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Delayed Neutron Detection with an Integrated Differential Die-Away and a Delayed Neutron Instrument

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Abstract

The Next Generation Safeguards Initiative (NGSI) of the U.S. Department Of Energy (DOE) has funded a multi-lab/university collaboration to quantify the plutonium (Pu) mass and detect the diversion of pins from spent nuclear fuel. The first two years of this NGSI effort was focused on quantifying the capability of a range of nondestructive assay (NDA) techniques with Monte Carlo (MCNPX) modeling and the second current phase involves measuring Spent Fuel. One of the techniques of interest in this paper involves measuring delayed neutrons. A delayed neutron instrument using 36 fission chambers and a 14 MeV neutron generator so called DT generator (Deuterium + Tritium) surrounding the fuel was previously studied as part of the NGSI effort[12].

This paper will quantify the capability of a standalone delayed neutron instrument using 4 ^{3}He gas filled tubes and a DT generator with significant spectrum tailoring, located far from the fuel. So that future research can assess how well a delayed neutron instrument will function as part of an integrated NDA system. A new design is going to be used to respond to the need of the techniques. This design has been modeled for a water media and is currently being optimized for borated water and air media as part of ongoing research. This new design was selected in order to minimize the fission of ^{238}U, to use a more realistic neutron generator design in the model, to reduce cost and facilitate the integration of a delayed neutron (DN) with a differential die-away (DDA) instrument.

Since this paper will focus on delayed neutron detection, the goal is to quantify the signal from ^{235}U, ^{239}Pu and ^{241}Pu, which are the isotopes present in Spent Fuel that respond significantly to a neutron interrogation. This report will quantify the capability of this new delayed neutron design to measure the combined mass of ^{235}U, ^{239}Pu and ^{241}Pu for 16 of the 64 assemblies of the NGSI Spent Fuel library in one of the three media, water.

Introduction

There is a range of motivations for measuring the Pu mass in Spent Fuel assemblies. Five safeguards motivations were listed in a recent publication[1] and are briefly listed here: Independently verify the mass of plutonium, Measure the shipper/receiver difference, Recover from lose of continuity of knowledge, determine input accountability mass, and determine Pu mass in Spent Fuel that is no longer considered "self-protecting." Four additional non-safeguards reasons were also listed: Provide confidence to the public in terms of nuclear management, enable optimal reloading of reactor cores, enable efficient fuel storage through "burnup (BU) credit", and enable assembly selection for reprocessing in order to obtain optimal compositions in reprocessing solutions.

The Monte Carlo effort has two main goals: Quantify the expected capability of each technique as an independent instrument and determine how to integrate a few techniques together in order to determine elemental Pu mass and pin diversion. This report is a part of the first goal; this report will quantify the capability of delayed neutron detection. Recent publications provide details of the motivations[2] and approach[3] being taken by to the NGSI research effort[13]. In order to give context to the delayed neutron

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research presented here, the other 12 techniques are listed below. A description of the basic physics of each of these techniques along with more detailed references are given in the following publications: Tobin et al[1], Menlove et al.[4] and Simpson et al.[5] The 12 additional NDA instruments are the following: Differential Die-Away, Differential Die-Away Self-Interrogation, Lead Slowing Down Spectrometer, Neutron Multiplicity, Passive Neutron Albedo Reactivity, Total Neutron (Gross Neutron), X-Ray Fluorescence, ^{252}Cf Interrogation with Prompt Neutron Detection, Delayed Gamma, Nuclear Resonance Fluorescence, Passive Prompt Gamma and Self-integration Neutron Resonance Densitometry.

Concept of Delayed Neutrons Assay

The delayed neutron technique is an active assay that consists in turning on a source, in this case the DT generator, already close to the exterior of the fuel, or in bringing a ^{252}Cf source close to it. The fundamental concept in the context of Spent Fuel is first to perform a passive neutron measurement to determinate the background neutron count rate (singles), mainly from spontaneous fission of ^{244}Cm in Spent Fuel, then to perform an active assay switching on the 14.1 MeV neutrons DT generator. These interrogating neutrons enter the fuel with a significant spectrum tailoring in this particular design and induce fission in the assembly. The DT source is then switched off or in the case of ^{252}Cf removed and the neutrons emitted are counted soon after. It is typical in Schuffler history to alternate between neutron interrogation and total neutron counting[6]. There are many isotopes produced when actinides fission, precursors. The time dependence of the delayed neutrons emitted from the β-decay of the precursors is often described by six groups, each with different half-lives. The half-lives of these groups vary from ~1/5th of a second to ~1 minute. The practice of alternating between neutron burst and total neutron counting produces a delayed neutron count rate that reaches an equilibrium value as illustrated in Figure 1 below. The following interrogation pattern was repeated 150 times in the production of Figure 1: 0.9 s. interrogation, 0.1s. pause, 1.0 s. count time in the model.

