LA-UR- 09-01188

Approved for public release;
distribution is unlimited.

Title:	A Monte Carlo Linked Depletion Spent Fuel Library for Assessing Varied Nondestructive Assay Techniques for Nuclear Safeguards
Author(s):	Michael L. Fensin (N-4), Steven J. Tobin (N-4), Nathan P. Sandoval (N-4), Scott J. Thompson (N-4), Martyn T. Swinhoe (N-1)
Intended for:	Advances in Nuclear Fuel Management IV

Los Alamos
NATIONAL LABORATORY
EST.1943

Los Alamos National Laboratory, an affirmative action/equal opportunity employer, is operated by the Los Alamos National Security, LLC for the National Nuclear Security Administration of the U.S. Department of Energy under contract DE-AC52-06NA25396. By acceptance of this article, the publisher recognizes that the U.S. Government retains a nonexclusive, royalty-free license to publish or reproduce the published form of this contribution, or to allow others to do so, for U.S. Government purposes. Los Alamos National Laboratory requests that the publisher identify this article as work performed under the auspices of the U.S. Department of Energy. Los Alamos National Laboratory strongly supports academic freedom and a researcher's right to publish; as an institution, however, the Laboratory does not endorse the viewpoint of a publication or guarantee its technical correctness.

Form 836 (7/06)

Advances in Nuclear Fuel Management IV (ANFM 2009)
Hilton Head Island, South Carolina, USA, April 12-15, 2009, on CD-ROM, American Nuclear Society, LaGrange Park, IL (2009)

A MONTE CARLO LINKED DEPLETION SPENT FUEL LIBRARY FOR ASSESSING VARIED NONDESTRUCTIVE ASSAY TECHNIQUES FOR NUCLEAR SAFEGUARDS

Michael L. Fensin, Stephen J. Tobin, Nathan P. Sandoval,
Martyn T. Swinhoe, Scott J. Thompson
Los Alamos National Laboratory
MS: E540, P. O. Box 1663, Los Alamos, NM 87545
mfensin@lanl.gov; tobin@lanl.gov; nathans@lanl.gov;
swinhoe@lanl.gov; scottt@lanl.gov

Keywords: Nonproliferation; MCNPX; Burnup; Safeguards

ABSTRACT

The objective of nuclear safeguards is the timely detection of the diversion of significant quantities of nuclear materials, and the deterrence of such diversion by the risk of early detection. In order to achieve this goal, one must be able to characterize the elemental quantity and isotopic ratios of actinide-containing nuclear materials in various components of the nuclear fuel cycle. Many nondestructive assay (NDA) techniques exist; however, no single NDA technique can, in isolation, quantify elemental plutonium in spent fuel. A research effort has been undertaken to determine the best integrated combination of cost effective strategies for fully characterizing elemental and isotopic content of plutonium and other actinides of interest in spent fuel. This paper briefly outlines the integrated process for assessing NDA techniques, specifically focuses on the development of the modeling approach for the burnup/enrichment dependent library and addresses a significant MCNPX 2.7.A enhancement for speeding up depletion calculations involving large amounts of burn materials.

1. INTRODUCTION

The objective of nuclear safeguards is the timely detection of diversion of significant quantities of nuclear materials, and the deterrence of such diversion by the risk of early detection.[1] In order to achieve this goal, one must be able to determine the elemental quantity and isotopic ratios of Pu and other actinides of interest, in nuclear materials for various components of the nuclear fuel cycle. Spent fuel, from the moment the reactor pressure vessel is opened to the ultimate reprocessing or disposal at a high level waste repository, offers a diversion potential of significant quantities of actinide-containing nuclear material. Therefore elemental and isotopic characterization of spent fuel, at the shipper site (i.e. power plant) and at the receiver site (i.e. reprocessing site or high level waste repository), is extraordinarily important to detecting, in a timely fashion, the diversion of actinide-containing nuclear material. There are also a variety of other motivations for characterizing elemental and isotopic content of spent fuel including the following: strengthening the capabilities of the International Atomic Energy Agency's (IAEA) ability to safeguard nuclear facilities, input accountability at reprocessing

facilities, optimizing the location of partially burned assemblies in the core and burnup credit at repositories.

The most cost efficient way to assay the plethora of spent fuel assemblies at any given site, in order to meet the objectives of nuclear safeguards, is through the use of nondestructive assay (NDA) techniques. At first, one might think that Cerenkov detection would seem to be the simplest NDA strategy for determining spent fuel tampering.[2] This strategy relies upon the IAEA inspector visually assessing the Cerenkov glow created by electrons, generated from fission product gammas. The inspector can, if sufficient measurements are taken, gain information on assembly burnup, cooling time, and tampering based on the characteristics of the Cerenkov glow. However, because the measurement relies on visual inspection with inspector interpretation (for digital systems with image subtraction the user must still position a hand-held instrument), different levels of experience, from inspector to inspector, or even measurement time to next measurement time, may not be sufficient for the future of nuclear safeguards inspections. Destructive analysis techniques are also not conducive to cost-efficient monitoring as they are both expensive and time consuming. NDA techniques, which can give isotopic information, are most conducive to standardized measurement at more stages of the nuclear fuel cycle because the techniques do gain information regarding the quantity of interest, Pu, and do not rely on visual interpretation that may be misinterpreted between measurements, or intrusive assembly destruction that results in costly loss and destruction of useful material.

