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Hybrid Gamma Emission Tomography for the Verification of Unirradiated Fuel: A Viability Study

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Abstract:

Current International Atomic Energy Agency (IAEA) methodologies for the verification of fresh low-enriched uranium (LEU) and mixed oxide (MOX) fuel assemblies are volume-averaging methods that lack sensitivity to individual pins. Further, as unirradiated fuel assemblies become more and more complex (e.g., heavy gadolinium loading, high degrees of axial and radial variation in fissile concentration), the accuracy of current IAEA instruments degrades and measurement time increases. Particularly in light of the fact that no special tooling is required to remove individual pins from modern fuel assemblies, new capabilities for the verification of unirradiated (i.e., fresh LEU and MOX) assemblies are needed to ensure that fissile material has not been diverted. Passive gamma emission tomography has demonstrated potential to provide pin-level verification of spent fuel, but gamma-ray emission rates from unirradiated fuel emissions are significantly lower, precluding purely passive tomography methods. The work presented here introduces the concept of Hybrid Gamma Emission Tomography (HGET) for verification of unirradiated fuels, in which a neutron source is used to actively interrogate the fuel assembly and the resulting gamma-ray emissions are imaged using tomographic methods to provide pin-level verification of fissile material concentration. This paper describes the status of a viability study on the HGET concept, including: envisioned use-case scenarios and corresponding definitions of fuel assemblies; modeling framework based on Monte Carlo and deterministic transport methods, and its validation; quantitative assessment of candidate HGET signatures with a focus on prompt fission gamma rays and delayed fission gamma rays; a nominal HGETv1 instrument design; candidate HGET-specific tomographic reconstruction methods that fully incorporate declared information; and examples of simulation-based predictions of HGET performance.

Keywords: safeguards; fuel verification; gamma emission tomography

1. Introduction

Current IAEA methodologies for the verification of fresh LEU assemblies at fuel fabrication facilities utilize active neutron interrogation with neutron coincidence counting; verification of fresh MOX fuel utilizes passive neutron coincidence counting with gamma-ray spectroscopy for Pu isotopics. These volume-averaging methods are not capable of individual-pin sensitivity and as fuel assemblies become more complex (e.g., heavy gadolinium loading, and axial variation in boiling water reactors [BWRs]), their accuracy degrades and measurement times increase. Particularly in light of the fact that no special tooling is required to remove individual pins from modern fuel assemblies, the IAEA needs new capabilities for the verification of unirradiated fuel assemblies that can provide high-precision fissile-mass quantification, ideally at the single-pin level. The IAEA has documented the need for new unirradiated-fuel verification tools in the IAEA Department of Safeguards Long-Term R&D Plan [1]. Other potential users of a new fuel verification tools include EURATOM, and State regulators.

Passive gamma-ray emission tomography (GET) is a promising candidate for verification of item integrity in fuel assemblies because it has the potential to directly image the spatial distribution of the active fuel material, without the need for operator-declared information [2]. In this sense, it is an absolute, rather than comparative verification method. In addition, the relative intensity of gamma-ray signatures can be used to verify declared attributes on a pin-by-pin basis (e.g., burnup in irradiated fuels; uranium enrichment or plutonium isotopics in unirradiated fuels). The viability of GET for the detection of missing pins in irradiated fuels, where relatively intense, higher-energy gamma emissions are available, appears promising based on findings of a recent IAEA study [3] and ongoing testing of a prototype passive GET instrument by the IAEA.

For unirradiated fuels with relatively weak and lower-energy emissions, the ability to see interior pins with purely passive tomography is less clear. The use of active neutron interrogation to stimulate gamma-ray emission can provide additional signal intensity for emission tomography, here referred to as Hybrid Gamma Emission Tomography (HGET). There are several candidate signatures for hybrid (i.e., tomographic imaging of an active interrogation signature) assay of unirradiated fuels, including prompt capture gamma rays in the isotopes of interest (e.g., 1.28 MeV from ^{235}U); prompt fission gamma rays (continuum peaked at ~ 1 MeV); and delayed gamma rays from short-lived fission products (discrete lines generally from 1 to 7 MeV).

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Each of the candidate signatures above has been studied previously, and sometimes in combination, for the assay of both irradiated and unirradiated fuels. For example, delayed-gamma methods have been studied for the direct assay of fissile isotopes in irradiated fuels [4, 5], but the delayed-gamma methods studied to date provide no spatial information about the origin of the signatures and therefore, localized neutron moderation effects and self-attenuation can produce biases in fissile isotope quantification. In addition, the high passive background in spent fuel forces the use of only the higher-energy (> 3 MeV) delayed-gamma signatures, while the most intense signatures are presented at lower energies. In unirradiated fuels, these more-intense, lower-energy delayed-gamma signatures are accessible, but information about their location of origin in the fuel assembly is needed.

