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MONTE CARLO CHARACTERIZATION OF PRESSURIZED WATER REACTOR SPENT FUEL ASSEMBLY FOR THE DEVELOPMENT OF A NEW INSTRUMENT FOR PIN DIVERSION DETECTION

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ABSTRACT

A novel concept to detect pin-diversion from spent fuel assembly is proposed and described. The instrument will use multiple tiny neutron and gamma detectors in a form of cluster (detector cluster) and high precision driving system to collect radiation signatures inside pressurized water reactor (PWR) assembly. In order to validate our concept, a Monte Carlo study was done using a Monte Carlo code MCNP5. MONTEBURNS, a computational tool that links MCNP and ORIGEN, was used to produce accurate PWR spent fuel isotopic compositions. Monte Carlo simulations, using realistic fuel geometry and actual fuel material information, were performed to study radiation field inside a PWR spent fuel assembly. The preliminary Monte Carlo simulation study shows that indeed 2 dimensional neutron data, when obtained in the presence of missing pins, have data profiles distinctly different from the profiles obtained without missing pins.

INTRODUCTION

A novel concept to detect pin-diversion from PWR spent fuel assemblies is proposed and described. The envisioned instrument would most likely require multiple tiny neutron and gamma detectors that can be inserted into the vacant guide tubes of a PWR bundle, perhaps in a "detector cluster" configuration which may also require a high-precision axial driving system to collect radiation signatures inside pressurized water reactor (PWR) assemblies.

In order to validate this concept, exploratory research studies using the Monte Carlo code MCNP5 [1] are herein reported. These studies have also required the use of the MONTEBURNS code [2], a computational tool that links MCNP and ORIGEN to help simulate the operational history, depletion, and cooling of the PWR spent fuel assembly to help characterize the spatial distribution of isotopic compositions. Currently there are no safeguards instruments that can detect a possible pin diversion scenario. The FORK detector [3] can characterize spent fuel assemblies using operator declared data, but it is not sensitive enough to detect missing pins from spent fuel assemblies. Likewise, an emission computed tomography system has been used to try to detect missing pins from a spent fuel assembly [4], which has shown some potential for identifying possible missing pins but this capability has not yet been fully demonstrated.

Monte Carlo simulations using a realistic 17x17 fuel geometry and depleted fuel material data were performed to study the neutron and gamma radiation field inside a PWR spent fuel assembly. The preliminary simulations herein reported show that indeed two-dimensional neutron and gamma data, when obtained in the presence of missing pins, can have data profiles distinctly different from the profiles obtained without missing pins. Ongoing studies (not included) continue to evaluate the possible benefits of a three-dimensional analysis, with the purpose of confirming further advantages from axially-dependent measurements.

METHODOLOGY

The first step to simulation studies was to construct a realistic MCNP5 model of a modern PWR fuel assembly. This was achieved by employing an OECD/NEA benchmark specification of a Takahama-3 17x17 PWR fuel assembly [5]. Subsequently, that study was followed by the depletion of the assembly using MONTEBURNS to approximate the isotopic distribution at EOC and after two years of cooling [6].

Figure 1 below shows a diagram illustrating the 39 independent regions depleted using MONTEBURNS (MB), in which the color red highlights primarily non-depletable regions of water and the guide tubes (larger diameter circles). As noted previously, the fuel geometry was the Takahama-3 17x17 PWR NT3G24 assembly loaded with 248 UO2 fuel pins, 4.1%wt U235 enriched, 16 UO2-GD2O3 pins (2.6% wt U235 and 6% wt Gd) and 25 water rods. The assembly was irradiated for three cycles with a power of 38.6 W/gU.

