.

Because of continuous fuel handling, a challenge for PBRs not present in LWRs could be timing. It will be necessary for the MC&A system and physical inventory approach to closely integrate with the fuel-handling system to establish a cutoff time. For physical inventory purposes, a cutoff time is a point in time when the physical location of nuclear material is clearly delineated as to where it is in the process to ensure that everything is counted, and that nuclear material is not double counted.

This could be done procedurally by stopping operations for a physical inventory. It could also be done using time stamps of events obtained from the spent fuel-handling system for transfers to determine material balances at each location at the cutoff time.

2.8.2.6 Types of accountable materials

Accountable nuclear material is usually described as bulk material or as an item. PBRs tend to display aspects of both not clearly fitting into either current definition, so this area has been the subject of a lot of discussion.

Bulk material is nuclear material that exists in such a way that it is impossible to distinguish individual items. Although this is technically true for pebbles because current plans do not include methods for pebbles to be individually identified, they do not behave as bulk material because they are distinct objects that, when moved, move their entire nuclear material content with them. Conversely, traditional bulk materials typically exist in tanks or processing lines as solutions or powders. Transfers or movements typically result in some residual material, and the amount of material transferred has some associated uncertainty due to measurement errors.

Pebbles, although not currently planned to be uniquely identified, are distinct objects. The nuclear material content in a pebble is expected to be well characterized with low uncertainties. All the nuclear material (with the possible exception of broken pebbles) moves with the pebble. Therefore, the traditional uncertainties in the nuclear material value for a transfer of bulk material are not present.

An item is the smallest subdivision of accountable nuclear material represented individually or uniquely identified in the accounting system. Again, this would not be the pebble because they are not uniquely identified. Rather, the item would be the container or the reactor/fuel-handling system they are contained within.

PBRs best fit an approach used by DOE termed *items and pieces*. In the DOE approach, using items and pieces the pebbles would be represented on an inventory as pieces. A fresh fuel shipping container and a spent fuel container would be the item, and the pebbles would be represented by a piece count or numbers of pebbles in the container. The reportable content would be the sum of the SNM in each pebble.

Depending on how this distinction is implemented, it would affect the accounting system implementation for how the loading and unloading of a fresh and spent fuel container is modeled. Two possible approaches for implementing these are containerization, or splits, combines, and transfers.

1. **Containerization**—Often, items are physically placed in a container, assembly, or part for storage, processing, or shipment. Current accounting systems have the capability to load or unload containers such as spent fuel casks or other types of containers to show this “nesting” of items within these containers. Once nested or grouped, any transactions such as changes to the location or account are made to the group. The number of nesting levels is currently unrestricted in many accounting systems.

Accounting systems are designed so that during the containerization/grouping, certain attributes like account (typically) and location (always) are the same for the grouping. Systems typically also record additional attributes associated with the container such as type, certification date, and tamper-indicating device (TID). The transaction process to load or unload a container will also record specifics about the transaction to assist in maintaining continuity of knowledge.

For PBRs if the item/piece count approach were used, in the US DOE LANMAS application the fresh and spent fuel containers would be modeled as items versus containers. Unloading and loading would be handled by a split/combine function. This would not affect the further nesting of the fresh or spent fuel container into another level of containment. For example, a spent fuel container modeled as an item under the approach used in the LANMAS application could be placed inside a spent fuel cask, which would be modeled as a container.

2. **Splits, combines, and transfers**—Splits, combines, or transfers are used to move material between items by specifying the "from" item, "to" item, and the amount of material to be transferred. Applied to operations within a PBR, the unloading of a fresh fuel container using the LANMAS design would be the transfer of the pebbles into the fresh fuel-handling system showing the transfer of "X" number of pieces and their nuclear material content from the fresh fuel container. The discharge of spent pebbles into a spent fuel container would be handled similarly.

2.8.2.7 External shipments and receipts

The NRC references and NMMSS instructions explain the requirements for reporting. Manual D-42 provides examples of the electronic XML format for reporting to NMMSS. All the systems highlighted in the previous section have this capability.

The Westinghouse, Standish Technologies, and LANMAS systems include additional data formats that are used to transmit fuel manufacturing information not required for regulatory reporting between the shipper and receiver. This additional capability would be applicable to PBRs and should be considered a functional requirement.