To the authors' knowledge, no prior work has demonstrated the ability to provide spatial information about the origin of the candidate signatures and therefore, verify fuel characteristics at the pin level. In the HGET concept, it is postulated that the collection of these candidate signatures through a tomographic lens will support pin-by-pin verification of fissile materials in the assembly.

Here we describe an ongoing modeling-based viability study of the HGET concept. This paper discusses potential IAEA use cases and implementation approaches, a novel method for modeling instrument response that couples Monte Carlo and deterministic transport methods, candidate signatures, and a method for extracting fissile isotope concentrations on a pin-by-pin basis. Example results for pin-level verification of fissile isotope concentrations in MOX fuel assembly are presented. The paper concludes with a discussion of the ongoing and planned analyses that are needed to more fully assess the viability of the HGET concept for safeguards verification.

2. Potential Use Cases in International Safeguards

The use case for an HGET instrument by safeguards inspectorates is presumed to be consistent with how current IAEA instrumentation is used for the verification of unirradiated fuels. For fresh LEU fuel, the IAEA uses the Uranium Neutron Coincidence Collar (UNCL); for MOX fuel the Passive Neutron Coincidence Collar (PNCL). Both instruments use neutron coincidence signatures to verify the total uranium or plutonium in the assembly--additional information about each method can be found in [6], with IAEA's International Target Values (ITVs) for verification of unirradiated assemblies in [7].

For fresh LEU fuel, the UNCL is used to measure the mass density of ^{235}U at a given axial location of the assembly. It is assumed that the ^{235}U is the only fissile isotope in the assembly and therefore, that all induced fission comes from ^{235}U. This localized ^{235}U mass density is translated to total ^{235}U mass for the assembly using an active length measurement (e.g., gamma scanning). The ITV for determination of total ^{235}U mass in an LEU assembly is 4.5% (one-sigma relative), assuming relatively low gadolinium (Gd) content. Count times are not specified in the ITV document, but other reports indicate thatfor fuels with Gd, count times for UNCL can approach one hour to reach the desired statistical uncertainty. Systematic uncertainties for high-Gd assemblies can be 10 or more times higher [8].

For MOX fuel, the PNCL is used to measure the mass density, at a given axial location, of the Pu isotopes with appreciable spontaneous fission yields (^{240}Pu dominates). High-resolution gamma-ray spectroscopy on exterior pins of the assembly is then used to infer the linear density of total Pu. An active length measurement (e.g., gamma scanning) is employed to translate that value to total Pu for the assembly. The ITV for determination of total Pu mass in a MOX assembly is 3.2% (one-sigma relative).

The use cases and ITVs for UNCL and PNCL provide useful context for the HGET viability study, and are the basis for the assumptions that were adopted to guide the first phase of this study:

- Verification of unirradiated fuel will occur in an air environment and the operator will position the fuel assembly in such a way that the HGET collar will assay one or more vertical segments of the assembly. As with UNCL and PNCL, it is assumed that some form of active-length measurement will inform the translation from the HGET-measured ^{235}U and total Pu linear densities to a ^{235}U and total Pu assembly mass value. Note that the HGET gamma-spectrometer array, operating in purely passive mode, could provide an active-length measurement similar to what is performed today using a handheld gamma-ray detector. (This assumes that the operator moves the fuel assembly through the HGET collar.)
- Total measurement time for HGET verification of unirradiated fuel assemblies should be on the order of 1-2 hours. While today's measurements may be shorter in duration for many fuel types, the fact that HGET will provide pin-by-pin verification of fissile content encourages a broader window of assay-time acceptability for the first phase of the study.
- The physical dimensions and mass of HGET should be comparable to existing IAEA instruments: for example, the JCC-71 PNCL/UNCL instrument sold by Canberra weighs approximately 40 kg [9]. A maximum neutron moderator/reflector weight of 100 kg was enforced during the design study, on the logic that this represented a reasonable size for a mobile instrument deployed at a fuel fabrication or reactor facility.

