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SPENT THERMAL REACTOR FUEL ASSEMBLY CHARACTERIZATION WITH SOLID STATE TRACK RECORDERS

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SPENT THERMAL REACTOR FUEL ASSEMBLY CHARACTERIZATION
WITH SOLID STATE TRACK RECORDERS

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ABSTRACT

A scoping experiment to characterize the neutron field generated from a Light Water Reactor spent fuel assembly has been successfully completed. Solid State Track Recorder (SSTR) neutron dosimeters were exposed at the surface of a spent fuel assembly from a Pressurized Water Reactor. Acceptable track densities were obtained. From these SSTR neutron dosimetry measurements, an absolute neutron flux of about 8×10^3 n/cm² sec was determined at the surface of the spent fuel assembly some three years after discharge from the reactor. This deduced neutron energy flux, with a mean neutron energy of about 1.3 MeV, is intimately dependent upon the actinide content of the spent fuel. Hence, the results of this preliminary experiment have demonstrated that, with suitable calibration, SSTR neutron dosimetry can be successfully applied for nondestructive spent fuel actinide assay and for characterization of the radiation environment associated with spent reactor fuel assemblies. Follow-on studies are described.

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SPENT THERMAL REACTOR FUEL ASSEMBLY CHARACTERIZATION
WITH SOLID STATE TRACK RECORDERS (FF028)

1.0 SUMMARY

A preliminary scoping experiment to characterize the neutron field generated from a Light Water Reactor (LWR) spent fuel assembly (FA) has been successfully completed. Solid State Track Recorders (SSTR) neutron dosimeters have been exposed at the surface of a spent FA from a Pressurized Water Reactor (PWR). Acceptable fission track densities were obtained from a five-day exposure using mica SSTR and isotopic fission deposits of ^{232}Th , ^{235}U (bare and cadmium-covered), ^{237}Np , and ^{238}U .

From these SSTR neutron dosimetry observations, an absolute neutron flux of roughly $8000 \text{ n}/(\text{cm}^2 \cdot \text{sec})$ was obtained at the surface of a spent thermal reactor FA approximately three years after discharge from a PWR. The deduced neutron energy spectrum, with a mean neutron energy of roughly 1.3 MeV, is intimately dependent upon the actinide content of the spent fuel. Hence, the results of this preliminary experiment have demonstrated that, with suitable calibration, SSTR neutron dosimetry can be successfully applied for nondestructive spent fuel actinide assay and for characterization of the radiation environment associated with reactor spent fuel assemblies.

Track data from this scoping experiment have been used to prepare fission deposits for more accurate SSTR neutron dosimetry in follow-on experiments. The Materials Interaction Test (MIT) is a particularly important follow-on study, in which a long term in-situ irradiation will be used to test material properties of prospective geologic storage media candidates. Fission deposits and mica SSTR have been prepared, encapsulated, and incorporated in various MIT specimen capsules scheduled for 3- to 5-year exposure inside a demonstration spent fuel storage facility.

2.0 INTRODUCTION

As spent fuel from nuclear power stations continues to accumulate, it becomes increasingly important to provide effective means for long-term disposal of this material. To this end, it is essential to assure 1) personnel safety during handling and transport, 2) long-term integrity of the storage medium, and 3) accurate accountability of the actinide content of the fuel. Accomplishment of these objectives is dependent to a considerable degree upon development of means to characterize the radiation field associated with spent fuel and to determine the elemental isotopic content of the spent fuel.

Direct radiochemical procedures can be used to determine the concentration of plutonium or other actinides in spent fuel. However, procedures dealing directly with highly radioactive spent fuel are complex and expensive. Hence, it is highly advantageous to either eliminate such procedures or reduce them to a bare minimum. A nondestructive procedure is, therefore, highly desirable. Since spent fuel is handled, transported and stored in the form of fuel assemblies (FA), nondestructive methods that can be applied directly to these assemblies are preferable. Comprehensive measurement of spontaneous neutron generation from spent fuel is also necessary for more accurate definition, prediction and control of personnel radiation exposures and for assessment of the potential for damage to candidate geologic storage facility media.

