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Nondestructive assay options for spent fuel encapsulation

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Abstract

This report describes the role that nondestructive assay (NDA) techniques and systems of NDA techniques may have in the context of an encapsulation and deep geological repository. The potential NDA needs of an encapsulation and repository facility include safeguards, heat content, and criticality. Some discussion of the facility needs is given, with the majority of the report concentrating on the capability and characteristics of individual NDA instruments and techniques currently available or under development. Particular emphasis is given to how the NDA techniques can be used to determine the heat production of an assembly, as well as meet the dual safeguards needs of 1) determining the declared parameters of initial enrichment, burn-up, and cooling time and 2) detecting defects (total, partial, and bias). The report concludes with the recommendation of three integrated systems that might meet the combined NDA needs of the encapsulation/repository facility.

Contents

Acronyms

2D	Two-Dimensional
BU	Burnup
BWR	Boiling Water Reactor
CDH	Calorimetric Decay Heat
CIPN	^{252}Cf Interrogation with Prompt Neutron Detection
Clab	Centralt mellanlager för använt bränsle, in English: Central interim storage facility for spent nuclear fuel
Clink	Centralt mellanlager för använt bränsle samt inkapslingsanläggning, in English: Central interim storage facility for spent nuclear fuel and Encapsulation Plant.
CN	Coincident Neutron
CT	Cooling Time
D/S	Ratio of the Doubles Count Rate to the Singles Count Rate
DCVD	Digital Cherenkov Viewing Device
DD	Deuterium-Deuterium
DDA	Differential Die-Away
DDSI	Differential Die-Away Self-Interrogation
DG	Delayed Gamma
DN	Delayed Neutron
DT	Deuterium-Tritium
GT	Gamma Tomography
HPGe	High-Purity Germanium
IAEA	International Atomic Energy Agency
IE	Initial Enrichment
KAERI	Korean Atomic Energy Research Institute
LEU	Low Enriched Uranium
LINAC	Linear Accelerator
LSDS	Lead Slowing Down Spectrometer
MOX	Mixed Oxide
NDA	Nondestructive Assay
NGSI	Next Generation Safeguards Initiative
NGSI-SF	Next Generation Safeguards Initiative Spent Fuel
NRF	Nuclear Resonance Fluorescence
NRTA	Neutron Resonance Transmission Analysis
PDET	Partial Defect Tester
PG	Passive Gamma
PNAR-FC	Passive Neutron Albedo Reactivity with Fission Chambers
PWR	Pressurized Water Reactor
SINRD	Self-Integration Neutron Resonance Densitometry
SKB	Swedish Nuclear Fuel and Waste Management Company
SMOPY	Safeguards MOX Python
SSM	Strålsäkerhetsmyndigheten, in English: Swedish Radiation Safety Authority
TN	Total Neutron
UGET	Unattended Gamma Emission Tomography
XRF	X-Ray Fluorescence

1 Introduction

This report describes the role that nondestructive assay (NDA) techniques and systems of NDA techniques may have in the context of an encapsulation and deep geological repository. The potential NDA needs of an encapsulation and repository facility include safeguards, heat content, and criticality. Some discussion of the facility needs is given, with the majority of the report concentrating on the capability and characteristics of individual NDA instruments and techniques currently available or under development. To the degree possible, currently attainable uncertainty estimates are given. The description of each instrument is also complemented with information regarding requirements on measurement times and needed infrastructure.

2 Spent fuel management at the Clink Facility

The Clink Facility will combine the existing interim storage facility Clab and the soon-to-be-built encapsulation plant Inka, resulting in an encapsulation plant with storage capacity. A detailed description of the clink facility can be found on-line at the following link: http://www.skb.se/upload/publications/pdf/Inka2008Eng.28.1NY.pdf

As long as the Swedish nuclear power plants are in operation, spent nuclear fuel will be arriving at the Clink Facility. Once at the Clink Facility, the fuel will go 50 m underground into storage pools before going to encapsulation. The age of fuel assemblies being encapsulated could range from about 10 to 80 years, with an average age of around 30 years (Liljenfeldt H 2012, personal communication).

After being removed from underground storage, the fuel is taken to the handling pool, where it will be moved from storage canisters into transfer canisters. Each transfer canister is filled up with the fuel that will later be going into the copper disposal canister. The fuel is being selected for the disposal canister at the encapsulation facility, so the safeguards and decay heat measurements also must be made there.

The possible positions for fuel measurements within the pool are the following (Liljenfeldt H 2012, personal communication):

- On the pool side, with instruments being either used from the surface or lowered into the pool.
- In a measurement room next to the pool via collimators. The room is ~3 × 10 m.
- Inside the pool or at fuel racks; the anticipated space is ~1.5 × 1.5 m.
- Integrated into the fuel handling equipment (transfer crane).

Because of the throughput of the Clink Facility of one canister per day, it is not reasonable to make more than one measurement of a given fuel assembly, which should be on the order of 30 minutes (Liljenfeldt H 2012, personal communication). It is anticipated by the operator (Liljenfeldt H 2012, personal communication) that boiling water reactor (BWR) assemblies will need to be measured at three locations on each assembly and pressurized water reactor (PWR) assemblies at one location, most likely near the axial centre.

Once the fuel has been selected, it is taken to the hot cell for drying. Then the assembly is transferred to the insert that is in the disposal canister. This is the last time the fuel assemblies are handled individually. When the insert that is inside the disposal canister is filled, it is covered by a steel lid. Up until the moment when the lid is placed up on the canister, the identification numbers of each individual assembly can still be read. Complementary measurements could be made within the hot cell for verification.

Finally the insert inside the disposal canister is filled with argon gas and the disposal canister is welded shut. The quality of the weld is then inspected for flaws. The canister then is decontaminated before transport by boat to the final repository, where it will be put 500 m down in the bedrock.

3 Heat load requirements

The ability to determine the decay heat of each individual assembly before loading the assembly into the canister has a large impact on the total cost of the final repository. A larger uncertainty in the heat in the canister means either a larger number of canisters need to be interred or more space is needed between the canisters in the repository, both resulting in a larger area for the repository.

Because the throughput of the encapsulation plant does not allow two different measurements to be made, it is important to determine if obtaining good results is possible when the safeguards measurements are combined with a burn-up code to determine the decay heat.

4 Safeguards requirements

In this section a brief summary of the safeguards requirements of the International Atomic Energy Agency (IAEA) are given. The Clink facility is also under the safeguards jurisdiction of Euratom and the Swedish Radiation Safety Authority (SSM). The three regulatory bodies are currently engaged in communication to assure effective and efficient safeguards across their varied jurisdictions.

From communication with the IAEA through October 2012, it is understood that spent fuel assemblies at Clink will be treated as "items" until final deposition in the repository. Under item accountancy, the primary goal of NDA measurements performed on the fuel is *the verification* of the operator-declared information. Under item accountancy, the IAEA needs to verify the "correctness" of declared parameters, (i.e. initial enrichment, burn-up, and cooling time) and the "completeness" of the item, (i.e. that the item is intact, meaning no pins have been removed). If verification methods indicate that operator's declaration is correct and complete, *then the operator's values for fissile material are adopted* for the purposes of the facility material balance.

The following quote from the 2010 IAEA publication by the title "Model Integrated Safeguards Approach for a Spent Fuel Encapsulation Plant" describes the need for a partial defect measurement as part of assuring the "completeness" of the declaration.

Each spent fuel assembly transferred into a disposal canister should have been verified by a partial defect test or, if not available, the best available method approved for inspection use, and continuity of knowledge should be maintained through the time of permanent closure of the disposal canister. If the spent fuel was verified by a partial defect test or best available method at the reactor or spent fuel storage installation, and successful dual containment and surveillance was maintained, reverification is not required (IAEA 2010).

For Clink it is anticipated that the majority of the assemblies need a partial defect measurements since they were not previously verified with a partial defect test approved for inspection and/or dual containment and surveillance was not maintained.

Additional clarity on the partial defect test were stated in the 2009 IAEA publication "IAEA Safeguards Criteria: Special Criteria for Difficult-to-Access Fuel Items" in which the quantitative standard for a partial defect was defines as assuring that "*at least half of the fuel pins are present*."

The requirement for partial defect tests applies… to irradiated fuel assemblies which can be dismantled at the facility, in which case the partial defect case used should assure that at least half of the fuel pins are present in each assembly (IAEA 2009).

Note that the partial defect determination needs to have a 90% or better confidence level for detecting the absence of pins. A point of clarification to the need for partial defects, only a gross defect test is needed for assemblies that cannot readily be disassembles such as those that were welded together. Because most of the assemblies in Sweden are not welded, a partial defect test is needed. Some additional considerations for the NDA system are the following: 1) it needs to be unattended for long periods of time, 2) data security needs to be considered as part of the overall design, 3) robust technologies are highly desirable, and 4) shared instruments are highly desirable, for reasons that include cost-sharing and maintenance.

Because the Clink Facility is unique and the final repository is the first of its kind, the safeguards requirements are yet to be fully defined. Currently the operator is interested identifying the best available technology that fits with the operational requirements of the Clink.

To verify the identity of the fuel assembly, the following parameters will be examined:

- Read assembly identification number.
- Burn-up (BU).
- Cooling time (CT).
- Initial enrichment (IE).
- Possible fissile content of multiplication.

Throughout most of this report the assumption is made that no declared information can be used in the analysis process. It is worth pointing out that declared information or acceptable knowledge may be deemed usable and will very likely reduce the uncertainty in the final determined quantity. Because the information that may be deemed as acceptable knowledge will vary with the applications and because the decision on acceptable knowledge for the Clink facility has not yet been determined, the more general assumption that no information is known has been made.

Another point of general relevance to this report, in the safeguards profession there is the need for both portable and permanently installed technologies. In general the toolkit of inspectors has primarily relied upon portable instruments. Yet, in the case of a facility such as Clink, it is very likely that a permanently installed piece of hardware can be used. Note that a change to non-portable hardware will require research on issues of data authentication.

5 Description of the Nondestructive Assay (NDA) options

5.1 Introduction

This section is a summary of techniques available for the NDA of spent nuclear fuel assemblies that are to be encapsulated for a deep geological repository. This section contains a brief description of each instrument currently available or under development. The measured signal and discussion of expected uncertainties are included. The description of each instrument is also complemented with information regarding requirements on measurement times and needed infrastructure. The time needed for handling fuel has not been included in the measurement times. All uncertainties in this report are specified for a 1 sigma confidence interval unless otherwise noted.

Since much of this report leverages the research done under the Next Generation Safeguard Initiative (NGSI) Spent Fuel Project (Humphrey et al. 2012), it is important to indicate how the goals of that project do, and do not, compliment the Safeguards needs of the Swedish encapsulation/repository facility. The NGSI Spent Fuel Project has the goal of quantifying Pu mass; also, this project has the philosophical approach that integration of a few NDA signals together is the best path to take in pursuit of Pu mass determination. Furthermore, the NGSI Spent Fuel project had a many year time horizon. As a consequence, the results extracted from the NGSI reports presented in this SKB sponsored report focus on intermediate quantities such as fissile content (also known as $^{239}Pu_{effective}$); generally the goal of the individual NDA technique reports was to connect the raw signal with the quantity that caused the raw signal to vary. The NGSI results presented are for non-integrated techniques focused on quantifying how the given NDA technique's signal varied with IE, BU and CT.

As stated elsewhere in this report, the safeguards needs of Swedish encapsulation/repository facility are "correctness" of the declaration (verify IE, BU, and CT) and "completeness" of the declaration (detect if there is a partial defect). The impact of IE, BU and CT on a wide range of NDA signals was researched by the NGSI project because the NGSI project research used a wide range of fuel that varied in IE, BU and CT; yet, additional work is needed on indicating how well and in what exact way IE, BU and CT can be determined with given NDA techniques and/or combinations of techniques. This is a research area being pursued during the fourth year (2013) of the NGSI project. The results presented here are generally those that were available after the second year of the NGSI project since that is when the individual NDA technique reports were completed. Preliminary work on partial defect detection capability will be started in 2013.

The NDA techniques described in this report involve photon or neutron detection. In the case of photons, high purity germanium (HPGe) is primarily used. The primary reason for this is that among the off-the-shelf technology, HPGe has the best spectral resolution. Other detection materials such as CdZnTe are also mentioned with a few NDA techniques because, in some applications, the fact that HPGe needs to be actively cooled, while CdZnTe crystals do not, become a tradeoff worth making.

In the case of neutron detection, fission chambers were most commonly used because they do not need gamma shielding to reduce the gamma dose. Also, in the spent fuel context there are so many neutrons to count that the low efficiency of fission chambers is not a problem in most applications described. ^{3}He technology is only needed in the case of correlated neutron detection because, in that application, the superior efficiency of ^{3}He is needed. Currently there is significant research being done to replace ^{3}He. "^{3}He replacement" technologies are a rapidly evolving field and an accurate summary of current progress was deemed beyond the scope of this document. Note that a typical 2.5 cm diameter fission chamber is roughly 100 times less efficient per unit length relative to a ^{3}He tube.

The selection of which photon or neutron technology to make is not a focus of this work. In most cases, several options will work and what is commonly done today is what was primarily discussed.

5.1.1 ^{252}Cf Interrogation with Prompt Neutron Detection (CIPN)

What is measured: Prompt neutrons emitted at the end of an induced fission chain that was initiated by a ^{252}Cf source (or neutron generator) placed on the far side of the assembly from the detectors, as illustrated in Figure 5-1.

What is quantified: Multiplication or fissile content (weighted sum of ^{235}U, ^{239}Pu, and ^{241}Pu); for fissile content to be determined, a neutron absorber correction is necessary. For thermal-induced prompt fission per unit mass, ^{239}Pu and ^{241}Pu produce ~1.5 and 2.0 times as many neutrons as ^{235}U, respectively.

Description of the basic physics: Two measurements of the total neutron (TN) count rate are made with a detector that is very similar to a fork detector, see Section 5.3. For the first measurement, the ^{252}Cf source is located far from the assembly. This first measurement quantifies the passive neutron count rate. For the second measurement, the only change is that the ^{252}Cf source is brought in close to the assembly, ~5 cm from the center of one side of the assembly that is opposite the detector. This second measurement quantifies the combined count rate of the background and the neutrons that induced fission in the assembly. Given the size of the assembly and the dimensions of the detector, the probability of a ^{252}Cf source being directly detected is small. The net signal above the background overwhelmingly is due to the fission chain reaction that occurs across the assembly. Fission chambers were selected for the thermal neutron detectors to keep the instrument light (Hu et al. 2011).

Spatial distribution of the signal: The induced fission density is a gradient across the assembly decreasing with distance from the ^{252}Cf source. The length and locations of the fission chambers about the assembly, as well as the positioning of a cadmium sheet on the side of the assembly near the ^{252}Cf source, were selected to make the signal response to each pin in the assembly similar.

Expected measurement time: A ~100-s count duration will produce a statistical uncertainty of ~0.2% for a ~0.5-m section of the assembly. For the CIPN design used by the NGSI Spent Fuel Project, a CIPN signal that was ~75% stronger than the background (net count rate of ~1×10^4 counts/s) was produced with a 2×10^8 n/s ^{252}Cf source (100 µg) for a fully burned assembly (45 GWd/tU, 4% wt % ^{235}U, 5 years cooled). Note that the largest commercially available ^{252}Cf source is 50 times stronger than the source used here. The count time can be shortened by using a stronger source. A deuterium-tritium (DT) or a deuterium-deuterium (DD) generator could be used instead of a californium source.

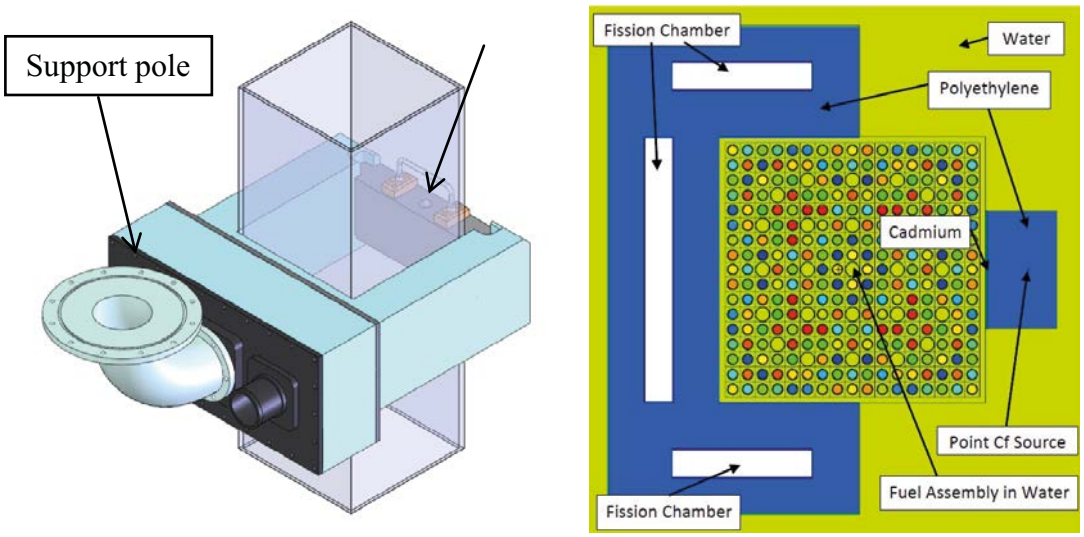

Figure 5-1. *Left: Mechanical drawing of a CIPN instrument (Polk P, personal communication); supported on a pole, the californium source either is inserted in the removable door or is part of the removable door. Right: Conceptual design of CIPN instrument as simulated in MCNP for the NGSI Spent Fuel Project (Hu et al. 2011).*

Discussion of calibration, precision and accuracy: The precision of a CIPN instrument is similar to that of differential die-away (DDA) and delayed neutron (DN) systems; the precision will be excellent because the net count rates will be very high: the net CIPN signal needs to be sufficiently large compared with the large background, and the ^{252}Cf source will be selected to give a good signal-to-background signal. In the context of accuracy in meeting Clink's safeguards needs, the "correctness" of the declaration (IE, BU and CT) and the "completeness" (detection of a partial defect) are the targets. It is expected that correctness will be determined by integrating the CIPN signal with PG and TN. More research is needed to assess the capability of such an integrated system. Part of such an assessment will include quantifying systematic uncertainties such as assembly positioning and isotopic spatial variation within the assembly; simulations performed to date indicate a positioning uncertainty of approximately 1% is expected for a 5 mm water gap. Part of future assessments need to investigate the utility of using fresh assemblies, or spent assemblies that may be considered as "working standard." The role of simulation and its uncertainty in the calibration process needs to also be considered.

Unique features relative to other NDA techniques: CIPN is similar to DDA in terms of prompt neutrons being detected, as well as the interrogating source and neutron detectors that are set up. The two techniques traditionally differ in that DDA normally does not count neutrons while the source is "on," whereas CIPN does.

Maturity of the hardware: The hardware is extremely robust (fission chambers and a ^{252}Cf source), and the safeguards field has over 30 years of experience. Essentially, CIPN is a fork detector with a californium source added; a fork detector has been used with spent fuel for over two decades. Moving the ^{252}Cf source rapidly is unnecessary. The source could be located at the end of a pole, at the end of a cable, or in the removable door, as indicated in Figure 5-1; in either case, moving the source rapidly is unnecessary.

Maturity of the analysis: French and British researchers have used CIPN with spent fuel (Simpson et al. 2008). However, depending on the information that is intended to be extracted from the signal, the analysis may or may not be mature. The simplest signal to extract is multiplication, which is essentially the difference between the two measurements. Quantifying the fissile content or plutonium mass involves more analysis and is less mature.

Considerations for implementation: Care must be used in working with a strong ^{252}Cf source. The US Department of Energy has experience with manipulating ~500-µg sources at several sites. These sources are used in DN instruments called shufflers (Rinard 2001). CIPN for spent fuel will likely use a ~200-µg source so that it is strong enough to use for ~5 years given the 2.6-year half-life of ^{252}Cf.

5.1.2 Calorimetric Decay Heat (CDH)

Decay heat in nuclear fuel is defined as the heat produced within the fuel assembly as a result of radioactive decay.

What is measured: The temperature increase in water surrounding the fuel assembly, placed within a calorimeter (see Figure 5-2).

What is quantified: The thermal power (energy released per time unit) in the fuel assembly.

Description of the basic physics: The technique consists of the following steps:

1. Establish a calibration between temperature increase in the calorimeter and a well-known (electrical) power input to the volume within the calorimeter.
2. Measure the temperature increase in the calorimeter with a nuclear fuel assembly that is positioned within the calorimeter. Use the calibration to get a decay heat value from the measured temperature increase.
3. Correct the decay heat value for losses due to radiation that escapes from the calorimeter.

Expected measurement time: One calorimetric measurement of one fuel assembly takes on the order of 4–5 hours (SKB 2006). The measurement time is nearly the same for BWR and PWR assemblies.