2.8.2.8 On-site movements or transfers between accounts and locations

Movements/transfers are typically defined as changes in the location or account attributes of a nuclear material item or container. Most systems have a single function that can change one or both attributes in a single transaction.

Account changes are a change to the account attribute of the item or container. These could be externally reportable and or not. For example, a change in RIS (or MBA under IAEA safeguards) would be a reportable event. Conversely, a change in sub-MBAs would not be externally reportable in any case. The transaction will record the from account, to account, the date/time the change occurred and the date/time it was entered into the system. Typically, there are business rules to ensure a logical order to account for changes.

Location changes include changes to the location attribute of the item or container. The transaction will record the from location, to location, the date/time the change occurred and the also the date/time it was entered into the system. Typically, there are business rules included in the system to ensure a logical order to location changes designed to assure continuity of knowledge of the physical location.

2.8.2.9 Burnup and production

For LWRs, adjustments to the fissile material content in the fuel due to reactor operations is accomplished through a system interface to reactor codes that computationally calculate changes to nuclear material inventories based on operational parameters. The MC&A system and associated transactions for recording this change create an audit trail that captures when the change occurred along with the documentation (calculations, power levels, code used, etc.) for the change.

PBRs will have the option to use the reactor code to predict nuclear material inventories, but they will also have the option to use BUMS. It remains to be seen which approach is the best estimator of the nuclear material content in the spent fuel. Either approach could be acceptable.

If the reactor code is used, the MC&A system would follow the same approach used in LWRs, creating a transaction that is linked to, and supported by, the reactor code. If BUMS measurement is used, the MC&A system would need to link to the measurement system and the individual measurement events for the spent pebbles that are discharged into the spent fuel container, summing those measurements for the nuclear material content to be declared for the spent fuel container. There are relevant Nuclear Quality Assurance 1 requirements for measurement and testing equipment that would be applicable to support the measurement results.

The MC&A system should also be designed to allow for subsequent updates to the nuclear material content in the spent fuel using this type of transaction. As operational experience improves, so will the ability to determine nuclear material content. Therefore, it would be logical to have the capability to update the spent fuel values in storage based on this new information.

2.8.2.10 Rounding [37]

As was discussed in a previous section, PBRs have many discrete objects (i.e., pebbles) containing small quantities of nuclear material that will require special considerations in an MC&A system to handle rounding for inventory and transaction reporting. The basis and approach for rounding adjustments should be covered in the facility's Fundamental Nuclear Material Control plan, which should describe how rounding adjustments are captured within the MC&A system and subsequently reported to NMMSS.

2.8.2.11 Inventory differences or Material Unaccounted For (MUF)

PBRs ideally will behave like item facilities, in that they will not have inventory differences except in cases of missing items. It remains to be determined whether the inventory approach for the reactor and fuel handling system will result in count discrepancies in the numbers of pebbles, which could result in positive or negative inventory differences.

The timing of and how broken pebbles captured within the spent fuel handling system are handled could also result in events that might be classified as inventory differences. Ideally, the objective should be to document or record nuclear material in broken pebbles as a transfer from the reactor to the broken pebble container. However, in this case, the integrity of the pebble is compromised, so the amount of nuclear material content and where it is located with the fuel-handling system may be difficult to fully determine.

In summary, although inventory differences are not likely except in rare cases, the MC&A system should have the capability to adjust the inventory to either add or remove items and their associated nuclear material values and to document these changes as an inventory difference or material unaccounted for.

Decay

PBR spent fuel will be subject to changes in nuclear material content due to radioactive decay, which is a reportable transaction. This calculation and adjustment transaction could be approached in two ways. One way is how LANMAS, the DOE application, approaches it, which is to maintain isotopic information and the creation date, which is then used to periodically check for decay and automatically update the SNM content and create the supporting transactions. Alternatively, spent fuel management systems typically maintain a more complete set of isotopic information to support requirements that extend beyond MC&A. These systems would also update isotopic information based on decay, which could be fed into the

MC&A reporting system. To maintain consistency between operational and MC&A records, the latter approach would probably be more appropriate for PBR applications.