NDA strategies have historically been employed at fuel fabrication facilities for fuel specification testing; however, NDA strategies are more complicated when examining spent fuel. For example, for fresh fuel pin assay one can use passive gamma ray detection where the emission of the 186-keV gamma ray is generally used for U-235 assay and the 384-keV complex is generally used for assaying plutonium.[2] In spent nuclear fuel, these signatures fade into the noise as the Compton continuum created from the fission products dominates the signal in this energy regime. Passive gamma ray measurements are also not conducive to entire assembly assay due to the significant attenuation of the inner fuel pin signature so that only the outer pins are seen by the detection equipment. Another example of an assay strategy complicated by spent fuel measurement is the coincidence fuel rod scanner. The coincidence fuel rod scanner is basically an active neutron coincidence counter used to measure fissile content through the detection of coincident counts from fission induced events.[2] When only one fissile species is present in the fuel matrix, the assay strategy is capable of quantifying the content of that fissile species. For example, in commercial UO_2 the counter can be used to effectively determine U-235 enrichment because U-235 is the only fissile species in the fuel matrix; however, spent fuel contains a variety of fissile species and therefore trying to isolate the content of any one fissile species can become very difficult.

Many nondestructive assay (NDA) techniques exist for spent fuel measurement; however, no single NDA technique can, in isolation, quantify elemental plutonium and other actinides of interest in spent fuel.[3, 4] Therefore, a study has been undertaken to determine the best integrated combination of cost effective strategies for characterizing the elemental and isotopic content of interest from spent fuel for nuclear safeguards.[3] This

study seeks to examine the merits and faults of 12 different detection techniques in order to down-select to a few necessary, bare minimum, detection techniques for an integrated spent fuel assay strategy. The 12 NDA techniques being researched include the following: Delayed Gamma, Delayed Neutrons, Differential Die-Away, Lead Slowing Down Spectrometry, Neutron Multiplicity, Nuclear Resonance Fluorescence, Passive Prompt Gamma, Passive Neutron Albedo Reactivity, Self-integration Neutron Resonance Densitometry, Total Neutron (Gross Neutron), X-Ray Fluorescence, ^{252}Cf Interrogation with Prompt Neutron Detection.[4] These techniques rely on the detection of either gamma or neutron emission from either induced or passive events occurring in the spent fuel, and therefore each technique has advantages and disadvantages for the nuclear safeguards objective of Pu assay in spent fuel.

To see which instruments can be integrated to quantify elemental Pu and other actinides of interest during the design down-select, a process has been developed for generating appropriate models for assessing the strengths and weaknesses of each NDA technique. The process involves a standardized coupling for the generation/automation of burnup/enrichment dependent fuel assembly library, development of diversion scenarios, and engineering of detector models for each NDA technique to be examined. This paper briefly outlines the integrated process for assessing NDA techniques, specifically focuses on the development of the modeling approach for the burnup/enrichment dependent library and reports spatially dependent isotopic ratios of importance to reactor safeguards NDA techniques.

2. INTEGRATED PROCESS

The down-select process for determining the optimum combination of cost effective NDA strategies, from the 12 assessed spent fuel NDA techniques, involves many varied types of calculations. First, a spatially dependent burnup library must be generated in order to create a basis for comparing each detection strategy. The burnup library is implemented as the source term for the fixed source detector calculations used to compare each detection system. In tandem with the generation of this library, detector models were developed and engineered to generally quantify the capability of each NDA technique for the assessment process. Little effort was put into optimizing any of the NDA techniques. Once each detector design has been engineered, fuel pin diversion scenarios, which mimic both what is "expected" to be diverted and also what is expected to the worst case scenario, must be generated and analyzed. Using the combination of the constructed high fidelity source term, engineered detector models, and developed diversion scenarios, fixed source detector calculations must be completed to assess the merits and faults of each technique for determining plutonium content in the fuel assembly.

MCNPX 2.7.A was chosen as the modeling tool for the assessment effort because of the code's ability to adequately model multi-particle physics necessary for detector calculations, achieve high spatial resolution without concern for spatial self-shielding approaches, and minimize assumptions necessary for standardized code-to-code coupling.[5] Thousands of output and input files must be managed for the analysis due to the vast variety of calculations required for the assessment. To automate and standardize

the managing of data transitions from one calculation to another, a graphical user interface (GUI) has been developed. The Burnup Automation MCNPX File Data Retrieval Tool (BAMF-DRT) is a VBA-Excel constructed tool used to automate the coupled process of extracting burnup data from MCNPX burnup output files to create MCNPX fixed source detector input files, with and without pin diversion, for the detector assessments.[6] BAMF-DRT completes the following tasks: (1) BAMF-DRT reads an MCNPX burnup output file to ascertain available burnups and cooling times containing nuclide buildup information; (2) The available burnups and cooling times are loaded into an easy-to-use list box so that the user may then select time steps and cooling times to generate fixed source input files; (3) Once the user selects burnups and cooling times, the program then proceeds to extract isotope concentrations for each selected time step and cooling time, loading the concentrations as material cards into subsequent fixed source detector calculation models; (4) In tandem the user also has the ability to manually create fuel pin diversion scenarios using various diversion material substitutions; (5) The user then chooses a detector to wrap around the fuel assembly, and the final fixed source detector file is generated. The user also has the ability to manipulate detector size, change cross section extensions, and develop a specific diversion scenario type in order to determine the limits of a given detection strategy.