3. MOX Fuel Assembly as Initial Case Study

While the HGET study is also investigating low-enriched uranium (LEU) fuels with and without burnable poisons, an initial use case was defined for a pressurized water reactor (PWR) assembly of 17x17 pins with mixed oxide (MOX) fuel. PWR fuel is at the more-challenging end of the continuum of fuel types under safeguards in terms of neutron and gamma-ray attenuation, due to its relatively dense pin-array geometry and overall large dimension. The age since separation for the reactor-grade Pu was assumed to be 5 years, an upper limit in terms of occupational health hazards (after about 5 years enough ^{241}Am has grown in to make handling difficult; this process has little impact on the amount of fissionable material for the HGET measurement), and the composition is shown in Table 1.

Generally speaking, the composition of MOX fuel pins varies with pin position. An IAEA technical report, provides Pu concentrations of each pin type in an example MOX assembly, as shown in Figure 1 [10]. Note that the overall Pu concentration varies from pin to pin but the relative Pu isotopics, as defined in Table 1 is consistent across all pins.

		MOX	LEU
Atom	Isotope	Atom Fraction (X3)	Atom Fraction (X3)
U	234	5.20×10^{-5}	3.12×10^{-3}
U	235	6.81×10^{-4}	4.05×10^{-2}
U	238	9.39×10^{-1}	9.60×10^{-1}
Pu	238	1.36×10^{-3}	
Pu	239	3.21×10^{-2}	
Pu	240	1.52×10^{-2}	
Pu	241	7.06×10^{-3}	
Pu	242	4.21×10^{-3}	
O	16	2	2
	density (g/cc)	10.4538	10.4538

Table 1: Initial composition (before decay) of the fuel assembly definitions used in the HGET viability study (atom fractions displayed are 3x the total atom fraction, such that the U/Pu isotopes add to approximately 1).

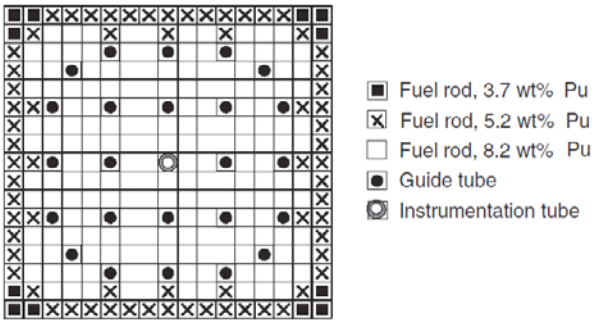

Figure 1: PWR MOX assembly design of the 17x17—24 type with assembly averaged plutonium concentration of 7.2 wt% Pu. (From [10])

4. Overview of HGET Modeling Methods

An overview of the HGET modeling methods is given in Figure 2 below. Neutron transport was performed using MCNP6 and the calculated fission rates in the fuel pins were used to generate the prompt- and delayed-gamma source terms. Those gamma-ray source terms were then used as input to a separate calculation for the transport of the photons out of the assembly and into the detector. Gamma-ray transport through a highly attenuating assembly can be prohibitively time-consuming with pure Monte Carlo methods. The gamma-ray transport was performed using a deterministic transport by the discrete-ordinates package Attila [11]. More detail on the HGET modeling method and validation can be found in [12].

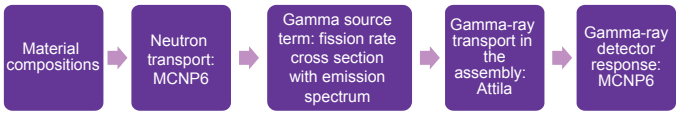

Figure 2: Schematic of HGET modeling approach for modeling neutron-induced gamma-ray signatures and detector response functions.

5. Candidate HGET Signatures

Gamma emission tomography is based on detecting gamma emissions selectively, sensitive to both their location and their angle of incidence. Detection of inner pins can be very difficult, since radiation from these pins must pass through a considerable distance of dense fuel. Gamma rays with energies of less than 500 keV have a very low probability of escaping from inner pins to the outside. Penetration increases with increasing energy to a broad maximum at around 3 MeV to 4 MeV, beyond which the pair production mechanism of absorption causes penetration to decrease. Isotopic specificity is also desirable; if a gamma-ray emission is uniquely tied to a given isotope (e.g., fissile isotope or fission product), it will likely be more useful in characterizing the fissile content of the assembly. Finally, methods based on excessively complex signatures may be difficult to implement, limiting their utility.

An order-of-magnitude comparison of typical spent-fuel assay signatures, to the candidate signatures for HGET assay of a nominal MOX fuel assembly is given in Table 2. The actively induced count rates were estimated using the neutron-gamma modeling methods described in the previous section, and the nominal HGETv1a design described later that employs a commercial, off-the-shelf deuterium-tritium (D-T) neutron generator producing approximately 10^8 n/s at 14.1 MeV.