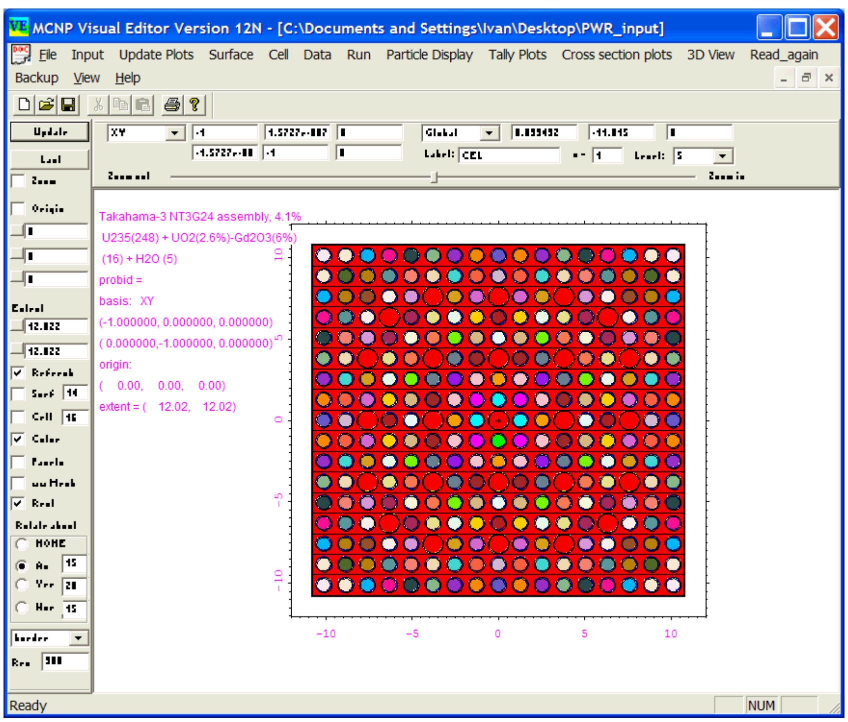

Figure 1. MCNP5 Visual Editor Image of the Takahama-3 17x17 PWR Bundle

Figure 2 illustrates the hot k-infinity trajectory of the depleted assembly as a function of burnup during the bundle's operational history. The assembly was assumed to be discharged at 50 GWd/TU and subsequently cooled for two years, during which the isotopic distribution changes only by decay. Due to the maximum material number limitations of MB (about 40), 1/8 bundle symmetry was used for the depletion process, as well as reflective boundary conditions surrounding the outer surface of the bundle (i.e., x and y outer boundaries were set to specular reflection to simulate the effect of being surrounded by similar assemblies).

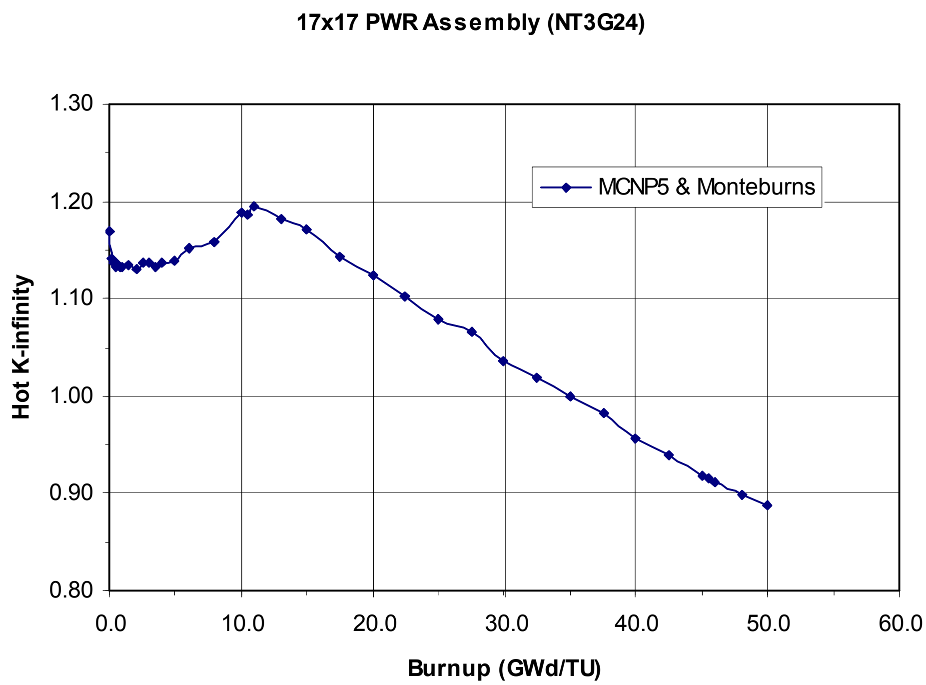

Figure 2. Reactivity Trajectory of Depleted NT3G24 17x17 Bundle

Separate MCNP5 cases were run for neutron and gamma studies, and these employed 10^7 histories to confirm adequate statistics. For the neutron flux source in these assemblies, we targeted the Cm-244 distribution in the assembly. This is because for typical commercial power spent fuel assemblies, the neutron flux inside spent fuel assemblies is expected to be dominated by the spontaneous fission from Cm-244 after two years of cooling time. Figure 3 illustrates the spontaneous fission neutron emission after 2 years of cooling. Accordingly, the neutron source strengths were established in the bundle in proportion to the Cm-244 relative accumulation illustrated in Figure 4. The neutron flux was calculated by the Watt fission spectrum and divided in 23 groups between 1.0E-05 and 20MeV, plus a total count.