BAMF-DRT uses visual control boxes with easy-to-use dropdown menus, list boxes and push-button execution in order to automate and standardize the coupling of data transitions from burnup output to fix source detector calculation for the abundance of calculation inputs to be generated. Figs. 1a-e display examples of the visual displays encountered in typical use of BAMF-DRT. BAMF-DRT is accessed within the BAMF-DRT excel workbook through an easy-to-access push button as displayed in Fig. 1a. The user is then able to access a MCNPX burnup output file, from any directory, using the control box displayed in Fig. 1b. BAMF-DRT then loads all available burn times/burnups and cooling times for generating fixed source inputs as displayed in Fig. 1c. The user then has the option to generate diversion scenarios using a visual diversion map. Fig. 1d displays an example visual diversion map. The user may then choose to replace the fuel containing cells within each fuel universe (red boxes) with any of the available fuel diversion materials. This task is completed by highlighting the cells in which a diversion is to be made and then clicking on the diversion material of choice to make a diversion. The user has the ability to select as many different types of diversion materials for a given diversion scenario, and also has the ability to manipulate, or accept the default, density and constituents of each diversion material. Finally, the user is taken to the Detector Options control box, Fig. 1e, where the user then selects a detector to wrap around the geometry to generate a complete fixed source detector input file. BAMF-DRT is a powerful tool that streamlines the process for assessing the effectiveness of a given technique by minimizing time spent generating input files and transitioning data therefore maximizing the time spent actually analyzing data.

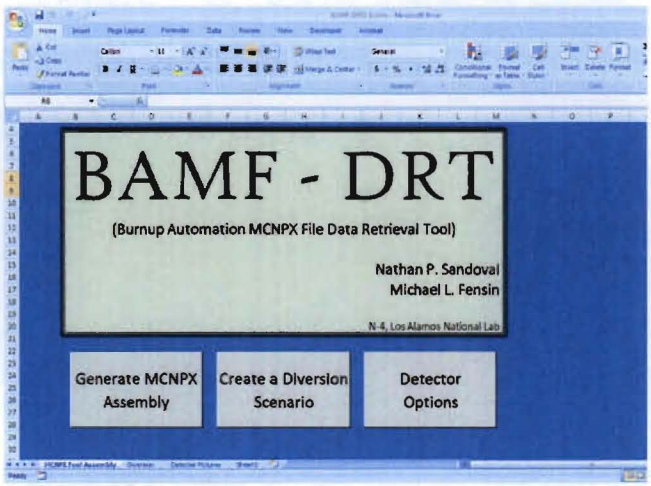

Fig. 1a BAMF-DRT start page.

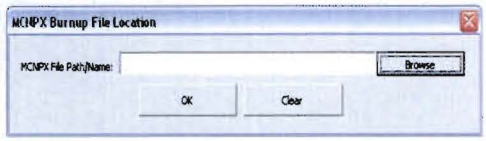

Fig. 1b Burnup file locator.

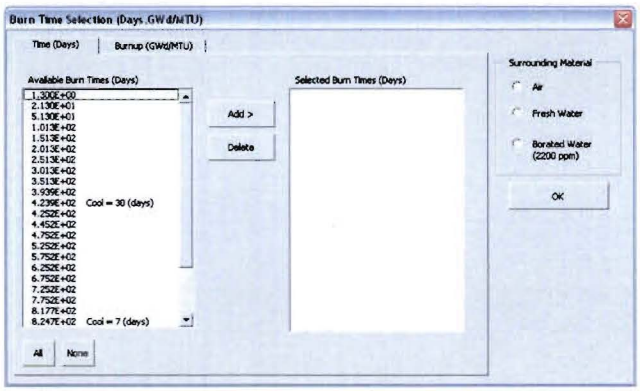

Fig. 1c Available burnup/cooling times menu.

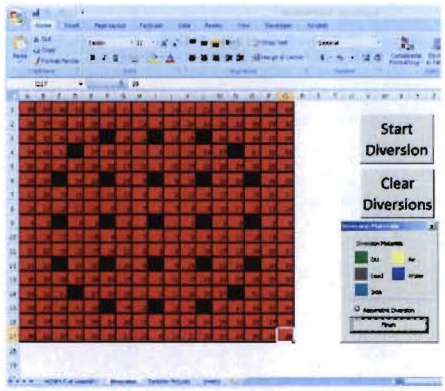

Fig. 1d Diversion map generator.