Technique	Emission rate (γ/pin/cm/s)
Spent fuel, 1 year CT, ^{154}Eu 1274 keV	1 x 10^{8}
Spent fuel, 30 year CT, ^{137}Cs 662 keV	5 x 10^{9}
^{239}Pu 414 keV	5 x10^{3}
Prompt fission > 1000 keV	1 x10^{4}
Delayed gamma individual lines	< 2 x10^{1}
Delayed gamma at 1 s, > 1000 keV	4 x 10^{2}
Delayed gamma at 1000s, > 1000 keV	5 x 10^{3}
Prompt capture gamma, U	Uncertain, ~1 x10^{2}
Prompt capture gamma, Pu	<1 x10^{3}
Activation gammas > 500 keV	Uncertain, small

Table 2: Order-of-magnitude comparison of signal intensities from passive tomography of PWR spent fuel variants (top three entries) and HGET for unirradiated MOX fuel. HGET signatures highlighted in gray are the most promising in terms of absolute emission intensity.

Clear from Table 2 is that the passive Pu emissions from unirradiated MOX fuel are many orders of magnitude less than from spent fuel, and have low penetrating power. The actively induced signatures offer somewhat higher intensities and importantly, their higher energies offer the promise of greater penetrability through the fuel assembly. Delayed gamma signatures are more complex, due to their time dependence, and no individual lines are observed with sufficiently high emission intensity for tomography. It is possible that a delayed gamma-ray signature summed over broad energy windows could be imaged – but fissile-isotope specificity would be lost. The assay of other activation products to infer fuel composition offers little promise both because of low intensity and limited direct connection to the fissile material that is the focus of IAEA verification. Given that no isotope-specific signatures are high enough in intensity for direct fissile isotope assay, the most useful signature for verifying the integrity of fuel assemblies and total fissile content appears to be the prompt fission gamma rays, possibly in combination with the delayed gamma rays. The relatively high production of these signatures at energies above 1 MeV is key, although still four orders of magnitude below emission rates typical of spent fuel.

This large gap in emission intensity points to the need for the development of HGET-specific tomographic designs and methods, for example neutron moderation and reflector designs that are efficient for inducing fission in the assembly, detector and collimator designs that balance gamma-ray collection efficiency with spatial resolution for imaging, and tomographic reconstruction methods that wring as much information as possible from the collected data by relying heavily on the declared, *a priori* information about the pin assembly geometry. These topics are discussed in the sections below.

6. Nominal HGET Design

A wide range of source/moderator/reflector designs and materials (e.g., poly, graphite, hydrided DU) were considered in the early stages of the HGET v1 design study. Both a D-D and a D-T neutron generator were considered; the lower energy neutrons from D-D produce a smaller background of ^{238}U fissions but D-D generators are generally significantly lower in achievable intensity, given similar form factors. The metrics for evaluating the various designs were: 1) uniformity of thermal and epithermal flux across the assembly cross-section, 2) total fission rate induced in the MOX fuel definition, and 3) relative contributions of fissile and ^{238}U fission. Several of the early designs were discarded based on these metrics; Figure 3 (left) depicts the design that demonstrated considerable promise: HGET-v1a. Figure 3 (right) shows the low-energy fission rate distributions for the HGETv1a designs, with an assembly present (each pixel in the image corresponds to an individual pin). Immediately evident is a relatively high fission rate on the generator side of the assembly, in the outer row of pins. The neutron self-shielding effect, which depresses the fission rate on the interior of the assembly due to interactions between the neutrons and the fuel pins, is also clear. The overall effect is a gradient of approximately 10X between the fission rates at the outermost to innermost pins – although if the outer row of pins is neglected, the fission rate in the rest of the assembly is within a factor of ~3X and has a predictable gradient structure, with no highly localized changes on the interior of the assembly.

For the collection of the prompt and delayed gamma rays produced by the induced fissions, an array of highly collimated gamma detectors is rotated around the assembly to build up the tomographic projection data, as a function of both energy and angle. A number of potential collimator/detector combinations are possible, but the nominal HGET design assumes a configuration founded on the IAEA's original Passive Gamma Emission Tomographer (PGET), as described more fully in [3] and depicted in Figure 4 below. Though PGET is intended for verification of spent fuel, a variant on PGET tailored for unirradiated fuels would benefit from a high degree of familiarity among tomography practitioners and the potential for leveraging of hardware components (e.g., detector arrays, pulse-processing electronics).