Hence, neutron dosimetry measurements with spent FA can serve two broad purposes, namely to characterize the composition and the radiation environment. This first purpose includes the need to understand and quantify spent fuel fissile isotopic compositions for reasons of economics, safeguards, or reactor physics. The second purpose encompasses health and safety requirements created by the spent fuel radiation environment.

The initial impetus for this work grew out of criticality safety studies with FA arrays stored for future use in the Fast Test Reactor (FTR). In

these studies, SSTR neutron dosimetry was successfully employed to determine the effective criticality factors of FTR storage facilities. On the basis of these investigations together with the known characteristics of SSTR, the feasibility of SSTR neutron dosimetry for spent FA characterization evolved. Ensuing efforts were also dedicated to serving the parallel need for SSTR scoping data that could be used to specify fission deposits for SSTR neutron dosimetry in a long-term (3- to 5-year) Materials Interaction Test (MIT).

Passive neutron dosimetry techniques to characterize composition can be based upon spontaneously-generated neutron emission from the plutonium and higher actinide content of the fuel. Neutrons are emitted from spontaneous fission as well as from (α, n) reactions in the irradiated fuel. Consequently, the presence of these spontaneously-generated neutrons is strongly correlated with the higher actinide content. At the same time, these neutron observations will contribute to characterizing the radiation environment in that fast neutrons are an important component of this environment.

Many alternative nondestructive dosimetry techniques may be possible candidates for this application. However, most of these alternatives require additional interrogating devices and/or facilities, such as accelerators, reactors, etc. These alternatives can prove to be very expensive compared to conventional passive neutron dosimetry techniques to observe spontaneously-generated neutrons. Two such conventional neutron dosimetry techniques possessing the requisite sensitivity and discrimination against background are Solid State Track Recorder (SSTR) Neutron Dosimetry and Mn-V Bath Neutron Dosimetry.

The sensitivity and absolute accuracy of the SSTR neutron dosimetry method are well established⁽¹⁾ as is the insensitivity of many SSTR candidates to other radiation field components. The Mn-V bath method of measuring absolute neutron source strength is also well documented.^(2,3) For this application, the spent FA would be inserted into a specially-fabricated bath system possessing a cylindrical void specifically designed to accept

such assemblies. Both SSTR and Mn-V bath methods appear to possess sufficient sensitivity for accurate characterization to well below 1% Pu by weight.

It should be noted that these two methods complement each other in several respects. SSTR dosimetry is passive, whereas the bath method is an on-line active monitoring system. Hence, both methods could be employed simultaneously. SSTR dosimetry could be applied to many FA simultaneously, whereas the bath system can be used for only a single FA at a time. A V-bath has a turn-around time of an hour or so, whereas SSTR exposures require a day or more.

One distinct advantage of SSTR dosimetry over the bath technique is the ability to apply SSTR over a wide range of neutron moderating environments with neutron reactions of different spectral sensitivity. Hence one can surround the spent FA with excellent moderator, or even none at all. In the latter case, the FA would be suspended in air, which is a good approximation of a vacuum boundary condition. SSTR measurements under such drastic changes of moderating environment with neutron reactions of different spectral sensitivity would provide further information on spent FA constituents.

Several attributes of the SSTR method cannot be overemphasized: cost effectiveness, ready availability, and direct applicability without the need for special facilities. Consequently, SSTR neutron dosimeters appear to offer ideal attributes for cost effective and precise nondestructive characterization.

These factors have permitted quick adaptation of the SSTR dosimetry in a spent FA scoping experiment, which is described in Section 3.0. Experimental results obtained from this scoping study are given in Section 4.0. Future plans and activities in spent fuel dosimetry characterization work are then discussed in Section 5.0.

The work of other groups must be mentioned, although the existence of such efforts only came to light after this scoping experiment was completed. A completely different on-line approach for spent fuel assay based upon neutron detection with BF_3 counters has just been published.⁽⁴⁾ However this latter method can only handle small quantities of fuel and so does not classify as a true nondestructive technique. It is amusing to note that a recent review implies that passive neutron dosimetry for this particular application is not feasible.⁽⁵⁾ Although this review is a general evaluation of nuclear waste assay techniques and does not concentrate on spent fuel assemblies per se, apparently these reviewers were unaware of the unique characteristics of SSTR. In contrast with this recent review, it has been discovered that the use of SSTR in spent fuel assay was actually suggested some years ago.⁽⁶⁾ However, this suggestion by the Hungarian SSTR group provides no specific details on the means to carry out such experiments, and no record of actual follow-on experiments by this group has been uncovered to date.