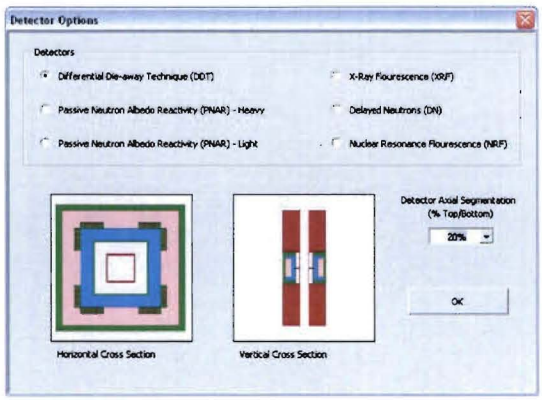

Fig. 1e Detector options menu.

3. BURNUP LIBRARY

To standardize the assessment of each NDA technique for spent fuel assay, a source term library has been developed. For spent fuel assay, IAEA inspectors will encounter a vast array of fuel assemblies containing varied initial enrichments, burnups, and cooling times. The inspector's responsibilities will include accounting for the special nuclear material (SNM) quantity of each fuel assembly and determining if fuel tampering has occurred.[7] One of the ways an inspector could achieve this objective is by monitoring the radiation signature of the fuel assembly. The radiation signature of a spent fuel assembly will depend on the assembly type, reactor operation characteristics, initial enrichment, burnup, cooling time, and diversion scenario. The case matrix addressing all the mentioned variables, for every reactor type in the world, would be too large to manage within the context of the current assessment; therefore a base burnup/enrichment/cooling time dependent library has been generated, based on pressurized water reactor (PWR) spent fuel, to be implemented as a source term for assessing each NDA detection strategy. The source term library consists of a typical 17 X 17 PWR spent fuel assembly burned at varied burnups, with typical initial enrichments, consistent with the range of PWR spent fuel available today. Many cooling times, consistent with typical spent fuel measurements times, are also available in the library.

Table 1 Generic Westinghouse 17 X 17 assembly parameters

Parameter	Data
Assembly general data	
Lattice	17× 17
Number of fuel rods	264
Number of guide tubes	24
Number of instrument tubes	1
Fuel rod data	
Type of fuel pellet	UO_2 (10.4538 g/cc)
Rod pitch	1.26 cm
Clad thickness	0.065 cm (no gap between fuel and clad)
Pellet diameter	0.410 cm
Active fuel length	365.76 cm
Fuel temperature	900 K
Clad temperature	620 K
Clad material	Zircaloy-4 (5.8736 g/cc)
Guide and Instrument tube data	
Inner radius	0.571 cm
Outer radius	0.613 cm
Material	Zircaloy-4 (5.8736 g/cc)

Due to the large 6.67 eV resonance of U-238, the mean free path, within the fuel pellet, of thermalizing neutrons around this energy is very small. This phenomenon results in a significant plutonium concentration gradient between the pellet surface and center. A significant plutonium density gradient is generated within the first 200 microns

of the fuel pellet surface, and therefore the density gradient of other higher actinides follows this trend as well. Nuclear safeguards based detection relies upon a variety of signatures in order characterize the spent fuel signal; therefore understanding how this spatial gradient of actinide buildup affects the emission signature is paramount to understanding the limitations of certain detection equipment. Of particular note in terms of the radial distribution of plutonium is X-ray fluorescence for which the detected signal primarily comes from the outer 200 microns of the fuel. Continuous energy Monte-Carlo linked depletion is an excellent candidate for addressing burnup dependent actinide buildup spatial gradients as it can easily segment the geometry into as many spatial zones as memory permits without concerns of spatial self-shielding treatment. The MCNPX 2.7.A depletion capability has been successfully benchmarked for LWR calculations and therefore the computational system was therefore chosen to generate the initial enrichment/burnup/cooling time dependent PWR spent fuel library.[8]

The simulations implemented a generic 17 X 17 Westinghouse infinitely reflected assembly. Though core operating strategies, such as bundle placement affecting bundle to bundle leakage, does affect the ultimate buildup and depletion of nuclides, this study chose to only examine a generic average bundle and therefore infinite reflection was deemed to be an acceptable boundary condition. The general assembly parameters are listed in Table 1. The UO_2 fuel pellets were encased in Zircaloy-4 cladding. The coolant was chosen to be 660 ppm average boron concentration at a density of 0.7245 g/cc and temperature of 575 K. In order to maximize the spatial resolution per fuel pin, the fuel assembly was modeled using repeated structures with eighth-assembly symmetry. Fig. 2 displays an axial slice of the modeled fuel assembly. Using eighth-assembly symmetry results in 39 repeated fuel pin regions (diagonal rows of pins are shared amongst the eighth-assembly divisions).

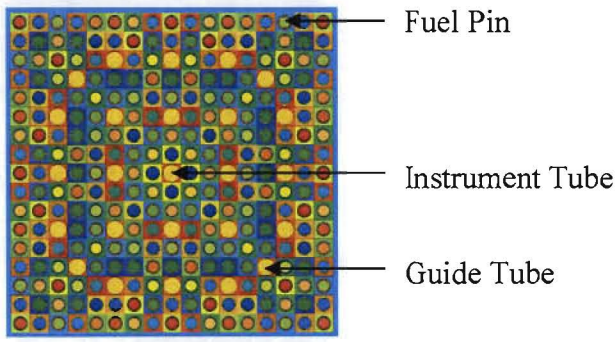

Fig. 2 Axial slice of generic Westinghouse 17 X 17 PWR assembly.