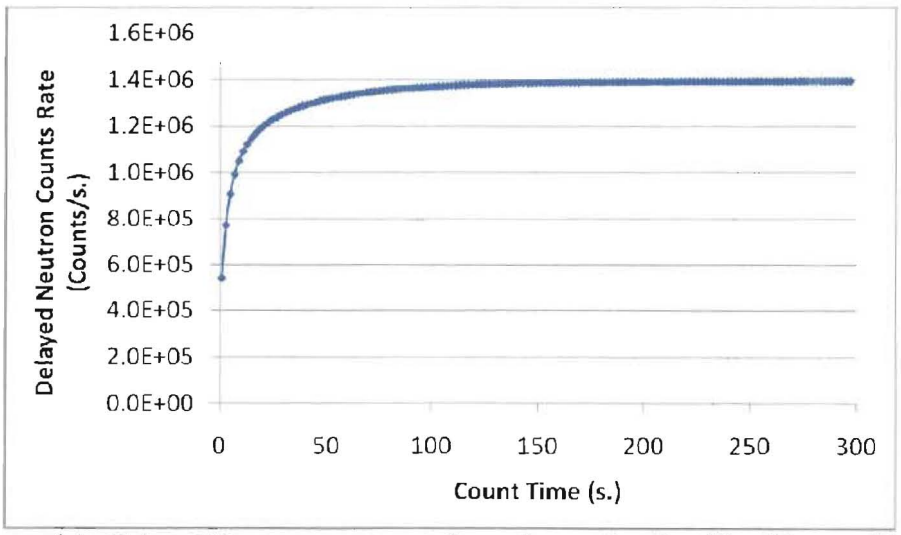

Figure 1: Sum-up of the Delayed Neutron counts rate for each counting time bin of 1 second for an assembly representing a BU of 15GWd/tU, 5% CT and 4 % IR over a period of time of 300 seconds – pattern repeated 150 times – Every 16 assemblies modeled have the same trend.

The total amount of neutrons after a burst is primarily a combination of spontaneous fission and delayed neutrons from the induced fission reactions (along with any multiplication those neutrons

experience). The intensity of the spontaneous fission background, mainly from ^{244}Cm in Spent Fuel, is not included in Figure 1; it will be quantified in a separate model in the future, especially since ^{244}Cm increases very quickly ($\sim\times10^4$) with burn up (BU). The neutron background is also eventually build over burst of cosmic rays, (α,n) representing a low portion (at its greatest 8% and most of the time less than 2% of the background[16] for all 64 assemblies of the Spent Fuel library), and (γ,n) emitted for gamma rays reaching at least 5 MeV energy where from fission they have a lower energy spectrum and irradiated materials around the assembly emit around 2.4 MeV thus usually negligible.

Design Requirements of the Model in the context of Spent Fuel and of an Integrated DDA Instrument

The delayed neutrons emitted by the fission fragments that are produced by induced fission with fissile isotopes are the quantity of interest. The production of background neutrons by spontaneous fission is dominated by ^{244}Cm. The energy spectrum of these neutrons is higher than that of delayed neutrons; however, this energy difference is not significant enough to discriminate between delayed neutrons and spontaneous fission neutrons. Furthermore, since we expect the multiplication for our design to be ~4 at low BU and ~1.5 at high BU in water according a previous design[12], however it will be quantify in the future, a significant fraction of the counts measured during the "delayed neutron counting" interval comes from prompt neutrons induced by delayed neutrons. In order to have meaningful statistics, the delayed neutrons count rate from fissile material must be ~10% percent or more relative to the spontaneous fission signal.

Both fissile and fertile isotopes in Spent Fuel produce delayed neutrons. ^{235}U, ^{239}Pu and ^{241}Pu are the prominent fissile isotopes of interest and ^{238}U is the dominant fertile isotope but those are not the only origin. Given that the goal of the NGSI Spent Fuel effort is to quantify Pu mass and that ^{238}U comprises more than 95% of the actinide mass, it is necessary to design the instrument to minimize fission in ^{238}U. This requirement is aided by three main factors: Effectively, fission can only be induced in ^{238}U if the incident neutron's energy is greater than ~1 MeV[15]. The amount of DN from other isotopes should be negligible; however where there is ^{241}Pu there is ^{241}Am for instance and this delayed neutron fraction would justify advanced future research concerning other contribution. Thermal fission cross sections of the fissile isotopes are roughly 3 orders of magnitude greater[15] than ^{238}U fission cross section in the MeV range and the amount of ^{238}U varies little among assemblies of all burnups. As the fuel assembly is burned from 15 to 60 GWd/tU, the ^{238}U mass decreases by between ~4% (IR=5%) and ~6% (IR=2%)[16]. ^{238}U can be a real issue since it's in the fuel in big quantity (table 2). The DN instrument is one of the only instruments that preferentially measure ^{235}U, since it has considerably more delayed neutrons per fission resulting in a discrimination ratio of 2.6 relative to ^{239}Pu (table 1) for a two component linear system. Because the ^{241}Pu has a delayed-neutron fraction that is very similar to ^{235}U and it will tend to wash out the discrimination factor[7]. Although in Spent Fuel the amount of ^{241}Pu is significantly smaller than ^{235}U what should allow keeping the discrimination factor.

