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APPLICATIONS OF SOLID STATE TRACK RECORDERS IN UNITED STATES NUCLEAR REACTOR ENERGY PROGRAMS

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APPLICATIONS OF SOLID STATE TRACK RECORDERS IN UNITED STATES NUCLEAR REACTOR ENERGY PROGRAMS†

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ABSTRACT

The domain of Solid State Track Recorder (SSTR) applications in United States nuclear reactor energy programs extends from the harsh high temperature environments found in high power reactor cores to very low flux environments arising in out-of-core locations, critical assemblies, or away from reactors (AFR) experiments. The neutron energy region arising in these applications is very broad, covering more than eight decades from thermal up to fusion energies. The range of neutron flux/fluence intensity is even greater, extending over more than thirteen decades. As a consequence, use of a variety of SSTR is entailed in U.S. Fast Breeder Reactor (FBR), Light Water Reactor (LWR), and Magnetic Fusion Energy Reactor (MFER) programs. A summary status is presented of selected SSTR experiments undertaken in these programs at the Hanford Engineering Development Laboratory (HEDL).

I. INTRODUCTION

High accuracy goals, typically 1-3% (1σ), have been established for in-situ measurement of nuclear parameters in U.S. reactor energy programs. These high accuracy goals have not been arbitrarily defined. The economics of nuclear power demands optimized reactor design. Safe and reliable reactor operation requires highly accurate knowledge of the spatial dependence of the neutron flux and key nuclear reactions, such as isotopic fission rates.

These high accuracy goals are not easily attained, as is attested to by the cost and effort already expended. A diversity of experimental and calculational procedures is required to achieve these goals. No single technique can adequately cover both the energy and intensity ranges of interest. Moreover the need for

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extensive basic nuclear data, such as cross sections, obligates a multi-method approach. Within this context, selected contributions of SSTR techniques to United States nuclear reactor energy programs will be described. Efforts in FBR, LWR, and MFER programs are discussed separately in sequel. Conference space and time allotments permit only a general review of selected applications, and in fact those applications that have been singled out can only be briefly touched upon. The reader may, however, consult the specifically cited references for further details.

Applications of SSTR in reactors arise in three broad power ranges: (1) Very Low Power (VLP), which is typical of critical assembly work; (2) High Power (HP), which is representative of the in-core region of power reactors; (3) Low Power (LP), which is an intermediate power range between critical assembly and high power reactor environments. The HP needs of reactor energy programs are the most urgent. In HP environments, where high fluence (HF) irradiations arise, the limitations of SSTR must necessarily be defined. To this end, emphasis has been placed on mica and natural quartz crystals, which are outstanding SSTR candidates for accurate HF applications. Three major effects which can limit SSTR-HF applicability are: (1) annealing, (2) radiation damage, and (3) background track production. It must also be noted that the existence of track pile-up produces accuracy limitations at higher track densities. Each of these effects can play a significant role in defining the high fluence limit of applicability of a given SSTR. It must be stressed that these different effects act in consort rather than independently to produce a high fluence limit. For example, radiation damage of the crystal lattice produced by a high fluence of fast neutrons in a given SSTR can alter SSTR annealing characteristics dramatically. Hence, track fading due to annealing can be considerably enhanced due to the damage of the SSTR crystal lattice that is produced at high fluence. In this regard, initial efforts to define the domain of applicability of SSTR for HP reactor environments have already been described[1,2] and further work on annealing effects will be presented in a companion paper at this conference.[3]

Although these HF effects are not actually separable, our work to date suggests that radiation damage of the crystal lattice is the most serious HP limitation of SSTR. However, the existence of this specific limitation for HF-SSTR fission rate observations is not a complete deficit. Indeed, it is on this very basis that the concept of the SSTR radiation damage monitor has been advanced. In HF applications, a number of SSTR attributes have exhibited significant fluence dependence. For minerals, SSTR characteristics depend intimately upon crystal lattice structure. Hence, for sufficiently high neutron fluence, radiation damage of the lattice alters SSTR properties. In particular, track registration as well as bulk and track etch rates undergo substantial change. Consequently, for HF irradiations, any of these SSTR observables afford a unique measure of radiation damage.