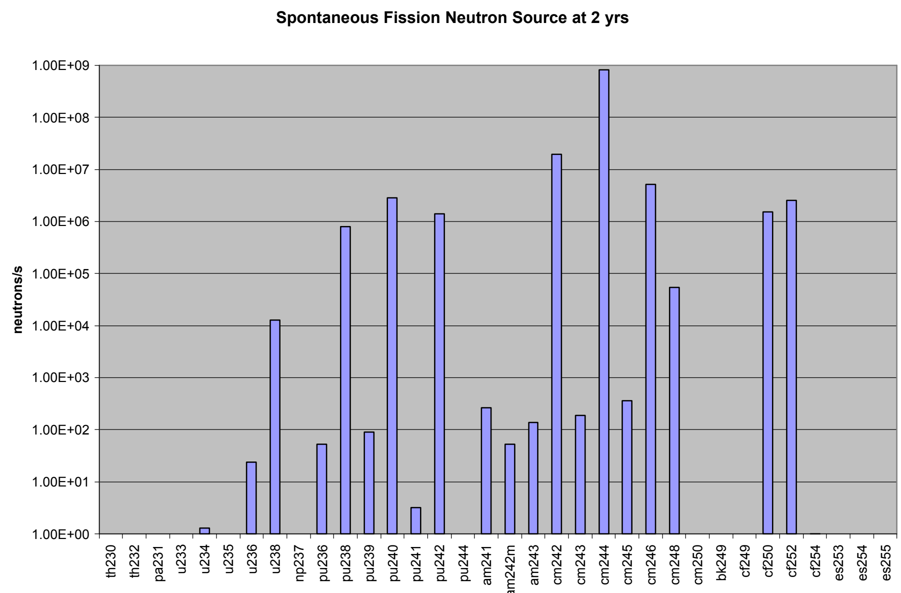

Figure 3. Dominance of Cm-244 Spontaneous Fission Neutrons After 2 Years of Cooling

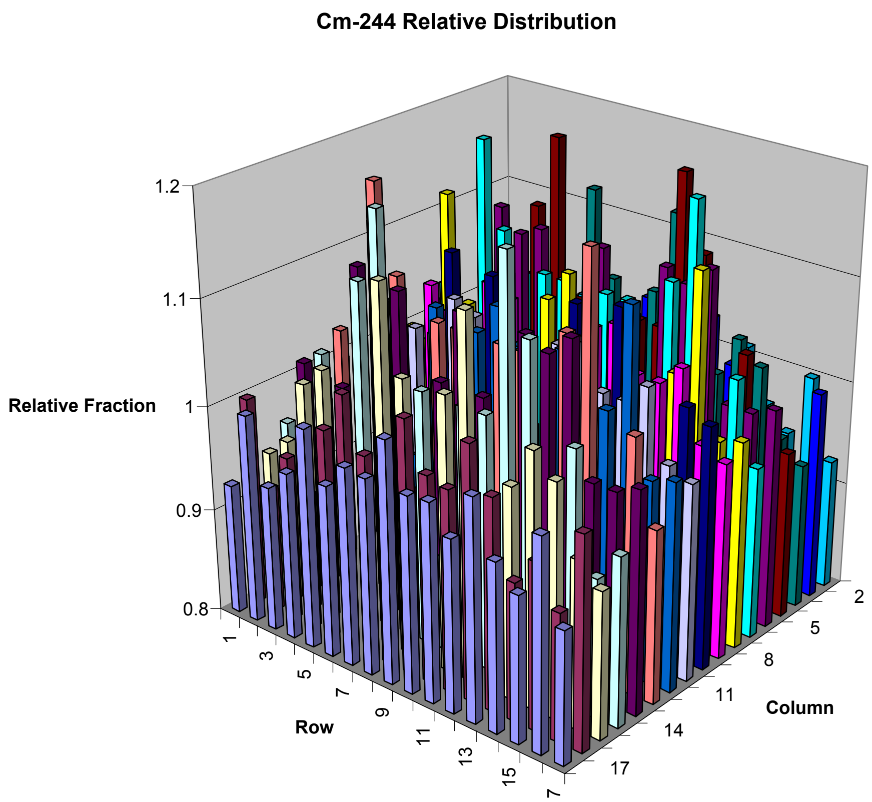

Figure 4. Relative Cm-244 Pin-by-Pin Accumulation in 17x17 Bundle Used to Establish Neutron Source

RESULTS

To determine the limits of detection, a single pin from the spent fuel assembly was removed and replaced with a fresh fuel pin. Two cases were considered: a pin replaced near the center of the assembly at row 10, column 9, and a pin replaced at the corner at row 17, column 1. Figure 5 shows these locations and their nearest guide tubes.