The change in inventories due to plutonium decay, although reportable, is small and driven primarily by the ^{241}Pu content. Using the PBMR-400 model with a burnup of 90 GWd/tHM, the plutonium content in a spent fuel container of 2,000 pebbles would decrease by 10% over 25 years (Table 12.).

Table 12. Example of Plutonium Decay Using PBMR-400 Model

Mass (g) in 2,000 pebbles	0	1 y	5 y	10 y	25 y
^{238}Pu	4.682	5.025	4.966	4.774	4.412
^{239}Pu	104.920	105.462	106.644	106.632	106.607
^{240}Pu	71.020	71.043	70.809	70.884	70.976
^{241}Pu	38.656	36.825	30.710	24.096	14.835
^{242}Pu	29.114	29.114	28.995	28.993	28.993
^{244}Pu	0.000	0.000	0.000	0.000	0.000
Total plutonium	248.392	248.694	242.125	235.379	225.823

2.8.2.12 Tamper-indicating devices

Tamper-indicating devices, or TIDs, are used to seal items, containers, and locations that hold accountable nuclear material. To ensure the integrity of measurements and continuity of knowledge, the issuance, movement, application, verification, and removal of TIDs must be controlled and documented. Most MC&A systems provide a TID management function to record the receipt, issuance, application, destruction, and transfer of TIDs. This function is typically integrated into the shipment, receipt, physical inventory, and containerization functions or any function where TIDs are used. Regulatory Guide 5.80, “Pressure-Sensitive and Tamper-Indicating Device Seals for Material Control and Accounting of Special Nuclear Material,” is a good reference for the facility to implement a tamper-safing program [38].

3. CONCLUSIONS AND RECOMMENDATIONS

This report addressed the following MC&A topics for PBRs:

- The NRC licensing basis for PBRs
- Current industry partnerships that are helping to inform challenges and potential solutions for applying MC&A to PBRs
- Description of fuel flows in a typical PBR
- Containerization methods used for the transport and loading of fresh fuel pebbles and retirement of spent fuel pebbles
- Development of PBR reactor fuel flow models to determine the movement of fuel inside the reactor core and to calculate the burnup of individual pebbles based on their irradiation histories

- Modeling of potential gamma signatures that could be used to facilitate nondestructive measurements of irradiated fuel pebbles

- Determination of possible neutron-based NDA systems that could measure irradiated fuel pebbles and withstand the harsh environments near the reactor systems

- Statistical approaches needed to address the unique accounting challenges of millions of mobile fuel pebbles with low densities of nuclear material

- Analysis of the features of potential inventory management systems that could be used in a PBR

Two PBR designs that are active in the US are the X-energy XE-100 helium gas cooled and the Kairos Power molten salt PBRs. Interactions with both companies have provided a context to consider MC&A during the design phase and plans for deployment. Although proprietary data and intellectual property were not included, the questions that were asked and issues addressed informed the topics and results of this report.

Recommendation 1: Continued engagement between the national laboratories and PBR designers is necessary to put into practice the results and recommendations of this report.

The licensing basis for PBRs is not entirely clear at this moment. Based on previous reports, it is not certain that the NRC will accept an MC&A program based solely on Regulatory Guide 5.29 and ANSI Standard N15.8-2009. This is because of the inability to rely on pure item counting as is done for the current licensed fleet of LWRs as well as the push to higher enrichments that will classify them as Category II facilities. There are currently no licensed Category II LWRs. Category II facilities are those that use uranium enrichments equal to or greater than 10% but less than 20%. The exceptions to the other requirements of 10 CFR Part 74 that are automatically included in the licensing process of 10 CFR Part 50 or exclusions that are granted as part of 10 CFR Part 52 may not fully apply to PBRs. In this case portions of NUREG 2159—Acceptable Standard Format and Content for the Material Control and Accounting Plan Required for Special Nuclear Material of Moderate Strategic Significance—may apply. Category II licensed facilities have tighter material and measurement controls and reporting thresholds, but these are based on fuel cycle facilities that process true bulk material in powders or solutions. Since fuel pebbles are distinct pieces, they do not fit the classification as true bulk material.

Recommendation 2: PBR designers should engage in early, pre-licensing discussions with the NRC to determine an acceptable format for the MC&A plan as part of the overall MC&A program.