II. FAST BREEDER REACTOR (FBR) APPLICATIONS

A. Characterization of the Fast Test Reactor (VLP, LP, HP)

In FBR work, extensive use of SSTR is planned for key experiments in the Fast Test Reactor (FTR) at startup. This 400 Mw FBR, located on the Hanford reservation, has been built by the U.S. Department of Energy (DOE) for irradiation testing and development of FBR fuels and materials. It is scheduled

for operation later this year. FTR will provide the means to develop and operationally test FBR components and to gain operating experience for future FBR.

A reactor characterization program has been planned and scheduled for FTR startup(4), which consists of VLP, LP, and HP irradiations. SSTR experiments at VLP will concentrate on absolute fission rates of the highest possible accuracy(5) using both vacuum evaporated and electrodeposited fission sources. In-situ comparisons and intercalibrations will be carried out with absolute NBS fission chamber(6) at core center using axial locations near midplane and approximately 40 cm below midplane. In the LP and HP irradiations, SSTR experiments will focus on fission product yields and integral fission rates for power level determination and neutron dosimetry. For the measurement of fission product yields, SSTR are used in conjunction with the radiometric fission samples which are then quantitatively analyzed by absolute beta or gamma counting after the irradiation.(7)

Dosimetry capsules are designed to fit inside of fuel pins in the FTR core matrix. A typical LP capsule is shown in Figs. 1(a,b). SSTR observations will generally be conducted in empty fuel pins, with the exception of a single experiment that has been designed to evaluate the perturbation created due to measurements in empty fuel pins. LP and HP measurements will be made at various locations throughout the FTR core with the following isotopes: ^{232}Th, ^{233}U, ^{235}U, ^{238}U, ^{237}Np, ^{239}Pu, ^{240}Pu and ^{241}Pu.

Fast fluence estimates for the LP and HP exposures are 10^{18} and 10^{21} neutrons/cm^2, respectively. As a consequence, LP and HP fission deposits have been prepared by electrodeposition and lie in the sub-nanogram/cm^2 regime. Fission deposits of such low mass have never before been quantitatively prepared for SSTR work. Deposit masses have been determined to better than 1% and variations in uniformity are generally less than 10%. For the LP irradiation at 400°F, mica SSTR have been incorporated and track densities of the order of 10^5/cm^2 are anticipated so that manual scanning techniques can be used. In the HP irradiation at 700-1000°F, natural quartz crystal (100 plane) SSTR have been incorporated and track densities of the order of 10^8/cm^2 are anticipated so that scanning electron microscopy (SEM) techniques will be necessary.

B. Criticality Safety Studies

1. Critical Assembly Experiments (VLP)

As part of the U.S. Advanced Fuel Recycle Program, critical mass measurements have been made of several lattices consisting of mixed oxide (PuO_2-UO_2) FTR fuel pins in water at selected inter-rod spacings. In addition to providing a reliable technological basis for nuclear criticality control, these criticality safety studies are designed to provide benchmark data for validation of cross sections and calculational techniques used for mixed plutonium and uranium systems. In order to characterize some of the lattices more thoroughly, spatial fission rate distributions were measured using SSTR.(8) These fission rate traverses cannot only be compared with calculated shapes, but since SSTR observations provide absolute fission rates, the data were also used as input to multiple-foil unfolding codes, such as SAND-II(9) to derive adjusted neutron spectra. The extremely high sensitivity of the SSTR method affords in-situ dosimetry for critical mass measurements with negligible spatial and spectral perturbations.

SSTR measurements were carried out using mica track recorders with isotopic fission deposits of ^{235}U, ^{238}U, ^{232}Th, and ^{237}Np. To measure spatial and isotopic fission rates simultaneously, deposit thicknesses were adjusted using thin deposits of negligible fission fragment self-absorption as well as asymptotically thick fission foils. The sensitivity of asymptotic fission foils is independent of the thickness and has been previously determined.(5,10)

An overall error of roughly 3-5% (1σ) was attained for SSTR absolute fission rate observations in these critical safety studies. These SSTR measured fission rates were input to the SAND-II code together with an initial spectrum calculated with the KENO Monte Carlo code.(11) The SAND-II code produced an adjusted calculated neutron spectral shape which agreed with the measured data. Results from two critical assembly exposures at midplane are shown in Table 1 together with SAND-II calculated neutron fluxes, which possess uncertainty estimates of 10-15% (1σ).