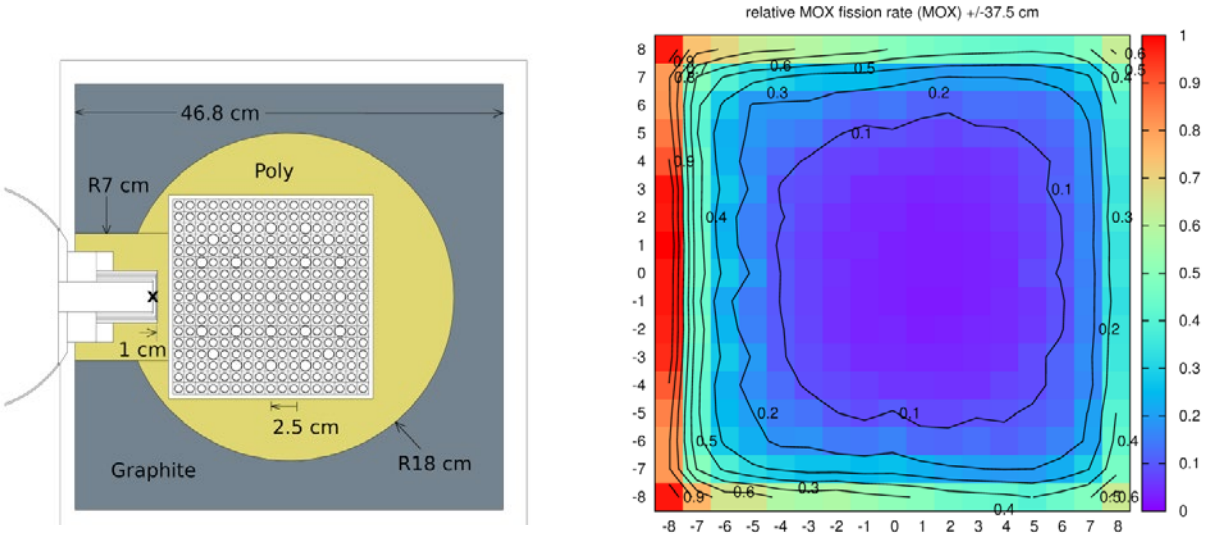

Figure 3: Left: Cross-section of the HGETv1a instrument geometry including a D-T generator (far left, with neutrons generated at the location of the x), a PWR assembly, and a combination of poly and graphite moderator/reflector. Right: Mapping of fission rate induced by low-energy neutrons, assuming the HGETv1a design (each pixel represents one fuel pin).

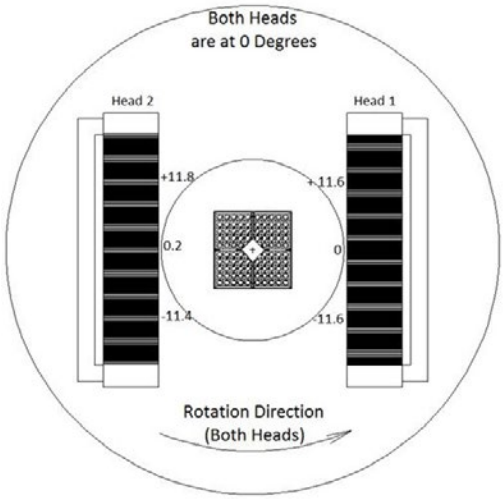

Figure 4: Rendering of the IAEA's PGET instrument design showing a vertical view of the detector heads containing 104 CdTe detectors in each head [3].

The central challenge in designing the HGET collimator/detector combination is to increase the collection efficiency significantly while preserving sufficient spatial resolution to resolve individual pins. To increase the collection efficiency for HGET gamma-ray signatures, the aperture's field of view was opened significantly in the horizontal and vertical directions. These adaptations produce an increase in the count rate at the detectors of approximately 30X compared to the PGET collimator.

The initial feasibility studies focused on a nominal design where the neutron generator and the gamma detectors lie in the same plane. Tests were also performed with designs that split the neutron reflector and placed the gamma detectors and staging in between, and with a stacked design with gamma detectors on a plane beneath the neutron reflector; both these configurations will result in lower count rates at the detector. An optimized design is expected to be intermediate between the ideal case shown here and the non-optimized tests.