3.0 SSTR SCOPING EXPERIMENT WITH TURKEY POINT SPENT FA

SSTR dosimeters were exposed at the surface of a spent fuel FA from the Turkey Point Reactor No. 3, a Pressurized Water Reactor (PWR) located in Florida City, Florida. The characteristics of the spent fuel FA are summarized in Table 3.1.

TABLE 3.1
CHARACTERISTICS OF SPENT FA FROM TURKEY POINT 3

Serial Number	B43
Discharge Date	12/12/75
Average Thermal Power	13.96 MW/Assembly
Operating Power Density	82.6 kW/ft (Core Average)
FA Linear Power Density	5.56 kW/ft (Core Average)
Burnup	
Peak	28,564 MWd/MTU
Average	25,665 MWd/MTU
Fuel Length	144 in.
Fuel Pellet Diameter	0.380-in.
Fuel Rod Diameter	0.422-in.
FA Array	15 x 15*
Fuel Composition	2.559 wt% ²³⁵ U
	UO ₂ , 92% theoretical density

*Since each FA contains 20 control rods and one instrument tube, there are only 204 rods per FA.

Figure 3.1 is a photograph of this FA through the hot cell window at the Battelle (Columbus) facility, where the exposures were conducted. A schematic drawing of the SSTR dosimetry measurements is depicted in Figure 3.2,

FIGURE 3.1. A Spent Fuel LWR FA from the Turkey Point 3 Reactor as Photographed Through a Hot Cell Window at the Battelle (Columbus) Facility. Neg 7807339-19cn

FIGURE 3.2. Location of SSTR Capsules for the Scoping Experiment. Neg 7813171-4

which reveals the FA locations chosen for the three dosimetry capsules that were exposed. A more detailed breakdown of these SSTR dosimetry capsules is given in Figure 3.3. The actual SSTR loadings for these three dosimetry capsules are summarized in Table 3.2 and Figure 3.4.

TABLE 3.2
CAPSULE LOCATIONS OF SSTR DOSIMETERS

Capsule No.	SSTR Deposit Location		
	1	2	3
1	^{238}U	^{232}Th	^{237}Np
2(a)	^{235}U	CN + CR-39	CR-39
3(b)	^{235}U	CN + CR-39	^{237}Np

(a) CN and CR-39 are plastic track recorder materials that were used to explore the possibility of measuring ^1H , ^6Li , and ^{10}B reaction rates.

(b) In this exposure, all SSTR were cadmium-covered.

Mica SSTR were used with fission deposits of ^{232}Th , ^{235}U , ^{238}U and ^{237}Np . Plastic SSTR, namely Cellulose Nitrate (CN) and CR-39 (a relatively new SSTR candidate)⁽⁷⁾ were included to explore the possibility of obtaining data from the following reactions:

(n,p) on hydrogen

(n, α) on ^6Li

(n, α) on ^{10}B

Original plans for these scoping studies called for three separate exposures of roughly 3-1/3 days for each SSTR dosimetry capsule, so that measurements could be sequentially carried out at the same FA location. However, shipping delays, hot cell operating schedules, and time schedule requirements for follow-on experiments (see Section 4.0) did not allow the luxury of the original 10-day scoping experiment forecast. Consequently, a

FIGURE 3.3. Detailed View of SSTR Neutron Dosimetry Capsules for the Scoping Experiment. Neg 7813171-1

FIGURE 3.4. SSTR and Fissionable Deposit Loading Diagram for the Turkey Point Spent Fuel Scoping Exposures. Neg 7813171-3

single exposure of roughly five days was carried out simultaneously with all three dosimetry capsules. The actual time of this single exposure extended from 10:50 a.m. on August 31, 1978 until 7:55 a.m. on September 5, 1978. The specific locations of dosimetry capsules 1, 2, and 3 have already been depicted in Figure 3.2.