Isotope	Thermal Fission Cross sections[7](barns) Fast for 238U	Discrimination Ratio[7]	Delayed Neutron Fraction (β, %): Average at all energies[6]
235U	584	-	0.67
239Pu	742	2.6	0.21
241Pu	1010	-	0.50
238U	0.7	-	1.48

Table 1: Thermal fission cross sections of a delayed neutron from reference[7] emitted as of a fraction β (average at all energies from reference[6]) are listed for the dominant three fissile isotopes in Spent Fuel. The fast fission data for ^{238}U are from reference[8].

Assembly properties	U-235 (g)	U-238 (g)	Pu-239 (g)	Pu-241 (g)
15 GWd/tU, 4% and 5 years	11,916	448,942	2,070	157
30 GWd/tU, 4% and 5 years	7,150	446,241	2,686	438
45 GWd/tU, 4% and 5 years	3,936	442,946	2,817	632
60 GWd/tU, 4% and 5 years	1,945	439,251	2,779	724

Table 2: Isotopes masses from MCNPX calculations for 4 of the 64 assemblies for a constant IE of 4%, 4 BU (15, 30, 45 and 60 GWd/tU) and a constant CT of 5 years. Masses obtained for all 64 assemblies[16].

We can easily see that in average over the 16 assemblies modeled, the mass of ^{241}Pu represents ~10.3% of ^{235}U mass and thus this instrument in Spent Fuel can preferentially measure ^{235}U, and as such a delayed neutron instrument is useful in combination with other instruments that preferentially measure Pu. Furthermore, in typical Spent Fuel there is at least twice as much ^{235}U as ^{239}Pu. And there is usually more than twice as much ^{239}Pu as ^{241}Pu as listed in table 2. If the fission of ^{238}U is sufficiently minimized, the DN signal will be produced by fission fragments from ^{235}U, ^{239}Pu and ^{241}Pu. Since a delayed neutron instrument isn't capable of discerning what delayed neutrons came from which isotope, the measured signal is a combination of neutrons from every neutrons detected disregarding of their origin in ^{3}He detectors. In Passive Neutron Coincidence Counting, PNCC, in analyzing coincident neutron signals, it is proved useful to introduce the concept of "$^{240}Pu_{eff}$"[9], here we are introducing a similar term called "$^{239}Pu_{DNeff}$" which stands for "^{239}Pu delayed neutron effective".

$$^{239}Pu_{DNeff} = C_1 \times {}^{235}U + {}^{239}Pu + C_2 \times {}^{241}Pu \qquad (1)$$

C_1 and C_2 are the contributions of ^{235}U and ^{241}Pu in terms of an equivalent amount of ^{239}Pu. We'll see in the processing of the tally section how do we account for these weighting factors.

There are essentially three interrogating sources of interest in the context of delayed neutron detection from Spent Fuel: a deuterium-tritium (DT), deuterium-deuterium (DD) neutron generators and a ^{252}Cf isotopic source. The research presented here indicates that a DD generator is definitely viable and that ^{252}Cf source will work as well particularly if more than one Cf source is used. However, for the case of MOX Spent Fuel, a DT generator will be needed. Given the initial expectation that a DT generator was needed, in addition to the significantly lower cost of such generators relative to DD generators, the majority of the results modeled started with the neutron energy (14.1 MeV) produced by DT generators. For a DT generator, the flux is expected to be highly increase through (n,2n) reactions in

this design. The maximum intensity expected possible for DT technology is ~2.10^{12} ns/s and the calculations will be made assuming that the source strength is 1.10^{11} ns/s.

Modeling of the Delayed Neutron Technique

The starting point for quantifying the expected performance of a delayed neutron instrument is the Spent Fuel library (64 cases plus about 40 diversions cases). A 17 by 17 Westinghouse PWR assembly was selected for quantifying the capability of all the techniques in the Spent Fuel effort. The differences among the assemblies emphasized isotopic, spatial, and diversion variability as described next[3].

The isotopic variability among 64 assemblies was obtained by using the Monte Carlo N-Particle eXtended (MCNPX) transport code that recently had the CINDER burnup capability added[11]. Each assembly has an isotopic mix determined from a unique combination of burnup, initial enrichment, and cooling time. The burnup cases were 15, 30, 45 and 60 GWd/tU; initial enrichment is here kept constant to 4%; cooling times are 1, 5, 20 and 80 years which makes it 16 assemblies studied in water only for design optimization reasons in other media to integrate this instrument with DDA instrument.

The spatial variability is only in the horizontal direction since no axial variation was quantified[16]. Each of the 264 pin rods were divided into four separate cells. This level of resolution within one pin is not expected to produce a significant effect on a delayed neutron instrument. However, the burnup difference among pins in the center and edge of an assembly should be more important. A future document will quantify these spatial differences within an assembly for the cases run.

Assemblies from which pins were diverted were created from three of the 64 assemblies. The majority of the diversion assemblies involve replacing pins from the center, mid and outer regions with depleted uranium (DU) pins. DU was selected since DU is not a safeguarded quantity and since DU pellets would maintain the weight of the assembly. The details of the diversion assemblies were presented in a paper by Fensin et al.[3] and diverted cases will be modeled in future studies with this design.