$Bi_4Ge_3O_{12}$ (BGO) was chosen as the nominal gamma-ray spectrometer material for the HGETv1a design. Although it has significantly poorer energy resolution than other candidates such as NaI(Tl), CZT or $LaBr_3$, energy resolution is not expected to be critical for the broad energy windows for collection of prompt-fission and delayed-gamma signatures. BGO's high density and atomic number translate to high stopping power for higher-energy gamma rays, whilst maintaining a relatively small form factor that can support a highly arrayed detector arrangement like the one used in PGET. Importantly, BGO is a very common material for positron emission tomography systems used in nuclear medicine, which means that large arrays of relatively small voxels are readily available in the commercial market. The BGO detector is observed to give a factor of three improvement in intrinsic efficiency for the collection of gamma-rays above 1 MeV, when compared to the CdTe detectors employed in the original PGET device.

The combination of higher collimator efficiency (~30 X) and greater intrinsic detector efficiency (~3X) results in an overall HGET gamma-ray collection efficiency that is approximately two orders of magnitude higher than the original PGET design, thereby helping to recover a significant portion of the signal discrepancy (several orders of magnitude) between the spent fuel applications for which PGET was originally designed and the HGET scenarios for unirradiated fuels.

7. Reconstruction and Analysis Methods for HGET

The simplest approach to tomographic reconstruction is filtered backprojection, which solves analytically for the distribution of emissions, assuming that measurements are spaced at equal angles and that attenuation is minimal. This approach has the advantage of being both fast and requiring few assumptions about the system [13], and has been successfully used to locate missing pins in spent nuclear fuel [3]. However, since FBP in its simplest form makes no assumptions about attenuation, it cannot correct for the highly attenuating pins that block emissions from the center of the assembly. This leads to a reconstructed image which is systematically lower in intensity inside the assembly.

The case of fresh fuel is different from the spent fuel application in two important ways. First, the emission intensity is much lower, as previously noted. Second, emission intensity is a function not only of fuel composition, but also of illumination by the neutron field. Achieving a high and relatively uniform flux of thermal neutrons in the center of a large assembly is difficult, as discussed previously. The difference in counts at the detector from inner pins to outer pins is already large in passive emission tomography, but neutron interrogation adds another significant gradient, on the order of 10X, between inner and outer pins. Reconstructing a dataset with such an extreme gradient results in poor image quality.

While the HGET application is challenging from the reconstruction standpoint, it is decidedly different from other tomographic applications (e.g., nuclear medicine) in that it is fundamentally a confirmatory measurement of the operator's declaration about the assembly (as opposed to a blind test in which nothing about the object is known). This means that *a priori* information about a declared assembly, perhaps after initial verification via FBP, can be used to extract as much information as possible from each collected gamma ray and thereby improve the quality of the resulting image reconstruction. There are a number of ways to incorporate this information, but one straightforward approach is to assume a declared assembly geometry and solve for average emission values for each pin. Mathematically, this is phrased as measurement data (g_α) with α as the sinogram angle/offset index according to

$$g_\alpha = \sum_k H_{\alpha k} f_k \tag{1}$$

where f_k is the reconstructed activity estimate, here with k as the pin index, and $H_{\alpha k}$ is the model-based system response matrix, in this case the detector response to each possible emitting pin in the presence of attenuation due to the whole assembly. This approach is described more fully in [3][12]. Reconstructing at the level of individual pins, rather than over a series of pixels, incorporates the assembly geometry and greatly decreases the number of unknowns, regularizing the inverse problem. This results in much lower relative statistical error, but contingent upon the accuracy of the model.

The model-based tomographic reconstruction methods translate the collected gamma-ray signature into the emission intensity of prompt and delayed fission gammas in each pin, and therefore the pin-wise fission rate. Next, the pin-by-pin fission rate produced by the tomographic inverse problem must be translated to the verification parameter of interest: fissile-material concentration. De-tangling the fissile concentration from the fission rate must recognize that fission from non-fissile isotopes, most notably ^{238}U, can contribute significantly to the total induced fission rate, but the concentration of the non-fissile isotopes is not the IAEA verification parameter of interest. Such a translation can be complex since the fission cross-sections for the fissile and fissionable isotopes are highly dependent on incident neutron energy (including resonance structures and threshold reactions), and the neutron energy spectrum varies by pin location—due to attenuation from surrounding pins and attenuation within the pin of interest due to its own fissile loading. The methods used to translate total fission rate in a pin into fissile-isotope concentrations in that pin are beyond the scope of this paper but are described fully in [12].