TABLE 1

CRITICALITY SAFETY MID-LATTICE FISSION RATES AND GROUP NEUTRON FLUXES

Lattice Pitch	0.77 cm	1.90 cm
Relative Fission Rates: (fissions/atom-sec)		
^{235}U (n,f)	1.28×10^{-15}	2.28×10^{-15}
^{238}U (n,f)	1.19×10^{-17}	2.00×10^{-18}
^{232}Th (n,f)	3.60×10^{-18}	4.00×10^{-19}
^{237}Np (n,f)	8.52×10^{-17}	9.50×10^{-18}
Relative Flux (n/cm^2-sec)		
Thermal	1.85×10^{6}	4.31×10^{6}
E > 0.1 MeV	9.04×10^{7}	9.32×10^{6}
E > 0.1 MeV	3.48×10^{7}	4.48×10^{6}
Total	1.25×10^{8}	1.92×10^{7}
Average Energy (MeV)	0.92	0.70

The extent of modifications in calculated spectra may be inferred by a comparison of measured and calculated ^{235}U (n,f) to ^{238}U (n,f) ratios. For the 0.77 cm pitch lattice this ratio was measured to be 107 as compared with the calculated (KENO) value of 119, an 11% difference. For the 1.90 cm pitch these were 1140 measured and 1236 calculated, an 8% difference. At other lattice positions, comparable differences were observed. As indicated by this one spectral index, therefore, the KENO calculations produce a reasonable approximation to the neutron spectrum. However, improvements in flux-spectrum values can be obtained using SSTR observations in unfolding codes, such as SAND-II, which may then be used for comparisons with lattice calculations or in perturbation calculations to derive effects of poison or geometry changes.

The results of this program to date have shown that accurate SSTR measurements of fission rates can be made in critical mass facilities at very low exposure levels. Such measurements when combined with the multiple foil technique have great potential for use as additional benchmark data for improvement and validation of calculational techniques.

2. Fuel Assembly Storage Facility Measurements (AFR)

In a comparable set of AFR safety experiments, SSTR were successfully employed to investigate the criticality safety of an FTR fuel assembly (FA) storage facility. Prior to actual use in FTR, these FA must be safely stored. A typical storage facility for unused reactor FA consists of a rectangular array of cylindrical retaining pits in a concrete medium, with each pit large enough to house a separate FA. For the FTR storage facility, center-to-center distances between FA of 15 cm and 30 cm were used for the two orthogonal axes of this rectangular FA array, respectively. For such sub-critical arrays, a very simple relation exists between the criticality factor, k_{eff}, and the neutron multiplication M, namely

$$k_{eff} = 1 - M^{-1} \tag{1}$$

The criticality factor for the storage facility can therefore be determined from the multiplication measurements which can, in turn, be obtained from SSTR observations at the surface of the FA. Here, however, only the fast neutron component Φ_f (i.e., neutrons above roughly 0.1 MeV) should be used, for it is only the fast neutron component that is directly correlated with the fission process. Hence, one can write

$$M = \Phi_f(\text{in-situ})/\Phi_f(\text{source}), \tag{2}$$

where Φ_f(in-situ) is the representative fast neutron flux within the storage facility and Φ_f(source) is the fast neutron source spontaneously generated in the fuel without any multiplication.

Asymptotically thick ^{235}U and ^{238}U fission foils were employed with mica SSTR for absolute fission rate measurements in the FTR storage facility. Exposures of about two days at the FA surface provided sufficient track density to determine absolute fission rates with roughly 10% (1σ) uncertainty. Absolute fluxes were derived by using these absolute SSTR fission rates as input data in the SAND-II computer unfolding code.[9] On this basis, an average axial FA value of 930 neutrons/(cm^2 · sec) was found for Φ_f (in-situ). To determine Φ_f(source) similar FA surface measurements were conducted with SSTR attached to isolated FA suspended in air. Under such conditions, return neutrons are negligible and SSTR observations should provide a first approximation of Φ_f(source). In this manner, an average axial FA value of 670 neutron/(cm^2 · sec) was found for Φ_f(source). Using these results in Equations (1) and (2), one finds a criticality factor of $k_{eff} \approx 0.28$.