In Figure 2 and 3 we can see a 17 by 17 PWR fuel assembly, 366 cm height and 21.58 by 21.58 cm. The medium surrounding the pins within the assembly is water. Surrounding the assembly on all sides is a 0.49999 cm gap filled with water and needed to provide the mechanical tolerance for moving an assembly through the detector. The exterior surface defining the border of this gap is the inner surface of the detector. It is made of small layers of Cadmium of about 4 cm long, 0.1 cm large and 21.58 cm height and a rectangle of lead surrounding the assembly on the 3 sides where the 4 detectors are located. The lead block is 5 cm large everywhere and 22.42 cm long on both sides where there is only 1 detector and 32.42 cm long where 2 detectors are located, the block is centered in the middle of the assembly and its height his 21.58 cm. The 4 detectors are composed of ^{3}He gas in an Aluminum cylinder along Z, 5.08 cm height, and 0.945 cm radius, centered in the middle of the assembly. ^{3}He tubes are surrounded by high density polyethylene (HDPE) blocks of 5.99 by 5.99 cm of 9.28 cm height, surrounded by a Cadmium liner large of 0.1cm and 9.28 cm height, included in a lead block of 5.89 by 5.89 cm 11.28 cm height and centered in the middle of the assembly. 14.1 MeV neutrons are emitted from the DT generator located in a 9.8 cm height and 2.4 cm radius cylinder surrounded by a Tungsten square 15 by 15 cm and 20 cm height surrounded by a Beryllium square 3.5 cm large and 27 cm height. In Figure 2 and 3 a XY and YZ slices through the modeled detector are depicted.

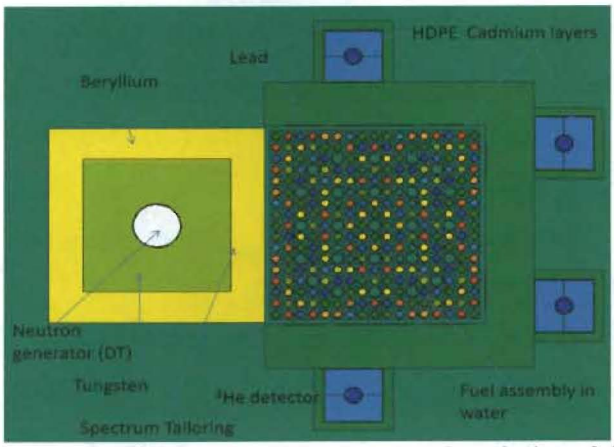

Figure 2: Horizontal cross section of the delayed neutron detector – a description of the material is given in the text.

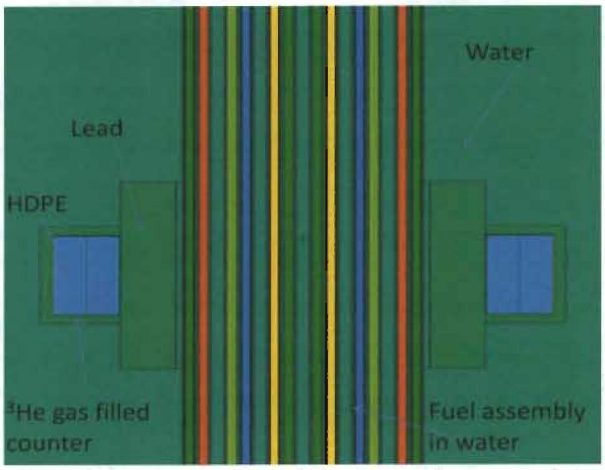

Figure 3: A vertical cross section of the delayed neutron detector is illustrated.

The Cd located around the fuel assembly and around the HDPE surrounding is there to minimize the detection of neutrons emitted by the assembly from outside the detector region. Tungsten, Beryllium and lead are used to increase the spectrum tailoring and lead to increase multiplication since it has a good cross section for (n,2n) reactions. HDPE surrounding the detectors is a moderator to allow thermal neutrons reactions with ^{3}He for (n,p) reactions in the gas.

Monte Carlo code specifics and Post Tally Processing

Geometry and physics were modeled with MCNPX[11] (version 27b). The ^{3}He were modeled using the F4 tally looking at the count rate averaged in each of the 4 detectors and summed (count rate (CR)/source neutron), the uncertainty in this tally is under 3%. Only one interrogation pulse of 0.9 s duration was modeled. Given this one pulse, the probability of a neutron being detected in a particular time interval per source neutron emitted was tallied from 1 until 300 seconds. Each tally bin was 1 second in duration. Given that the instrument functions by repeatedly pulsing and counting, it is necessary to sum probabilities to get the actual probability of detecting a neutron in any given bin. To get the tally in the 3 to 4 second interval for the case of continuously repeated bursts, one would add the probability for the 1 to 2 s. interval to that of the 3 to 4 s. interval. This progressive summation results

in a probability that keeps increasing up until a nearly steady state probability is reached as is illustrated in Figure 1. Once the summation of probabilities described above is completed, it is simply a matter of multiplying the chosen source intensity (10^{11} ns/sec) times the probability of any bin to get the counts in that bin. The total number of counts in an assay involves summing up the bins. The rate is obtained by dividing by the total count time.