8. Example Performance Prediction Results

In the early rounds of performance prediction studies, only the prompt-fission gamma-ray signal has been considered, and a straightforward model-based reconstruction on a pin-by-pin basis was used to bound the tomography inversion problem. The primary question to be addressed was: Based on the HGET v1a design, the "MOX A" assembly definition, anticipated operator declarations, and simulated prompt-gamma signatures, can reasonable statistical uncertainties be achieved for fissile-mass concentration on a pin-by-pin basis within 1-2 hours? The end-to-end HGET analysis process is shown in Figure 5 for this MOX A case study and reflects the discussions in the previous sections of this paper. In the example case-study results presented in Figure 6, it is assumed the operator declares the Pu isotopics for each pin.

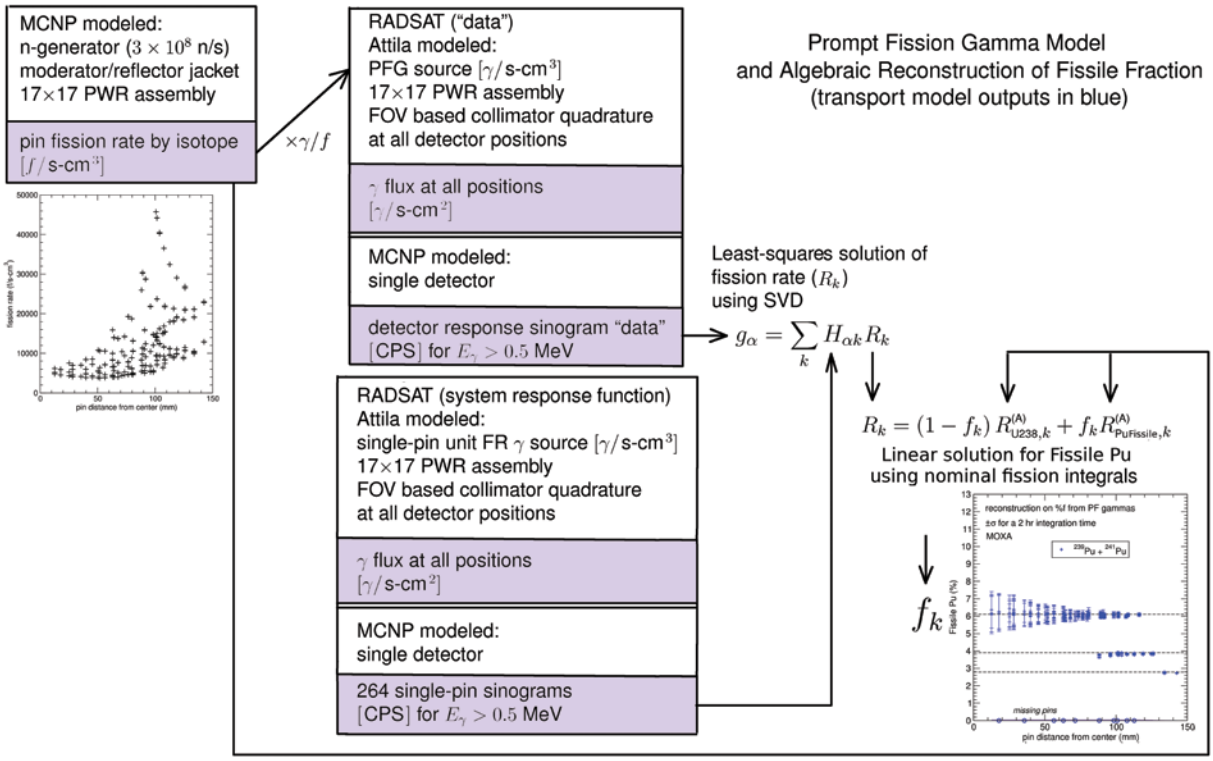

Figure 5: Overview of HGET performance-prediction methodology (assuming a PWR MOX assembly) that begins with forward calculations of induced fission rate (upper left) and culminates in quantification of fissile Pu concentration in each pin (lower right). Starting from left, MCNP calculates pin fission rates by isotope, which is used as a source term for gamma transport models (top center) and for interpreting pin-by-pin fission rates as fissile isotope fractions f_k (bottom right). Top center: sinogram data as a function of lateral position and angle g_α is calculated in counts per second (CPS) using RADSAT. Bottom center: RADSAT is used to calculate a pin-by-pin system response function using unit fission rates (FR) and a monte carlo detector response function (DRF). Singular value decomposition (SVD) is used to provide a least squares solution to invert the transport equation and recover individual pin fission rates R_k, which in turn provide fissile isotope fractions in each pin.

