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Report on Design of Spent Fuel Mockup Differential Dieaway Experiments

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INTRODUCTION

The conventional differential dieaway technique is typically used to measure the fissile content of items such as waste drums or baskets of fuel rod cladding remaining after dissolution. The principle of the technique is that an initial neutron burst of fast neutrons is moderated to thermal energies in a chamber that allows a long thermal neutron lifetime (for example with graphite walls). After the initial neutrons are thermalized the only origin for fast neutrons is then fission in fissile material. Neutron detectors that are only sensitive to fast neutrons are used to detect these neutrons in the presence of the long-lived thermal neutron flux.

In the NA24 spent fuel project, Monte Carlo simulations have been made to apply the differential dieaway technique to the measurement of spent fuel assemblies in water. The neutron behavior in this arrangement is dominated by the spent fuel assembly itself (instead of being dominated by the specially designed chamber in the conventional case). These simulations have shown the potential to extract quantities of interest (burnup, initial enrichment and plutonium content) from the time behavior of the differential dieaway signal [2, 3, 4].

The present experimental program has two goals. The first is to confirm that the Monte Carlo simulations are close to reality. The second is to demonstrate that the data that is required to generate the fuel assembly results is practically measureable. That is to say that if the simulations indicate the use of data from 200 μ s after the initial burst to 1000 μ s after the burst, then we need to demonstrate that we can acquire data of sufficient quality during this period. There are potential deadtime and detector recovery issues that need to be addressed. The experiments planned here will use unirradiated uranium fuel rods in water to reproduce the neutron behavior in the spent fuel. It is not planned in these experiments to simulate the high passive neutron and gamma background of a spent fuel assembly. For the first goal of this project this is not a limitation. The passive rates in the final instrument only affect the statistical uncertainty of the result. For the second, measurement practicality, goal these passive rates are important. The passive neutron rate will increase the required counting rate capability and the gamma dose will significantly affect the detector behavior. It is planned to use equivalent gamma shielding in some of the current experiments that would reduce the gamma dose rate at the detector to acceptable limits. Separate testing with a strong gamma source will be considered later in the project. It is not absolutely necessary to achieve both of the projects goals simultaneously. It may be convenient to confirm the Monte Carlo calculations in one set of experiments and demonstrate the practicality of the technique in another set. This may prove to be necessary depending on the challenge associated with the preamplifiers and data acquisition system.

EXPERIMENTAL DESIGN

In order to satisfactorily achieve the first goal of the project, the neutron transport conditions need to be close to those considered for the conceptual instrument. However there are some features of the design that can be simplified in order to make the experimental arrangement more convenient. The essential features include uranium fuel rods in water and a fast neutron detector arrangement moderated by polyethylene. The neutron detectors will be wrapped in cadmium to produce fast neutron detector packages. The conceptual design to date has concentrated on ^3He detector. These have the advantage of high efficiency but do not tolerate a high gamma dose. The alternative would be to use fission chambers that can withstand a much higher dose rate but have much lower efficiency. The experimental design itself will have to be modeled with Monte Carlo in order to compare the measured and simulated results.

From a point of view of demonstrating the practicality of the measurement technique we will need to design a layout that would be feasible for actual spent fuel. For example if we use ^3He detectors we need to shield them sufficiently to make the dose at their location less than about 10 R/hr for the case of an actual fuel assembly. There is a considerable range of dose rates from different fuel types and fuel histories, but we need to accommodate the majority of potential spent fuel measurement cases. A starting point will be the amount of shielding that was used for the conceptual instrument.

The geometry of the conceptual design is shown in Figure 1 [1]. There is 5cm of lead shielding provided between the fuel assembly and the fast neutron detection packages. This is expected to have a significant effect on the neutron transport of the system. This shielding will be simulated with lead bricks.

Figure 1 Diagram of the neutron spectrum tailoring assembly, 17x17 PWR fuel assembly, ^3He detectors, and lead shielding for the integrated DDA-DN system. (From reference 1).

Another significant feature of the setup is the large amount of tungsten and stainless steel around the neutron generator. The purpose of this material in the conceptual design is to lower the average neutron energy that is incident on the fuel assembly (“spectrum tailoring”) and thus reduce the contribution to the induced fission signal from ^{238}U fission. This is particularly important for the measurement of delayed neutrons (not considered here). The final design of an instrument may or may not include this large amount of material. In fact, since the intended target of this instrument is to measure spent fuel underwater, it may prove acceptable to use the water itself as a spectrum tailoring material. For the purposes of showing the validity of the Monte Carlo simulation, this material is not necessary as it is expected to have a relatively small effect on the neutron time behavior. The initial design of the experiment therefore will not include this material. Some tests may be made later in the project.