A tally multiplier Fm4 in the fuel provides results concerning the neutron flux in the fuel and allows to determining C_1 and C_2 from Equations (3), below, and the uncertainty in this tally is under 0.6%. This tally gives several elements in two time bins, one from 0 to 1 s., which is considered composed of prompt neutrons and from 1 to 300 s. considered composed of the delayed neutron signal. To get C_1 and C_2 we use results from this tally that gives among other data the total number of fission neutrons in the first time bin for each of the 4 isotopes, i, of interest and allow us to determine their contribution, ^{239}Pu, ^{235}U, ^{241}Pu and ^{238}U as shown below:

$$Total\ \#\ of\ fission\ neutrons_i(MCNPX) = \iint_{E\,V} \Phi(\nu\sigma_f)_i\, dEdV \quad (2)$$

$$C_1 = \frac{A_{239}\beta_{235}\iint_{E\,V}\Phi(\nu\sigma_f)_{235}dEdV}{A_{235}\beta_{239}\iint_{E\,V}\Phi(\nu\sigma_f)_{239}dEdV} \quad and \quad C_2 = \frac{A_{239}\beta_{241}\iint_{E\,V}\Phi(\nu\sigma_f)_{241}dEdV}{A_{241}\beta_{239}\iint_{E\,V}\Phi(\nu\sigma_f)_{239}dEdV} \quad (3)$$

C_1 and C_2 being the weighting factors of Equation (1). The terms in equation (2) and (3) are the following: Φ is the flux in the fuel. Since multiplication in the fuel and absorption by poisons both alter the interrogating flux, multiplication and poison population both influence Φ. V is the volume of the fuel over which the measurement takes place and E is the energy of the neutrons energy spectrum. σ_f is the fission cross section and β is the average number of delayed neutrons emitted per fission as given in table 1. In this study all terms of this equation will be determined thanks to results from MCNPX calculations[10] apart from the atomic number and the fraction β used as given in table 1 and therefore as nuclear data and β will be determined more accurately through the code in future studies as a function of energy.
These two coefficients lead to the DN effective mass of ^{239}Pu. Multiplying the total number of fission neutrons by the fraction of delayed neutrons gives us the amount of delayed neutron contribution from each isotope. Moreover this tally is useful to determine a relative contribution from each fissile isotope and in addition the fertile ^{238}U contributing to the delayed neutron signal thus ^{238}U, ^{235}U, ^{241}Pu and ^{239}Pu contributions as shown in Equation (4) below:

$$DNCR_i = \beta_i\left(\iint_{E\,V}\Phi(\nu\sigma_f)_i dEdV\right) \times (Source\ Strength) \times Correction\ Factor \quad (4)$$

For a given assembly the efficiency, ε of the detectors is constant therefore isn't accounted in the correction factor of Equation (4). This Correction factor has been estimated as a first estimation as the ratio of the weight fraction (WF) of each isotope to the total weight fraction of the 4 isotopes quoted above as shown in Equation (5) below:

$$(WF)_i = \frac{\left(\frac{Mass_i}{Molar\ Mass_i}\right)\times N_A}{Volume} \quad (5)$$

In the future other ways to correct for the atom density of each density will be seek. Some of the isotopic masses are provided in table 2 and the other are referenced. The volume is the sum of the volume of each 264 fuel pins, each 193.16 cm^3. N_A is the Avogadro number.

Impact of Fission Fragments Absorption on Delayed Neutrons

As fuel burns in a reactor, fission fragments are produced. Some of these fission fragments have large thermal absorption cross sections. If all other factors remained constant, one might expect the delayed neutron count rate to be reduced by fission fragment absorption at elevated burnups as compared to lower burnups due to their accumulation. Or one might be concerned that fission fragment absorbers decay away or are being produced by the decay of a parent nucleus. For the cases researched, the impact of this isotopic trend is that the delayed neutron count rate appears to be relatively insensitive to fission fragment absorbers. Furthermore, their production and decay are smooth function of burnup and time, hence it is expected that they will be "absorbed" into a calibration factor. So far the two main conclusions are that 2 vary significantly with time, ^{155}Gd and ^{241}Am and 20 have a significant variation with burnup.

Delayed Neutron Signal as a function of Burnup & Cooling Time

The initial enrichment will be held constant to illustrate the variation of the delayed neutron count rate as a function of burnup and cooling time. In Figure 4 the delayed neutron count rate as a function of burnup (15, 30, 45 and 60 GWd/tU) and cooling time (1, 5, 20, 80 years) is illustrated for an assembly with an initial enrichment of 4%.