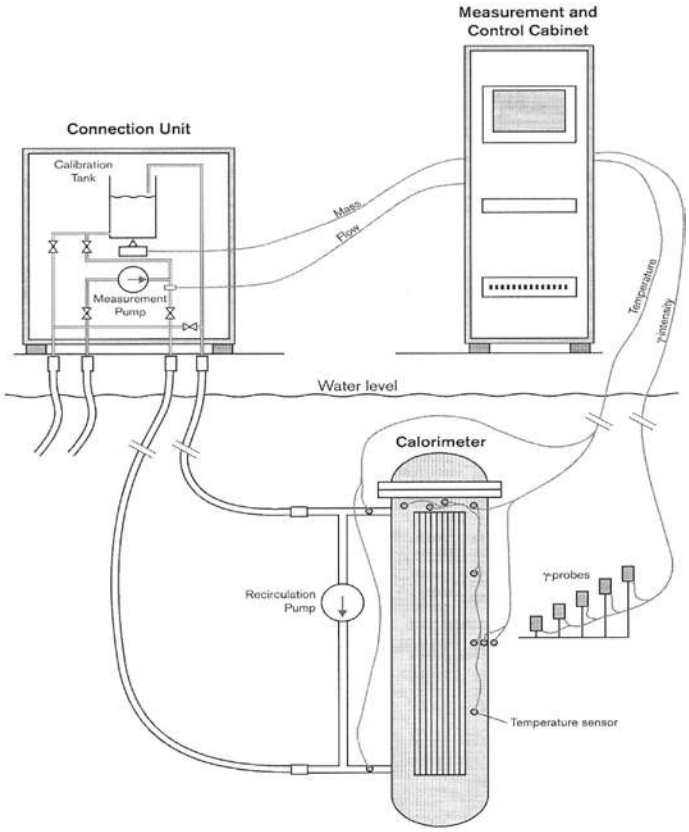

Figure 5-2. *Schematic illustration of the calorimetric measurement system in use at the Clab Facility, Oskarshamn, Sweden.*

Discussion of calibration precision and accuracy: SKB (2006) reports that the decay heat in BWR and PWR fuel assemblies measured with a calorimeter at the Clab Facility was determined with an uncertainty on the order of 1%–2%.

Unique features relative to other NDA techniques: No other technique in this work measures the decay heat directly.

Maturity of the hardware: The hardware is composed of off-the-shelf items. The calorimeter itself is made of stainless steel. Other calorimetric systems have been in used in science for many tens of years. The hardware is mature.

Maturity of the analysis: Analysis of thermal power by a temperature increase has been used for many tens of years. The correction for escaping radiation has been used at the Clab Facility for about 10 years. The analysis is mature.

Considerations for implementation: The size of the calorimeter and supporting structures is on the order of 1 × 1 × 5 m.

Development: The infrastructure used and the fuel handling needed to measure with a calorimetric technique could be optimized for speed.

The calibration used should be verified for fuel types other than those measured so far. For example, the electrically heated fuel dummy should be modeled accordingly, and the impact on thermal conductivity and flow within the calorimeter should be understood.

Further, the correction for escaping radiation is currently performed using measurements of dose rate around the calorimeter. The analysis of escaping heat from these measurements is based on a fit of measurement data to an assumed dose rate distribution, which could be developed further to take into account the effect of different assembly types on the dose rate distribution. However, the heat loss due to escaping radiation is in the order of a few percent of the total heat, implying that uncertainties in the dose rate distribution contribute little to the total uncertainty of the decay heat.

5.1.3 Delayed Gamma (DG)

What is measured: Photons emitted from fission products in the seconds to minutes following an active interrogation burst. The ~3- to ~6-MeV energy range is of primary interest (Mozin et al. 2011).

What is quantified: Multiplication, fissile content (weighted sum of ^{235}U, ^{239}Pu, and ^{241}Pu), or relative masses of four main isotopes that fission. For fissile content to be determined, a neutron absorber correction is necessary. The role of ^{238}U can be minimized by lowering the energy of the interrogating neutrons by both measuring in water and placing judiciously selected material between the neutron generator and the fuel. However, if desirable, the contribution of ^{238}U can be maximized by keeping the neutron energy elevated. By locating several detectors around the interrogated region, each with different collimators, spatial information for the emitting isotopes can be obtained; also these additional detectors can reduce the count time and/or the neutron generator strength.

Description of the basic physics: An active interrogation source such as a neutron generator or linear accelerator (LINAC) is used to produce neutrons for the purpose of inducing fission in the assembly, as illustrated in Figure 5-3. The fission products produced by induced fission are the source term for the DG measurement. The majority of the detectable DGs are emitted from fission products with half-lives in the 1.0- to 1,000-s time interval (Beddingfeld and Cecil 1998). The NGSI Spent Fuel Project is currently researching the optimal combination of interrogation and count time; results to date indicate that the interrogation scheme selected for the 2011 Review Committee Report is a reasonable choice. In that report, a 15-minute interrogation, 1.0-minute cool down, 15.0-minute count time scheme was selected (Mozin et al. 2011). The ability of a DG measurement to discern among ^{235}U, ^{238}U, ^{239}Pu, and ^{241}Pu is derived from the data depicted in Figure 5-4. This figure illustrates per fission what percentage of total fissions, for each of the four main isotopes, results in a particular fission product.

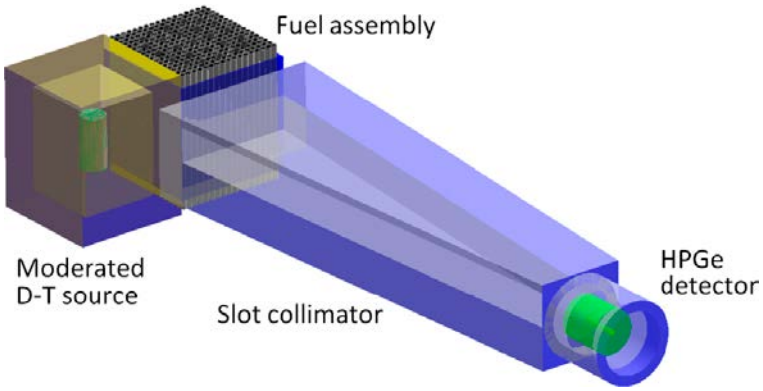

Figure 5-3. *Conceptual design of DG instrument as simulated in MCNP for the NGSI Spent Fuel Project (Mozin et al. 2011).*

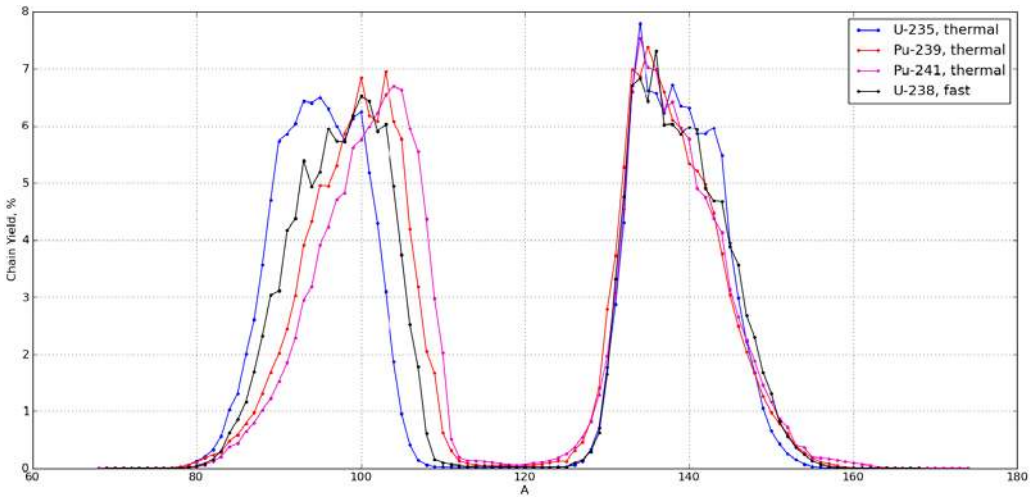

Figure 5-4. *Illustration of the fission product yield per fission for each of the four main isotopes (Mozin et al. 2011).*

The basic concept for quantifying the relative mass of each isotope rests on detecting DG rays from several fission products so that the relative intensities of the emitted lines allow the separation of the relative contribution of each of the four major isotopes. A calibration is needed to determine absolute masses from the relative measurements. The calibration options are a point of active research in the NGSI Spent Fuel Project. A relatively high-resolution detector is used to detect DGs; $LaBr_3$ and high-purity germanium (HPGe) are the crystals of primary interest.

A ~3-MeV lower limit to the detection window was selected because ~3.2 MeV appears to be the upper limit of the relatively intense passive gamma (PG) background as determined through simulation. This limit is still a point of active research, but the current anticipation of the NGSI Spent Fuel Project is that the background above ~3.2 MeV is less than 1 count/s. The promise of extracting meaningful spectra in the ~2.2- to ~3.2-MeV range is a topic of active research. The very large passive photon flux below 3.2 MeV is a major consideration in the design of this instrument. Attenuation and collimation are both used to keep the total count rate of the detector within the limits of current technology. Compton suppression technology could be used; however, since the collimator limits the incident photons to the central area of the crystal, the usefulness of Compton suppression technologies is limited.

Spatial distribution of the signal: The fission products are produced in a gradient across the assembly, decreasing with distance from the neutron generator. The detection probability of the photons is determined by the location of the fission product in the assembly; unlike other induced fission techniques, multiplication cannot help the signal "get out" of the assembly. It is worth noting that the attenuation coefficient is at a minimum for the DG energy window. The intensity of a 3-MeV peak in UO_2 (10 g/cm^3) is reduced by ~50% in ~1.7 cm; however, given streaming between rods, DG photons will be able to exit the assembly. It is expected that DG will not be very sensitive to photons from the center of the assembly. The sensitivity of the analysis to assembly alignment is in need of research.

Expected Measurement time: Given that the analysis approach is a point of active research, this measurement time value is a rough approximation: ~30 minutes for measuring a ~0.5-m section of the assembly with a 1×10^{12} DT neutron generator (Mozin et al. 2011). An average strength of $\sim 1 \times 10^{11}$ n/s is expected to be a lower limit to the neutron generator strength when ~10 detectors are used. The neutron generator intensity will need to be measured on a pulse by pulse basis by a dedicated detector to assure the accurate intensity of the pulse.

Discussion of calibration, precision and accuracy: The precision of the DG system is largely determined by how large a neutron generator or LINAC is used. In the context of accuracy in meeting Clink's safeguards needs, the "correctness" of the declaration (IE, BU and CT) and the "completeness" (detection of a partial defect) are the targets. It is expected that correctness will be determined by integrating the DG signal with any of the other neutron generator techniques (DN and DDA, both can be available since they need a lower intensity generator), PG and TN. More research is needed to assess the capability of such an integrated system. Part of such an assessment will include quantifying systematic uncertainties such as assembly positioning and isotopic spatial variation within the assembly. Part of future assessments need to investigate the utility of using fresh assemblies, or spent assemblies that may be considered as "working standard," as well as the role of simulation in the calibration process.

Unique features relative to other NDA techniques: The DG technique can detect photons from multiple fission products; each peak is indicative to varying degrees of the mass of ^{235}U, ^{238}U, ^{239}Pu, and ^{241}Pu. This physics enables the de-convolution among the various peaks to measure the relative or absolute masses of each. Among the NDA techniques considered for this review, only DG has "neutrons-in/photons-out," which means that the interrogating source uses neutrons to induce fission, whereas the detected signal is composed of photons emitted from the fission location. Compared with "neutrons-in/neutron-out" techniques, the uniformity of detection across the assembly is less. This difference in spatial sensitivity may be useful in diversion detection in an integrated system.

Maturity of the hardware: A custom-built neutron generator or LINAC is anticipated because the source intensity is roughly an order of magnitude greater than the largest commercially available neutron generator; thus, the neutron source is of medium/low maturity. Photon detection with HPGe is considered to be a mature technology.

Maturity of the analysis: The measurement of DGs from spent fuel is immature. The discussion of analysis maturity needs to consider the end quantity of interest. Determining the relative mass of the isotopes that fission is less challenging than determining the fissile content or plutonium mass. DG analysis has not been used regularly in non-spent fuel applications in the safeguards profession.

Considerations for implementation: The primary issue of concern for implementation is the need for a very large neutron generator. It is anticipated that a ~$4 million and ~3 years will be needed to fabricate the necessary generator, although using a LINAC would not involve such a development cycle. Using a LINAC would require that the LINAC be designed into the facility so that the neutron source could be located adjacent to the pool and the fuel brought to the edge of the pool next to the LINAC.

5.1.4 Delayed Neutrons (DNs)

What is measured: Neutrons emitted from fission products in the seconds to minutes following an active neutron generator burst.

What is quantified: Fissile content (weighted sum of ^{235}U, ^{239}Pu, and ^{241}Pu) emphasizing the presence of ^{235}U.

Description of the basic physics: A neutron generator is used to produce neutrons for the purpose of inducing fission in the assembly, as depicted in Figure 5-5. The fission products are the source term for the DN emission. The majority of the detectable DNs are emitted from fission products with half-lives in the 2- to 22-s time interval (Rinard 2001). The interrogation scheme selected for the 2011 Review Committee Report (Blanc et al. 2011) used a 0.9-s interrogation interval, followed by a 0.1-s pause (for the burst neutrons to die away), followed by a 1.0-s DN count interval (Blanc et al. 2011); the timing is flexible: faster or slower will work. Optimization of timing and neutron generator design can reduce the strength of the neutron generator needed; however, it is not expected that an improvement of more than a factor of 2 can be practically obtained. The net DN count rate is the difference between the passive background count rate measurement made before interrogation and the DN count rate determined during active interrogation. The desired precision is obtained by repeating the interrogation/pause/count cycle.

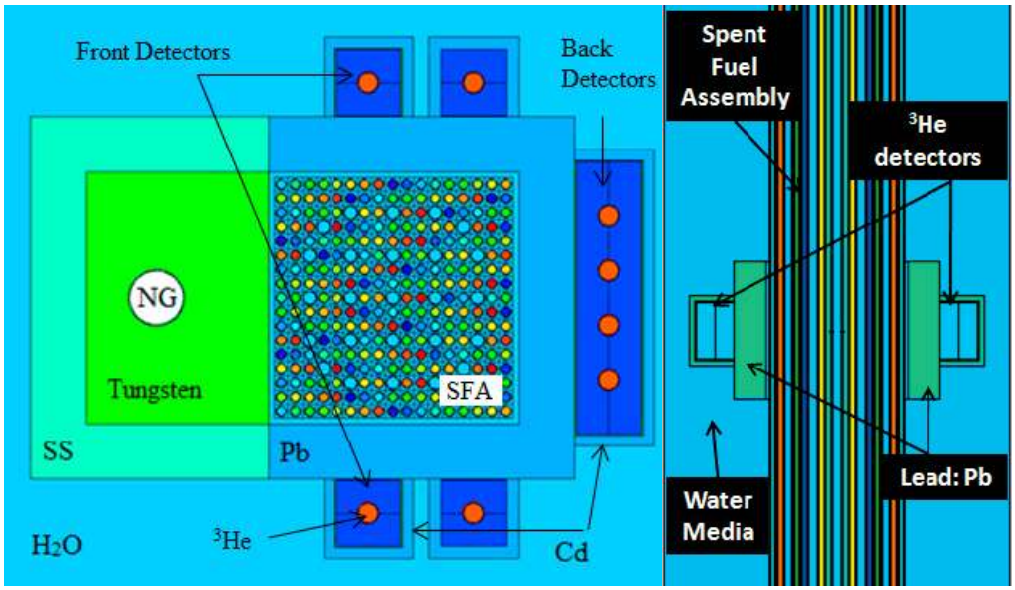

Figure 5-5. *Conceptual design of the DN instrument as simulated in MCNP for the NGSI spent fuel project (Blanc et al. 2011) for which SS means "stainless steel," NG means "neutron generator" and SFA means "spent fuel assembly".*

In Table 5-1, the fission cross section (σ), DN fraction (β), and yield per fission (υ) are listed for the four main isotopes of relevance. This table emphasizes the point that DNs preferentially measure ^{235}U relative to ^{239}Pu by a factor of ~2 per unit mass. Several of the other techniques have the opposite weighting; for prompt fission induced by thermal neutrons, per unit mass, ^{239}Pu produces ~1.5 times as many neutrons as ^{235}U and ^{241}Pu produces ~2.0 as many neutrons as ^{235}U. Table 5-1 also indicates that ^{241}Pu and ^{238}U could be significant contributors. The role of ^{241}Pu is important but not dominant because the mass of ^{241}Pu is generally 4 to 10 times less than that of ^{235}U. The role of ^{238}U is minimized to a few percent by keeping the interrogation energy below ~1 MeV.

Spatial distribution of the signal: The fission products are produced in a gradient across the assembly, decreasing with distance from the neutron generator. The detection efficiency gradient for the DN instrument was designed to be essentially the opposite of the fission product gradient so that the combined effect is a relatively uniform signal across the assembly. The uniformity of response is enhanced by multiplication, both during the interrogation phase and during the DN counting phase.

Expected measurement time: For the ^{3}He design used by the NGSI Spent Fuel Project, a fully burned assembly (45 GWd/tU, 4 wt % ^{235}U, 5 years cooled) produced a net DN signal of ~5×10^5 counts/s, with a background signal of ~15×10^5 counts/s for a neutron generator that produced 1×10^{11} n/s (Blanc et al. 2011). For such a setup, only ~1 s is needed to obtain less than 1% statistical uncertainty. It is not reasonable to reduce the DN signal much because a healthy signal–to-background ratio is needed; thus, a reasonable design change is to use fission chambers instead of ^{3}He tubes. Even with fission chambers, a count time of ~10 s is expected for one ~0.5-m section.

Discussion of calibration, precision, and accuracy: The precision of a DN system will be excellent because the net count rates will be very high: the DN signal must have significant strength relative to the large background, and the neutron generator will be selected to give a good signal-to-background signal. In the context of accuracy in meeting Clink's safeguards needs, the "correctness" of the declaration (IE, BU and CT) and the "completeness" (detection of a partial defect) are the targets. It is expected that correctness will be determined by integrating the DN signal with DDA, PG and TN. More research is needed to assess the capability of such an integrated system. Part of such an assessment will include quantifying systematic uncertainties such as assembly positioning and isotopic spatial variation within the assembly. Part of future assessments need to investigate the utility of using fresh assemblies, or spent assemblies that may be considered as "working standard," as well as the role of simulation in the calibration process.

Unique features relative to other NDA techniques: DN is the only technique that preferentially weights ^{235}U on a gram-for-gram basis relative to the other fissile isotopes. The DN instrument cannot discern among ^{235}U, ^{239}Pu, and ^{241}Pu isotopes; as such, if the goal is to quantify plutonium mass instead of fissile content, it is important to integrate DN with other instruments for separation of the contribution among the fissile isotopes.

Maturity of the hardware: It may be possible to use one to four commercial neutron generators for the source, or it may be necessary to have a custom generator built. The first step in advancing from a conceptual design to a fabrication design is determining if a commercial generator(s) can be used. Research into minimizing the neutron generator strength needs to consider optimal timing, spectrum tailoring, and generator-to-assembly distance. The hardware for a custom neutron source is of medium/low maturity.

Table 5-1. Fission Cross Section (σ), DN Fraction (β), and Yield per Fission (υ) for the Four Major Sources of DN. The Final Column Gives a Weighting of Each Isotope in the DN Signal on a Per Atom Basis (Blanc et al. 2011).

Isotope	Fission Cross Section (barns)	DN Fraction as a Yield per Fission ($\beta\nu$, %)	$(\sigma\beta\upsilon)_{isotope}/(\sigma\beta\upsilon)_{Pu\text{-}239}$
^{235}U	584 (thermal)	1.65	2.03
^{239}Pu	742 (thermal)	0.64	1
^{241}Pu	1,010 (thermal)	1.58	3.44
^{238}U	0.7 (~2 MeV)	4.39	6.86 (note: fast/thermal ratio)

Maturity of the analysis: DN analysis in safeguards is mature. However, in the context of spent fuel, the analysis is relatively immature. The type of research needed has not yet been determined. Quantifying multiplication, fissile content, or plutonium mass requires different analyses.

Considerations for implementation: The primary issue of concern for implementation is determining if a custom-built neutron generator is needed.

5.1.5 Differential Die-Away (DDA)

What is measured: Traditional DDA measured the prompt neutrons emitted by induced fission during a time interval when the active generated interrogating neutrons were thermal in energy (Caldwell and Kunz 1982). The NGSI Spent Fuel Report (Lee et al. 2011) produced for the NGSI Review Committee implemented traditional DDA for which the count interval started 200 μs after the burst ended so that the thermal neutrons from the burst would be ~1% of the neutrons produced by induced fission. The updated DDA reports by the NGSI Project (Henzl et al. 2011), expanded the scope of the DDA research by looking at the signal at times between the termination of the burst and 200 μs. This more recent research also looked at the spatial response between different detectors at different time intervals. These recent results are still being called DDA, although strictly speaking the research has strayed from the traditional mode of counting the TN count rate only when thermal neutrons are remaining in the sample of interest.