4.0 EXPERIMENTAL RESULTS

Data obtained from the mica SSTR are summarized in Table 4.1. Note that the track density is quite acceptable for all the mica SSTR exposures. Overall experimental error in these fission rates is approximately 10% (1σ), which is primarily due to track counting statistics with a small contribution from the uncertainty in deposit mass or asymptotic sensitivity.⁽⁸⁾

It is interesting to note that the ^{237}Np data from Capsules 1 and 3 lie outside one standard deviation, but agree within a two standard deviation error bound. Thus a high probability exists that this observed difference is due to a real effect, rather than merely the result of statistical uncertainty.

Two apparent explanations of this difference in ^{237}Np fission rate can be related directly to the scoping exposure configuration. Since dosimeters in Capsule 3 were cadmium-covered, sub-threshold fission in ^{237}Np below the cadmium cut-off energy (~ 0.5 eV) could be responsible for the observed difference. On the other hand, Capsules 1 and 3 were exposed on different faces of the FA, in fact on faces A and D, respectively, as shown in Figure 3.2. Further inquiry revealed that face A was closer to core center than face D during the last in-core irradiation cycle of FA B43. Consequently, the most recent burnup for face A should be slightly higher than that for face D. This suggests that passive SSTR neutron dosimetry could be sensitive enough to reflect the previous history of a given FA, which has undergone a specific reactor fuel assembly management plan.

While quantitative data was not obtained from any of the plastic SSTR included in this scoping experiment, tracks were definitely observed in CR-39. Figure 4.1 presents a microphotograph of the CR-39 dosimeter exposed in location 2 of Capsule 3. Although the surface has been damaged by the intense gamma-ray component, $\sim 10^7$ Rad, tracks are clearly evident. These tracks are most likely ^7Li nuclei generated from the $^{10}\text{B}(n,\alpha)$ reaction. Further study is required in order to apply this relatively new SSTR material

TABLE 4.1

SCOPING EXPERIMENT RESULTS FOR MICA SSTR

<u>Dosimetry Position</u>		<u>Isotope</u>	<u>Deposit Density^(a)</u>	<u>Track Density^(b)</u>	<u>Fission Rate^(c)</u>
<u>Capsule</u>	<u>Location</u>				
1	1	^{238}U	Asymptotic ^(e)	4.37×10^3	1.01 (0.398)
1	2	^{232}Th	Asymptotic ^(e)	1.34×10^3	0.322 (0.124)
1	3	^{237}Np	132	1.45×10^3	11.6 (4.55)
2	1	^{235}U	Asymptotic ^(e)	6.89×10^4	16.1 (6.27)
3	1(d)	^{235}U	Asymptotic	2.90×10^4	6.77 (2.64)
3	3(d)	^{237}Np	133	1.02×10^3	8.07 (3.18)

(a) Units of $\mu\text{g}/\text{cm}^2$

(b) Units of tracks/ cm^2

(c) Units of fissions/ μg ; the number in parenthesis is the fission rate in fissions/atom $\times 10^{15}$

(d) Cadmium-covered SSTR exposures

(e) The asymptotic sensitivity for ^{238}U , ^{232}Th and ^{235}U is 1.0981×10^{19} atoms/ cm^2 , 1.0841×10^{19} atoms/ cm^2 , and 1.0985×10^{19} atoms/ cm^2 , respectively.⁽⁸⁾

FIGURE 4.1. Microphotograph of Tracks Observed in the CR-39 Type SSTR from the Scoping Experiment Exposure at Location 2 of Capsule 3. Neg 7811613-1

for quantitative neutron dosimetry. To this end, such characteristics as efficiency, sensitivity, angular dependence, uniformity, ... etc. must be determined. Investigation of these characteristics is currently in progress.