In MCNPX depletion, all the tracked fission and transmutation products for a specific burn material are automatically placed in the burn material cards at the start of the calculation. This procedure involves allocating space for arrays required for transport and isotope buildup tracking. Multi-processor calculations can be run successfully if the total amount of space allocated for a calculation, buffer + array sizes, does not exceed the RAM limit of the computer. For large burnup problems, this task involves allocating hundreds of megabytes of space for the mentioned information and therefore limits the ability to run calculations in multiprocessor mode which is necessary to enhance

computational performance in Monte Carlo calculations. Preliminary calculations examined combinations of compiler options and versions of MPI in order to determine the maximum amount of burn materials, and thus maximum allowable divisions per fuel pellet, that could be examined per 1GB of RAM. The INTEL 9.1 and Portland Group 5.3 FORTRAN90 compilers and both the OPENMPI and MPICH1 multiprocessor standards were examined in different combinations. Examination determined that a combination of the Portland Group 5.3 compiler using the MPICH1 standard resulted in a maximum of 156 burn materials with 40 burn steps per 1 GB of RAM. It is important to realize that a tradeoff exists between amount of materials analyzed and the amount of burn steps as arrays are dimensioned by the product of these quantities. Dividing 156 allowed burn materials by the 39 fuel pin regions results in 4 radial subdivisions per fuel pellet. The gradient in plutonium density is most significant within the first 200 microns of the fuel pellet, after the first 200 microns the plutonium density if roughly constant. The radii of the fuel regions were chosen to be 0.3900, 0.4020, 0.4075 and 0.4100 cm. This division would result in regions containing varied densities of plutonium and minor actinides that would best represent the spent fuel signal.

Table 2 Case matrix for the burnup library.

U-235 Enrichment	Burnup (GWD/MTU)	Available Cooling Times
2%, 3%, 4%, 5%	15, 30, 45, 60	1 and 4 weeks, 1, 2, 5, 20, and 80 years

The case matrix for the generated assemblies is listed in Table 2. This case matrix encompasses the range of enrichments, burnups, and cooling times to be used in the spent fuel assessment work. The assembly power was assumed to be 17.896 MW which is similar to the power used in the OECD/NEA IVB benchmark (this benchmark looked at a typical 17 X 17 Westinghouse PWR for MOX applications, and the power history was supposed to represent that of a typical assembly in a PWR).[9] In order to enhance nuclide prediction, short time steps were taken at the beginning of each cycle in order to account for equilibrium buildup of Xe and Sm fission product poisons. After equilibrium Xe and Sm is achieved, longer time steps, ~ 2 GWD/MTU were used. The time durations per time step per cycle are given in Table 3. Each KCODE time step utilized 10000 particles per cycle, skipping the first 25 cycles for 155 cycles.

Table 3 Time durations per time step per cycle.

Cycle	**Time Durations (days)**
1	1.3 30 65 70 80 80 67 27 30*
2	1.3 30 57 70 70 71 67 54 30*
3	1.3 20 62 67 67 62 60 81 30*
4	1.3 20 62 62 62 62 43
Cooling Times	7 21 337.25 1095.75 5478.75 21915

* Represents downtime

Some type of neutron based counting system will be required for an integrated spent fuel assay strategy. Unlike the gamma signal, the neutron signal is not as heavily attenuated and therefore can be used to assay a larger portion of the fuel assembly such as inner pins. For passive assay, not considering multiplication contributions, the passive neutron signal is primarily from the Cm isotopes Cm-242 and Cm-244.[10] For 5 year cooling times and higher burnup (>30 GWD/MTU), excluding multiplication, the neutron signal is largely dominated by Cm-244. One can determine the Cm-244/Pu-total ratio from a burnup code, or destructive analysis, and divide this ratio by the measured count rate, Cm-244 dominated, in order to determine total Pu.[11] For methodologies that do not rely on this method for ascertaining Pu concentration, understanding the Cm-244 signature versus enrichment, burnup, and cooling time is also important for determining how the source term affects the counting technique.

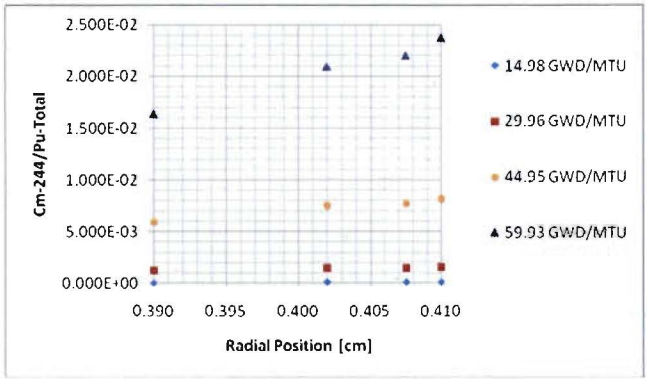

Fig. 3a Cm-244/Pu-total ratio for a corner pin of a 4% enriched fuel 17 X 17 generic PWR fuel assembly.

