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Measurement Science Section

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ABSTRACT

If developed, a nondestructive method would be useful for verifying canister contents without requiring the canister to be opened. This paper addresses the application of the ^{252}Cf-source-driven frequency analysis measurements for verification of the fissile material content of sealed spent fuel canisters. The cross-power spectral density (CPSD) between the ^{252}Cf source in an ionization chamber and external neutron detectors depends only on the induced fission rate in the fissile system and is independent of inherent sources. Thus the source-to-detector CPSD is ideal for determination of fissile material content of spent fuel. This paper evaluates the application of this method to a 125-ton spent fuel canister that contained 21 pressurized-water reactor fuel elements. The results demonstrate that the fissile material content of a sealed spent fuel canister could be obtained using the ^{252}Cf frequency analysis method if calibration standards were available. The results also indicate that a measurement could be performed in less than a day for burnups up to 36 GWd/MTU and in less time for lower burnups.

INTRODUCTION

The verification of the contents of the spent nuclear fuel in sealed canisters and casks while the casks are loaded would be useful for material control and accountability. If developed, a nondestructive method would be useful for verifying canister contents without requiring the canister to be opened. This paper addresses the application of the ^{252}Cf-source-driven noise analysis method[1] for this task. As a first step in the application of this method for monitoring of spent fuel, an experiment was performed with a mockup of up to 17 fresh pressurized-water reactor (PWR) fuel assemblies in borated water.[2] This measurement in an ~5-ft-diam tank with borated water was performed to assess the capability of the method to measure the subcritical neutron multiplication factor. Based on the early success of these measurements, an evaluation was done to investigate the use of this method to justify burnup credit.[3] The recent development of the Monte Carlo code MCNP-DSP[4] that directly calculates all the measured parameters in the measurement allowed for the interpretation of configurations even as small as a single fuel element. MCNP-DSP was used to evaluate the feasibility of measuring the fissile mass of a single PWR fuel element as a function of the fuel element length.[5] Because the cross-power spectral density (CPSD) between the source and the detector only correlates neutrons from induced fissions by the ^{252}Cf source, this CPSD does not depend on the background from inherent sources in the spent fuel. This analysis showed that measurement of the CPSD between an array of moderated ^{3}He chambers on one side of the spent fuel element and a ^{252}Cf source on the other side was sensitive to the fissile mass of the fuel. Thus, this method could be used to scan a fuel element to obtain the fissile mass directly rather than measure the burnup. With these satisfactory results, a limited feasibility analysis was performed to assess if the ^{252}Cf-source-driven noise analysis method could be used for nondestructive assay of spent nuclear fuel canisters of interest to the International Atomic Energy Agency.

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This paper presents a brief discussion of the ^{252}Cf-source-driven noise analysis method; provides a short description of the MCNP-DSP Monte Carlo code, which simulates the source-driven measurement; provides assessments of the practicality of performing nondestructive assay of spent nuclear fuel in canisters; presents some conclusions about the results of this initial analysis; and describes areas of future analysis.

MEASUREMENT METHOD

The ^{252}Cf-source-driven noise analysis method was developed to characterize subcritical configurations of fissile material. This measurement method has been applied to initial loading of reactors,[2] refueling of reactors,[6] fuel preparation and processing facilities,[7,8] nuclear weapons indentification,[9] and nondestructive assay for nuclear material control and accountability.[10] In these latter applications, the signatures provided by the measurement can be used to identify nuclear components or provide assay of special nuclear materials by comparison with known standards. One advantage of this method is the high sensitivity of the measured parameters to fissile mass, which has been observed experimentally and demonstrated theoretically for a variety of configurations of fissile materials.[11] Consequently, the sensitivity of the measured parameters to fissile mass for the spent fuel applications has been reported.[12,13] Another advantage of this method is that some of the parameters are independent of detection efficiency and source intensity. Some of the measured quantities depend only on the induced fissions by the ^{252}Cf source, while others are dependent on all fission sources.

This method generally measures the CPSD, $G_{23}(\omega)$, between a pair of detectors (detectors 2 and 3) located in or near the fissile material and measures the CPSDs, $G_{12}(\omega)$ and $G_{13}(\omega)$, between these same detectors and a source contained in an ionization chamber (detector 1) that is also located in or near the fissile material. The autopower spectral densities (APSDs) of the detectors, $G_{11}(\omega)$, $G_{22}(\omega)$, and $G_{33}(\omega)$, are also measured. The CPSDs G_{12} and G_{13} depend only on detected particles from induced fissions by the ^{252}Cf source, the CPSD G_{23} depends on all fission sources, and the APSDs G_{22} and G_{33} depend on all radiation sources. The CPSD between the source and detector, G_{12}, is ideal for fissile mass assay because the magnitude of G_{12} does not depend on the inherent sources. However, the measurement time required for G_{12} to converge does depend on the inherent sources. The effect of the background radiation on the measurement time is well known and can be estimated from calculations.