Some concern existed that this criticality factor value was too conservative owing to the neglect of self-multiplication in a single isolated FA. First-flight fission probabilities were calculated for an isolated FA which revealed that the fast neutron population could be increased by at least 15% due to such a self-multiplication effect. Consequently, the spontaneous fission rate in FTR fuel was measured directly with SSTR. The ends of FTR fuel pellets were polished to provide smooth, flat surfaces for contact with mica SSTR. An exposure period of about two days provided fission track densities in excess of 10^5/cm^2. An asymptotic sensitivity of 3.0 mg/cm^2 was calculated(10) for the mixed plutonium oxide-uranium oxide FTR fuel, with an uncertainty estimated to be about 10% (1σ). This latter uncertainty contribution is the dominant overall uncertainty in this FTR spontaneous fission observation.

These spontaneous fission observations led to a value of 475 neutrons/(cm^2·sec) for Φ_f(source), which includes a small contribution from neutrons generated by (α,n) reactions in the fuel. Using this result, one finds a criticality factor of $k_{eff} = 0.5 \pm 0.1$ for the FTR storage facility. (Actually for an isolated FA, the self-multiplication comes to M = 1.4, which corresponds to an isolated FA criticality factor of $k_{eff} \approx 0.29$). Monte Carlo analyses using KENO II and GEM IV computer codes of storage facilities[12,13] similar to the FTR facility have yielded a value of $k_{eff} = 0.66$. Moreover, these analyses are expected to be somewhat higher since fuel of 31 weight percent plutonium was assumed for the Monte Carlo calculations, whereas SSTR observations were conducted with FTR Core-1 fuel, which is only 24.2 weight percent plutonium.

III. LIGHT WATER REACTOR (LWR) APPLICATIONS

A. Pressure Vessel Surveillance Program

Aging light water reactor pressure vessels (LWR-PV) are accumulating significant neutron fluence exposures, with consequent changes in their steel embrittlement characteristics. Recognizing that accurate and validated measurement methods are needed to periodically evaluate the metallurgical condition of these reactor vessels, the U.S. Nuclear Regulatory Commission has established the LWR-PV Surveillance Dosimetry Improvement Program. The major benefit of this program will be a significant improvement in the accuracy of the assessment of the remaining safe operating lifetime of light water reactor pressure vessels. The primary concern of this program will be to improve, standardize, and maintain neutron dosimetry and damage analysis procedures used for predicting integrated damage exposure of LWR pressure vessels. A vigorous research effort attacking the same measurement problems exists world-wide and cooperative links with the NRC supported activity have been established.[14-16]

Neutron dosimetry validation and calibration will be conducted in a variety of neutron irradiation test facilities, including LWR-PV mockups, power reactor surveillance positions, and related benchmark neutron fields. The key benchmark fields will be two "controlled environment" PV mockups, namely the Pool Critical Assembly (PCA) and the Pool Side Facility (PSF) at the Oak Ridge Reactor (ORR). The necessity for a pressure vessel mockup facility for dosimetry investigations and for irradiation of metallurgical specimens was recognized early in the formation of the NRC program. Low-flux and high-flux versions, i.e., PCA and PSF, of a single pressure vessel mockup have been constructed, as shown schematically in Figure 2. As a specialized benchmark, this facility will provide a well-characterized neutron environment where active and passive neutron dosimetry, various types of LWR-PV neutron field calculations, and temperature-controlled PV damage exposure are brought together for validation. SSTR work in each of these mockups is discussed separately below.

1. Critical Assembly Studies (VLP)

Dosimetry measurements in the PCA have been underway for about a year now. In the very first SSTR experiments, "free-field" scoping studies were conducted. The term "free-field" signifies an unperturbed water environment, i.e., without the presence of the thermal shield and pressure vessel simulator. After these initial scoping experiments, further SSTR irradiations were carried out with the PV mockup installed. Emphasis was placed on the following radial locations: one-quarter, one-half, and three-quarters of the way into the PV simulator and in the void box behind the pressure vessel simulator.

Absolute fission rates of ^{232}Th, ^{235}U (bare and Cd-covered), ^{237}Np, and ^{238}U were obtained with mica SSTR. Fission deposits in the $\mu g/cm^2$ region or asymptotically thick foils were employed. In most instances, track densities are quite adequate and experimental error lies in the 5-10% range (1σ). A series of 15 irradiations were conducted resulting in more than 160 exposed mica SSTR. Consequently, analysis of these experiments awaits complete scanning of all of these mica SSTR dosimeters.