Absolute fission rates obtained from these SSTR measurements have been used as input data for the SAND-II computer code⁽⁹⁾ to derive absolute neutron fluxes. Assumptions utilized for the initial trial neutron spectrum in these data analyses are: 1) a thermal Maxwellian spectral form with a temperature of 90°C was assumed for the energy region below about 0.4 eV; 2) a 1/E intermediate slowing down spectrum was assumed for the 0.4 eV to 0.5 MeV region; 3) a Maxwellian fission form with a temperature of $T = 1.29$ MeV was assumed above 0.5 MeV. The SAND-II procedure adjusted these three segments of the spectrum to give absolute integral flux results consistent with the measured bare and cadmium-covered SSTR absolute fission rates.

Table 4.2 summarizes the SSTR based fission rates used as input data for the SAND-II code. The cadmium-covered ^{235}U fission rate in Table 4.2 was obtained by scaling the face D result to face A by using the ^{237}Np face A-to-face D fission rate ratio. Based on the data cited in Table 4.1, this ratio is 1.44 ± 0.2 . This value of cadmium-covered ^{235}U fission rate gave the most consistent output results from the SAND-II code.

TABLE 4.2
ABSOLUTE SSTR FISSION RATES

<u>Isotope</u>	<u>Fission Rate*</u>
^{232}Th	2.94×10^{-22}
^{235}U (bare)	1.49×10^{-20}
^{235}U (Cadmium-covered)	8.97×10^{-21}
^{237}Np	1.08×10^{-20}
^{238}U	9.44×10^{-22}

*Units of fission/(atom x sec)

Table 4.3 presents the absolute neutron data generated from the SAND-II code. Overall flux error estimates depend not only upon the error in the SSTR absolute fission rate data, but also upon uncertainties entailed in the unfolding process, i.e., cross sections, initial spectrum, degree of convergence..., etc. In addition, flux perturbation effects due to the finite thickness of the SSTR fission foils and cadmium covers have been neglected in this treatment. Consequently, only estimated overall errors are given in Table 4.3 for these absolute group fluxes.

TABLE 4.3
ABSOLUTE GROUP NEUTRON FLUXES AT THE SURFACE OF SPENT FA B43

<u>Energy Interval</u>	<u>Flux n/(cm² x sec)</u>	<u>Estimated Uncertainty 1σ(%)</u>
Thermal (<0.4 eV)	14	50
Epithermal (0.5 eV - 0.1 MeV)	140	50
Fast (0.1 MeV - 2 MeV)	6520	30
High Energy (>2 MeV)	1230	20
Total Flux	7900	30
Mean Energy	1.3	15

It can be seen from Table 4.3 that the absolute intensity at the surface of this spent FA is still significant even three years after discharge. The energy distribution of these neutrons is quite hard in the hot cell environment, where measurements were carried out essentially in air. This approximates a vacuum boundary condition reasonably well, so that one expects very little neutron moderation and hence a very low intensity of slow neutrons.

The rather large neutron flux uncertainties given in Table 4.3 reflect the preliminary nature of this scoping experiment. While only ten percent track scanning statistics was justified in these initial efforts, future track scanning could be performed to obtain absolute fission rates of 1-3%

accuracy (at the 1σ level). Hence, considerable reduction in neutron flux uncertainty could be realized. It must be stressed, however, that one need not deduce neutron (total or group) flux for actinide assay work. In particular, either absolute isotopic fission rate or isotopic fission track density are correlation parameters of much higher accuracy than can be used for actinide assay work.

5.0 FUTURE WORK

5.1 SSTR DOSIMETRY FOR THE MATERIALS INTERACTION TEST (MIT)

The Materials Interaction Test (MIT) design objective is to obtain material science information on candidate geologic storage media in prototypic temperatures and irradiation fields during 3- to 5-year irradiation. The MIT will investigate materials' compatibility, structural property change, and chemical transformations. One of the test purposes is to determine changes in the actinide content of the fuel during the storage period.

The test train for this experiment will cover all but four inches of the fueled region of the assembly. Figure 5.1 shows the arrangement of the test capsules. As illustrated, capsules II, IV, and VI contain SSTR dosimeters. Fission deposit masses for these SSTR dosimeters were based upon track data obtained from this scoping experiment. A photograph of the SSTR dosimetry subcapsule is shown in Figure 5.2. Four dosimeters, ^{232}Th , ^{235}U , ^{238}U and ^{237}Np , are used in each SSTR subcapsule. Figure 5.3 provides a detailed view, with specific dimensions for both SSTR dosimeters and subcapsules.