MCNP-DSP

MCNP-DSP is an analog Monte Carlo code developed from MCNP4a[14,TM] which calculates the time and frequency analysis parameters of the ^{252}Cf-source-driven noise analysis measurement. In MCNP-DSP, average quantities like the number of neutrons from fission have been removed and replaced with the appropriate probability distributions because average quantities reduce the statistical fluctuation in the fission chain populations. These distributions and others when available are used in the calculation of the noise measurement and in the calculation of the neutron multiplication factor.

For the frequency analysis calculations, the particle tracking begins with the source fission. The source particles and their progeny are tracked throughout the system until the particles are either absorbed or escape from the system. The time dependence of the neutron track is followed for each particle from the birth time of the source particle. The source particles are started randomly within a specified period of time. The time of detection is determined from the neutron time of flight to the detector and the source birth time. The detector time history is then formed into sequences that are sampled into blocks of 512 or 1024 data points. These blocks of data are Fourier transformed using standard algorithms, and then the Fourier transformed blocks are complex multiplied to obtain estimates of the APSDs and CPSDs. This process is repeated until the specified number of blocks have been calculated.

SPENT FUEL CANISTER DESCRIPTION

The canister chosen for this analysis is the 125-ton multipurpose canister (MPC) that was filled

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[TM] MCNP is a trademark of the Regents of the University of California, Los Alamos National Laboratory.

with 21 Westinghouse PWR fuel elements with the same average burnup. Seven axial burnup zones were used for each fuel element whose fuel isotopics were estimated using the PDQ code.[15] For the purposes of this analysis, the source was located in a 2-in.-thick polyethylene moderator exterior to the canister, and 15 unshielded 2.5-in.-diam., 10-ft-long ^{3}He proportional counters, surrounded by 2.5-in.-thick polyethylene moderators, were positioned on the opposite side of the canister. The type and number of detectors were chosen to minimize computation time. A schematic of this canister configuration is shown in Fig. 1. The 60-in.-OD canister had a 1-in.-thick steel shell, a 2.5-in.-thick steel base plate, and was 193 in. long. The lid of the canister was comprised of steel and depleted uranium and was 10.5-in.-thick.

RESULTS

Because the source-to-detector CPSD, G_{12}, depends on the induced fission rate, it should change as the fissile material content in the canister changes. In this analysis, G_{12} was estimated for six different average burnups from fresh to 32 GWd/MTU with only 1.6 million ^{252}Cf fissions for each burnup (a 1-μg ^{252}Cf source fission at a rate of 614,000 per second). The low-frequency average of the magnitude of G_{12} is related to the induced fission rate in the system and the roll off of the signature can be related to the prompt neutron decay constant of the system. The low-frequency average of the magnitude of G_{12} was calculated for each burnup and is plotted as a function of fissile material content (^{235}U and ^{239}Pu) in Fig. 2. The low-frequency average of the magnitude of G_{12} for the 32-GWd/MTU burnup (0.15-g/cm^3 fissile material) loaded canisters is ~40% lower than that for the fresh fuel (0.31-g/cm^3 fissile material) loaded canister.

The effect of background does not change the value of G_{12} but does affect the measurement time for convergence of G_{12}. The inherent source contributions available to us were for 36 GWd/MTU.[15] The estimated measurement time to achieve a 5% uncertainty in G_{12} for a fuel assembly irradiated 3 years (36 GWd/MTU burnup) is presented in Table 1 for various source sizes and for several cooling times. The measurement times for this burnup would be approximately one day or less using a 200-μg ^{252}Cf source. These measurement times account for spontaneous fission and alpha-n inherent sources and would be shorter for lower burnup fuel. The source sizes presented in Table 1 could be achieved by using multiple sources with 25 μg of ^{252}Cf or larger. Sources have been fabricated at Oak Ridge National Laboratory with as much as 15 μg of ^{252}Cf; therefore, a 25-μg ^{252}Cf source is only a small extrapolation. For large sources the chambers could be operated in the current mode if the alpha particle contribution to the current is known.

CONCLUSIONS

This preliminary analysis has shown that the ^{252}Cf-source-driven frequency analysis measurement method may be useful for fissile mass assay of spent fuel canisters if calibration standards were available. The measurement time for obtaining accurate estimates of the CPSD between the source and the detector array was shown to be dependent on the source size and on the spent fuel burnup. For a canister filled with 36-GWd/MTU spent fuel, a measurement of the CPSD between the source and the detector array could be performed in one day or less using eight 25-μg ^{252}Cf sources. For lower burnup fuel, the measurement time would be significantly lower, and smaller sources may be used. A more extensive future study to be performed would address varying the fuel burnup for each canister position, investigating the amounts of shielding for the ^{3}He detectors, and addressing the application of this method to shielded casks.