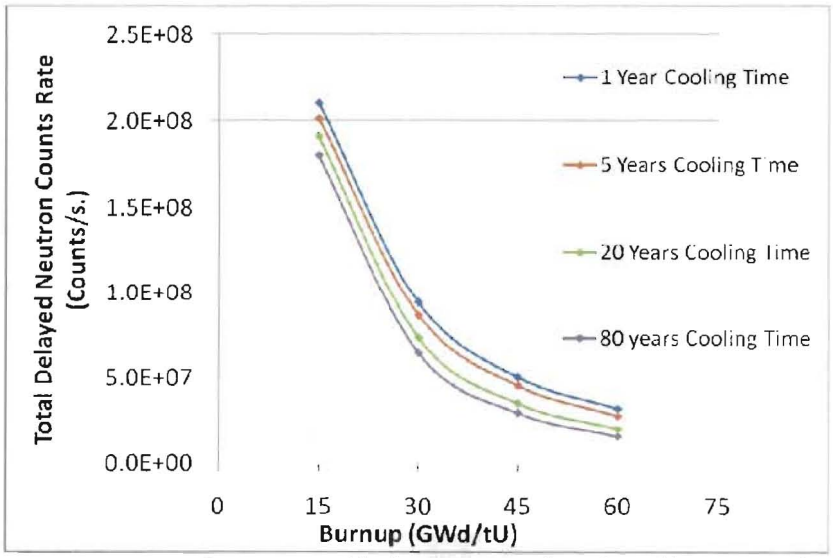

Figure 4: Delayed neutron counts rate for an assembly in water as a function of burnup and cooling time for a case when the fuel had an initial enrichment of 4%.

In Figure 4, the shape of variation in the delayed neutron count rate as a function of burnup indicates the dominance of ^{235}U as compared to the fissile isotopes of Pu. The change is delayed neutron count rate is more rapid at low burnup since ^{235}U is the dominant fissile isotope, and as such, it is being depleted at a faster rate. Later in time (when the burnup is higher) ^{239}Pu, ^{241}Pu and ^{235}U are being

consumed to maintain the power output of the reactor. Note that the percentage change in the delayed neutron count rate with cooling time at 15 GWd/tU is significantly smaller than at 60 GWd/tU. This factor would even be more pronounced if the contribution of ^{238}U were subtracted. The significant reason for this is the decay of ^{241}Pu. As the fuel burns up, the contribution of ^{241}Pu to the delayed neutron signal increases[14]. As a result, the loss of ^{241}Pu (14 year half-life) with time is more noticeable at high burnup.

Delayed Neutron Intensity as a Function of $^{239}Pu_{DNeffective}$

A key design criteria for a delayed neutron instrument for Spent Fuel is to minimize the fission of fertile isotopes ^{238}U so that the delayed neutron count rate will scale with the weighted mass of the fissile isotopes ^{235}U, ^{239}Pu, and ^{241}Pu. Equation (4) was used to calculate the $^{239}Pu_{DNeffective}$ using C_1 and C_2 from equation (6) for each assembly modeled in water. Φ is the interrogating flux, which depends on the number of neutrons produced by the interrogating source, the multiplication of those neutrons and the loss of those neutrons to absorbers. The flux is determined by MCNPX calculation from the tally multiplier together with the total number of fission neutrons and β is given in table 1. In Figure 5 below, the delayed neutron count rate is graphed as a function of "$^{239}Pu_{DNeffective}$" for each of 16 assemblies in the Spent Fuel library.

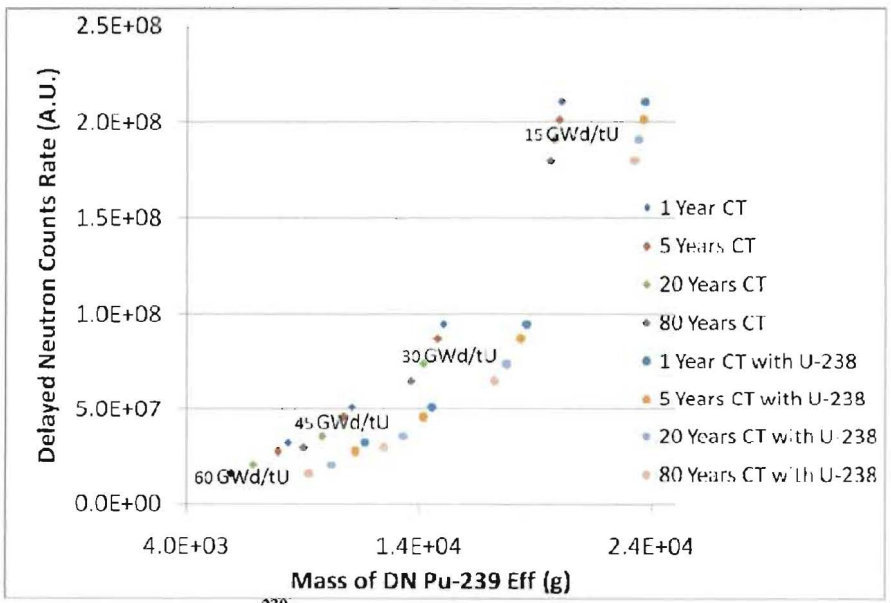

Figure 5: Delayed neutron count rate vs. $^{239}Pu_{DNeffective}$ from Equation (1) using Equations (2) and (3) and (9) when adding the ^{238}U contribution is illustrated for all 16 assemblies of the Spent Fuel library in water modeled for a constant IR, 4% as a function of BU (15, 30,45 and 60 GWd/tU) and CT (1,5,20 and 80 years) – Furthermore DN eff mass of ^{239}Pu has also been estimated including the contribution of ^{238}U as explained below.