What is quantified: Multiplication and fissile content (weighted sum of ^{235}U, ^{239}Pu, and ^{241}Pu) emphasizing the presence of ^{239}Pu and ^{241}Pu. For prompt fission induced by thermal neutrons, per unit mass, ^{239}Pu and ^{241}Pu produces ~1.5 and 2.0 times as many neutrons as ^{235}U, respectively. Preliminary results indicate that the elemental plutonium mass may be determined by combining the DDA signal with the TNs count rate (Henzl 2012).

Description of the basic physics: A measurement begins with the burst of neutrons (~10 μs in duration was used for the NGSI research) produced by a neutron generator, as illustrated in Figure 5-6. A DD (2.2-MeV) or DT (14-MeV) generator would work. Those burst neutrons slow down to near thermal energies (0.025 eV). Because the cadmium-covered detectors detect neutrons only above ~0.5 eV, for the NGSI setup with ^{238}U and oxygen in the fuel, after ~200 μs the detected count rate for the burst neutrons is low. Subsequent simulations indicated that the count rate of the burst neutrons was ~1% of the count rate from a fully burned assembly at 200 μs (Lee et al. 2011). Note that even though the direct count rate from the burst neutrons is very low, the burst neutrons are still present in the fuel as thermal neutrons – they are just very unlikely to penetrate the cadmium and thus are very unlikely to produce a count. If fissile material is present in the fuel, induced fission by thermal neutrons will occur and neutrons will be produced with much higher energy (~2 MeV on average). Some of the induced fission neutrons will have energies above the cadmium cutoff energy when they arrive at the detector and will be detected. Traditional DDA functions by count neutrons during a time window when the burst neutrons are negligible compared with the induced fission count rate. For the NGSI Review Committee Report (Lee et al. 2011), a time window of 200 to 1,000 μs was used. It is expected that a DDA system with spent fuel would pulse the generator with a frequency of ~100 Hz.

For traditional DDA, data are collected only after the neutrons from the neutron generator have moderated to below ~0.5 eV; thus, the neutrons primarily detected are produced by fissions induced by the thermal neutrons. Because the thermal fission cross sections of ^{235}U, ^{239}Pu, and ^{241}Pu are 584 b, 742 b, and 1,010 b, respectively, and because the number of prompt neutrons emitted per thermal fission is 2.41, 2.88, and 2.8, respectively, the prompt neutron signal on a per-atom basis will weight these three fissile isotopes as 1.0, 1.5, and 2.0, respectively.

Because the DDA involves interrogating the fuel with thermal neutrons, a logical concern is self-shielding. However, because a fully burned assembly is significant multiplying, results to date for PWR assemblies indicate that the entire fuel assembly is interrogated.

Spatial distribution of the signal: The induced fission density during the DDA counting interval is a gradient across the assembly that decreases with distance from the neutron generator. The detection efficiency for the DDA instrument was designed to be essentially opposite to that of the induced fission gradient so that the combined effect is a relatively uniform signal from all pins across the assembly. The uniformity of the DDA signal is significantly enhanced by multiplication, both during the interrogation phase and during the DDA counting phase.

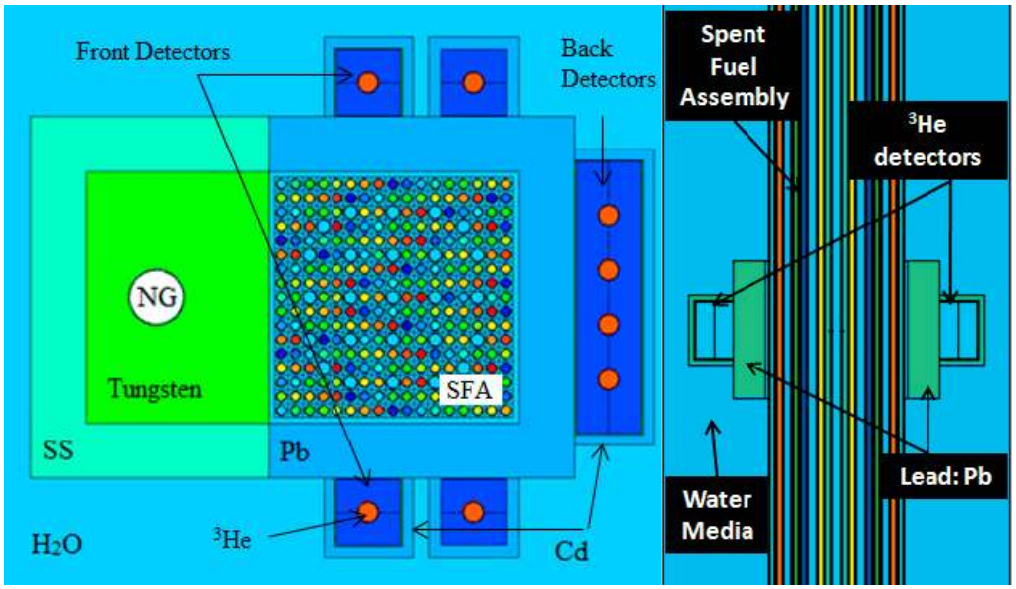

Figure 5-6. *Conceptual design of the DDA instrument as simulated in MCNP for the NGSI spent fuel project (Blanc et al. 2011) for which the acronyms are defined in Figure 5-5.*

Expected measurement time: A few seconds count time is expected for a 0.5-m section of the assembly. For the ^{3}He design used by the NGSI Spent Fuel Project, a DDA signal (0.2- to 1.0-ms integration window) that was ~50% stronger than the background was produced with a 5×10^8 n/s neutron generator (10-µs burst, 100-Hz repetition rate) for a fully burned assembly (45 GWd/tU, 4 wt %, ^{235}U, 5 years cooled). Because the background for such an assembly is ~15×10^5 counts/s, excellent statistics are obtained in a second. For this reason, it is likely that fission chambers would be used instead of ^{3}He tubes. Note: Since the NGSI Review Committee Spent Fuel Report was written, a variety of integration windows have been explored. For some of these windows, a weaker generator will be needed, and for others, a stronger generator will be needed. Recent results indicate that an uncooled neutron generator of ~1×10^8 n/s will suffice.

Discussion of calibration, precision and accuracy: The precision of a DDA system is similar to that of a DN system; it will be excellent because the net count rates will be very high: the count rate needs to be comparable to the large background, and the neutron generator will be selected to give a good signal-to-background rate. In the context of accuracy in meeting Clink's safeguards needs, the "correctness" of the declaration (IE, BU and CT) and the "completeness" (detection of a partial defect) are the targets. It is expected that correctness will be determined by integrating the DDA signal with PG and TN. More research is needed to assess the capability of such an integrated system. Part of such an assessment will include quantifying systematic uncertainties such as assembly positioning and isotopic spatial variation within the assembly. Part of future assessments need to investigate the utility of using fresh assemblies, or spent assemblies that may be considered as "working standard," as well as the role of simulation in the calibration process.

Unique features relative to other NDA techniques: A rich variety of information is available in the ~1 ms following a burst. This richness has been explored only in the past year, and significant work remains to be done. The analysis goes beyond traditional DDA and should perhaps have a different name given to it. Preliminary results indicate that multiplication in the assembly can be quantified by three different means: elemental plutonium can be determined, IE and BU can be determined if the CT is known, and calibration likely can be done with fresh assemblies (Henzl et al. 2011, 2012a, Henzl 2012).

Maturity of the hardware: It is possible to use one commercial neutron generator for the source because less than 1×10^9 n/s is needed (Lee et al. 2011, Henzl et al. 2012a). The detector tubes can use ^{3}He or fission chambers. In summary, the hardware is considered to be *mature*.

Maturity of the analysis: DDA analysis in safeguards is mature. However, it has traditionally been applied in the context of waste measurements for which very small quantities of material (often gram and sub-gram levels) are present in a large volume. In spent fuel we have a very different situation; kilogram quantities of highly multiplying material. In the context of spent fuel, the analysis is *not mature.*

Considerations for implementation: Putting a neutron generator into a spent fuel pool may not be trivial. Issues surrounding this activity have not been addressed. However, commercial generators of sufficient output were designed to go down well-logging holes; therefore, putting them under water is not expected to be a major challenge.

5.1.6 DDA Self-Interrogation (DDSI)

What is measured: The time and location at which each neutron is detected, with an accuracy of ~0.1 µs. Singles and doubles count rates are calculated from these raw data.

What is quantified: Multiplication or fissile content (weighted sum of ^{235}U, ^{239}Pu, and ^{241}Pu); for fissile content to be determined, a neutron absorber correction is necessary. For thermally induced prompt fission per unit mass, ^{239}Pu and ^{241}Pu produce ~1.5 and 2.0 times as many neutrons as ^{235}U, respectively.

Description of the basic physics: Two possible designs of a DDSI detector are depicted in Figure 5-7. Traditional DDA begins with a neutron generator burst; the burst neutrons interrogate the sample, and data are collected only after the burst neutrons have become thermal in the sample. DDSI has DDA in its name because DDSI also has a burst; however, in the case of DDSI, the burst is a spontaneous or induced fission event that liberates nubar neutrons. DDSI has two signals of interest in the context of spent fuel. One signal uses the ratio of the count in an early time gate to the counts in a late time gate. The other signal uses the ratio of the doubles count rate in the late gate to the singles count rate (D/S) (Schear et al. 2011). The D/S ratio is a standard quantity used in classical coincidence counting. What makes the DDSI doubles calculation unique is the use of a very long delay between the measurement of a neutron trigger and the opening of the gate (Schear et al. 2011). In traditional coincidence counting, an integration interval in time, a gate, is opened as soon after the detected trigger neutron is detected as possible within the limits of the electrical system. With DDSI the gate is delayed for the purpose of separating the passive interrogating signal, composed primarily of ^{244}Cm, from a signal that is primarily induced fission. The first induced fission, in a chain of induced fissions, is delayed in time by ~10 µs from the time when the initiating neutron was born. This delay is approximately the time required for the neutron that initiates a fission to thermalize. Note that in spent fuel, ~80% of induced fissions occur at thermal energies. Given that the multiplication in a fully spent assembly is ~2, a series of induced fissions is common and the induced fission signal can be largely separated from the initiating burst event (often a ^{244}Cm fission).

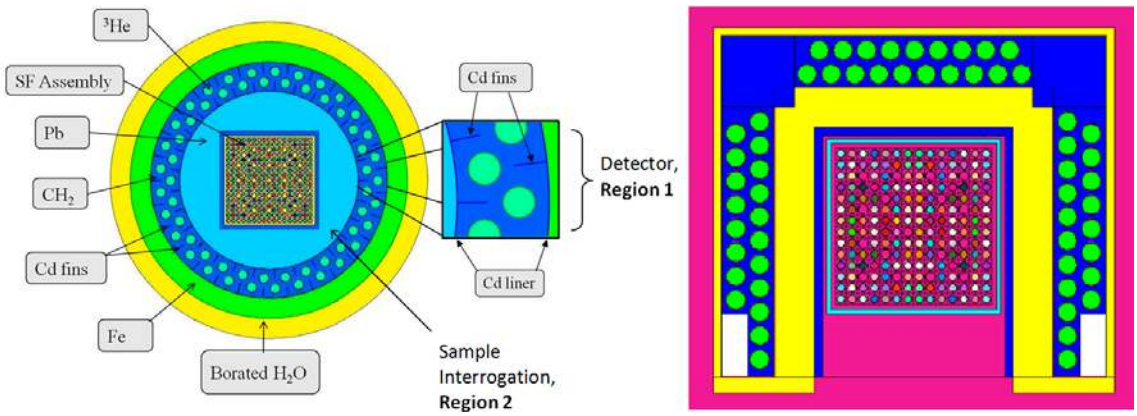

Figure 5-7. *Left: Conceptual design of DDSI instrument as simulated in MCNP for the NGSI Spent Fuel Project (Schear et al. 2011). Right: Modified DDSI design to enable fuel to be loaded from the side (Belian et al. 2012).*

Spatial distribution of the signal: Of all the photon and neutron techniques, preliminary results indicate that DDSI is the most spatially uniform in response. The interrogating source is spread throughout the assembly. From a diversion detection perspective, DDSI is more sensitive to a change in multiplication from the center of the assembly than from the edge. From a "first-generation" perspective, the response is very uniform, with a slight emphasis on the outermost row (Blanc et al. 2012).

Expected measurement time: The measurement time for the doubles dictates the overall measurement time. For a 45-GWd/tU, 4 wt %, ^{235}U, 5-years-cooled assembly, a statistical uncertainty of 1% for the doubles can be obtained in 2 minutes for a 20-µs pre-delay and a 32-µs gate width. For this same assembly with a 60-µs pre-delay and a 32-µs gate width, it takes 16 minutes to obtain the same uncertainty. By changing the pre-delay from 20 to 60 µs, the sensitivity doubled for a change in the fissile content, but the count time increased by a factor of 8 (Schear et al. 2011). Thus, the count time can be varied significantly, depending on the sensitivity required.

Discussion of calibration, precision and accuracy: The precision of a DDSI system is determined by the efficiency of the instrument, multiplication of the fuel, and intensity of the passive neutron source. In the context of accuracy in meeting Clink's safeguards needs, the "correctness" of the declaration (IE, BU and CT) and the "completeness" (detection of a partial defect) are the targets. It is expected that correctness will be determined by integrating the DDSI with PG and TN. More research is needed to assess the capability of such an integrated system. Part of such an assessment will include quantifying systematic uncertainties such as assembly positioning and isotopic spatial variation within the assembly. Part of future assessments need to investigate the utility of using fresh assemblies, or spent assemblies that may be considered as "working standard," as well as the role of simulation in the calibration process.

Unique features relative to other NDA techniques: The uniformity of response is the primary unique feature of DDSI. Noteworthy also is the large sensitivity to changes in fissile content for a passive instrument.

Maturity of the hardware: The hardware is very mature for assemblies up to ~30 GWd/tU for a 17 × 17 assembly; below that, BU ^{3}He tubes and current list mode can handle the count rate. Above the ~30-GWd/tU BU level, a new data acquisition capability may be needed. A tradeoff between efficiency and BU of the assembly can be measured. The instrument being fabricated for testing can tolerate assemblies up to the ~45-GWd/tU BU level (Belian et al. 2012). The availability of ^{3}He could be an issue, yet safeguarding a repository may make this system a high priority, thus enabling the acquisition of ^{3}He.

Maturity of the analysis: Moderate maturity. Research is ongoing to determine if the relative amount of ^{239}Pu and ^{235}U can be determined from time-correlated neutrons. Also, the count rate issue could be minimized by using a lower-efficiency system; however, it is not known how low is acceptable.

Considerations for implementation: It is expected that current technology is close but may not be able to handle the high count rate of all the Swedish fuel. Using a less-efficient system, reducing tube diameter, and optimizing electronics all lead to the ability to function at higher count rates. A project began in 2012 to test out reduced-diameter tubes and optimized electronics.

5.1.7 Digital Cherenkov Viewing Device (DCVD)

What is measured: The Cherenkov (ultraviolet) light produced by the gamma radiation escaping the fuel assembly (see IAEA 1997 and Figure 5-8).

What is quantified: The device is used to identify an assembly with gross or partial defects.

Description of the basic physics: When the gamma radiation from the fuel assembly is absorbed in the surrounding water, recoil electrons are produced, with a velocity exceeding the speed of light, and therefore lose energy by emitting Cerenkov radiation. The Cherenkov viewing device is optimized to view the ultraviolet light produced in the water surrounding the fuel assembly. The glow of the Cherenkov light is bright in the regions close to the assembly, where fuel rods are present.

Expected measurement time: The DCVD is a camera, so one image is collected within 1–2 s. The image is saved for offline analysis. Chen et al. (2003) reported that verification of 12 fuel assemblies was performed in 82 minutes.

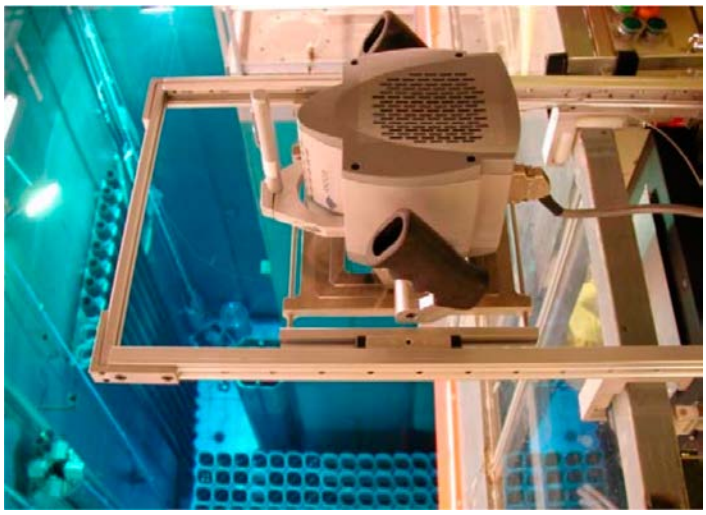

Figure 5-8. *The DCVD mounted above a fuel storage pool.*

Discussion of calibration precision and accuracy: Chen et al. (2010) and Lindberg et al. (2006) conclude that defects on the order of 50% can be detected in PWR fuel assemblies. No information has been found regarding uncertainties in the defect level that can be detected.

Unique features relative to other NDA techniques: No other technique uses the Cherenkov light produced in the water surrounding the fuel assembly.

Maturity of the hardware: The DCVD is a digital version of the older analog Cherenkov viewing device. The hardware is under development.

Maturity of the analysis: The analysis of DCVD images is under continued development. Yet since it is a commercially available product, the analysis is more mature than the majority of techniques in this report.

Considerations for implementation: The DCVD device is handheld and contains a platform for mounting it on a facility railing or fueling machines. The device must be placed above the fuel assembly to be able to view down on it.

Development: The analysis of images from the DCVD is an ongoing research topic. The applicability of the technique to partial defect verification should be verified on fuel types other than those measured so far.

5.1.8 Gamma Tomography (GT)

What is measured: The two-dimensional (2D) intensity distribution of gamma radiation of one or more energies at one axial level of the fuel assembly. The distribution is measured over many lateral and angular positions around the fuel assembly (see Figure 5-9).

What is quantified: The 2D emission distribution at one axial level of the fuel assembly is quantified. The two main applications of the technique are

1. integrity verification and
2. determination of the pin-power distribution to validate production codes for core simulation at nuclear power plants.

Note that using tomography for measuring the ^{137}Cs distribution throughout the assembly should give a better correlation to decay heat than ordinary PG scanning (where only the outer pins are effectively seen by the detector).

Description of the basic physics: Tomographic reconstruction techniques are used to calculate the emission distribution using the measured intensity distribution.

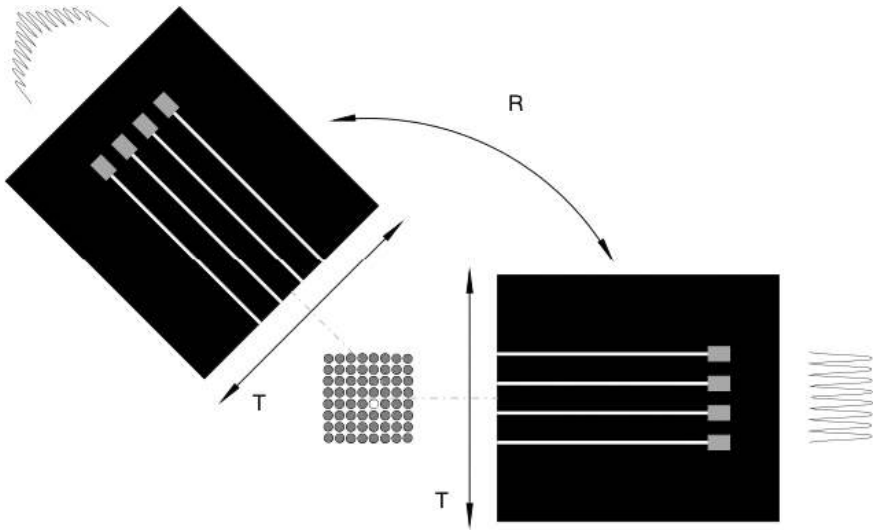

Figure 5-9. *Schematic image of the translational (T) and rotational (R) movements involved in a tomographic measurement. The intensity pattern of radiation measured in one translation scan is indicated behind the detector/collimator package.*

Expected measurement time: Jacobsson Svärd et al. (2005) estimated that 25 axial positions of a BWR assembly with ~1 month of CT could be measured in ~8 hours using 1,596 keV of gamma radiation from ^{140}Ba. Note that the spent fuel of interested to Clink will be significantly older (10 to 70 years cooled) and that all the ^{140}Ba will have decayed away; for this reason an isotope such as ^{137}Cs (662 keV) will be needed. Count time estimates for assembly of interest to Clink are need and are underway as part of an IAEA coordinated research effort. The area of a PWR assembly is about 1.5 × 1.5 times larger than a BWR assembly, which would imply a 2.25 times higher measurement time. It should be noted that the measurement time is inversely proportional to the number of detectors used in the equipment. For instance, using the system described in Jacobsson Svärd et al. (2005) but with 16 detectors, the measurement time per axial position would be ~10 minutes. Note that these measurement times are specified for the application to determine pin-power distribution, which needs better accuracy than the application to verify fuel integrity.