TABLE 1. ESTIMATED MEASUREMENT TIME (DAYS) TO ACHIEVE 5% UNCERTAINTY IN G_{12} FOR 36-GWD/MTU BURNUP FUEL ASSEMBLIES[a]

	Cooling Time After Discharge		
^{252}Cf source (μg)	Discharge	5 yr	10 yr
25	10	5	4
50	5	2.5	2
100	2.5	1.3	1
200	1.3	0.6	0.5

[a]Less time for lower burnup

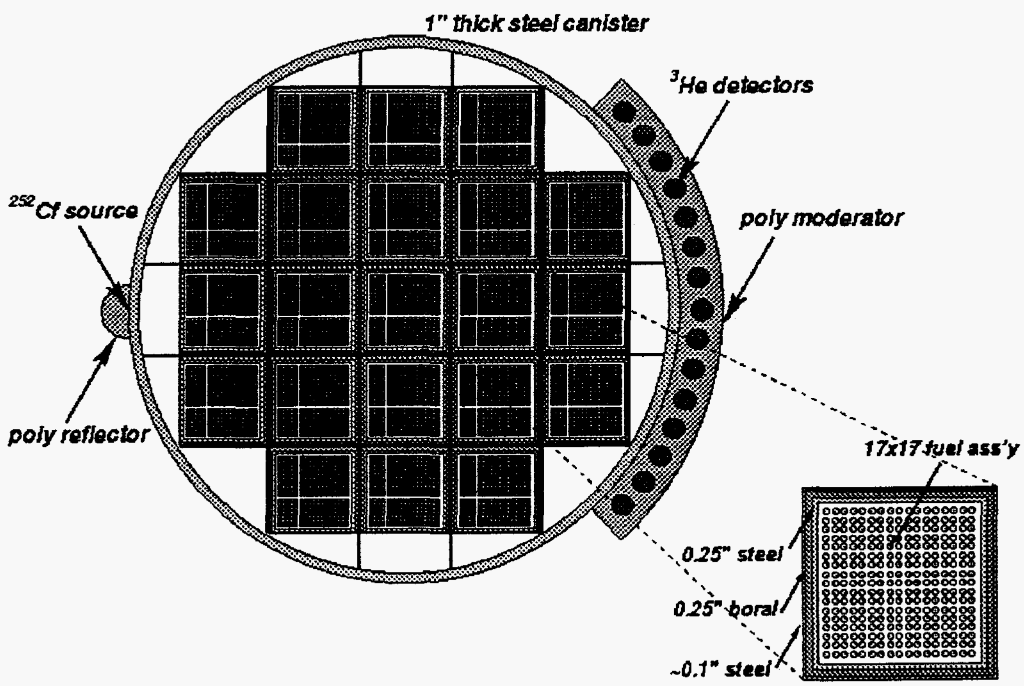

FIGURE 1. SCHEMATIC OF SOURCE AND DETECTOR CONFIGURATION FOR MULTIPURPOSE CANISTER

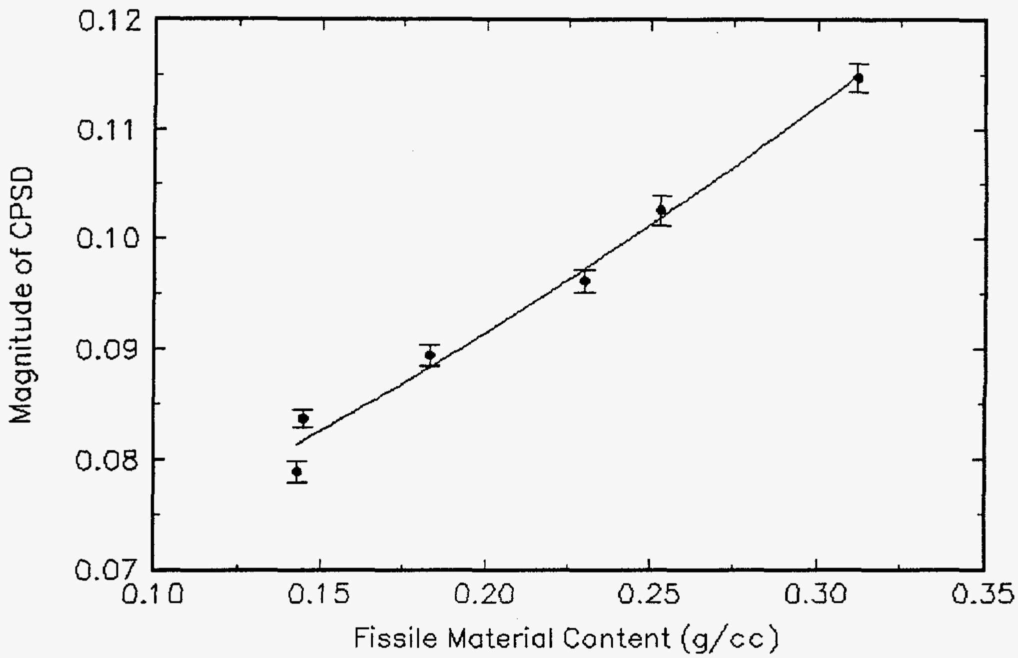

FIGURE 2. MAGNITUDE OF G_{12} AS A FUNCTION OF FISSILE MASS

REFERENCES

1. Mihalczo, J. T., Paré, V. K., Ragan, G. L., Mathis, M. V., and Tillet, G. C., "Determination of Reactivity from Power Spectral Density Measurements with Californium-252," *Nucl. Sci. Eng.*, **66**, 29 (1978).

2. King, W. T., Mihalczo, J. T., and Blakeman, E. D., "Preliminary Investigation of the ^{252}Cf-Source-Driven Noise Analysis Method of Subcriticality Measurement in LWR Fuel Storage and Initial Loading Applications," *Trans. Am. Nucl. Soc.*, **47**, 239 (1984).

3. Mihalczo, J. T., and Broadhead, B. L., "Feasibility of Spent LWR Fuel Subcriticality Measurements by the Cf Noise Method," *Trans. Am. Nucl. Soc.*, **62**, 322 (1990).

4. Valentine, T. E., and Mihalczo, J. T., "MCNP-DSP: A Neutron and Gamma Ray Monte Carlo Calculation of Source Driven Noise-Measured Parameters," *Ann. Nucl. Energy*, **23**, 1271 (1996).

5. Mihalczo, J. T., Valentine, T. E., and Mattingly, J. K., "Feasibility of Subcriticality and NDA Measurements for Spent Fuel by Frequency Analysis Techniques with ^{252}Cf," Proc. of NPIC and HMIT '96, pp. 883–891 in Penn State University, 1996.

6. Mihalczo, J. T., and King, W. T., "^{252}Cf-Source-Driven Neutron Noise Method for Measuring Subcriticality of Submerged HFIR Fuel Elements," *Trans. Am. Nucl. Soc.*, **43**, 408 (1982).