The structure of the MBAs and KMPs for a generic PBR is straight forward. The fresh fuel is received in containers that are sealed at the fuel fabrication facility. The shipper values will likely be accepted because remeasurement would be costly. The concept of reporting in batches of fresh fuel drums with the individual pebble count listed as “pieces” within the batch are accepted MC&A reporting practices to the NRC. The same is true for the spent fuel canisters. For spent fuel containers, the dose rate at 1 m will initially be higher than for a typical LWR fuel assembly but will decay more rapidly. This is because of the higher burnups that will be achieved resulting in more fission product loading per unit volume but with lower overall actinide content of TRISO.

Recommendation 3: Designers should consider how fresh and spent fuel will be accounted for and reported when developing MC&A programs, and designers should employ containerization whenever possible in the design and in nuclear material fuel flows.

Recommendation 4: The self-protecting nature of spent TRISO fuel should be considered when developing the theft and diversion scenarios and as part of the MC&A and physical protection planning.

To adequately account for the production and loss of nuclear material inside the reactor, new models were developed to simulate the actual pathways that individual fuel pebbles take in the reactor system. The continuous on load feature combined with possible pathways through the reactor must be captured in the modeling approach. The neutron flux that a fuel pebble will experience in the reactor core will vary based on radial and axial positions as well as the aggregate fuel pebbles themselves and the operation of the reactor. This in turn affects the burnup per pass and how many passes a fuel pebble will experience based on the BUMS measurements. The design chosen to model was the PBMR-400 design. It adequately represents PBR designs, and the data is public and readily available. Based on this modeling the average and maximum total uranium loss and plutonium production were determined. Based on an initial loading of approximately 9 g total uranium at an enrichment of 9.6%, the total plutonium mass was calculated as approximately 0.1242 g per pebble, on average, after reaching a target burnup of 90 GWd/tHM. The total plutonium in a spent fuel cannister holding 2,000 retired pebbles was 248.392 ± 0.26 g. The relatively low uranium and plutonium content is an inherent feature of TRISO-based fuels. The low uranium and plutonium content requires the diversion or theft of hundreds or thousands of pebbles to result in a significant amount of nuclear material. However, from a security perspective, the theft of only a small number of pebbles is a major concern because of the threat from a dirty bomb or irradiation device that can cause significant harm to workers or the public.

Recommendation 5: Models should be developed for each design to adequately represent the production and loss of nuclear material based on the specific features and operations of the reactor.

A central component of PBR operations is the BUMS determines when a fuel pebble is either reinserted into the reactor or retired as spent nuclear fuel. Determining the burnup of each irradiated fuel pebble requires measuring gamma signatures that represent the burnup level reached. Measuring short-cooled nuclear fuel is very challenging because of the emanations of high-energy photons from many different fission products. Designing BUMS requires adequate considerations of the conditions where the instrument will be placed, the necessary shielding and collimation, and the geometry of the gamma measurement system. One main challenge is determining what fission products can be measured and which can provide a representation of the fuel pebble's burnup. Output from the reactor modeling was input into a GADRAS model to determine that measuring the absolute quantity of ^{137}Cs in each pebble should be possible, which is a good indicator of the burnup. Cesium-137 is a reliable burnup indicator because the fission yields of ^{137}Cs from ^{235}U and ^{239}Pu, the two primary fissioning nuclides in PBRs, are nearly identical. This assumes that the gamma instrument (e.g., a high-purity germanium detector) can be calibrated properly to determine its absolute efficiency and the geometry of the measurement system is controlled.

Recommendation 6: Work should continue to develop gamma measurement systems in collaboration with the national laboratories, vendors, and measurement equipment manufacturers.