Discussion of calibration precision and accuracy: For the safeguards application, detection of partial defects on a single-rod level has been demonstrated in Jacobsson Svärd (2004). For the pin-power application, the distribution of pin power inferred from the distribution of ^{140}Ba within a BWR fuel assembly has been determined with an uncertainty of 3% (see Jacobsson Svärd et al. 2005) using bismuth germanium oxide (BGO) scintillator detectors.

Lundqvist et al. (2007) has shown that tomographic reconstruction techniques can be used to determine the position information and fuel assembly type information needed for a full reconstruction whereby missing fuel pins can be determined without prior knowledge of the assembly.

Measurements on PWR (15 × 15 or 17 × 17) assemblies have not yet been performed. Computer simulations, documented in Andersson (1997) demonstrated that the tomographic technique would be able to detect single pin partial defects, even in 17 × 17 assemblies by using gamma radiation from ^{154}Eu. However, the relatively short half-life of ^{154}Eu restricts its use to assemblies with short CT. The replacement of spent fuel with fresh fuel or fuel-like material leads to more confident detection of the manipulation compared with the case where rods have been replaced with water.

Unique features relative to other NDA techniques: The tomographic reconstruction technique is unique among the methods considered in this work in the sense that it can provide a cross-sectional view of the assembly on different axial levels. The DCVD can provide a cross-sectional view from the top of the assembly, and the view can be blocked by hardware on top of the assembly.

Maturity of the hardware: The technique uses off-the-shelf commercial equipment. The combination of devices used in the equipment in the high-radiation field from a fuel assembly has been tested only a few times; therefore, the hardware is considered to be of low maturity.

Maturity of the analysis: Tomographic image reconstruction techniques have been used for decades in medicine. The image reconstruction techniques are mature. Tomographic reconstructions that take into account attenuation within the assembly are subject to research and are not as mature.

Considerations for implementation: Jacobsson Svärd et al. (2005) and Jansson et al. (2006) demonstrated the tomographic technique and used a prototype tomographic device that was on the order of 2 × 2 × 5 m. Lundqvist Saleh et al. (2010) has shown that a more compact device can be constructed that is on the order of 50 × 50 × 50 cm that would presumably be stationary, with the fuel assembly movement performed by other equipment. The collimator opening size in Lundqvist Saleh et al. (2010) was proposed to be 1 × 10 mm.

Development: The International Atomic Energy Agency (IAEA) has previously coordinated the development of a cylindrical-shaped device with the assembly being inserted in the center of the cylinder; this research was conducted by the Finnish, Hungarian and European Commission support programs. This instrument contains about 200 CdTe detectors; however, development is ongoing, and information as to the performance of the system is unknown to the authors. Although the selection of CdTe was not optimum from an energy detection perspective, CdTe does have the positive attribute of room temperature operation and sufficiently minimal neutron damage at the rates typical to spent fuel assay.

The setup of a member state support program with Sweden, Finland, USA and the European Commission has been proposed by the IAEA, in which unattended gamma emission tomography (UGET) is to be further developed for back-end purposes.. In this context the use of detector materials and pulse-processing electronics with properties appropriate for the detection of gamma-ray energies up to several MeV, and high-rate operation in intense radiation fields, will be considered.

It should be emphasized that using gamma tomography on fuel assemblies with a CT between 50 and 80 years have never been investigated. Due to the decay of gamma emitting fission products, most of the radiation will originate from ^{137}Cs at such long CTs. It remains to be investigated if ^{137}Cs can be used on, e.g. PWR fuel. This will be studied within the proposed member state support program to the IAEA mentioned in the previous paragraph.

The Organization of Economic Cooperation and Development's Reactor Halden Project at Institute for Energy in Norway is beginning to use the tomographic technique for in-house purposes. It has also shown interest in the technique in a broader context.

Work currently being performed at Uppsala University shows that verification of fuel completeness is possible with the tomographic technique (see Figure 5-10) without prior knowledge of fuel type or geometry. Figure 5-10 shows an image produced by tomographic reconstruction techniques using no prior knowledge of the type of fuel assembly or other geometrical information. Using image analysis techniques, a histogram of rod positions was established whereby assumptions of "wrongly positioned fuel rods" could be detected as outliers in the histogram (red-colored rods in Figure 5-10).

5.1.9 Lead Slowing Down Spectrometer (LSDS)

What is measured: Prompt neutrons from induced fission as a function of incident neutron energy.

What is quantified: Conceptually the mass of ^{235}U, ^{239}Pu, and ^{241}Pu can be determined although the current analysis approach to such an absolute mass determination depend significantly on using fission chambers of each of these isotopes, only ^{235}U fission chambers are readily available. The relative mass of each of these isotopes is an easier goal than the absolute mass of any particular isotope.

Description of the basic physics: As illustrated in Figure 5-11, the spent fuel is positioned near the center of a large cube of lead (~1.5 meters on a side). An active neutron source sends out a burst (~10 μs in duration) of neutrons from near the center of the lead cube (Abdurrahman et al. 1993, Smith et al. 2011). These neutrons slow down gradually, given that they mostly collide with lead. During the time interval when the neutron energy is below the fertile fission cross sections, the prompt neutrons from the fission of ^{235}U, ^{239}Pu, and ^{241}Pu are measured. The unique features in the cross section of these three isotopes are used to unfold the amount of each isotope from the total measured prompt neutron signal. A key design goal is to keep the slowing down neutrons "tight" in energy, which is easy when the interrogating object is small but a challenge for a spent fuel assembly.

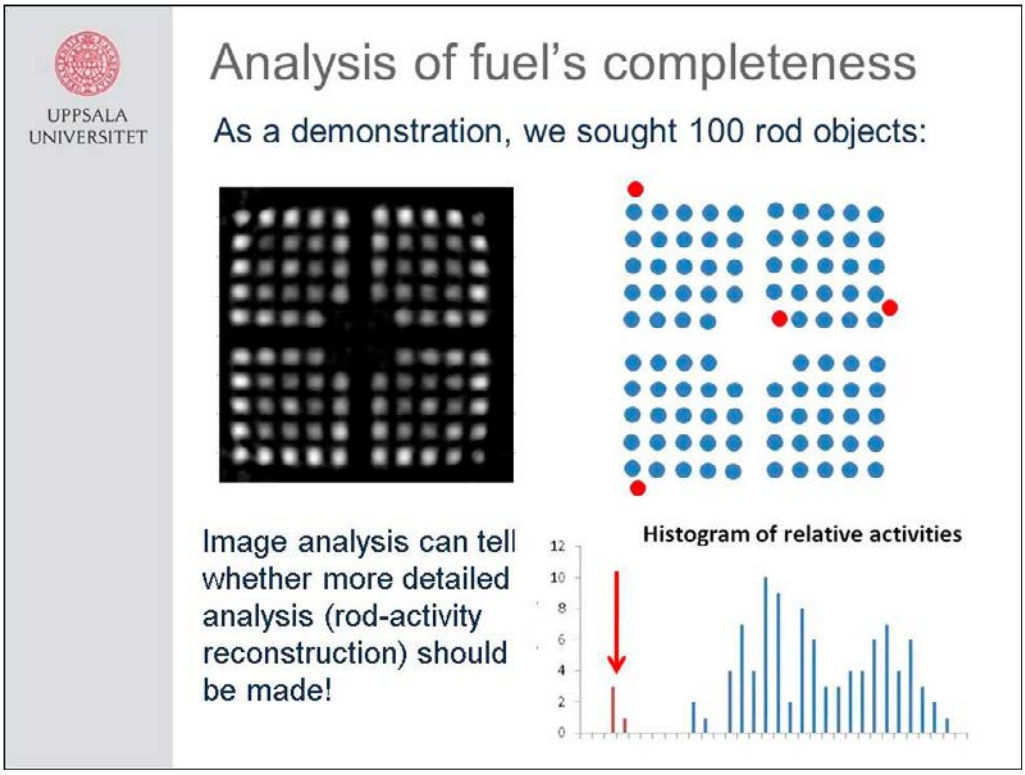

Figure 5-10. *Using image analysis of tomographic measurements without prior knowledge of the fuel geometry or composition can provide enough information to determine both the geometry (which can be used for refined tomographic analysis) and verification of fuel completeness.*

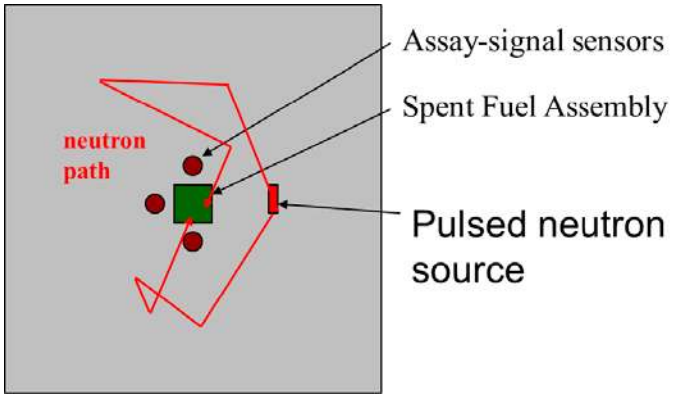

Figure 5-11. *Conceptual design of an LSDS, indicating the location of detectors, fuel, and neutron source within the lead cube (Smith et al. 2011).*

Of note, the presence of hydrogen in the cladding at levels of a several hundred parts per million makes a noticeable impact on the assay results. Another concern for LSDS assays is the penetrability of the neutrons into the assembly, particularly at lower neutron energies; the current analysis approach incorporates a self-shielding correction.

Spatial distribution of the signal: Spatial distribution is a topic of ongoing research. Because the information on the mass of the various isotopes is obtained from the entire neutron energy spectrum and because the penetration of the neutrons through the assembly is a strong function of neutron energy, it is challenging to assess from what depth into the assembly the information is obtained. What is clear is that self-shielding is a challenge for current LSDS researchers, and thus, the center of the assembly is not as well sampled as the exterior.

Expected measurement time: A rough estimation is that a 3×10^{12}-n/s neutron generator or accelerator is needed for ~1 hour to measure a ~1-m axial length of fuel.

Discussion of calibration, precision, and accuracy: The accuracy and precision are difficult to assess at this time. In the context of accuracy in meeting Clink's safeguards needs, the "correctness" of the declaration (IE, BU and CT) and the "completeness" (detection of a partial defect) are the targets. It is expected that correctness will be determined by integrating the LSDS with PG and TN. More research is needed to assess the capability of such an integrated system. Part of such an assessment will include quantifying systematic uncertainties such as assembly positioning and isotopic spatial variation within the assembly. Part of future assessments need to investigate the utility of using fresh assemblies, or spent assemblies that may be considered as "working standard," as well as the role of simulation in the calibration process.

Unique features relative to other NDA techniques: Similar to neutron resonance transmission analysis (NRTA) and nuclear resonance fluorescence (NRF), the LSDS signal measures cross-sectional properties of the ^{235}U, ^{239}Pu, and ^{241}Pu – the fission cross-section in the case of LSDS. Thus, isotope specific data are produced for the purpose of measuring individual isotopes.

Maturity of the hardware: The neutron source is likely to be a custom-built neutron generator or a neutron-producing accelerator system. The ^{238}U, ^{239}Pu and ^{241}Pu fission chambers are possible to obtain, but considerable effort is needed in their production. The ^{232}Th fission chambers can substitute for the ^{238}U fission chambers with factor of ~3 reductions in efficiency. The need for ^{239}Pu and ^{241}Pu fission chambers is a topic of ongoing research.

Maturity of the analysis: Several decades of experience exist with LSDS in the context of small, relatively pure samples. The maturity is considered medium in the context of spent fuel.

Considerations for implementation: Measurements must be made in air. Because several hundred parts per million of hydrogen in the cladding has a noticeable impact on the assay, other sources of hydrogen must be examined, such as coolant as part of the neutron generator and any water that might linger on the fuel after removal from a pool. Current analysis uses ^{239}Pu and ^{241}Pu fission chambers, yet neither of these is available commercially. The use of ultra-pure ^{238}U fission chambers may be an issue, as well.

5.1.10 Coincident Neutron (CN)

What is measured: Time-correlated neutrons from which doubles and triples count rates are calculated.

What is quantified: Multiplication or fissile content (weighted sum of ^{235}U, ^{239}Pu, and ^{241}Pu); for fissile content to be determined, a neutron absorber correction is necessary. Because the measured signal originates primarily from prompt induced fission for which nubar and the fissile cross section are greater for fissile plutonium, the plutonium fissile isotopes are emphasized relative to ^{235}U.

Description of the basic physics: A conceptual design of a CN detector is depicted in Figure 5-12 and is identical to the conceptual DDSI detector. CN counting is a subset of multiplicity counting. For multiplicity counting using shift register logic, the number of counts in two different time windows is quantified. The first window is opened *very soon after each detected neutron* so that a neutron produced from the same initiating fission as the triggering neutron is more likely to be detected. The second gate is significantly separated in time from when the triggering neutron was detected such that any neutrons in the second gate are not correlated with the triggering event. A distribution is formed from the difference between the total counts in these two gates. From this distribution, the count rate for detecting coincident events, the doubles count rate, can be determined. The rate at which three correlated neutrons are detected or the triples count rate can also be quantified. The text "very soon after each detected neutron" is italicized above since this is the key statement that distinguishes the gates used by CN and DDSI. With DDSI the gate is intentionally opened long after each detected pulse.

To perform shift register logic-correlated neutron detection, it is necessary to have a relatively efficient detector – on the order of several percent at the very least. For such a system, the count rate in the context of spent fuel is so large that the accidental count rate for triples becomes very significant relative to true triples events such that the uncertainty on the triples count rate is excessive (Croft et al. 2011). For this reason, only coincident counting, determination for the doubles count rate, is useful with spent fuel.

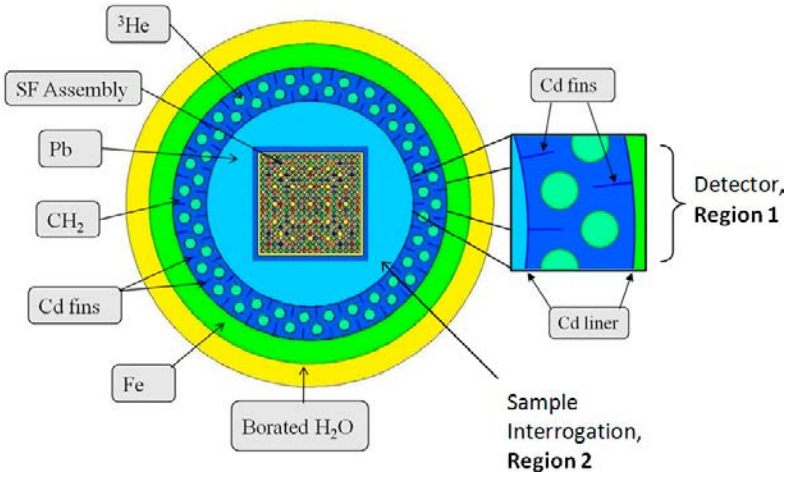

Figure 5-12. *Conceptual design of multiplicity instrument as simulated in MCNP for the NGSI Spent Fuel Project (Schear et al. 2011).*

Spatial distribution of the signal: Of all the techniques, DDSI can be the most spatially uniform in response. The interrogating source is spread throughout the assembly. From a diversion detection perspective, DDSI is more sensitive to a change in multiplication from the center of the assembly. From a "first-generation" perspective, the response is very uniform, with a slight emphasis for the outermost row.

Expected measurement time: The measurement time for the doubles dictates the overall measurement time. For a 45-GWd/tU, 4 wt %, ^{235}U, 5-years-cooled assembly, a statistical uncertainty of 1% for the doubles can be obtained in 2 minutes for a 20-µs pre-delay and a 32-µs gate width. For this same assembly with a 60-µs pre-delay and a 32-µs gate width, it takes 16 minutes to obtain the same uncertainty. By changing the pre-delay from 20 to 60 µs, the sensitivity to a unit fissile mass doubled, but the count time increased by a factor of 8 (Schear et al. 2011). Thus, the count time can be varied significantly, depending on the sensitivity required.

Discussion of calibration, precision, and accuracy: The precision of a CN instrument is determined by the efficiency of the instrument, multiplication of the fuel, and intensity of the passive neutron source. In the context of accuracy in meeting Clink's safeguards needs, the "correctness" of the declaration (IE, BU and CT) and the "completeness" (detection of a partial defect) are the targets. It is expected that correctness will be determined by integrating CN with PG and TN. More research is needed to assess the capability of such an integrated system. Part of such an assessment will include quantifying systematic uncertainties such as assembly positioning and isotopic spatial variation within the assembly. Part of future assessments need to investigate the utility of using fresh assemblies, or spent assemblies that may be considered as "working standard," as well as the role of simulation in the calibration process.

Unique features relative to other NDA techniques: The uniformity of response is a unique feature CN shares with DDSI. Noteworthy also is the large sensitivity for a passive instrument.

Maturity of the hardware: The hardware is very mature for assemblies up to ~30 GWd/tU for a 17 × 17 assembly; below that, BU ^{3}He tubes and the current list mode can handle the count rate. Above the ~30-GWd/tU BU level, a new data acquisition capability is needed. Conceptual designs are promising for improving the state of the art. The availability of ^{3}He could be an issue; yet given the importance of safeguarding a repository, access to ^{3}He is expected.

Maturity of the analysis: Moderate maturity. Research is ongoing to determine if the relative amount of ^{239}Pu and ^{235}U can be determined from time-correlated neutrons. Also, the count rate issue could be minimized by using a lower efficiency system, but it is not known how low is acceptable.

Considerations for implementation: It is uncertain if current technology can handle the high count rate of all the Swedish fuel. Using a less-efficient system, reducing the tube diameter, and optimizing electronics all lead to the ability to function at higher count rates. A project began in 2012 to test out reduced diameter tubes and optimized electronics.

5.1.11 Neutron Resonance Transmission Analysis (NRTA)

What is measured: The intensity of neutrons that have traversed the assembly as a function of energy. Because the intensity of neutrons incident on the assembly is known, the measured quantify is the percentage reduction in the neuron intensity as a function of energy.

What is quantified: The mass of four plutonium isotopes (238, 239, 240, and 242), four uranium isotopes (234, 235, 236, and 238), ^{241}Am, and several fission fragments (Sterbentz and Chichester 2010).

Description of the basic physics: The NRTA assay starts with a burst from a pulsed high-energy particle accelerator, as illustrated in Figure 5-13. This burst of charged particles initiates a several-step process that results in the creation of neutrons with a range of energies; of particular interest to NRTA are the neutrons in the a 0.1- to 40-eV energy range. The neutron burst is short enough in time and the neutron source is separated from the assembly far enough in space that a nearly uniform neutron energy arrives at the assembly at a given moment in time. These mono-energetic neutrons can scatter out of the beam as they interact with individual fuels pins in the assembly through low-energy elastic scattering, neutron-capture absorption, and neutron capture fission. The interaction of these quasi mono-energetic neutrons with the assembly can be measured by placing the neutron detector on the far side of the assembly from the neutron source. This setup provides the intensity of the transmitted beam as a function of neutron energy, which can be used to quantify how much of each isotope is in the assembly, provided the features of the spectra are detectable and do not interfere significantly with each other. Experimental results performed with spent fuel pins (Behrens et al. 1984) indicate that interferences are not significant.

Spatial distribution of the signal: The approach advocated by the NGSI Spent Fuel researchers (Sterbentz and Chichester 2010) is to make a narrow beam of neutrons such that the interrogation can be used for a single row of pins at a time. Preliminary results indicate that the limit for assaying multiple pins is expected to be approximately 8; thus, it is not expected that NRTA can assay the center of large assemblies. For the pins that are assayed, the signal represents the average properties of those pins.

Expected measurement time: The measurement time is expected to be ~40 minutes for an axial slice of one assembly when a ~1 × 10^{13}-n/s accelerator source is used to obtain a statistical uncertainty of ~5%. Note that several assemblies could be measured in parallel to use the neutron source more efficiently (Chichester D 2012, personal communication).