Some scoping calculations on the design of DDA measurement systems were made by Niklas Lundkvist while he was a summer intern at LANL. There were several designs that are useful as background information when we consider the features necessary for the design of the current experiment. The first design (Figure 2) would be suitable for the measurement of fresh fuel. There is no gamma shielding for the tubes.

Figure 2 Possible DDA design for Fresh Fuel (^3He)

Figure 3 Possible DDA design for Spent Fuel (fission chambers)

The second design (Figure 3) replaces the ^3He tubes with fission chambers and is a potential design for a spent fuel instrument.

Figure 4 Possible design for fresh fuel Measurement (55gal drum)

Figure 4 shows a possible test setup for a measurement using a 55 gal drum to contain the water. In this configuration the detectors would have to be fission chambers to measure spent fuel.

Based on these designs we developed a model for an experimental setup based on a 55 gallon drum. The first option is shown in Figure 5. This shows that the fuel and initial set of neutron detectors will fit into the drum. At present it is envisaged that the neutron generator will stay out of the water. This configuration does not include the lead shielding that would be necessary in the real instrument, but does allow us to determine the effect of the thickness of water between the tube and the fuel assembly by moving the detector setup.

Figure 5 Experimental geometry using 55 gal drum

In order to have a setup which is more representative of a real spent fuel instrument, we can add the lead shielding as shown in Figure 6 and Figure 7. In this case, because of the size of the assembly, shielding and detectors, we would need to immerse the neutron generator in the water if we wanted to vary the amount of intermediate water to see the effect on the instrument response.

Figure 6 Experimental geometry with lead shield (horizontal cross-section)

Figure 7 Experimental geometry with lead shield (vertical cross section)

Preliminary calculations have been performed to assess the count rate of these configurations. Figure 8 shows the time dependence of the counting rate in the side block (sum of 4 ^3He tubes).

Figure 8 Time dependence of count rate in side block for with and without lead shielding

The effect of the lead can be seen in making the initial dieaway more steep but making the flux dieaway more slowly at longer times. This is the kind of behavior we expect to be able to reproduce

experimentally. Figure 9 shows the time-dependence of the flux, for the case with the Pb shield, for the side and back detector blocks. The data from the two side blocks cannot be distinguished from one another. The ratio of the counting rate in the back block to one of the side blocks is shown in Figure 10.

Figure 9 Time dependence of side and back detector blocks.

Figure 10 Ratio of Back to Side detector block count rates as a function of time after the pulse.

Table 1 Count Rates from side block for different time intervals

Start time (μs)	End time (μs)	Response (counts/ source neutron)	Count rate for $2\text{e}8$ n/s
0	100	0.000812	1.62×10^5
0	1000	0.001672	3.34×10^5
100	200	0.000277	5.54×10^4
200	1000	0.000583	1.17×10^5
300	1000	0.000408	8.17×10^4

Table 1 shows the count rates from one of the side detector blocks for various time intervals. If it were possible to record the full count rate without deadtime effects, the overall count rate would be 3.34×10^5 cps with the neutron generator producing 2×10^8 source neutrons/sec. In practice we will initially only record the rates after 200 or 300 μs , giving maximum count rates of around 100,000 cps (for one block). This is a relatively high rate and we will have to carefully consider deadtime effects, either by reducing the source intensity or improving the electronics or a combination of the two.

A similar calculation has been carried out for fission chambers in order to estimate the counting rate to be expected. The geometry used was that of Figure 5, because no lead shielding would be needed in the real case. With 4 fission chambers the same physical size at the ^3He tubes (each with 190mg of uranium) the count rate is about 50 times smaller than the ^3He case. This should still result in satisfactory measurement times for reasonable statistical precision.

Figure 11 show the ratio of the counting rates in the fission chamber case to the ^3He case as a function of time after the neutron pulse.

Figure 11 Ratio of Fission Chamber Count Rate to ^3He Count Rate (with no Pb).

SUMMARY

We have developed a basic design for an experimental test of a differential dieaway instrument for the measurement of spent fuel. The design (Figure 6) that includes lead shielding has all of the important features that are required for a spent fuel measurements. A 3D view of the experiment is shown in Figure 12. This experiment with fresh fuel rods will be in a domain which is suitable for benchmarking the performance of the simulations. This is the first objective of the project.

The count rate estimates show that the expected rates with the standard configuration of ^3He tubes in this geometry will be relatively high. On the other hand, the response of the fission chambers is 50 times less and should therefore be relatively easy to record. The second objective of the project is to demonstrate that we can record data at least $200\mu\text{s}$ after the burst if not earlier.

Figure 12 3D view of experimental geometry

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