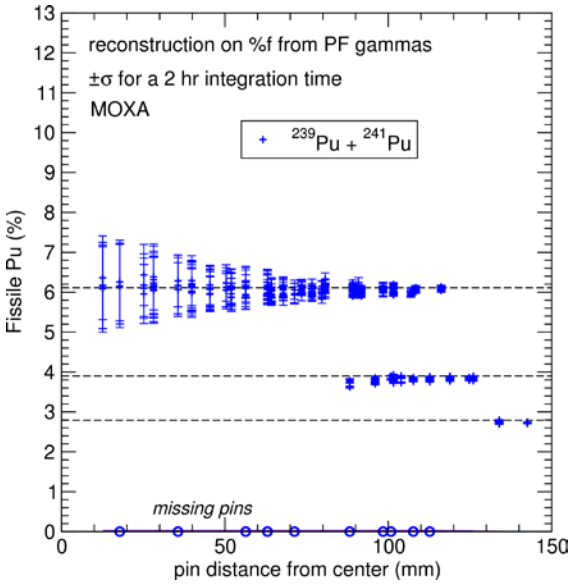

Figure 6: Example results for the determination of Fissile Pu (^{239}Pu + ^{241}Pu) fraction in MOX A, when incorporating assembly geometry and Pu isotopics data in the analysis process. This simulated assembly excluded 11 pins at different lattice positions from the middle to the edge of the assembly. Calculated values are based on 2-hour total assay time; one-sigma error bars on statistical uncertainty are shown. Dashed lines are the actual fissile fraction (^{239}Pu + ^{241}Pu) for the three Pu loading levels. (Note that Pu isotopics are identical for all loading levels; given the correct isotopic composition these numbers are equivalent to the wt% quoted in Fig. 1).

The preliminary results shown in Figure 6 indicate that: given a careful system design, a COTS portable neutron generator and reconstruction and analysis algorithms that fully acknowledge operator-declared information about the assembly, HGET has the potential to verify fissile-mass concentration on a pin-by-pin basis in total assay times of approximately 2 hours for a representative PWR MOX assembly. These initial performance estimates assumed that assembly type, missing-pin locations and Pu isotopics were accurately declared by the operator and incorporated in the analysis process. Under those assumptions, the relative standard deviation of the fissile-Pu quantification was less than a few percent for most pins, but approached 20% for the interior pins. The uncertainty in fissile concentration is small compared to the reconstructed contrast for the 11 missing pins.

9. Summary and Next Steps

While the initial HGET performance predictions presented in this paper are encouraging, further investigation is required to fully establish feasibility. From an operational standpoint, a neutron generator with 3 x 10^8 n/s is much higher flux than Am-Li sources currently used for active coincidence counting – typically 10^4 – 10^6 n/s [14]. This

raises concerns for exclusion area needed during a measurement. The measurement will induce some additional radioactivity in the assembly, although preliminary results indicate that the induced activity should be small relative to the original assembly activity. From a technical standpoint, the results here were generated using simulated data that does not include background terms, assuming full availability and accuracy of declared data for the assembly of interest, and assuming a near-perfect system response function in the tomographic reconstruction. Considerably more analysis is needed to more fully understand the potential of HGET and its viability for IAEA verification scenarios. The highest priority is the extension of the feasibility analysis to LEU, and to LEU with burnable poisons (Gd rods). The case of Gd loading in particular is challenging for coincidence counting assay, and the possibility of using the high-energy and high-intensity gamma rays from Gd to account for the burnable poisons offers potential advantages for HGET in overcoming the burnable poison effects. The performance predictions presented here were performed using only the prompt-gamma signature but delayed-gamma signatures become significant as the active-interrogation measurement progresses, and could be included in the reconstruction algorithms as a smoothly varying time dependent term. Continued study of HGET-specific reconstruction algorithms, particularly those that can identify the perturbation patterns created by missing pins and have robustness to imperfections in the system response function (e.g., undeclared or inaccurately declared missing pins), is needed. Perhaps most importantly, the HGET viability study needs to move into empirical space. The challenges of high-fidelity simulation for this relatively complex active interrogation approach (for which even basic cross-section data are not always available), and the inability of simulations to accurately capture the background terms that may arise in this active-interrogation scenario, strongly encourage proof-of-principle laboratory measurements using a representative tomographic device and objects (e.g., LEU fuel rodlets), to benchmark the predictive modeling tools and guide refinement of the nominal HGET instrument design.

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