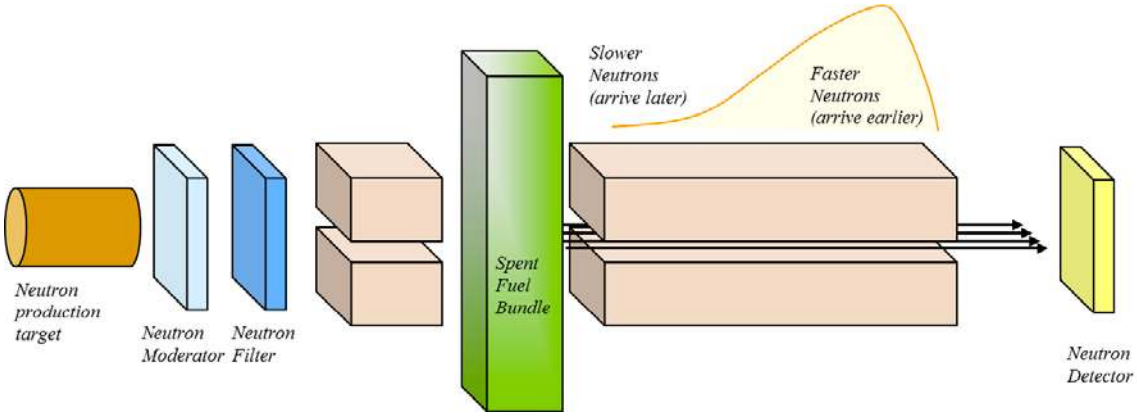

Figure 5-13. *Schematic of a conceptual NRTA system.*

Discussion of calibration, precision and accuracy: Because an active source is involved, conceptually better precision can be obtained with a stronger source. Because the source is expected to be large, it is likely that obtaining good precision will be a challenge. In the context of accuracy in meeting Clink's safeguards needs, the "correctness" of the declaration (IE, BU and CT) and the "completeness" (detection of a partial defect) are the targets. It is expected that correctness will be determined by integrating NRTA with PG and TN. More research is needed to assess the capability of such an integrated system. Part of such an assessment will include quantifying systematic uncertainties such as assembly positioning and isotopic spatial variation within the assembly. Part of future assessments need to investigate the utility of using fresh assemblies, or spent assemblies that may be considered as "working standard," as well as the role of simulation in the calibration process.

Unique features relative to other NDA techniques: The isotopic variety and quality of the data obtained from NRTA are excellent.

Maturity of the hardware: The ~1×10^{13}-n/s accelerator-driven source is commercially available. It is expected that the feasibility of the 1) size of the accelerator and 2) time needed to obtain adequate statistics will be the key issues for NRTA.

Maturity of the analysis: Several decades of experience with NRTA exist in the context of small, relatively pure samples. The maturity is considered medium in the context of spent fuel.

Considerations for implementation: Measurements must be made in air. A key question involved determining how complex is the system. The size and cost of an NRTA instrument are a major concern.

5.1.12 Nuclear Resonance Fluorescence (NRF)

What is measured: Given the thickness of a spent fuel assembly, the application of NRF to spent fuel studied by the NGSI Spent Fuel project focused on the NRF transmission measurement approach as opposed to backscattered NRF (Ludewigt et al. 2011). With transmission NRF, the absence of milli-electron-volt-level photons at a very specific energy are measured; note these resonance are significantly narrower than the energy resolution of typical detectors. The absence of photons at the resonance energy is indicative of the presence of the specific isotope.

What is quantified: The mass of any isotopes with a significant NRF cross section and sufficient mass to be detected; isotopes researched in the NGSI Spent Fuel Effort included ^{239}Pu, ^{240}Pu, and ^{235}U.

Description of the basic physics: As illustrated in Figure 5-14, NRF is a two-stage process that involves the excitation of a nucleus by the absorption of a photon, which is then followed by the de-excitation of the nucleus to the ground state by the emission of one or more photons. In the transmission approach to NRF researched for spent fuel assembly assay, a relatively flat photon spectrum is incident on the assembly from a bremsstrahlung source. If a particular isotope of interest is present in the fuel, it will absorb photons at the resonant energy from the incident beam, then will re-radiate photons into all space. As a result, the photon intensity in the incident (nearly flat spectrum) beam will be depressed

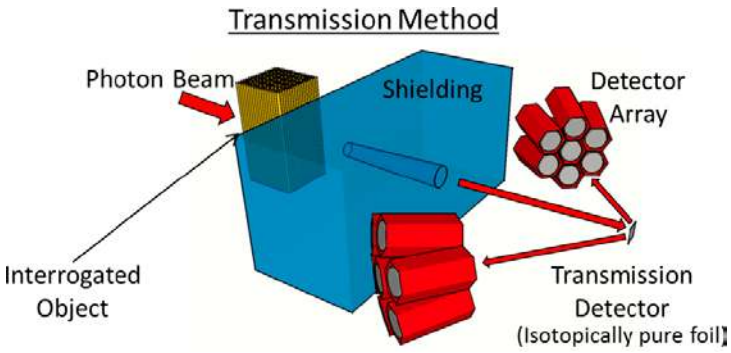

Figure 5-14. *Conceptual design of an NRF measurement of spent fuel using the transmission method (Ludewigt et al. 2011).*

at the resonant energy of the isotope of interest. Thus, as the incident continuum traverses the assembly, the presence of a specific isotope is indicated by a depression in the continuum – the greater the amount of an isotope present, the greater the depression in the spectrum.

Spatial distribution of the signal: Conceptually, the NRF signal uniformly weights each pin.

Expected measurement time: Because it is an active interrogation technique, the measurement time depends on the intensity of the interrogating source. Practically speaking, a very strong source is needed to obtain reasonable statistics in <1 hour.

Discussion of calibration, precision and accuracy: The end conclusion of the NGSI Spent Fuel Project research is that NRF is not viable for spent fuel assemblies with a bremsstrahlung source. As the ^{239}Pu mass changed from 0 to its maximum, the NRF signal changed by only ~3%.

Unique features relative to other NDA techniques: Isotope-specific data with uniform sensitivity for each pin. Unfortunately, with current technology, the "uniform sensitivity" essentially had very little sensitivity.

Maturity of the hardware: A technical breakthrough in the interrogating source is needed to enable this technology. Specifically, a mono-energetic source that can be tuned to the energy of the NRF excitation energy is needed.

Maturity of the analysis: Not very mature; fundamental cross sections such as ^{240}Pu measured only in the past few years.

Considerations for implementation: Application in air is significantly more practical than in water.

5.1.13 Passive Gamma (PG)

What is measured: The axial profile of the intensity of gamma radiation of one or more energies (see Figure 5-15).

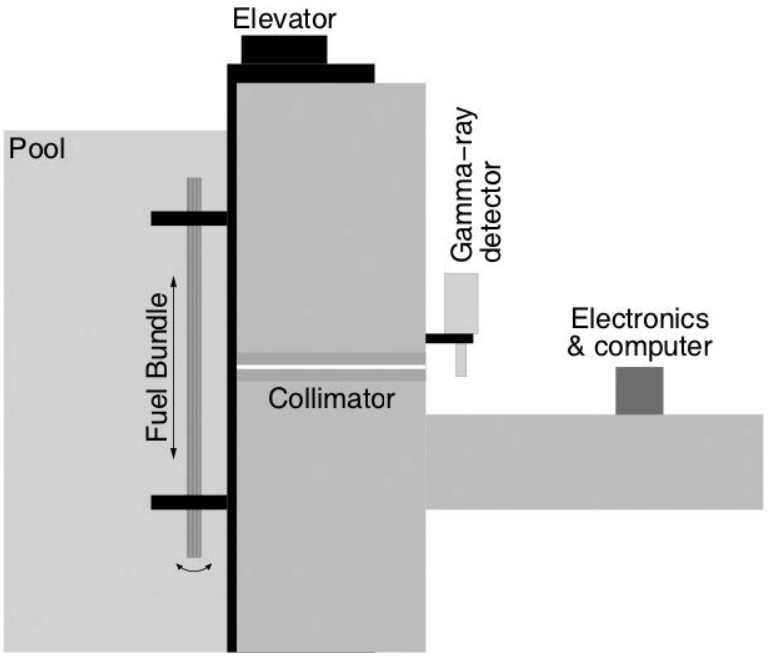

Figure 5-15. *A schematic image of a PG scanning system in use at the Clab Facility, Oskarshamn, Sweden.*

What is quantified: Isotope specific gamma radiation from ^{134}Cs, ^{137}Cs and ^{154}Eu can be used to determine the fuel parameters BU, IE, and CT for spent fuel with a CT less than ~20 years. Beyond that time, ^{134}Cs and ^{154}Eu have decayed significantly, and IE and CT have to be determined by other means. The longer half-life (~30 years) of ^{137}Cs implies that it can be used to determine BU for a longer time (see Jansson 2002 and Min et al. 1988). Decay heat can also be inferred from gamma scanning data using a calorimetric calibration (see Section 5.1.2). IAEA (1991) complements the applicability of PG scanning with the ability to indicate the following:

- Determine the concentrations of fission products and their distribution within the assembly and thus the comparison between calculated and experimentally determined power distribution parameters.
- Use the fission product distribution for accurately locating the fuel stack within the fuel rods and for determining dimensional changes in the fuel, [e.g., axial fuel swelling and gaps in the fuel pellet stack within the fuel rods (relevant for gamma scanning of single fuel rods)].

Expected measurement time: A complete gamma scan of a fuel assembly takes on the order of 15 minutes, which includes spectra-resolved information on the gamma intensity reaching the detector.

Discussion of calibration precision and accuracy: Using gamma scanning, the above-mentioned fuel parameters have been determined with the following uncertainties. Note that the uncertainties on BU are related to values calculated by the operator, which is performed with the same uncertainty or worse. Note also that the uncertainty on IE is reported in only one reference, which contains data on a limited set of PWR assemblies. It should also be noted that the uncertainties will probably be higher for old fuel. The calibration between e.g. ^{137}Cs and BU need to be established with small uncertainty for all fuel assembly types to fully utilize this technique in Clink.

Unique features relative to other NDA techniques: Axial scanning equipment used for this technique has already been developed and is in use. The technique is passive (as compared with some of the NGSI techniques).

Maturity of the hardware: The hardware, based on a system with HPGe detectors, has been used for more than 10 years and can as such be considered mature. Research needs to be performed regarding what uncertainties can be obtained with types of detectors other than the one HPGe used until now at the Clab Facility. CdTe or $LaBr_3$ are additional options to consider.

Maturity of the analysis: Analysis of gamma-ray energy spectra (peak positions and areas) is routinely performed in very many applications and can as such be considered mature. However, the analysis procedure needs to be properly documented for industrial purposes. For example, the fact that different fuel geometries require different calibration curves needs to be built into the analysis and documentation. Furthermore, the data generated until now are scarce with regard to some fuel types.

The analysis used in Min et al. (1988) to determine IE, BU and CT has been performed only on one set of PWR measurements and is therefore considered not to be mature. The results should be verified experimentally on a large data set involving more types of fuel assemblies.

Considerations for implementation: The equipment has dimensions on the order of 1 m^3. The uncertainties in the referenced work have been achieved using the equipment mentioned therein. If other types of detectors are to be used, which would influence the dimensions of the equipment, studies of achievable uncertainties should be performed. In designing the collimation for a PG instrument, the sensitivity to alignment must be considered. Such sensitivity can be minimized by a wide collimator that views a thin slice of the entire assembly if an, e.g. axial intensity profile is of interest. The sensitivity to alignment can be made smaller by measuring the average intensity from a scan of the assembly.

Parameter	Uncertainty	References
Burn-up	1.2% to 4%	Willman et al. (2006), Osifo et al. (2008), Guardini (2004)
Cooling Time	1.5% to 4%	Willman et al. (2006), Osifo et al. (2008), Guardini (2004)
Decay heat	2% to 3%	Simpson et al. (2008), Jansson (2002), Osifo et al. (2008)
Initial enrichment	<1% (~25% of ^{235}U content)	Min et al. (1988)

From a Clink planning perspective it is important to keep in mind the need for calibration of the passive gamma and gamma tomography methods to be able to determine decay heat. The calibration procedure includes both calorimetric measurements, of which some have been performed already, and measurements with passive gamma or gamma tomography in the measurement geometry to be used at the facility. The calibration needs to be performed on all types of fuel assemblies for which decay heat is to be determined.

Development: It should be noted that a tomographic measurement device would give the same (and more) information about the fuel assembly as PG scanning (see Section 5.1.8).

Because a relatively low-energy 662-keV gamma from ^{137}Cs is measured mostly from the outer pins of the fuel assembly, work has been performed regarding optimizing the "pin row depth" by adjusting the angle by which axial scanning is performed (Sihm Kvenangen 2007). This work could be further developed and considered if this technique is implemented.

Studies of very week signals as would be expected from old fuel, such as ^{134}Cs or ^{154}Eu, should be performed in order to quantify the uncertainties in the determined BU, CT or IE.

5.1.14 Passive Neutron Albedo Reactivity with Fission Chambers (PNAR-FC)

What is measured: TN count rate for two different physical setups, one setup designed to maximize multiplication and the other designed to minimize multiplication.

What is quantified: Multiplication or fissile content (the weighted sum of ^{235}U, ^{239}Pu, and ^{241}Pu); for fissile content to be determined, a neutron absorber correction is necessary. Because prompt fission based multiplication is measured, PNAR-FC emphasizes the presence of ^{239}Pu and ^{241}Pu per unit mass. For thermally induced prompt fission per unit mass, ^{239}Pu and ^{241}Pu produce ~1.5 and 2.0 times as many neutrons as ^{235}U, respectively.

Description of the basic physics: PNAR-FC, the conceptual hardware illustrated in Figure 5-16, uses the intrinsic neutron emission of the fuel to self-interrogate the fissile material in the fuel itself. Two separate measurements of the spent fuel are made. The primary difference between the two measurements is the neutron energy spectrum and fluence in the spent fuel – this difference was primarily achieved by surrounding the fuel with cadmium for one of the two measurements (Menlove and Beddingfield 1997, Tobin et al. 2006, Conlin et al. 2010). By varying the material around the spent fuel, a high and a low neutron-energy-measurement condition can be produced (low and high multiplying setups, respectively). The ratios of the count rates obtained for these two situations scale with the multiplication and fissile content in the spent fuel case. The primary difference between the two PNAR-FC measurements from an energy spectrum perspective is the presence of reflected neutrons with an energy below the cadmium cutoff energy (~1 eV); the PNAR-FC instrument can be considered to be an interrogation technique for which the interrogating source is essentially thermal neutrons incident from all sides of the assembly.

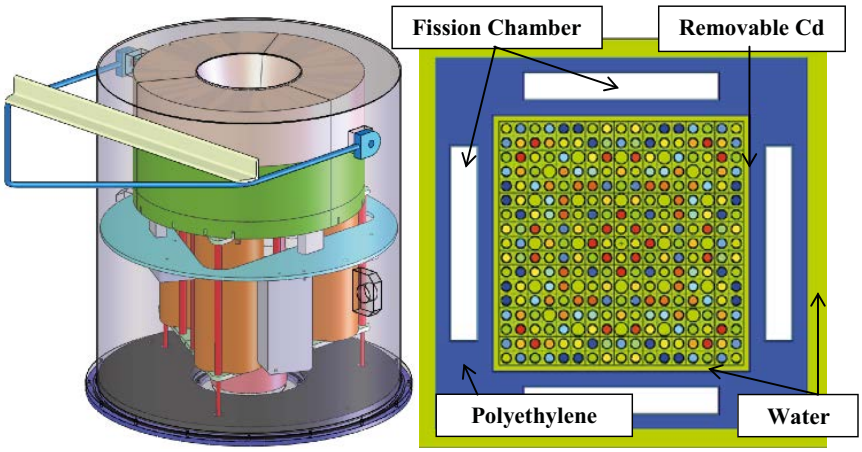

Figure 5-16. *Left: Mechanical design of a PNAR instrument for measuring circular fuel (Polk P, personal communication). Right: Conceptual design of a DG instrument as simulated in MCNP for the NGSI Spent Fuel Project (Conlin et al. 2010).*

Spatial distribution of the signal: The PNAR-FC technique is sensitive to the removal of pins anywhere in the assembly, with the sensitivity being largest in the center and the least near the edge (Conlin et al. 2010). The induced fission density of the interrogating neutrons (neutrons that contribute to the "first generation") (Blanc et al. 2012) decreases with a gradient sloping away from the exterior of the fuel such that ~80% of the first generation signal comes from the exterior three rows of the fuel.

Expected measurement time: The count rate for a fully burned 45-GWd/tU, 4 wt %, ^{235}U, 5-years-cooled assembly is ~1×10^5 counts/s; for a similar assembly after one cycle, the count rate is roughly 100 times lower (Conlin et al. 2010). Thus, counting statistics are excellent in ~100 s or less. Approximately 0.5 m of the assembly (all sides) is measured at the same time.

Discussion of calibration, precision and accuracy: Although PNAR-FC measurements have excellent precision, the instrument does not have as much of a dynamic range relative to the other techniques that measure multiplication. Of particular interest is the magnitude of the systematic uncertainties relative to the dynamic range in the PNAR-FC signal. In the context of accuracy in meeting Clink's safeguards needs, the "correctness" of the declaration (IE, BU and CT) and the "completeness" (detection of a partial defect) are the targets. It is expected that correctness will be determined by integrating PNAR-FC with PG and TN. More research is needed to assess the capability of such an integrated system. Part of such an assessment will include quantifying systematic uncertainties such as assembly positioning and isotopic spatial variation within the assembly. Part of future assessments need to investigate the utility of using fresh assemblies, or spent assemblies that may be considered as "working standard," as well as the role of simulation in the calibration process.

Unique features relative to other NDA techniques: Combining simplicity and the measurement of multiplication.

Maturity of the hardware: The hardware is very mature. The primary components are fission chambers and sheets of cadmium.

Maturity of the analysis: PNAR has been applied in several different safeguards relevant contexts. However, the technique has never been used with spent fuel – medium maturity.

Considerations for implementation: The PNAR-FC could be designed with a movable cadmium liner or, more likely, two sections – one section with a cadmium liner and one section without such a liner.

5.1.15 Self-Integration Neutron Resonance Densitometry (SINRD)

What is measured: The neutron intensity in four different parts of the TN spectrum. If the material in the fission chamber matches that of the material being quantified, then the sensitivity to the presence of the material of interest is enhanced because of the resonance energy structure (Menlove et al. 1969, LaFleur et al. 2011, Freeman et al. 2012, LaFleur 2011). For example, a ^{239}Pu fission chamber is more sensitive to the presence or absence of ^{239}Pu than a ^{235}U fission chamber although both will work for SINRDE given the use of absorptive filters. The utility of matching the isotope of interest to the material in the fission chamber is due to the fact that a ^{239}Pu fission chamber is particularly sensitive to the presence or absence of neutrons at 0.3 eV because this is a resonance of ^{239}Pu; if there is a significant amount of ^{239}Pu in the fuel, then there will be relatively few neutrons leaving the fuel with an energy of 0.3 eV.

What is quantified: The mass of ^{239}Pu for medium and full BU fuel; for low BU fuel, a correction is needed, which is the topic of current research.

Description of the basic physics: In the right-hand side of Figure 5-17, the locations of the various fission chambers in the SIRND unit are depicted. In the left-hand side of Figure 5-17, a SINRD unit built for deployment is illustrated. In Figure 5-18, the neutron energy spectrum for five 4% IE assemblies is illustrated as a function of energy, one fresh assembly, and four spent assemblies, each with a different BU.

The fundamental physics of SINRD is captured in Figure 5-18, which depicts the flux averaged over all the pins in the assembly such that the area under the curve is proportional to the flux. The largest "peak" at ~2 MeV is the "fast" birth energy of most neutrons following fission. These fast neutrons moderate by colliding in the water and fuel. The second major peak is at thermal energy and is formed by the neutrons that manage to "survive" all the collisions they underwent in the thermalization process and still reside in the fuel. The structure in the spectrum is the result of particularly prominent

absorption processed; of particular note are a few of the resonance absorption due to ^{238}U, ^{240}Pu and ^{239}Pu, which are illustrated in Figure 5-18. The SINRD detector comprises four fission chambers. By surrounding the fission chambers by absorbing material (cadmium, gadolinium, hafnium, and boron) of specific thicknesses, each fission chamber detects a different part of this spectrum. By calculating the difference and ratio among the count rates in these fission chambers, the SINRD signal is determined. This signal is proportional to the ^{239}Pu and ^{235}U content in the fuel. The sensitivity of a given fission chamber to the resonant structure of a given isotope is enhanced by using the same material in the fission chamber to match the resonance of interest.

Spatial distribution of the signal: The SINRD signal that is sensitive to the presence of ^{239}Pu, and ^{235}U in the fuel is sensitive only to the mass in the outer approximately three rows.

Expected measurement time: The count time for SINRD is largely determined by the ambient neutron emission of the fuel, which scales roughly as the third or fourth power of the BU. For fully burned assemblies, this emission can result in count times of between 5 and 20 minutes for a ~20-cm axial length along one side of the assembly. For one cycle of fuel, it may take 2 hours. Note: An active neutron source can be used to reduce the count time.