Table 5.1 gives the actual fission deposits used in the SSTR subcapsules. Dosimetry subcapsules II and VI are used in test train capsules II and VI, respectively, while dosimetry subcapsules IV-1 and IV-2 are used at the upper and lower end of the test train capsule IV, respectively. At the completion of the 3- to 5-year irradiation, these in-situ SSTR dosimeters will be retrieved, processed, and scanned manually to provide neutron dosimetry data for the MIT experiment.

5.2 RECOMMENDATIONS FOR FOLLOW-ON HOT CELL EXPERIMENTS

More definitive follow-on SSTR neutron dosimetry hot cell exposures have been planned. Axial SSTR measurements will be carried out using ^{235}U (bare and cadmium-covered) ^{238}U , ^{232}Th and ^{237}Np . At least two different faces

20

CAPSULES WITHIN TEST TRAIN

FUELED REGION 144"

	DOSIMETER*	TEMPERATURE* MONITOR	TEST NAME	MATERIAL TESTED	CAPSULE LENGTH INCHES
I	_____		STRUCTURAL PROPERTY	BASALT	38.1
II		_____	CHEMICAL TRANSFORMATION	BASALT MINERAL ARGILLITE	10.9
III	_____	_____	FLOODED REPOSITORY	BASALT METAL	11.7
IV			MATERIALS COMPATIBILITY	BASALT GRANITE ARGILLITE METAL	20.1
V	_____	_____	FLOODED REPOSITORY	GRANITE METAL	11.7
VI		_____	CHEMICAL TRANSFORMATION	GRANITE MINERAL WELDED TUFF	10.9
VII	_____		STRUCTURAL PROPERTY	GRANITE	38.1
*DOSIMETERS OR TEMPERATURE MONITORS HAVE BEEN INCORPORATED ONLY WITHIN THOSE CAPSULES DENOTED BY SHADED AREAS.					

FIGURE 5.1. MIT Capsule Contents and Relative Position within the Test Train. Neg 7813171-2

FIGURE 5.2. Photograph of the SSTR Neutron Dosimetry Capsule for the MIT Experiment. Neg 7811071-1cn

CYLINDER $0.349" \pm 0.005"$ O.D. $\times$ $0.350" \pm 0.005"$ LONG
 $\times$ $0.319" \pm 0.005"$ I.D. $\times$ $0.060" \pm 0.005"$ BASE

LID $0.317" \pm 0.005"$ O.D. (SLIP FIT INTO CAN) $\times$ $0.287"$
 $\pm 0.005"$ I.D. $\times$ $0.060" \pm 0.005"$ BASE $\times$ $0.150" \pm 0.005"$ LONG

FIGURE 5.3. SSTR Dosimetry Subcapsure for the In-situ 3- to 5-Year Irradiation in a Spent Turkey Point FA. Neg 7813171-5

TABLE 5.1
LOADING DIAGRAM FOR MIT DOSIMETRY SUBCAPSULES

<u>Dosimetry Subcapsule</u>	<u>SSTR Fission Deposits*</u>	
Subcapsule IV-1	^{237}Np	29.2
	^{238}U	106.8
	^{235}U	1660.0
	^{232}Th	172.0
Subcapsule IV-2	^{237}Np	18.3
	^{238}U	99.1
	^{235}U	951.0
	^{232}Th	101.0
Subcapsule II	^{237}Np	14.6
	^{238}U	92.4
	^{235}U	455.0
	^{232}Th	84.0
Subcapsule VI	^{237}Np	9.18
	^{238}U	31.8
	^{235}U	91.0
	^{232}Th	64.8

*Deposit masses in units of $\mu\text{g}/\text{cm}^2$, except for ^{235}U which is in units of ng/cm^2 .

of a spent fuel Turkey Point FA will be investigated simultaneously. This will allow a more quantitative evaluation of the possible differences in burnup experienced by different faces of the same FA. A hypothesis tentatively advanced in Section 4.0, on the basis of the scoping experiment data.