As observed in Figure 5, a third coefficient C_3 has been estimated in order to quantify the DN effective mass of ^{239}Pu including ^{238}U contribution. C_3 has been calculated as C_1 and C_2, as a weighted factor of ^{238}U to ^{239}Pu. C_3 weights the mass of ^{238}U given in table 2 and $^{239}Pu_{DNeff}$ equation becomes:

With: (9)

It is worth emphasizing the isotopic diversity of each data point in Figure 5. Each data point represents an assembly with a wide range of actinides and fission fragments. The primary conclusion in Figure 5 is that $^{239}Pu_{DNEffective}$ scales in a uniform manner with the delayed neutron count rate with or without ^{238}U contribution. Furthermore we can quantify as explained in the post processing of the tally, the contribution of each isotope in the fuel.

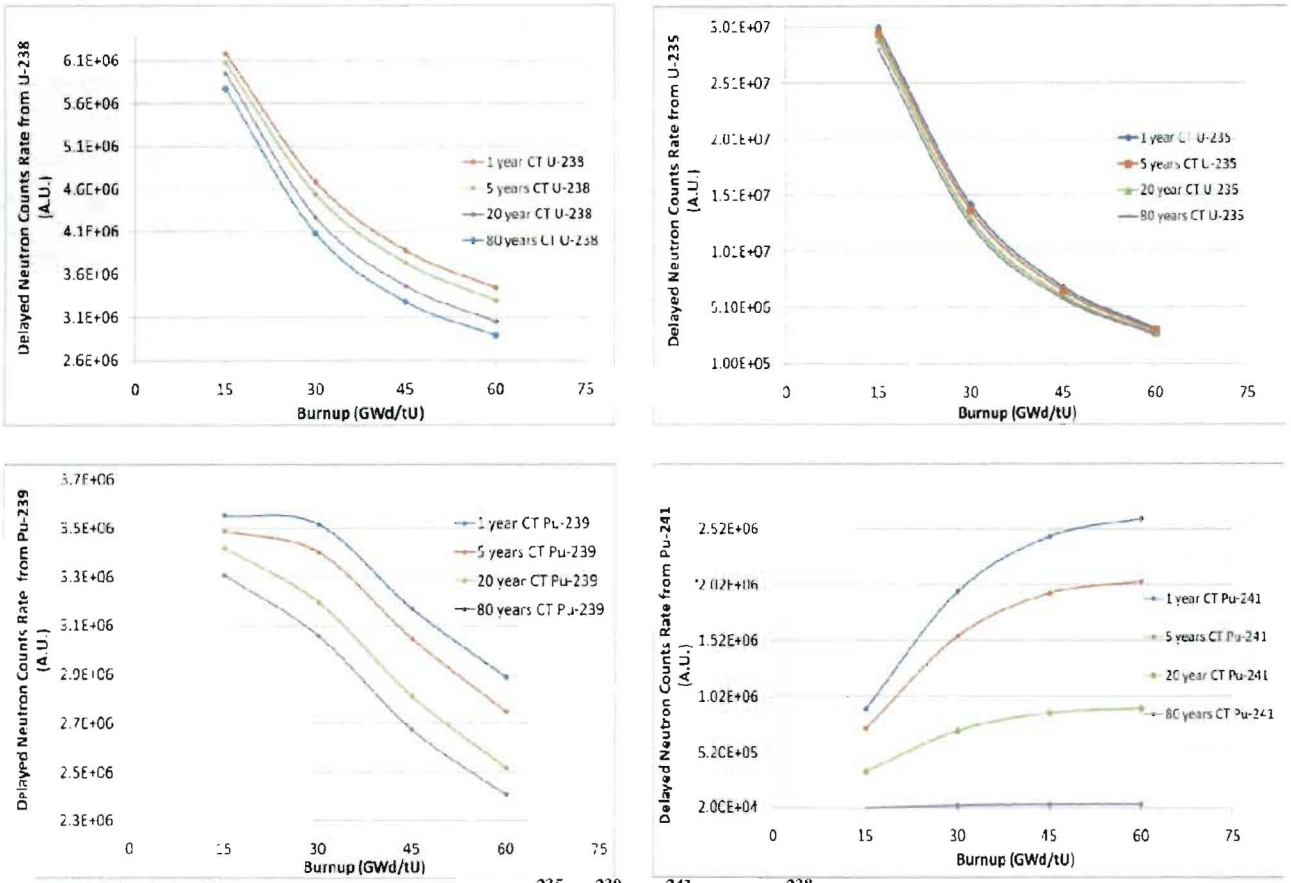

Figure 6: Delayed Neutron count rate of each, ^{235}U, ^{239}Pu, ^{241}Pu, and ^{238}U calculated in the fuel through Equation (4) as a function of all BU (15,30,45 and 60 GWd/tU), all CT (1,5,20 and 80 years) and for a constant IR of 4%.

The trend of ^{235}U follows exactly the trend of its population in a reactor. $1/4^{th}$ of the neutrons come from ^{238}U which doesn't represent a huge portion considering its amount (table 2) in the fuel assembly, thus this design seems to decrease fission in ^{238}U significantly. ^{238}U delayed neutrons decrease a lot as a function of burnup due to the increasing amount of poisons with burnup significantly absorbing neutrons. Also fissile isotopes population decrease with burnup as we can as well observe in the behavior of ^{235}U and ^{239}Pu in Figure 6 and therefore neutrons at the energy of the watt fission spectrum of induced fission neutrons inducing fission in ^{238}U is of smaller amount which with poisons explain the behavior of the signal coming from ^{238}U. Although ^{241}Pu population increases with burnup, its amount is negligible compared to the two other fissile isotopes amount.