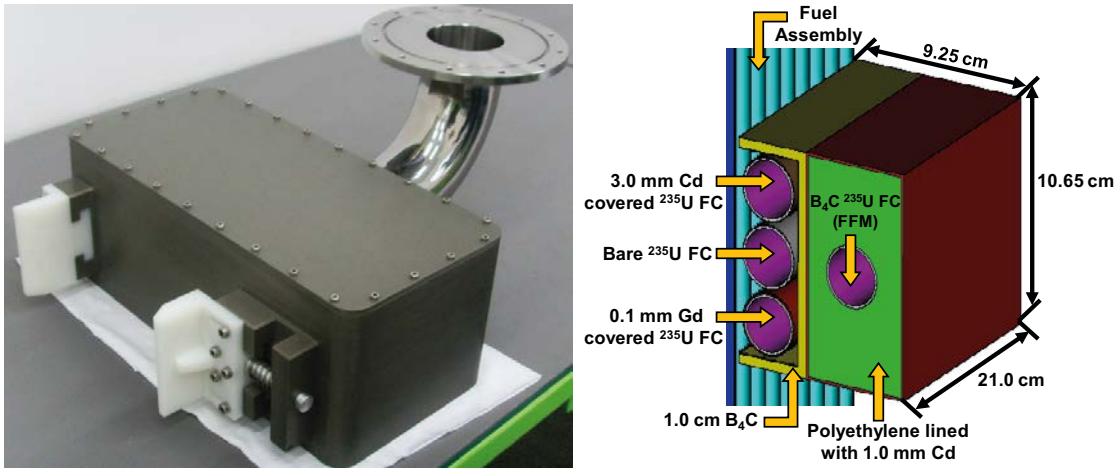

Figure 5-17. *Left: Fabricated SINRD prototype for spent fuel measurement (Park 2012). Right: Conceptual design of SINRD instrument as simulated in MCNP for the NGSI Spent Fuel Project (LaFleur 2011).*

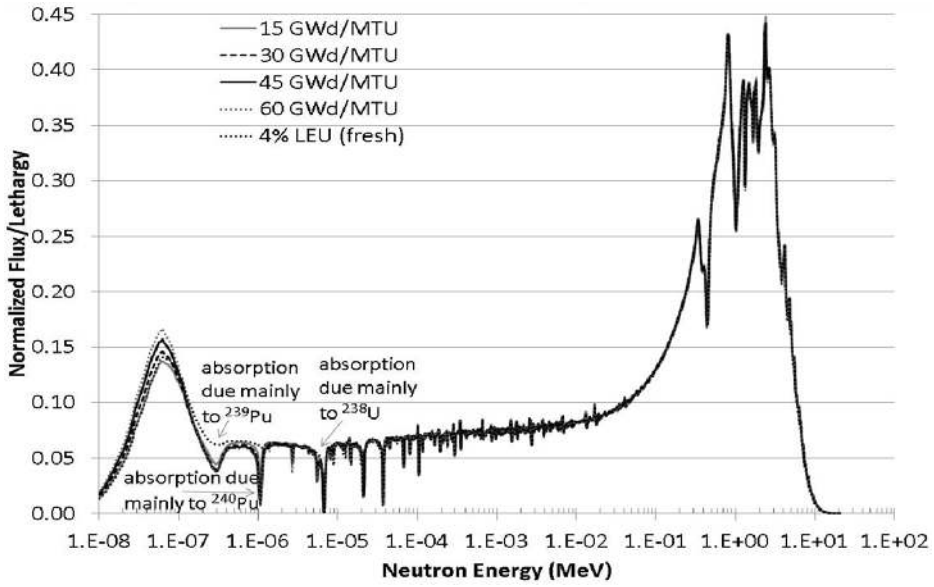

Figure 5-18. *Normalized neutron energy spectrum in the fuel rods for five different fuel assemblies (Freeman et al. 2012).*

Discussion of calibration, precision and accuracy: Calibration of SINRD is a topic of ongoing research. It is anticipated that the unit will be calibrated with a fresh assembly and that corrections will be determined through simulation and will be possible through measurement with working standards to account for the impact of significant absorber. Until calibration is advanced further, it is premature to discuss accuracy. Precision in the context of SINRD is unique relative to other techniques; it depends significantly on the BU of the fuel; less than one percent precision on SINRD ratios is expected for most fully burned assemblies in less than 5 minutes.

Unique features relative to other NDA techniques: SINRD is unique from most other techniques in that it measures the energy-specific perturbation due to the presence of specific isotopes. This technique is similar to LSDS and NRTA; however, SINRD is roughly a million times lighter.

Maturity of the hardware: The hardware is very mature. Fission chambers and sheets of cadmium, gadolinium, and hafnium are used.

Maturity of the analysis: The maturity of the analysis is low. One experiment in air on a fresh low-enriched uranium (LEU) assembly was performed. In 2013 both fresh and spent assemblies will be measured.

Considerations for implementation: The neutron energy structure, which is the foundation of the SINRD measurement, changes significantly on a millimeter level as the detector is moved away from the assembly (Freeman et al. 2012). Therefore, it is necessary to know very well the location of the SINRD unit relative to the fuel. In a practical sense, this effectively means that the SINRD unit needs to be flush up against the assembly. The impact of bent rods on positioning and accuracy needs to be determined. Of note in this implementation context is that the SIRND unit is very small and maneuverable; it is a box ~15 cm × ~15 cm × the width of the assembly. Given the count time per measurement and the small region measured (both in the axial and radial directions), it is unlikely that SINRD will be used to measure the entire assembly.

5.1.16 Total Neutron (TN)

What is measured: TN emission, also known as singles counting.

What is quantified: Provides information about coupled parameters of IE, BU, and CT; the three parameters can be determined in combination with PG. The signal is proportional to the product of the multiplication and the passive neutron source, which is dominated by ^{244}Cm for most spent fuel assemblies. The TN rate can also be a rough indicator that the assembly is whole.

Description of the basic physics: Radioactive material in spent fuel emits neutrons. The dominant spontaneous fission isotopes are generally ^{244}Cm, ^{242}Cm (for short CT), and ^{240}Pu. The (α,n) sources also contribute, particularly for low BU. In Henzl (2012), the neutron source term for the IE, BU, and CT combinations of the NGSI Spent Fuel Library is quantified. In Figure 5-19, three fission chambers are depicted inside a SINRD unit; ^{3}He tubes are also used and look essentially the same (LaFleur 2011).

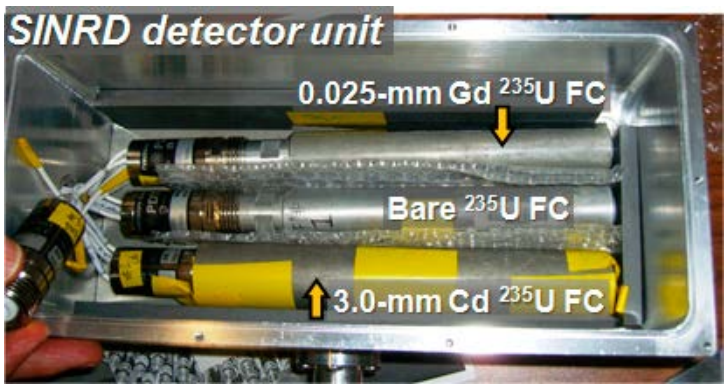

Figure 5-19. *Three fission chambers as depicted from the inside of a SINRD detector (LaFleur 2011).*

Spatial distribution of the signal: Generally uniform contribution among the pins because of multiplication.

Expected measurement time: Less than 1% counting statistics uncertainty is obtained in less than 10 s.

Discussion of calibration, precision, and accuracy: Precision is excellent. For many applications, a relative measurement among assemblies is used; therefore, a database of measurements is needed. Cross checking a few assemblies with a Fork detector would enable access to the wealth of Euratom data on many assembly types. It is anticipated that TN will be used with almost all systems to determined completeness and correctness.

Unique features relative to other NDA techniques: The unique property of TN is that it has been used so much. A database of past measurements can be used in interpretation.

Maturity of the hardware: Very mature; a fission chamber will suffice.

Maturity of the analysis: For comparing the BU among assemblies from the same reactor type, the analysis is very mature; the general scaling is known. A rich database already exists among Euratom inspectors because of Fork measurements. The determination of IE, BU, and CT is less mature, but some work has been done (Simpson et al. 2008).

Considerations for implementation: A TN detector is a mature, inexpensive, robust instrument; it will almost surely be a part of any NDA system selected for the Swedish repository.

5.1.17 X-Ray Fluorescence (XRF)

What is measured: Uranium and plutonium x-rays from a volume that is a few millimeters square in surface area and ~1 millimeter deep into the fuel from an individual exterior rod. It is likely that multiple detectors would be used to measure the same location from each rod and that the same size volume on multiple exterior rods of an assembly would also be measured.

What is quantified: The elemental plutonium mass of the assembly. XRF is of no use for fissile mass or diversion detection.

Description of the basic physics: Plutonium to uranium X-rays are stimulated equally by the radiation emanating from the spent fuel: both photon and charged-particle excitation. A conceptual design of an experimental setup for XRF detection from spent fuel is depicted in Figure 5-20. The elemental ratio of plutonium to uranium in the edge layer of the spent fuel can be determined by measuring these x-rays and taking the ratios of their intensity. A correction needs to be made to account for the radial profile of particularly plutonium which ramps up by a factor of 2 to 3 in the outermost ~0.2 mm to enable the average elemental plutonium-to-uranium ratio to be determined for that part of the rod (Rudy 2005, Freeman 2011). The absolute plutonium can be determined by multiplying the average elemental plutonium-to-uranium ratio by the total amount of uranium in the rod. The mass of elemental uranium can be well estimated in a spent fuel rod. When an assembly is fresh, ~88% of the mass is elemental uranium; at the end of life, elemental uranium is ~82% of the total mass in the rod. The change between these two extremes can be accurately determined from gamma or neutron measurements such that the uncertainty in the amount of uranium can be estimated to less than 1%. The final step involves extrapolating from the measured plutonium mass in the edge rods all around the exterior of the assembly to the entire assembly. This step is done through simulation, and preliminary results presented in Galloway et al. (2011) indicate that the uncertainty in this process is likely a few percent. The general conclusion from the preliminary research is that if the boundary plutonium mass is known, the center plutonium mass can be predicted accurately. In all this discussion, it must be emphasized that because the mean free path of the ~100-keV x-ray photons is ~0.5 mm in fuel, the extrapolation assumes that no diversion of rods from the assembly exists. XRF is completely blind to the diversion of pins from anywhere but the exterior rods of an assembly.

Spatial distribution of the signal: Given the low penetrability of plutonium and uranium x-rays in spent fuel, penetrability is only a few millimeters. The ability to extend an edge measurement to the entire assembly is simulation based. Another NDA technique is necessary to ensure that diversion has not taken place.

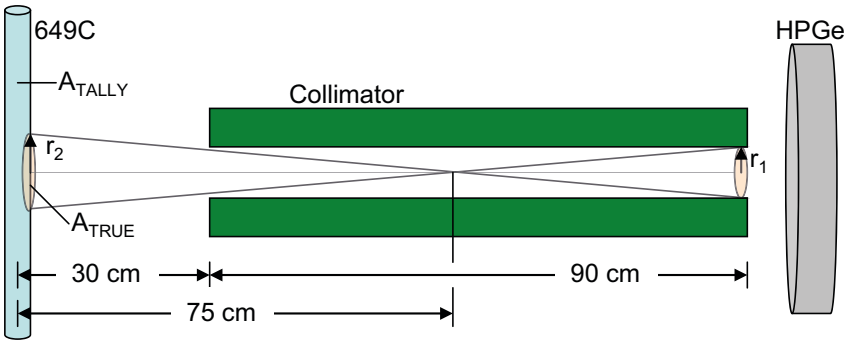

Figure 5-20. *Conceptual design of an XRF setup (Freeman 2011).*

Expected measurement time: In Freeman (2011) a counting duration of approximately 10 hours was determined to obtain a ~3% uncertainty in plutonium x-ray intensity for an assembly when a single planar detector was used. According to Charlton W (2011, personal communication) a ~2% uncertainty was obtained when measuring a single pin for ~2 hours. It is not clear that the difference between these two can be explained by the difference between assembly and pin measurements. It will be necessary to use either multiple crystals or use some additional techniques to improve the signal-to-background ratio.

Discussion of calibration, precision and accuracy: Unlike most other techniques, precision is a major concern for XRF. The plutonium x-ray peak is very small compared with the continuum. A few approaches for improving the signal-to-background are listed later in this section. The plutonium/uranium ratio does not require a calibration. Accuracy analysis will likely focus on the accuracy of the simulations used to extrapolate from the measured uranium/plutonium x-ray intensities on the edge to total assembly mass. The Pu mass could conceivable be used as part of determining the completeness and the correctness of a declaration. For example inconsistency between the edge Pu mass estimated with XRF and the TN or PG signals could indicate that the declaration was not correct or that some fuel rods had been removed from the assembly.

Unique features relative to other NDA techniques: The physics of XRF is unique from all the other techniques in that its signal is a measure of the amount of *elemental* plutonium. It is not influenced in any way by neutron absorbers. Furthermore, there is essentially no need to calibrate. This simplicity comes at a price of being blind to diversion and relying on simulation to extrapolate from an edge measurement to the entire assembly.

Maturity of the hardware: With sufficient collimation, current planar detectors can measure the XRF signal from spent fuel. Bent crystal techniques, Compton suppression, or active excitation of the plutonium x-rays will improve the signal-to-background ratio, as will more experimental technologies such as micro-calorimetry.

Maturity of the analysis: Although we are not aware of much research involved in extrapolating from the measured elemental plutonium on the edge to total plutonium in an assembly, it is expected to be relatively straightforward because it is primarily concerned with simulating the plutonium distribution in an assembly, which is a mature topic of research.

Considerations for implementation: XRF is of interest primarily to those quantifying plutonium mass. A single crystal will likely have a long measurement time of ~1 hour per section – although this count time question can be reframed as engineering questions: "How many detectors can be packed around the assembly? How much can the signal be improved relative to the continuum? All sides will need to be measured, and likely many crystals will be needed for each ~1 mm^2 spot. Optimization of the hardware in the context of "Pu mass determination by XRF for spent fuel assemblies" is important as is the analysis need to extrapolate from edge measurements to the entire assembly.

5.2 Summary of NDA techniques

The following characteristics of each of the NDA techniques described in this report are summarized in Table 5-2: 1) the impact of changing the fuel type from PWR to BWR assemblies, 2) the maturity of the hardware and the impact of this hardware in the facility, 3) the degree to which the signal is proportional to mass located at various depths inside the assembly, and 4) the count time per unit length.

Table 5-2. Summary of the 1) Relative Impact of Changing the Fuel Type from PWR to BWR Assemblies, 2) Maturity of the Hardware and the Impact of This Hardware in the Facility, 3) Degree to Which the Signal Is Proportional to Mass Located at Various Depths inside the Assembly, and 4) Count Time per Unit Length.

Techniques	Uncertainty introduced in due to transitioning from PWR to BWR (no, some, med., med. -high)	Maturity of Hardware (H) Analysis (A) and Impact (I) on Facility	Penetration of Signal inside Assembly	Measurement Time per Axial Unit Length and General Comments
CIPN	Some	High H, Med-Low A, Low I	Good	<100 s per 20 cm
CDH	No impact	High H, High A, Med. I	Excellent	4–5 hours per assembly
DG	Medium	Med. H, Med.-Low A, Med. I	Signal weighted to exterior	~30 minutes per ~50 cm, system outside pool likely ~2-m^2 footprint
DN	Some	Med. H, Med. A, Med. I	Good	~100 s per ~50 cm, system outside pool likely ~1-m^2 footprint
DDA	Some	High H, Med. A, Med. I	Good	~100 s per ~50 cm
DDSI	Some	Med. H, Med.-Low A, Med. I	Very Good	~15 minutes per ~50 cm
DCVD	No significant impact	High H, Med. A, Low I	Poor	<100 s for entire assembly
Passive GT	BWR easier than PWR	Med.-Low H, Med. A, High I	Excellent	About 10 minutes per axial position
LSDS	Some	Med.-Low H, Med.-Low A, Med.-High I	Good	Must measure in air, moderately large footprint
CN	Some	Med. H, Med.-Low A, Med. I	Good	~5 minutes per ~50 cm
NRTA	BWR easier than PWR	Med.-Low H, Med.-Low A, High I	Signal weighted to exterior	Must measure in air, large footprint[a]
NRF	BWR easier than PWR	Low H, Med.-Low A, High I	Conceptually Excellent	Not considered a viable option with currently available technology[b]
PG (total and spectral)	Medium-High, the metal box of a BWR may reduce signal somewhat	High H, High A, Low I	Signal from outer 2 or 3 rows	Total gamma, ~10 s per ~20 cm; spectral resolved gamma, ~10 s for ~1 cm axial length[c]
PNAR-FC	Some	High H, Med. A, Low I	Good	~100 s per ~50 cm
SINRD	Medium-High	High H, Med.-Low A, Low I	Signal from outer 2 or 3 rows	<15 minutes for 20 cm for most spent fuel[d]
TN	Some	High H, High A, Low I	Good	~10 s per ~20 cm
XRF	Medium-High for PWR, the metal box of a BWR may significantly reduce signal	Medium H, Med.-Low A, Medium I	Signal from outer few mm of exterior pins	Moderately large footprint, count time largely dependent on number of detectors.

[a]Could measure multiple assemblies in parallel, several hours per meter for one assembly.

[b]Sensitivity is very low with a Bremsstrahlung source and thus not considered a viable option until mono-energetic photon sources of sufficient technology are available.

[c]Many variables can impact this parameter: number of detectors, attenuator thickness, and collimation. Note: If desirable, the detection of the 60-keV gamma from 241Am would need a separately designed collimator and no significant attenuation.

[d]Count time can be more than 1 hour for one cycle fuel.

The simulation results on which much of these reports are based were performed on PWR assemblies. Because Sweden has a mixture of PWR and BWR assemblies, an important question is how the various NDA techniques perform for BWR assemblies. In transitioning from PWR to BWR assemblies, the following changes are of note for most NDA techniques: 1) There is a greater axial variation in all isotopes along the assembly (fission products as well as fissile isotopes); 2) the IE and pin geometry can vary within one assembly (axial and radial variation); and 3) the cross-sectional area of the BWR (8×8, 9×9, and 10×10) assemblies is less than the 17×17 PWR assemblies; 4) a zircaloy sheet surrounds the bundle for PWRs; and 5) the absorber blades, which can be thought of as a zircaloy cross, will be inserted into some assemblies.

The first two points increase the uncertainty in making the connection between a measured signal and a particular quantity, such as plutonium mass, fissile content, or diversion detection. The increase in the isotopic spatial variation impacts both the BU calculations and the interpretation of measured data. The BU calculations are expected to be less accurate for BWRs; thus, any analysis that uses the BU calculation is less accurate. The interpretation of the measured values is more uncertain because it is more important to know accurately the origin of the signal. The signals that propagate through multiplication in the assembly are not expected to be very sensitive to the BWR-introduced spatial variation given that the DN instrument was rather insensitive to spatial variation in PWR assemblies that was created by burning fuel assemblies asymmetrically (Broadhead et al. 2012, Trellue H R 2012, personal communication). Effectively, multiplication averages over the isotopic variation within the detector.

The inclusion of additional zircaloy in BWRs is not expected to be of significant concern for neutron techniques in terms of perturbing the actual measurements; the neutrons will easily penetrate through the zicaloy just as they did in the reactor. The presence of neutron absorber in the zircaloy is not expected to be a significant problem provided the absorber concentration evolved in a consistent way with the fuel BU. For photon techniques the zircaloy will not impact high energy photons, above ~1 MeV, much and the attenuation that is experience can be corrected for. However, low energy photons, particularly in the X-rays and the 60-keV peak from ^{241}Am may experience very significant attenuation.

The presence of burnable poisons in the fuel is not, in and of themselves, a problem. All the neutron techniques "work" with the absorbers that burn into the fuel (^{240}Pu, ^{143}Nd, ^{155}Gd, ^{149}Sm, ^{241}Am, etc). Because the concentration of burnable poisons is of the order of magnitude as these "natural thermal absorbers," the neutron NDA techniques are expected to give strong signals. Yet, the presence of neutron absorbers has not been researched in detail. Two primary paths of analysis are envisioned to accommodate the presence of neutron absorbers: 1) If the burnable poisons are loaded into the fuel at its fabrication and a large group of assemblies all start with the same burnable poison loading and experience the same neutron fluxes in the reactor, then the isotopics content of all the assemblies in this large group is expected to evolve together as a function of IE, BU and CT in a predictable manner. Hence, a predictable signal response for these assemblies can be determined through coupled measurements and simulation. 2) The more general path is more challenging, and a path that has not received much research yet. This path involves quantifying the absorber content in the fuel, something like a "neutron absorber figure of merit" that could be used to correct for the absorber content in the fuel. Conceptually this should be possible.

The hardware of an instrument was considered to have a "high maturity" if all parts are currently commercially available. The impact of an instrument was considered "low" if it could be retrofitted into a facility with little or no effort. The penetration was considered "good" if the signal had roughly the same sensitive to pins removed from any region of the assembly.