Present plans call for nine dosimetry locations. These dosimetry locations are depicted in Figure 5.4. Six SSTR dosimeters will provide an axial distribution on one FA face using one-foot spacing from spent fuel midplane to end of the fuel. The remaining three SSTR dosimeters will be simultaneously exposed on an adjacent FA face at every other axial position to observe any possible difference in neutron emission from adjacent FA faces, and thereby assess the hypothesis that burnup differences can exist for different spent FA faces.

The in-situ temperature of the MIT long-term exposure is estimated at roughly 300°C. At such temperatures, the use of plastic SSTR is precluded as is the use of Thermoluminescent Dosimetry (TLD) for gamma dose measurements. Consequently, a more limited SSTR dosimetry effort with plastic track recorders will be pursued in these follow-on hot cell experiments. The extent of this effort will be based, in part, on the progress effected in establishing absolute neutron dosimetry with selected plastic SSTR candidates.

Since gamma rays are the dominant component of the spent fuel radiation field, TLD will be used for hot cell measurements of this component. Hence, TLD gamma dosimeters will also be exposed at all nine positions used for neutron dosimetry. In addition, TLD dosimeters will be simultaneously exposed in a control rod guide tube of a Turkey Point spent fuel FA. In these latter exposures, a mock-up of the MIT test train will be used so that in-situ gamma doses can be measured. These data will provide the initial point gamma dose distribution at the beginning of the long-term MIT experiment. Known decay characteristics of spent fuel radioisotopes can then be used to generate the time-dependent gamma dose over the course of the long-term 3- to 5-year MIT irradiation.

FIGURE 5.4. Location of SSTR Neutron Dosimetry Capsules for the Follow-On Hot Cell Experiments.
Neg 7813171-6

5.3 RECOMMENDED CALIBRATION EXPERIMENTS

Application of this SSTR neutron dosimetry method for nondestructive actinide assay requires calibration through a limited number of destructive experiments. For such calibration experiments, spent fuel pellets can be obtained directly from that FA used in the follow-on hot cell dosimetry experiments (see Section 5.2). These pellets should be obtained as close as possible to each axial SSTR dosimetry location used in the follow-on hot cell irradiations. Polished metallurgical specimens will be prepared from each pellet for use in SSTR and radiochemistry analyses.

Prior to radiochemical analysis, mica SSTR will be exposed in firm surface contact with each fuel specimen to obtain the absolute spontaneous fission rate. Following these exposures, further irradiations in firm surface contact with these spent fuel specimens will be conducted with plastic SSTR to obtain the absolute alpha activity of the spent fuel. These two SSTR observations will permit independent estimates of the absolute source intensity of both spontaneous fission neutrons and (α ,n) generated neutrons.

Actually, these surface contact measurements serve three broad purposes:

- To offer a general method of spent fuel characterization.
- To provide the absolute source term for neutron dosimetry.
- To serve as a general measure of the actinide content of spent fuel by in-situ integration over all such isotopes.

From the standpoint of nondestructive methodology, the difficulty of SSTR surface contact measurements lies somewhere between passive SSTR neutron dosimetry and the complete destructive analysis required by most radiochemical techniques. On the other hand, these SSTR surface contact measurements provide data that can be simultaneously used for both neutron dosimetry and actinide characterization of spent fuel.

After these SSTR exposures, each fuel specimen will be quantitatively analyzed for actinides using radiochemical techniques, including mass spectrometry, low geometry alpha particle counting and alpha particle energy analyses. When necessary, isotopic separation of actinides will be performed. Observed actinide concentrations can then be correlated with SSTR neutron dosimetry data for each axial location employed. In this manner, calibration of this nondestructive SSTR neutron dosimetry assay method can be achieved over a limited range of fuel burnup.

State-of-the-art SSTR accuracy goals will be sought for absolute fission rate observations in these calibration experiments. For most isotopes of interest, SSTR state-of-the-art absolute fission rate accuracy lies in the 1-3% range (at the 1σ level). Comparable experimental accuracy will be available from the destructive radiochemical analyses performed for the most abundant actinide isotopes of the spent fuel. On this basis, an accurate calibration can be forecast.

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