Figs. 3a-d displays the Cm-244/Pu-total ratio calculated results versus various dependencies. Fig. 3a displays the burnup dependent spatial gradient of the Cm-244/Pu-total ratio for a corner pin of a 4% enriched fuel assembly. Because Cm concentration increases as a function of burnup, at 59.93 GWD/MTU the ratio of Cm-244 to total Pu buildup is many orders of magnitude larger than at 14.98 GWD/MTU. The slope of the ratio as a function of radial position is actually constant as a function of burnup. Fig. 3b shows the Cm-244/Pu-total for an entire fuel assembly at 30 GWD/MTU and 4% enrichment for varied cooling times. For the most part, the Cm-44/Pu-total ratio decreases as a function of the combination of the Cm-244 and Pu-241 half-lives (Cm-244 half-life is 18.1 years and Pu-241 half-life is 14.4 years). Therefore as the Cm-244 decays, decreasing the ratio, Pu-241 also decays, slightly increasing the ratio from what is expected from pure Cm-244 decay; however, because the initial percentage of Pu-241, relative to total Pu, in the burned assembly is small, the deviation from pure Cm-244 decay is not very significant. Fig. 3c displays the Cm-244/Pu-total for an entire fuel assembly at 30 GWD/MTU and 1 year cooling for varied enrichments. Because each fuel assembly was burned assuming the same power, more transmutation, as opposed to fission, occurs in the lower enrichment fuel assemblies leading to a larger buildup of Cm-244. Fig. 3d displays the Cm-244/Pu-total for an entire fuel assembly at 4% and 5 year cooling for varied burnups. As burnup increases, transmutation increases leading to a larger buildup of Cm-244; furthermore, at ~30 GWD/MTU Pu-239 buildup comes into

equilibrium, other Pu isotopes come into equilibrium at much higher burnups; however, Cm-244 will not come into equilibrium until all the other Pu isotopes have come into equilibrium leading to a larger buildup of Cm-244 as compared to total Pu as a function of burnup.

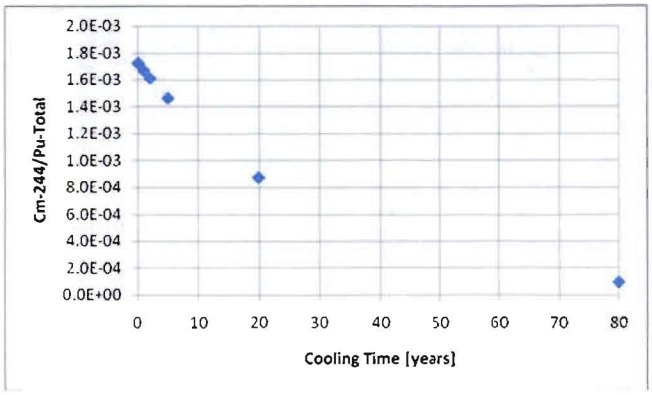

Fig. 3b Cm-244/Pu-total ratio for an entire fuel assembly at 4% enriched and 30 GWD/MTU.

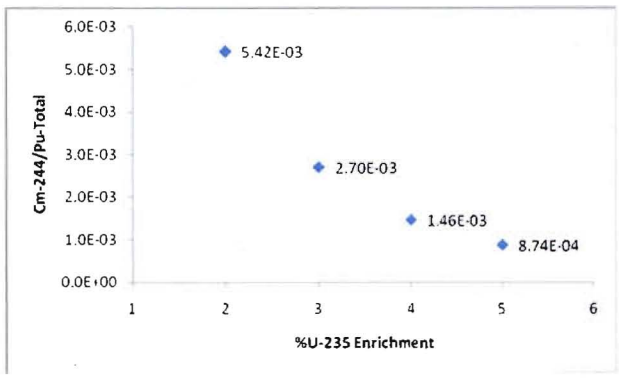

Fig. 3c Cm-244/Pu-total ratio for an entire fuel assembly at 30 GWD/MTU and 1 year cooling for varied enrichments.

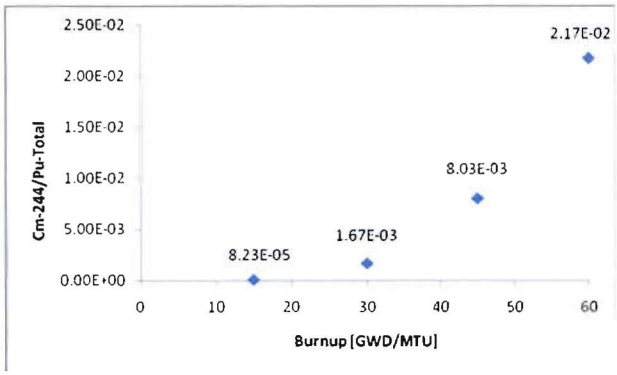

Fig. 3d Cm-244/Pu-total ratio for an entire fuel assembly at 4% enriched and 5 year cooling time for varied burnups.