5.3 Currently deployed integrated systems

In Section 5.1, numerous individual NDA techniques are described. The signature of each technique is unique for either the signature that it measures or the means by which a given property of the assembly is quantified. Given the 1) isotopic complexity (numerous isotopes varying in space and time) of the fuel and 2) challenge of using a given signature to answer a particular question (partial defect, multiplication, fissile content, plutonium mass, etc), it is expected, and to a limited extent demonstrated, that more accurate conclusions about the content of the fuel can be obtained by integrating a few NDA signatures together.

The remainder of this section describes two instruments, the Fork and SMOPY, that integrated the two most easily measured signatures emitted by the fuel: the total gamma and TN intensity. The PG signal quantifies the photon intensity emitted by a few fission products, whereas the TN signal primarily quantifies the neutrons that originated from curium. Both instruments vary somewhat in their implementation, application, and analysis.

5.3.1 Fork detector

What is measured: The gross neutron and gamma intensity using the Fork Detector Irradiated Fuel Measuring System (FDET) (see Figure 5-21). An enhanced version of FDET has a CdZnTe detector, which provides spectrally resolved gamma data.

What is quantified: The TN and total gamma counts are used for the gross defect detection and verification of declared data. The ratio between neutron and gamma-ray counts can be used to characterize a fuel assembly (i.e., the in-core neutron exposure, the initial fissile content, and its irradiation history). The measured neutron count rate is related to the BU and CT of the used fuel.

Description of the basic physics: The neutron detectors are gas-filled fission chambers, and the gamma detectors for the traditional Fork detector are gas-filled ionization chambers. The signal from these detectors is proportional to the gross (i.e., without energy spectral information) neutron and gamma intensity; the enhanced Fork detector has energy-resolved spectral information.

Expected measurement time: Measurements that were performed at the Clab Facility in 1997 lasted 2 minutes and resulted in better than 1% statistical uncertainty.

Discussion of calibration precision and accuracy: Rinard and Bosler (1988) reported that after making a correction for CT, the neutron count rate is correlated to the BU, with ~10% scatter being the average. It was also reported that the "removal of large fractions (perhaps 20% or more) of an assembly is easily detected."

In Borella et al. (2011) a correlation between the measured neutron count rate and declared BU was established to determine BU with a 2% uncertainty for a BU range of 30–55 GWd/tU and for a CT of at least 3 years.

Unique features relative to other NDA techniques: Compared with some of the NGSI techniques, the Fork detector and associated equipment have long been used, with a correspondingly long history of development of the infrastructure needed to support the measurement technique.

Maturity of the hardware: Fission chambers, ionization chambers, and associated electronics have been used in radiation detection for decades. The hardware is mature.

Maturity of the analysis: The analysis of neutron and gamma counts is relatively straightforward and has been thoroughly documented. The analysis is mature. However, work integrating a BU code with a fork measurement is less mature (Gauld et al. 2006, Smejkal et al. 2012).

Figure 5-21. *An FDET.*

Considerations for implementation: The system includes a detector head attached to an extension pipe, detector electronics, and a computer. The detector head is equipped with gamma-ray-insensitive neutron detectors and gamma-ray detectors. A measurement is conducted by positioning the detector fork so that it straddles the fuel assembly. The Fork detector and associated equipment are portable, with an overall size of ~1 m^3 when packed.

Development: An enhanced Fork detector, equipped with an additional CdZnTe detector, was used (Tiita et al. 2002) to evaluate the possibility of performing partial defect verification. It was concluded that a 50% defect level could not be verified without using operator-declared data. However, ongoing work (Van der Meer and Coeck 2006) questions that conclusion. The capability of partial defect verification, including the level of defect, should be verifiably established before implementing this technique for that purpose.

5.3.2 Partial Defect Tester (PDET)

What is measured: Total neutron (fission chambers) and total gamma (ion chamber) count rates measured by small detectors that move down guide tubes within an assembly (the hardware and a sample of the data are depicted in Figure 5-22 (Ham and Sitaraman 2011, Ham et al. 2010). This measured signal can be obtained only if guide tubes exist for the detectors to go down, which eliminates some assemblies (most notably BWR assemblies) from measurement.

What is quantified: Primarily detecting if pins are missing by detecting a localized variation in the neutron-to-photon ratio in the assembly. Information regarding BU and CT is also obtained. It is expected that diversion of ~10% of the mass can be detected (Ham and Sitaraman 2011).

Description of the basic physics: This integrated system combines PG and TN, as do the Fork and the SMOPY integrated systems. What makes PDET unique is the spatial information that is obtained by putting the detectors down the multiple guide tubes of a PWR assembly. In the right-hand side of Figure 5-22, both simulated and measured PDET data are illustrated. The "normalized ratio" is the normalized ratio of the gamma-to-neutron count rates. Each point on the "Detector" axis represents a different guide tube location. The green "J14" curve is the expected signal for the assembly if no pins were missing. In the case of the J14 assembly that was measured at the Korean Atomic Energy

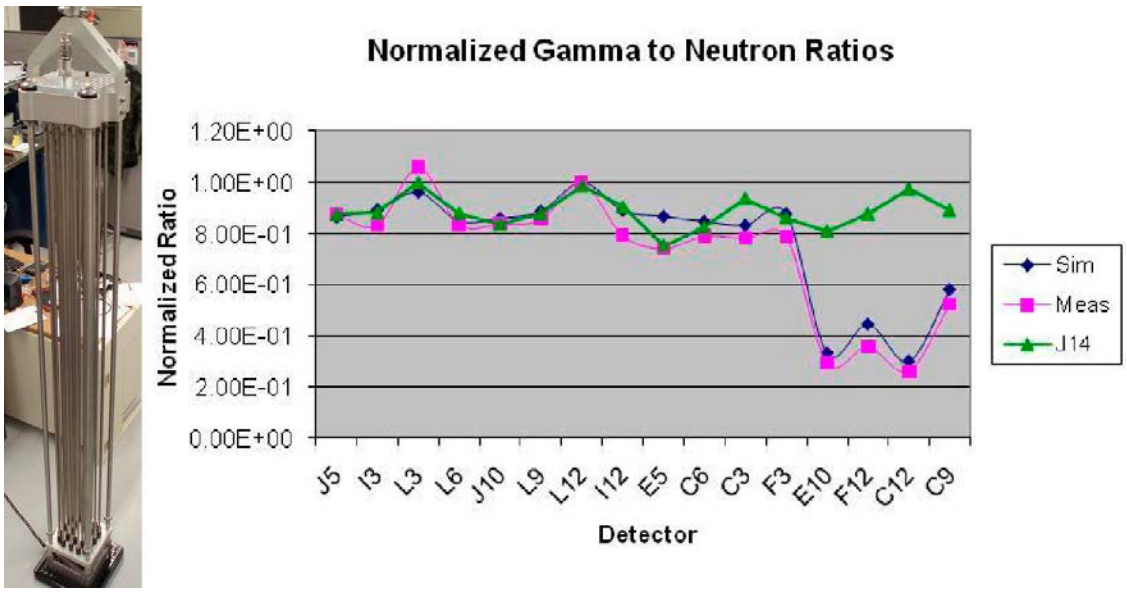

Figure 5-22. *Left, (Ham et al. 2010): Photograph of a prototype PDET system. Right, (Ham and Sitaraman 2011): Simulated and measured PDET data. The "normalized ratio" is the normalized gamma-to-neutron count rate. Each point on the "Detector" axis represents a different guide tube location. The "Sim" and Meas" curves are the measured and simulated results for a particular assembly that had missing pins. The "J14" curve is the expected signal for the assembly if no pins were missing.*

Research Institute (KAERI), some pins were not present (Ham and Sitaraman 2011). The “Sim” and Meas” curves in Figure 5-22 are the simulated and measured results with 22 pins (12% of the mass) missing, respectively. The difference between the “J14” and “Sim” and Meas” quantifies the change in the ratio when a diversion has occurred.

Spatial distribution of the signal: The PDET approach of putting detectors inside the assembly gives valuable spatial information about the nuclear material distribution in the assembly all along the axial length.

Expected measurement time: The time needed to measure one assembly will likely be dominated by the time it takes to attach/align the detector structure to the assembly. It is expected that the actual measurement of neutron and photon will take less than 5 minutes for most assemblies.

Discussion of calibration, precision and accuracy: No calibration is needed to produce the results illustrated in Figure 5-22. Additional information can be obtained by using the absolute signal values of the PDET detectors; relative comparison among assemblies will likely be most useful for the inspectorate.

Unique features relative to other NDA techniques: The unique feature of PDET is obtained by introducing the detectors into the assembly. This act of introducing fission chambers and ion chambers into the assembly provides very useful information regarding missing pins along the axial length of the fuel.

Maturity of the hardware: The fission chambers and ion chambers used are very mature. The mounting structure depicted in Figure 5-22 is the first prototype.

Maturity of the analysis: The analysis is moderately mature. Dr. Ham has several publications and has looked at the impact of near neighbors, as well as symmetric diversion; he will be performing tests in Sweden in 2012 or 2013.

Considerations for implementation: The largest detracting factor for inspectorates is that PDET can work only for a fraction of the fuel needing inspection; in the case of Sweden, less than 15% of fuel assemblies can be measured with PDET because most of Sweden’s assemblies are BWRs.

5.3.3 Safeguards Mixed Oxide (MOX) Python (SMOPY) Detector

What is measured: Gross neutron intensity and spectra resolved gamma intensity. The gamma intensity is measured with relatively poor energy resolution, but the peaks of interest are resolved.

What is quantified: A shielded CdZnTe gamma spectrometer and a fission chamber are used to distinguish MOX fuel from LEU fuel and to verify the BU and CT (Lebrun et al. 2003). A partial defect test of the used fuel can be performed using operator-declared data for depletion calculations (see Figure 5-23).

Figure 5-23. *A SMOPY detector.*

Description of the basic physics: Passive neutron and gamma spectrometry is combined (see the physics description for PG scanning and passive neutron measurements).

Discussion of calibration precision and accuracy: No information has been found regarding the uncertainties in CT verification. In Lebrun et al. (2003) a 5% BU verification uncertainty was reported for LEU fuel.

No information on the sensitivity of partial defect testing has been found.

Unique features relative to other NDA techniques: The system incorporates the use of room-temperature gamma spectroscopy using CdZnTe detectors, which has relatively low detection efficiency compared with an HPGe detector.

Maturity of the hardware: Mostly off-the-shelf hardware is used in the device. The hardware is mature.

Maturity of the analysis: An iterative procedure using measurement data, combined with operator-declared data and isotope depletion codes, is used for analysis. The maturity can be considered medium.

Considerations for implementation: The SMOPY device measures ~60 cm on one side and weighs 50 kg (Lebrun et al. 2003). The device is placed in water and positioned beside the fuel assembly to be measured. The CdZnTe detector was selected for the following reasons: 1) it can operate at room temperature, 2) it can tolerate elevated neutron fluxes and 3) it has good enough resolution for the few dominant spectral lines emitted from typical spent fuel.

Development: In Lebrun et al. (2003) the device was in the process of being upgraded to be able to measure gamma radiation from fuel assemblies with short CTs.

5.4 Integrated system of NDA techniques

5.4.1 A general discussion of integrated systems

In Section 5.1 a wide breadth of NDA options are described, which include calorimetric, active and passive neutron, and active and passive photon. In Section 5.3 the two integrated systems deployed for safeguards applications are described; these two systems combine PG and TN. The purpose of this section is to discuss the motivation for integrating techniques beyond what has already been done to meet past safeguards needs.

Commercial fuel starts out to be very simple from an isotopic perspective. Almost ~99.9% of the mass comprises three isotopes when it is first placed in the reactor – ^{238}U, ^{235}U, and ^{16}O. However, once the fuel is located in the reactor and exposed to the high neutron flux, a very diverse and complicated process of isotope creation and destruction begins. Multiple isotopes impact NDA signatures through the following mechanisms; a few of the most significant isotopes in a fully burned, few-year-cooled assembly are listed:

- Neutron production [^{244}Cm, ^{242}Cm ^{240}Pu, (α,n) sources, etc].
- Neutron multiplication (^{235}U, ^{239}Pu, ^{241}Pu, ^{238}U, etc).
- Neutron absorption (^{240}Pu, ^{143}Nd, ^{155}Gd, ^{149}Sm, ^{241}Am, ^{151}Sm, etc).
- Photon production (^{137}Cs, ^{134}Cs, ^{154}Eu, uranium x-rays, etc).
- Photon attenuation (all isotopes, uranium-dominant isotope).
- Heat production (^{90}Y, ^{90}Sr, ^{137}Cs, ^{134}Cs, ^{241}Am, etc).

The task in the context of the Swedish encapsulation/repository facility is to satisfy the heat, criticality, and safeguards requirements simultaneously in a timely manner. By combining a few NDA techniques, along with an analysis approach that includes the physics of isotope evolution of the fuel, both in a reactor and outside a reactor, the most accurate understanding of the fuel composition can be obtained within time and cost constraints.

5.4.2 A description of classes of NDA techniques and how they may be used

The remainder of this section describes positive and negative attributes of each of the three broad classes of NDA techniques, primarily from a safeguards perspective, as well as describes the quantities that these broad classes of NDA techniques can quantify.

Active and passive neutron: A positive attribute of most of the neutron-based techniques is that the entire assembly is interrogated due to neutron multiplication; this is true for fully burned commercial assemblies for which multiplying techniques are generally more sensitive to a central pin being removed than for an edge pin. Additionally, a significant fraction of the signal from the multiplication-based techniques comes from isotopes of plutonium. On the negative side, neutron absorbers, both fission fragments and actinides, alter the total count rate. Furthermore, because ^{235}U, ^{239}Pu, and ^{241}Pu are all involved in thermal multiplication, additional information is needed to separate the relative contribution among the three isotopes.

Numerous options are available regarding what to quantify using neutron techniques for safeguards. For mass accountancy, plutonium mass is the clear target, and some neutron techniques obtain a significant fraction of their signal from isotopes of plutonium. In the context of item accountancy, determining the integrity of the item is the end goal; this item accountancy is often implemented by making either a gross or partial defect determination through measurement. Some flexibility exists as to what would be quantified to assess the integrity of an assembly. Any of the following signal/analysis/integration of signals might be chosen as desirable by an inspectorate, listed in order of increasing complexity:

1. Fingerprint: the raw count rate of the instrument, the signal is unique for each NDA technique, comparison between measurements of the same assembly implied.
2. Multiplication: the combined effect of neutron absorbers and fissile material.
3. Fissile content or $^{239}Pu_{eff}$: "remove" the impact of neutron absorbers from multiplication.
4. Plutonium mass: discern among the isotopic component of fissile content and likely use a plutonium isotopic correlation to determine the elemental plutonium mass from a subset of the five main isotopes.

The issue of partial defect detection is significantly complicated if pin replacement scenarios are considered. A sampling of conceivable variations includes rods of the following composition: hollow, metal, depleted uranium, natural uranium, and enriched uranium. In addition, a neutron (curium or californium) or photon (cesium or cobalt) source could be added to any of the previous scenarios.

Passive photon: A positive attribute of passive photon techniques is that they provide information about what the fuel experienced in the core. The measured radiation after ~5 years of cooling is dominated by the decay of ^{137}Cs, ^{134}Cs, and ^{154}Eu, as well as radiation initiated by the activation of the cladding. Additionally, the strongest intensity line in the most fuel is ^{137}Cs, which is also an isotope that contributes roughly one-third of the heat emanating from the fuel. On the negative side, attenuation means that the measured signal is significantly stronger from the outer pins of an assembly; Generally, the vast majority of the photons come from the outer three rows. With intelligently designed collimation and longer count times, tomography can effectively detect the absence of fuel rods deeper in the assembly, particularly for BWR assemblies. An important aspect to verify experimentally is if tomography is blind to the center of assemblies for the very largest assemblies. An additional point is that no PGs are detectable from fissile material. The low-energy, 60-keV peak from the outer pins of an assembly is the only detectable peak from an actinide. Cherenkov techniques merit mention here because the PG intensity drives the light emitted from the water around the fuel.

The question of what to quantify for PG techniques is either spectrally resolved lines or the total intensity of gammas. Spectrally resolved gamma data can be useful in the context of mass accountancy by providing input data to BU code calculations. In the context of item accountancy, any of the following might be chosen to be quantified for assemblies, listed in order of increasing complexity:

1. Cherenkov image from the top of the assembly.
2. Fingerprint: photon intensity comparison between measurements of the same assembly.
3. Total gamma with an ion chamber.
4. Spectrally resolved single-collimator gamma spectra.
5. Tomographic gamma: pin-by-pin resolution.
6. Plutonium mass: input data to BU code calculations.

A sampling of conceivable replacement or diversion scenarios includes the following pin substitutions: hollow, metal, and uranium. The inclusion of gamma sources such as cesium or cobalt in the replacement pin are the most problematic but may be considered too complex.

Calorimetric measurements: Measuring the heat emitted by the assembly is the most accurate method of quantifying the heat. The challenge is the multiple hours generally needed to measure an entire assembly. Although calorimetric measurements are generally the most accurate measurements for safeguards, the isotopic uncertainty and complexity of spent fuel preclude applying calorimetric measurements for safeguards purposes.

5.4.3 Next generation safeguards initiative integrated systems

This section gives an overview of a few integrated NDA systems that were selected by the NA-241 management team to meet the goals and objectives of the NGSI Spent Fuel Project (Humphrey et al. 2012). The NA-241 management team down-selected among the NGSI Spent Fuel Techniques to the four NDA systems that seemed particularly promising to improve the toolkit of safeguards inspectors within a timeframe of ~5 years. Below, the four selected systems are listed by the sum of the acronyms from which they are composed. This order of listing is roughly in order of increasing complexity.

NGSI Spent Fuel Systems:

- PNAR-FC + SINRD + PG + TN
- CIPN + SINRD + PG + TN
- DDSI + SINRD + PG + TN
- Neutron-Generator-Based Techniques + PG + TN

System 1: PNAR-FC + SINRD + PG + TN

Of significant appeal to safeguards inspectorates for this system is that it is robust, inexpensive, and simple to deploy. The primary limitation is that the penetrating technique of the integration, PNAR-FC, is less sensitive to changes in fissile mass than the other penetrating techniques (CIPN, DDSI, DN, or DDA) used in the other systems. A key research question to answer is if it is sensitive enough for the task of interest. The hardware cost for building an already-designed instrument is less than $250k**.**

PNAR-FC is the “central” instrument in this integration because it is a “multiplication-based” instrument, which enables it to be sensitive to the absence of fuel everywhere in the assembly. Because PNAR-FC’s signal is proportional to a weighted sum of ^{235}U, ^{239}Pu, and ^{241}Pu, the signal is proportional to the multiplication. Further integration is unnecessary if that signal suffices. To determine more about the content of the fuel, it is likely that PG and TN measurements would be used to quantify the IE, BU, and CT. With knowledge of IE, BU, and CT, a correction for the impact of neutron absorbers is possible, which means that the fissile content or plutonium mass can be determined. It is anticipated that SINRD may be useful in the plutonium mass determination.

The most likely implementation of a PNAR-FC + SIRND + TN + PG detector is as a collar detector that sits on, or is built into, a spent fuel rack under water. PNAR could also be useful if at any point spent fuel management results in passing the fuel through a cylinder; for example, some facilities have a protective cylinder around the fuel when the fuel is on the crane as part of protecting the fuel – this protective layer could be a detector. One vertical half of the detector is lined with cadmium,

whereas the other is polyethylene or water. The TN measurement is already part of the PNAR-FC detector. To be consistent with the "robust, inexpensive, simple" appeal of this system, a few ion chambers would likely be used to measure the total PG signal, although a high-resolution detector could be used.

System 2: CIPN + SINRD + PG + TN

Of significant appeal to safeguards inspectorates for this system is that it is robust and inexpensive. This system is less easy to deploy than the PNAR-FC-based system because a ~200-µg californium source is needed (a 100-µg californium source likely is strong enough, but to have a reasonable lifetime, a stronger californium source must be purchased to allow for decay). The californium source renders this system appropriate for a fixed installation. The hardware cost for building an already-designed instrument is less than $250k, with the californium source costing ~$100k; essentially, CIPN is a Fork detector with a californium source added**.**

CIPN is the "central" instrument in this integration because it is a "multiplication-based" instrument, which enables it to be sensitive to the absence of fuel everywhere in the assembly. Because CIPN's signal is proportional to a weighted sum of ^{235}U, ^{239}Pu, and ^{241}Pu, the signal is proportional to the multiplication. Further integration is unnecessary if that signal suffices. If more needs to be determined about the content of the fuel, it is likely that PG and TN measurements would be used to quantify the IE, BU, and CT. With knowledge of IE, BU, and CT, a correction for the impact of neutron absorbers is possible, which means that the fissile content or plutonium mass can be determined. It is anticipated that SINRD may be useful in the plutonium mass determination.