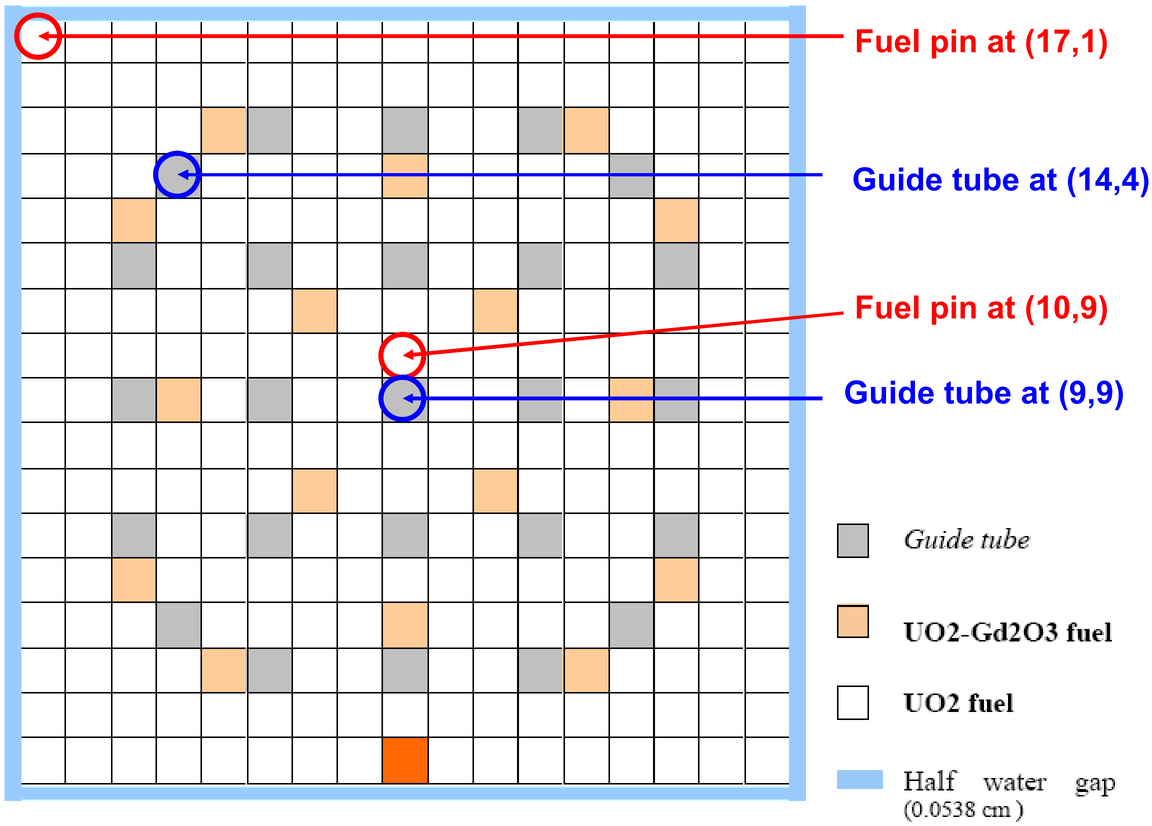

Figure 5. Diverted Pins Considered in this Study and Nearest Guide Tubes

Results from pin-diverted cases were compared against runs done with all spent fuel pins present, and the absolute difference and percent difference were calculated. Using standard error propagation [7], relative errors were calculated for both quantities. A threshold of 1.0 (100%) was set for both. Another indicator used was whether the difference was greater than could be accounted for by the margin of error of the results. Since the actual counting will use Maxwell statistics, a hypothetical count was constructed from the MCNP5 flux by multiplying it by 60 seconds. The absolute error was calculated as the square root of the count. The relative error of the difference from the case where all spent fuel pins are present was calculated.