We have determined through Equation (4) the delayed neutron counts rate from each of the 4 following isotopes, ^{235}U, ^{239}Pu, ^{241}Pu and ^{238}U, and making the assumption that the delayed neutron signal only comes from the fission products from fission in these isotopes, we have determined their own relative contribution to the total signal from the addition of their 4 $(DNCR)_i$.
Next we have used this relative contribution to make an approximation of the contribution of each isotopes in the total signal counted in the detectors (from tally 14) as we can observe it in Figures (7) and (8). Figure (7) and (8) represent the total delayed neutron counts rate from tally F4 divided by each isotope contribution respectively for an assembly of 4% IR and 60 GWd/tU as a function of cooling time and for an assembly of 4% IR and 5 years CT as a function of burnup.

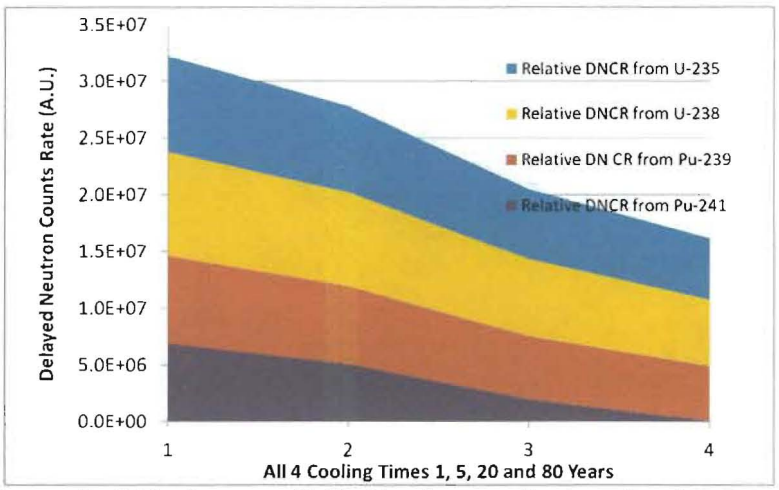

Figure 7: Total DNCR from tally F4 in detectors divided by the contribution of each isotope as an approximation of the relative contribution $(DNCR)_i$ from Equation (4) of each isotope to the total DNCR from Equation (4) of all 4 isotopes as a function of CT, 4% IR and 60 GWd/tU.

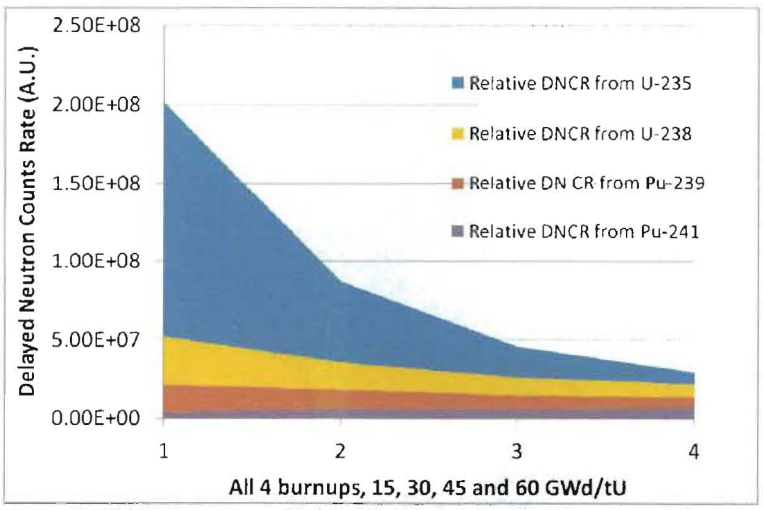

Figure 8: Total DNCR from tally F4 in detectors divided by the contribution of each isotope as an approximation of the relative contribution $(DNCR)_i$ from Equation (4) of each isotope to the total DNCR from Equation (4) of all 4 isotopes as a function of BU, 4% IR and 5 years CT.

Let's point out that in the graph of Figure (7) that concerning the X axis is linear approximation of the cooling time has been made where the units 1, 2, 3 and 4 represent the 4 cooling times respectively 1, 5,

20 and 80 years and in the graph of Figure 8, the X units 1, 2, 3 and 4 represent the 4 burnups, 15, 30, 45 and 60 GWd/tU.

Graphs from Figure (7) and (8) gives a good approximation of the contribution of each isotope but we have to keep in mind that we determine these contributions through a tally Fm4 that measures the flux in the flux and thus doesn't take in consideration the efficiency in the detectors from tally F4 and also these graphs do not account for the contribution of other isotopes that would contribute to the delayed neutron signal. In future studies these parameters will be deeply looked at.
Conclusions of these two last graphs are that as a matter of fact the prominent contribution as a function of burnup is from ^{235}U and not as prominently from ^{238}U as a function of cooling time.

In summary the DN instrument to quantify the fissile content in a spent fuel assembly considering the described design for an integrated system with a DDA instrument. In the future statistical study and a larger amount of assemblies will be performed.

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