Because the Cm-244/Pu-total ratios in Figs. 3a-d display significant dependencies upon enrichment, burnup, and cooling time, using the Cm-244/Pu-total ratio to calculate total Pu will also require knowledge of enrichment, burnup, and cooling time. Surely this knowledge may be attainable from the plant operator; however, it is the IAEA inspector's responsibility to suspect operator-reported information and be able to independently verify its validity. Therefore in assessing spent fuel to determine total Pu, a combination of many types of NDA techniques that can help ascertain this needed information will be required.

4. PARALLELIZING CINDER90 INTERFACE

Initial calculations demonstrated that a significant computational cost was incurred in the interface routines linking MCNPX and CINDER90. Several independent serial looping structures within the burnup interface prepare the depletion coefficients and isotopic information into a format suitable for CINDER90 execution. CINDER90 is then executed by a serial looping structure over each burn material. The computational expense associated for this serial execution increases nonlinearly with the number of burn materials. For example, burning a single material may only require one minute of CINDER90 interface execution time; however, burning 150 materials may require 300 minutes.

Efficient parallelization results from being able to efficiently execute independent processes on separate processors with minimal communication between processors. Because the depletion of each burn material is an independent calculation, the computational cost associated with the CINDER90 interface execution was remedied by parallelizing the CINDER90 interface routines for multi-processor execution. Each processor executes the CINDER90 interface routines for a subset of burn materials and then rendezvous at the end of the CINDER90 interface routines to determine total system burnup as well as atom densities for the subsequent reaction-rate calculation. Because the CINDER90 interface routines are independent for each burn material and minimal communication across processors is necessary to execute the CINDER90 interface routines, the computational enhancement scales almost linearly with the number of processors. This feature is now available in MCNPX 2.7.A.[5]

5. CONCLUSIONS

In order to better assess the detection of diversion of significant quantities of nuclear materials, a research effort is underway to determine the best integrated combination of cost effective NDA strategies for assaying spent nuclear fuel. In this paper we have outlined our integrated/automated approach, using BAMF-DRT, for managing the large quantities of data transitions necessary for assessing the different detection strategies for varied burnups, enrichments, cooling times, and diversion scenarios. The BAMF-DRT automation program is a powerful tool that streamlines the process for assessing the effectiveness of a given technique by minimizing time spent generating input files and transitioning data therefore maximizing the time spent actually analyzing data. We have also specifically described the source term burnup/enrichment dependent library used for our assessment. The Portland Group 5.3 compiler using the

MPICH standard resulted in a maximum of 156 burn materials with 40 burn steps per 1 GB of RAM, and therefore using eight-assembly symmetry in a 17 X 17 Westinghouse based lattice we were able to use 4 radial segment zones per each fuel region. Attributes of the burnup, enrichment, and cooling time dependence of the Cm-244/Total-Pu ratio have also been described further exemplifying the necessity for an integrated approach for assaying Pu content in spent fuel. The multi-processor implementation of the MCNPX/CINDER interface routines in MCNPX 2.7.A, greatly reduced the computational burden associated with the large burn material calculations necessary for the library generation. This integrated process and spent fuel source term library offers a significant mechanism for generating appropriate models for assessing the strengths and weaknesses of NDA techniques in order to down select to an ultimate optimized and cost effective solution for safeguarding spent nuclear fuel. This study is currently limited to generic PWR assemblies incorporating axially homogenous geometries using MCNPX 2.7.A depletion. More assembly types, of various reactor designs, with axial segmentation, may be examined in future studies.

5. FUTURE EFFORTS

Future efforts for this study are classified into two separate efforts: short term or long term. Short term efforts focus on the immediate down select to a few NDA techniques to develop a spent fuel detection strategy based on the spent fuel library mentioned in this paper. Long term efforts involve addressing broader issues related to the plethora of different types of fuel designs and reactor types available around the world.

The immediate short term efforts involve generating detector models for each of the 12 NDA techniques, developing fuel pin diversion scenarios, and determining the objective functions for assessing which techniques to combine into an ultimate integrated spent fuel assessment strategy. As previously mentioned, the strengths and weaknesses of 12 NDA techniques will be assessed. Currently, we have developed detector models for about 1/3 of the techniques and therefore models will have to be developed for the rest of the techniques. Developing models of each technique is not solely based on geometry development and detection efficiency optimization but also includes developing variance reduction strategies, such as time, energy, and spatial biasing, in order to improve statistical uncertainty in tallying for rare events such as delayed particle emissions. A standardized fuel pin diversion library is also under development and is the topic of another paper.[12] This diversion library includes both partial and gross diversions, using primarily depleted uranium fuel pins, within different regions of the fuel assembly, in order to best assess the limits of each detection strategy. The diversions were generated in such a way as to compliment the limitations of a typical Cerenkov field detector. The diversion library itself was developed based on "inverse radiation transport calculations," completed using the tally tagging feature available in MCNPX 2.7.A, from passive emission of a fuel assembly at varied burnups.