	r.e.<1	Maxwell	total<1	Maxwell	tot > overlap	Maxwell
GT (3,6)	3	7	yes	yes	no	no
GT (3,9)	1	4	no	no	no	no
GT (3,12)	2	4	yes	yes	yes	yes
GT (4,4)	1	2	no	no	no	no
GT (4,14)	1	3	no	no	no	no
GT (6,3)	3	5	no	no	no	yes
GT (6,6)	2	6	yes	yes	no	yes
GT (6,9)	5	10	yes	yes	no	yes
GT (6,12)	2	8	no	no	no	yes
GT (6,15)	2	6	no	no	no	no
GT (9,3)	1	6	yes	yes	no	yes
GT (9,6)	7	10	yes	yes	yes	yes
GT (9,9)	11	17	yes	yes	yes	yes
GT (9,12)	9	11	yes	yes	yes	yes
GT (9,15)	2	6	yes	yes	yes	yes
GT (12,3)	1	2	no	no	no	no
GT (12,6)	3	8	yes	yes	yes	yes
GT (12,9)	6	11	yes	yes	yes	yes
GT (12,12)	3	9	yes	yes	no	no
GT (12,15)	6	8	yes	yes	no	no
GT (14,4)	3	6	yes	yes	no	no
GT (14,14)	5	6	yes	yes	no	yes
GT (15,6)	1	2	no	no	no	no
GT (15,9)	2	4	yes	yes	no	no
GT (15,12)	1	2	no	no	no	no

Table 1. Neutron Results with Fresh Fuel Pin at (10,9)

Figure 6 illustrates the percent differences in the neutron flux at the closest guide tube (9,9). Absolute differences, as well as the same combinations or results for the outer pin substitution at (17,1) and guide tube at (14,4) were also collected.

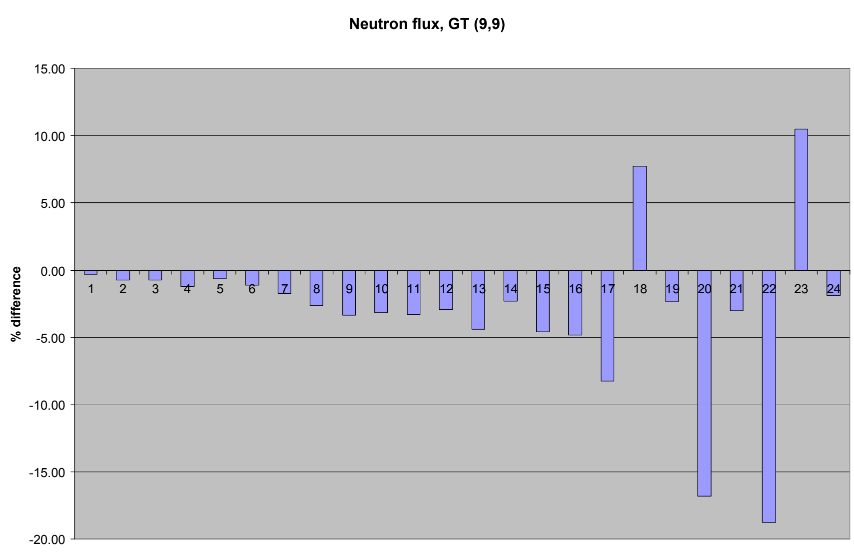

Figure 6: Percent Difference in Neutron Flux at Guide Tube (9,9)

CONCLUSIONS

These preliminary Monte Carlo simulation studies show that indeed two dimensional neutron data, when obtained in the presence of missing pins, have data profiles distinctly different from the profiles obtained without missing pins. Replacing a single spent fuel pin in the assembly resulted in detectable differences in the neutron flux greater than the designated threshold in at least one energy group for most of the guide tubes, as summarized in Table 2.

Pin replaced	MCNP5 statistics	Maxwell statistics
(10,9)	> Threshold in 7 tubes	> Threshold in 13 tubes
(17,1)	> Threshold in 3 tubes	> Threshold in 5 tubes

Table 2: Summary of Neutron Results

A 60 second count provides Maxwell statistics that are adequate; using a neutron detector in each of the guide tubes should be able to determine possible pin diversion.

Ongoing work at this time includes the following areas:

- Uncertainty analysis related to the operational and cooling history, and the type of PWR assembly (15x15 or vintage models), in particular, assessment of asymmetric depletion upon detection ability.
- Assembly depletion in 3D using TRITON and/or MCNPX/CINDER'90
- Monte Carlo analyses performed in 3D for both gammas and neutrons
- Study of detector design and efficiencies associated, such as study of thin fission chamber position within guide tube, and axial displacement of chamber.

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