Passive neutron measurements were also considered to measure the burnup of individual pebbles. Neutron NDA techniques could replace or augment gamma measurements to determine a pebble's burnup value. Neutron measurements can potentially provide a more sensitive measurement of the burnup than the gamma measurements can because the neutron emission of an irradiated pebble is a power function of burnup, whereas the photon emission is usually a linear function of burnup. Additionally, the coincidence neutron signal of an irradiated pebble can be explored because the coincidence neutron signal is usually linear with the fissile content of the spent fuel, with the caveat that the coincidence neutron signal from a pebble's fissile content is likely to be small because of the small fissile content in a pebble; therefore, the usefulness of coincidence neutron signal for a pebble remains to be proven. One challenge of neutron measurement next to a PBR is the environment is a higher temperature than a conventional light water reactor. The commonly used neutron moderators in neutron instruments, such as high-density polyethylene, would not withstand much greater than 104 °C [1]. An alternate neutron-moderating material, PolyEtherEtherKetone (PEEK), is a semicrystalline thermoplastic with excellent strength and ductility, has good neutron-moderating power, and is suitable for continuous use at temperatures up to 260 °C [2]. This new moderating material is promising and could allow the use of passive neutron NDA techniques to be used next to a PBR as part of BUMS.

Recommendation 7: Passive neutron detectors should be explored to see if they can either perform better than gamma detectors or if they can complement gamma detectors for MC&A purposes. They may also be considered as confirmatory measurements for retired pebbles as they exit the reactor system before being placed in spent pebble storage canisters.

Accounting for the nuclear material in a PBR will include several millions of pebbles that will be received, circulate in the reactor, and retired as spent fuel over the life of the reactor. This will require careful consideration of how the loss and production of nuclear material is measured and calculated and will require statistical approaches to adequately account for Type I and Type II errors. BUMS is an integral part of PBR operations and is required to fully utilize the fuel while preventing reinserting fuel that does not produce enough power, thereby degrading reactor performance. BUMS is also needed to prevent degradation of the fuel pebble due to damage from excessive neutron flux resulting from excessive residence time inside the reactor. The combination of BUMS measurements and the reactor core modeling provides a powerful combination of tools that can reinforce each other. During early deployment, it should be assumed that fuel pebbles will require destructive analysis, NDA, or a combination of both to verify fuel performance, and this analysis can be used to precisely quantify the actinide content and can be compared to the measured and calculated values. Sampling of fuel pebbles for destructive and NDA should be based on a statistical approach to select a representative population. These results can be used to both calibrate the BUMS values and validate the reactor code predictions for burnup and special nuclear material content. Future refinements can be made to both approaches to result in a robust measurement system and reactor burnup models. The rate of sampling can then be decreased significantly or eliminated altogether.

Recommendation 8: Designers should consider how statistical sampling of spent fuel pebbles will accommodate BUMS and the reactor core burnup models.

Recommendation 9: Additionally, a comparison of reactor models, the BUMS, and statistically based destructive analysis should be performed to validate the models and improve the BUMS performance.

Inventory management systems will be required for every PBR to facilitate the MC&A program. Currently available MC&A systems will likely need to be modified or adapted to use in a PBR. The MC&A system will be required to electronically import/ export inventory information for both operations needs and regulatory reporting. They should also have the ability to interface with the applicable reactor codes or measurement systems for updates to fissile material from reactor operations.

Recommendation 10: Future PBR owners and operators consider the MC&A software systems that are currently available to determine which one most closely meets their business and operational needs. Some modifications or adaptations will be required for PBRs. These can be performed in-house or outsourced to the software system vendor or a third-party software developer.

Recommendation 11: If it is determined that the MC&A software will be developed in-house, adequate preparation and understanding of the functional and interface requirements will be needed. Designers should plan accordingly.

There is still work to be done in modeling the reactor core and fuel handling systems due to the complexity of operations. Pebbles with different enrichments are used during the startup and run-in operations as well as graphite pebbles (nonfuel pebbles). This complicates the modeling and accounting of the nuclear material inventories. The consistency of the fuel from the fuel fabrication facility is also a consideration and will affect both the receipt verification as well as reactor operations. Non-steady-state and off-normal conditions have not yet been modeled, including reactor maintenance where the used fuel is off-loaded from the reactor, stored, and then reintroduced to restart the reactor. Development of the measurement systems and approaches still requires work. The difficulty in accurately and quickly measuring the actinide content in highly irradiated fuel must still be overcome. Measurement systems using both gamma and neutrons are being considered that can withstand the challenging radiation and thermal environments as well as space constraints. The reliability and robustness of such measurement systems must be ensured to achieve operational and economic goals.

As such, all of these issues are being addressed by industry, with support from the national laboratories, and all indications are that they will be successful and will result in a robust MC&A system that meets domestic requirements.

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