Although this system could be implemented at the end of a pole such as with a Fork detector, the use of a strong californium source means that the most probably use of this system will be at a fixed installation. For this reason, a possible implementation could have a Fork detector positioned on the side of the pool. The fuel would be brought into the detector from the side and the californium source brought in after the fuel was in place. A door slowly closing is one option for positioning the californium source. If such a setup were used, a collimated gamma detection system could be built in with a collimator through the side wall of the pool.

System 3: DDSI + SINRD + PG + TN

Of significant appeal to safeguards inspectorates for this system is that it is robust and provides higher-quality data than the previous two systems. Unlike any other technique, the interrogating source is embedded in the fuel. This technique results in a more uniform spatial sensitivity than any other technique, and the use of ratios between early and late gates makes the technique about five times less sensitive to positioning of the fuel. Also, there is the potential of measuring the relative amount of ^{235}U and plutonium fissile material because of the temporal response of the correlated neutrons. The hardware cost for building an already designed instrument is less than $600k. The weight of this system, due to lead shielding of the ^{3}He tubes, renders it appropriate for a fixed installation. The ^{3}He availability may limit this instrument to a few high-value facilities. High count rates will be an issue for high BU fuels.

DDSI is the central instrument in this integration because it is a multiplication-based instrument, which enables it to be sensitive to the absence of fuel everywhere in the assembly. The DDSI signal is proportional to a weighted sum of ^{235}U, ^{239}Pu, and ^{241}Pu; as such, the signal is proportional to the multiplication. The use of gamma data in the analysis could be similar to that of the previous two systems, unless it is possible to separate ^{235}U and plutonium fissile material. The DDSI detector can allow side entry, such as with a Fork detector, or it could be a cylinder structure with fuel inserted from above. Either way, the large weight of the instrument will require a standalone support structure. The TN measurement is already part of the DDSI detector. A high-resolution detector (HPGe, $LaBr_3$, or CZT) would likely be integrated into the DDSI unit.

System 4: Neutron-generator-based techniques + PG + TN

The appeal to safeguards inspectorates for this system is the richness of the physical signatures that is anticipated to improve the overall accuracy of isotopic determination. All three neutron-generator-based techniques produce signatures that are proportional to the weighted sum of ^{235}U, ^{239}Pu, and ^{241}Pu. However, each technique weights the three isotopes differently, providing the potential to determine each fissile isotope uniquely. The DDA signal is unique in that it can be used to measure the multiplication in three distinct ways, and simulation indicates promise for the determination of IE, BU, and CT. The DN signal is unique because it is the only technique that preferentially weights ^{235}U relative to ^{239}Pu per unit mass, whereas DG is unique because it provides the relative amount of each of the three fissile isotopes. Because of neutron multiplication, the signals from DDA and DN penetrate the assembly throughout; the response of DG is less uniform due to attenuation of the photons as they travel out of the assembly. The differing spatial response among these techniques may assist in detecting diversion.

In the context of IE, BU, and CT determination, it is worth pointing out that the traditional means of quantifying IE, BU, and CT use a PG signal that primarily comes from the outer two or three rows. By combining IE, BU, and CT determined with neutron-generator-based techniques with PG/TN determination of these same parameters, a more robust indication of IE, BU, and CT is expected. Any removal of fuel rods from the assembly would likely produce a variation between the edge and bulk determination of IE, BU, and CT; thus, the combination of PG would be a good indication of tampering with the assembly.

The hardware cost for building an already-designed instrument varies with the techniques primarily due to the strength of the neutron generator needed. A DDA instrument could be fabricated for less than \$500k (for an off-the-shelf generator), a DN instrument for less than \$1,200k, and a DG instrument for less than \$2,000k (assuming that the custom-designed neutron generator is already designed).

6 NDA options for quantifying heat

The two primary documented techniques for estimating the decay heat from assemblies with measurements are calorimetric and PG measurements. The calorimetric measurement of an assembly measures the actual heat leaving the assembly and corrects for the small amount, generally less than 10%, of energy that streams through the calorimeter. The PG determination involves correlating the PG signal and the heat produced. GT, with its greater spatial resolution of the gamma source, can also be used to correlate to the decay heat because the same isotopic information about the fuel assembly is measured as with PG. Table 6-1 indicates some of the key parameters for approaches to determining the heat production of an assembly (both BWR and PWR).

A calorimetric instrument directly measures the quantity of interest. This instrument is the most accurate option, with the primary drawback being a long measurement time.

The remainder of this section gives more detail on the connection between the PG signatures and heat production. To quantify the percentage of the total heat source term that can be estimated by correlating the PG signal to the total heat, the total heat produced was calculated for all assemblies in the first NGSI Spent Fuel Library (Broadhead et al. 2012, Trellue H R 2012, personal communication). This library contains the simulated isotopic inventory of 64 assemblies that vary over a wide range of IE, BU, and CT. For some isotopes the isotopic mass was used directly from the BU code simulation, whereas for the mass of $^{137}Ba_m$ and ^{90}Y, the Batesman equation was used because the BU code as implemented did not track these isotopes (Gerhart 2012). To quantify the PG signal, gamma rays were emitted from every pin in an assembly from all the primary isotopes known to produce detectable gamma rays (Broadhead et al. 2012). If a peak was detectable, then the heat from that peak could be correlated to the intensity of the peak. The simulated results from the NGSI Spent Fuel Library are depicted in Table 6-2; in addition to ^{137}Cs and ^{241}Am, the contribution of ^{134}Cs was also determined. ^{154}Eu contributes only a few % in some cases. The column labeled “Heat from ^{137}Cs” indicates that 1) gamma rays from ^{137}Cs are detectable and 2) the heat liberated by the decay chain containing ^{137}Cs emits the listed percentage of the total heat. As such, the final column represents the percentage of the total heat liberated by the sum of the decay chains containing ^{137}Cs, ^{134}Cs and ^{241}Am.

Fortunately, the two isotopes that produce between approximately one-third and two-thirds of the heat in a spent fuel assembly also produce detectable gamma rays. Those isotopes are ^{137}Cs and ^{241}Am. Of these two isotopes, ^{137}Cs is the most interesting because it emits a detectable gamma ray from essentially all fuel and ^{137}Cs is the dominant gamma source for fuel cooled longer that ~5 years. Also, the 662-keV gamma from ^{137}Cs is penetrating enough so that the outer two to three rows contribute ~90% of the measured signal for a typical detector setup. These rows account for ~50% of the fuel; thus, the measured signal is obtained from a large fraction of the fuel, providing some justification for the correlation between the PG signal and the heat generated in the assembly.

Table 6-1. Measurement Time, Uncertainty, and Equipment Time for Determining the Decay Heat with Colorimetric and PG Measurements for Both BWR and PWR Assemblies.

Technique	Measurement Time[a]	Uncertainty	Equipment Size	Notes
Calorimetric measurement	~4 h	~1%–2%	Full length of assembly, <1 m across	Requires calibration with electric heater
PG scanning	~15 min	~2%–3%	Collimated view of assembly	Requires calorimetric calibration
GT	~8 h	~2%–3%	~0.5 m axial length, ~2 m horizontally	Requires calorimetric calibration. Can do single–pin-level partial defect verification

[a]For complete assembly.

Table 6-2. Heat Generated by Isotopes that Emit Detectable Gamma Rays for the First NGSI Spent Fuel Library. Columns 1 to 3 Indicate the Properties of the Spent Fuel from the First NGSI Spent Fuel Library (Broadhead et al. 2012, Trellue H R 2012, personal communication). The Percentage of Total Heat Liberated by Specific Isotopes for Which Gamma Rays Are Detectable Are Listed in Columns 4 to 6.

Initial Enrichment (%)	Burn-Up (GWd/tU)	Cooling Time (years)	Heat from ^{134}Cs (%)	Heat from ^{137}Cs (%)	Heat from ^{241}Am (%)	Heat from ^{134}Cs, ^{137}Cs, ^{241}Am (%)	Total Heat (W)
2	15	1	31%	23%	0%	54%	638
2	15	5	11%	31%	2%	44%	442
2	15	20	0%	38%	8%	46%	254
2	15	40	0%	38%	18%	57%	157
2	15	80	0%	30%	39%	69%	80
3	15	1	27%	24%	0%	51%	628
3	15	5	10%	30%	1%	42%	449
3	15	20	0%	38%	6%	44%	257
3	15	40	0%	40%	14%	54%	153
3	15	80	0%	34%	33%	67%	72
4	15	1	25%	24%	0%	49%	637
4	15	5	9%	30%	1%	40%	461
4	15	20	0%	37%	5%	43%	264
4	15	40	0%	41%	12%	53%	154
4	15	80	0%	36%	29%	65%	69
5	15	1	22%	24%	0%	47%	623
5	15	5	8%	30%	1%	39%	459
5	15	20	0%	37%	4%	42%	262
5	15	40	0%	41%	10%	51%	149
5	15	80	0%	39%	25%	63%	64
3	30	1	39%	20%	0%	59%	1,481
3	30	5	16%	28%	2%	46%	942
3	30	20	0%	37%	8%	45%	510
3	30	40	0%	38%	17%	55%	312
3	30	80	0%	31%	36%	67%	156
4	30	1	36%	20%	0%	57%	1,453
4	30	5	15%	28%	1%	44%	948
4	30	20	0%	37%	7%	44%	517
4	30	40	0%	38%	16%	54%	313
4	30	80	0%	31%	35%	67%	153
5	30	1	38%	20%	0%	57%	1,749
5	30	5	15%	28%	1%	44%	1,128
5	30	20	0%	37%	7%	43%	606
5	30	40	0%	39%	15%	53%	362
5	30	80	0%	32%	33%	66%	173
4	45	1	39%	19%	0%	58%	2,288
4	45	5	16%	27%	2%	45%	1,460
4	45	20	0%	36%	7%	43%	790
4	45	40	0%	37%	15%	52%	481
4	45	80	0%	30%	33%	63%	237
5	45	1	41%	18%	0%	59%	2,349
5	45	5	17%	27%	1%	45%	1,469
5	45	20	0%	36%	7%	43%	778
5	45	40	0%	38%	15%	52%	469
5	45	80	0%	31%	33%	64%	227
5	60	1	44%	17%	0%	61%	3,394
5	60	5	19%	25%	1%	46%	2,050
5	60	20	0%	35%	6%	41%	1,060
5	60	40	0%	36%	13%	49%	639
5	60	80	0%	30%	29%	59%	308

The 60-keV peak from ^{241}Am may be of interest for older fuels (>40 years). Compared with ^{137}Cs, the use of the 60-keV peak from ^{241}Am is of less interest because 1) most fuel has CTs less than 40 years; 2) a dedicated detector, planar (not coaxial) with very small collimation, is needed to measure in the low-energy range of spent fuel; and 3) the detected gammas come primarily from the outer ~1 mm of the fuel. It may be necessary to displace water with an air guide to be able to measure the 60-keV peak. The attention due to an additional metal layer around the pins of a BWRs is also a concern. What is certain is that the 60-keV peaks was observed in single spent fuel pin measurements made in hot cell conditions (White et al. 2011).

The primary conclusion from Table 6-2 is that between ~40% and ~70% of the heat source term can be estimated from the PG signal for the wide range of fuels listed above. The fact that the signal comes from the outer few rows with traditional PG measurements may motivate the application of GT.

7 Options for safeguards parameters

Under current safeguards practice, NDA measurements of spent fuel are primarily used in the context of item accountancy. With item accountancy, the inspectorate is generally interested in determining if a defect is *gross* or *partial*. With a gross defect, the inspectorate is verifying that a particular assembly is indeed a spent fuel assembly. Detection of the 662-keV gamma ray emitted from the decay of the ^{137}Cs nucleus from the top of the assembly or the detection of glowing water around the pins in an assembly suffices. With a partial defect, the goal is to detect if a significant amount of nuclear material has been removed from a spent fuel assembly and possibly replaced by dummies; with "significant amount," the value of 50% is generally used.

Among the 17 individual NDA techniques listed in Section 5.1, two techniques (DCVD and GT), by the nature of their signal, automatically give information about the presence of individual pins. Both of these techniques, particularly DCVD, have a body of research quantifying their capability. The three integrated systems used by safeguards, all of which integrate PG and TN, also have focused research on their partial defect capability. The remainder of the individual NDA techniques has not been researched in much detail regarding partial defect detection.

The integrated systems of the NGSI Spent Fuel Project start, in a sense, with the same base information as the three current integrated systems – TN and total gamma. Each system adds more signatures to that: multiplication in the case of PNAR, CIPN, DDSI, and the neutron generator techniques; and ^{239}Pu mass in the case of SINRD. It is anticipated that the partial defects detection threshold will be noticeably improved through this integration.

The following is an example of how quantifying multiplication helps safeguards in the context of partial defect detection. The TN count rate is proportional to the neutron source term (^{244}Cm primarily in most cases) times the multiplication (primarily the cumulative effect of ^{235}U, ^{239}Pu, and ^{241}Pu, resulting in a multiplication of ~2 for a fully spent assembly). If the multiplication is quantified separately, then the ^{244}Cm is known. With a multiplication–based technique as part of the integration, the diversion of taking central rods and putting in any neutron source to match the overall neutron output of the assembly is detectable. Furthermore, quantifying the ^{244}Cm and the multiplication caused by ^{235}U, ^{239}Pu, and ^{241}Pu will improve the overall isotopic understanding of the assembly. This improved understanding of the isotopic content of the assembly should reduce the ability to change the content of the assembly without being detected.

The tables below summarize the techniques and combinations of techniques that can be used to determine safeguards-related fuel parameters.

7.1 Verify total, partial, or bias defects

Defect	Technique	Notes
Total	• All NDA techniques	
Partial >50% of fuel rods missing will be detected	• GT • All NDA techniques in Section 5.1, except CDH, NRF, PG, TN, and XRF. Note that several of these might function in an integrated system	DCVD and Fork need to be verified for all fuel types. DCVD can be fooled by activated rods with, e.g. ^{60}Co
Partial >25% of fuel rods missing will be detected	• GT • By integrating various arrangements of NDA techniques from Section 5.1, 25% is expected likely	
Partial >10% of fuel rods missing will be detected	• GT • By integrating various arrangements of NDA techniques from Section 5.1, 10% is expected possible, research needed to verify capability.	
Partial >5% of fuel rods missing will be detected	• GT • GT excluded; unless an assembly has previously been measured with one of the system techniques described in Section 5.1, 5% is going to be challenging. Research very much needed before claims can be made with confidence	
Single pin missing will be detected (bias defect)	• GT	Tomography needs verification on PWR fuel

7.2 Determine burn-up, initial enrichment, or cooling time

The primary NDA approach for quantifying IE, BU, and CT have used PG and TN and is reflected in the table below. The useful lines for safeguards in the PG spectra come from the decay of ^{137}Cs, ^{154}Eu, ^{134}Cs (CT less than ~20 years), ^{106}Ru (CT less than ~10 years). Each of these PG spectra and the TN signal are a function of BU and CT, and to differing degrees IE. Ratios among the spectral lines of these different isotopes are often used.

For the vast majority of the NDA techniques described in Section 5.1, little to no research has been done for the purpose of quantifying IE, BU, and CT; there is one publication on using DDA for quantifying IE, BU, and CT of which the author are aware (Henzl et al. 2012b). Yet, for most of the techniques, their signatures vary noticeably as a function of IE, BU, and CT. For the techniques that measure multiplication, IE, BU, and CT matter not only in terms of how the fissile content changes, but also in terms of how the neutron absorber population varies (Henzl et al. 2012a).

One potential benefit of measuring IE, BU, and CT with both a multiplication technique and a PG/TN is that the signal for the multiplication-based techniques penetrates the entire assembly, whereas the PG signal primarily comes from the outer two or three rows. Variation between the IE, BU, and CT determined from the edge and from the bulk could be an indication of tempering with an assembly. The table below lists some BU, IE, and CT situations and applicable NDA options.

Parameter	Technique(s)	Notes
BU without CT and IE	• PG scanning • GT • Fork • All NDA techniques in Section 5.1, likely a few techniques integrated together, except CDH and NRF	Gamma measurements need intensities from two isotopes and calibration curves for same fuel type
BU when only IE known	• PG scanning • GT • Fork • All NDA techniques in Section 5.1, likely a few techniques integrated together, except CDH and NRF	Gamma measurements need intensities from two isotopes and calibration curves for same fuel type
BU when CT and IE known	• PG scanning • GT • Fork • All NDA techniques in Section 5.1, likely a few techniques integrated together, except CDH and NRF	Gamma measurements need calibration curves for same fuel types
CT without BU and IE.	• PG scanning • GT • All NDA techniques in Section 5.1, likely a few techniques integrated together, except CDH and NRF	Gamma measurements need intensities from two isotopes and calibration curves for same fuel type. Fork can determine BU
CT when only IE known	• PG scanning • GT • All NDA techniques in Section 5.1, likely a few techniques integrated together, except CDH and NRF	Gamma measurements needs intensities from two isotopes and calibration curves for same fuel type. Fork can determine BU
CT when BU and IE known	• PG scanning • GT • All NDA techniques in Section 5.1, likely a few techniques integrated together, except CDH and NRF	
IE	• PG scanning • GT • All NDA techniques in Section 5.1, likely a few techniques integrated together, except CDH and NRF	Procedure according to Chichester D (2012, personal communication) to determine IE should be verified experimentally (using a large data set).

8 Conceptual description of a few systems customized for the Clink facility

A range of relevant factors should be considered in selecting an NDA system for future encapsulation and repository facilities. In selecting among the characteristics of the various options, a degree of subjectivity inevitably is involved. In this section, the authors of this report first list some of the key characters being used to make their recommendations, as well as list a few systems that appear to best meet these characteristics.

In the context of selecting NDA systems for the encapsulation and repository facilities, it is important to take a system-wide perspective; it may be possible to find a synergistic solution among the many requirements. From a *safeguards technical requirements perspective*, it is anticipated that two primary characteristics will be required: 1) accurate partial defect detection to assure the "completeness" of the declared assembly; and 2) accurate determination of IE, BU, and CT to assure the "correctness" of the declared assembly. From a *facility implementation perspective*, the following requirements are anticipated: 1) robust, reliable technologies and 2) a rapid-enough measurement time to meet the processing requirements of the Clink Facility. From a *heat load requirements perspective*, the heat of each assembly must be determined accurately. From a *criticality perspective*, measurement of the reactivity may be desirable.

Below, three NDA systems are described that may meet the NDA need of the combined encapsulation and repository facilities. For each of the three systems, the NDA techniques are listed, followed by the merits and caveats for that system (Note: A BU code, or results thereof, will be a part of each system). It may be desirable to mix and match among the three systems, but the following three systems provide a structure for highlighting some of the most promising techniques working as a system:

- System A: 1) full-length calorimeter + 2) tomographic gamma scanning + 3) TN.
- System B: 1) active neutron (DDA or CIPN), TN comes for free with an active neutron technique + 2) PG + 3) Cerenkov (DCVD).
- System C: 1) PG + 2) TN.

System A provides the most accurate measure of heat and, arguably, the most accurate diversion detection technique. The fact that calorimetric measurements are the most accurate determination of heat is not in question. The term "arguable" is inserted before "most accurate diversion detection technique" for two reasons: 1) the performance of GT still must be tested for the largest spent fuel assemblies (17 × 17) for the range of IE, BU, and CT to be encountered by the encapsulation facility; and 2) the diversion scenario of a rod filled with fission fragments would not be detectable. This diversion scenario on its own seems very unlikely. Yet, if a State were to remove rods and reprocess them for the plutonium, returning the waste products into a metal rod may not be so difficult. The TN detection was added to enable IE, BU, and CT to be determined; it was also included because it is inexpensive and because TN captures a unique signature relative to GT and calorimetric measurements. The primary concern with System A is the long count time; calorimetric measurements take multiple hours, and the count time of GT for large, long-cooled assemblies must be determined. It may be desirable to add a DDA- or CIPN-like measurement to System A to counter the fission-fragment-laced diversion scenario or to accurately measure the reactivity of the assembly for criticality concerns.

System B is a fast measurement time system. This system provides the determination of heat with PG. The existence of a partial defect is checked by two independent approaches: 1) combining the active neutron, TN, and PG and 2) DCVD. The partial defect detection approach 1) is vulnerable to a diversion that matches multiplication and matches TN source strength; whereas 2) is vulnerable to fuel not visible from above the assembly or substitution, including gamma-emitting isotopes. Both diversion detections systems use robust commercial, off-the-shelf technology. IE, BU, and CT are quantified with PG and TN; active neutrons can also determine IE, BU, and CT; however, this approach is not as mature.

System C is the status quo for inspectors. It could be satisfied with a Fork (enhanced of traditional) or SMOPY detector. The heat can be determined with the PG detection. The partial defect performance is that of a Fork or SMOPY detector. System C is simpler than System B in that an active neutron or DCVD is not included. Including either the active neutron or DCVD improves the partial defect detection. The addition of an active neutron capability measures the reactivity of the assembly, as well as provides confidence in the BU code calculation because the active neutron measurement will provide additional information about the isotopic composition of the assembly.

Note: TN, Fork, DDA, and CIPN can be the same passive hardware. CIPN adds an active californium source, and DDA adds an active neutron generator.

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