The ultimate objective functions for assessing each technique have yet to be completely defined; however, we have quantified how Pu and Cm content vary inside a fuel pin and other practical concerns for spent fuel measurements. We know that for each

technique we will need to assess the ability of that technique to determine burnup, initial enrichment, and cooling time. Looking forward we need to quantify the (alpha,n) signal and other background signals to assess performance of each of the techniques. For each technique we will also be developing criteria for assessing measurement limitations based on enrichment, burnup, cooling time, and diversion path way leading to an extremely data rich analysis for the entire project. Therefore we will need to develop data minimization techniques for assessing these criteria, and developing visualization strategies to assist in the final decision making process.

The spent fuel library mentioned in this paper only assesses a basic Westinghouse PWR 17 X 17 geometry with radial segmentation in the fuel pellet region and axial homogenization. Axial homogenization of the burnup is not true reality as a flux shape is invoked, axially across the fuel assembly, from the boundary conditions of the fuel height and moderator density change due to heat up. In a boiling water reactor (BWR), many pins within the fuel assembly will have separate enrichments leading significant radial heterogeneity of plutonium buildup. Significant axial plutonium heterogeneity will also exist because BWRs preferentially alter the boiling length during the cycle in order to develop plutonium to be burnt late cycle to increase cycle length. Other reactor types also have their own operational idiosyncrasies that will complicate the monitoring process. This would lead one to believe that 3-d scanning considerations should be evaluated in the future. However, for the purpose of the down select process, the Westinghouse PWR 17 X 17 geometry is suitable to demonstrate base weaknesses and strengths of each technique. The base down-select will seek to determine an optimum integrated NDA strategy that allows the inspector the best possible method for assaying spent fuel, and the concerns of operational scheme peculiarities of each reactor type will be evaluated in the future.

ACKNOWLEDGMENTS

The authors would like to thank all three of the sections of the Office of Nonproliferation and International Security (NA-24) of the National Nuclear Security Administration of the US DOE as well as the Safeguards Campaign of the Office of Nuclear Energy at US DOE for sponsoring this work.

REFERENCES

1. IAEA, INFCIRC/153 (Corrected), "The Structure and Content of Agreements Between the Agency and States Required in Connection with the Treaty on the Non-Proliferation of Nuclear Weapons," June 1972, IAEA (1972).

2. R. SHER and S. UNTERMYER II, *The Detection of Fissionable Materials by Nondestructive Means*, Ch. 5-6, American Nuclear Society, LaGrange Park, IL (1980).

3. D. REILY, N. ENSSLIN, H. SMITH, and S. KREINER, *Passive Nondestructive Assay of Nuclear Materials*, United States Nuclear Regulatory Commission: NUREG/CR-5550, U. S. Government Printing Office, Washington, DC (1991).

4. S. J. TOBIN, S. F. DEMUTH, M. L. FENSIN, J. S. HENDRICKS, H. O. MENLOVE, M. T. SWINHOE, "Determination of Plutonium Content in Spent Fuel with NDA—Why an Integrated Approach," Los Alamos National Laboratory Report, LA-UR-08-0376, Los Alamos, NM (2008).

5. D. B. PELOWITZ, J. S. HENDRICKS, J. W. DURKEE, M. R. JAMES, M. L. FENSIN, G. W. MCKINNEY, S. G. MASHNIK, L. S. WATERS, "MCNPX 2.7.A Extensions," Los Alamos National Laboratory Report, LA-UR-08-07182, Los Alamos, NM (2008).

6. N. P. SANDOVAL and M. L. FENSIN, "Burnup Automation MCNPX File Data Retrieval Tool (BAMF-DRT): Data Extraction and Input File Creation Tool Users Manual," Los Alamos National Laboratory Draft Report, Los Alamos, NM (2008).

7. J. E. DOYLE ed., *Nuclear Safeguards, Security and Nonproliferation*, Ch. 5: chapter written by Brian Boyer, Elsevier Inc., Burlington, MA (2008).

8. M. L. FENSIN, J. S. HENDRICKS, S. ANGHAIE, "The Enhancements and Testing of the MCNPX Depletion Capability," Proc. International Congress on Advances in Nuclear Power Plants, Anaheim, CA, (2008).

9. Organisation for Economic Co-operation and Development: Nuclear Energy Agency, "NEA Expert Group on Burn-up Credit Criticality Safety," September 2003, <http://www.nea.fr/html/science/wpncs/buc/index.html> (December 2007).

10. N. MIURA and H. O. MENLOVE, "The Use of Curium Neutrons to Verify Plutonium in Spent Fuel and Reprocessing Wastes," Los Alamos National Laboratory Report, LA-12774-MS, Los Alamos, NM (1994).

11. H. O. MENLOVE, D. H. BEDDINGFIELD, P. M. RINARD, T. R. WENZ, "The Curium Tagging Approach for Enhanced Safeguards for Spent Fuel Handling and Reprocessing," International Atomic Energy Agency Symposium on International Safeguards, Vienna, Austria (1997).

12. M. L. FENSIN, S. J. TOBIN, M. T. SWINHOE, H. O. MENLOVE, N. P. SANDOVAL, "A Monte Carlo Based Spent Fuel Analysis Safeguards Strategy Assessment," To be published in the Proc. for Global '09, Paris, France (2009).