7. Mihalczo, J. T., King, W. T., and Blakeman, E. D., "^{252}Cf-Source-Driven Neutron Noise Analysis Measurements for Coupled Uranium Metal Cylinders," *Trans. Am. Nucl. Soc.*, **49**, 241 (1985).

8. Mihalczo, J. T., Blakeman, E. D., and King, W. T., "Subcriticality Measurements for Two Coupled Uranyl Nitrate Solution Tanks Using ^{252}Cf-Source-Driven Noise Analysis Methods," *Trans. Am. Nucl. Soc.*, **52**, 640 (1986).

9. Mihalczo, J. T., and Paré, V. K., "Nuclear Weapons Identification System (NWIS)," p. 24 in *Arms Control and Nonproliferation Technologies*, Report on Nuclear Warhead Dismantlement, DOE/AN/ANCI-94C, Lawrence Livermore National Laboratory, Third Quarter 1994.

10. Mihalczo, J. T., and Paré, V. K., "NWIS Signatures for Confirmatory Measurements with B33 Trainers," *Proceedings of the 36th Annual Meeting of the Institute of Nuclear Materials Management*, Palm Desert, California, July 1995.

11. Mihalczo, J. T. and Valentine, T. E., "Calculational Verification and Process-Control Applications Utilizing the High-Sensitivity of Noise Measurement Parameters to Fissile System Configuration," *Nucl. Sci. Eng.*, **121**, 2 (1995).

12. Krass, A. W., Valentine, T. E. , and Mihalczo, J. T., "Sensitivity of the ^{252}Cf-Source-Driven Noise Analysis Method to Fission Product Content of Spent LWR Fuel," *Trans. Am. Nucl. Soc.*, **65**, 253-254 (June 1992).

13. Mihalczo, J. T., Krass, A. W., and Valentine, T. E., "Feasibility of Validation of Calculations for Burnup Credit for Spent Fuel with ^{252}Cf-Source-Driven Noise Measurements," Institute of Nuclear Materials Management Meeting, Orlando, July 19–22, 1992.

14. Briesmeister, J. F., Ed., *MCNP--A General Monte Carlo Code for Neutron and Photon Transport*, LA-12625-M, Los Alamos National Laboratory, 1993.

15. Personal communication with John Paul Renier, 1996.