In the upcoming PCA irradiations, SSTR will be used for perturbation evaluation studies. In the LWR-PV environment, voids created in the water by the need to insert active spectrometers, fission chambers, or surveillance dosimetry capsules can produce serious neutron field perturbations, not only in the measurements themselves, but in neighboring locales. The advantages of miniaturized SSTR together with microscopic readout techniques will be capitalized upon in evaluating these perturbations and minimizing interactions introduced by necessarily much larger detection systems. For these reasons, fission rate observations at the PCA will emphasize comparisons between SSTR and miniaturized fission chambers, rather than the absolute NBS fission chamber.

2. Accelerated Metallurgical Test (HP)

The PSF mockup will be operated for approximately one year to provide fast fluences in excess of 10^{19} $n/(cm^2 \cdot sec)$. Special features for the metallurgical testing will be temperature control, measured flux profiles and flux levels, and easy access and flexibility in changing the relative distances between the metallurgical capsules. This highly specialized mockup will contain PV steel materials for metallurgical tests such as Charpy, fracture toughness and tensile specimens, dosimetry capsules, and heating and cooling elements to obtain even and constant temperatures during irradiation. Because the PSF represents such a unique opportunity, metallurgists, radiation damage specialists and dosimetrists from all over the world are actively participating in this experiment.

In the PSF irradiation, SSTR capsules containing fissionable deposits of ^{226}Ra, ^{230}Th, ^{232}Th, ^{235}U, ^{238}U, and ^{237}Np have been incorporated. Two different dosimetry capsules called advanced and exploratory capsules are employed. There will be ten advanced capsules which will be used chiefly for dosimetry comparisons with radiometric monitors and helium accumulation fluence monitors (HAFM). Advanced capsule locations will range from the surveillance position through the PV mockup block back into the void box region. There will be four exploratory capsules, two at the surveillance position and two at the surface of the PV mockup. These exploratory capsules will contain both fission rate SSTR dosimeters as well as SSTR damage monitors (DM).

B. Non-Destructive Assay of Spent LWR Fuel Assemblies (AFR)

On the basis of the demonstrated sensitivity of SSTR in FBR criticality safety studies, it became apparent that SSTR could be applied for the characterization of spent reactor fuel assemblies.[16-19] Clearly, such efforts possess high priority as spent fuel from LWR nuclear power stations continues to accumulate and it therefore becomes increasingly important to provide effective means for long-term storage or 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. The 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 non-destructive method is therefore highly desirable. Since spent fuel is handled, transported and stored in the form of fuel assemblies (FA), it is preferable to use non-destructive methods that can be applied directly to these assemblies. 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.

Attributes of the SSTR method for spent fuel applications are outstanding and cannot be overemphasized, namely cost effectiveness, ready availability, and direct applicability without the need for special facilities. Hence, these factors have permitted quick adaptation of the SSTR dosimetry in a spent FA scoping experiment. 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. Mica SSTR were exposed for roughly 3-1/3 days at the surface of a spent FA near fuel midplane. Fission deposits of ^{235}U (bare and Cd-covered), ^{238}U, ^{232}Th, and ^{237}Np were used.

Acceptable fission track density was produced in all mica SSTR. In fact, all track densities were in excess of $10^3/cm^2$, with the ^{235}U track densities exceeding $10^4/cm^2$. On this basis an overall experimental error of 10% (1σ), was obtained in the absolute SSTR fission rates, with the dominant uncertainty due to track counting statistics and a small contribution arising from the uncertainty in deposit mass or asymptotic sensitivity. Table 2 presents absolute fluxes obtained from the SAND-II unfolding code[9] using these absolute SSTR fission rate data. 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 2 for these absolute group fluxes.

It can be seen from Table 2 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 2 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 that can be used for actinide assay work.

TABLE 2

ABSOLUTE GROUP NEUTRON FLUXES AT THE SURFACE OF SPENT FA B43

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

IV. MAGNETIC FUSION ENERGY REACTOR (MFER) APPLICATIONS

In support of materials development for the MFER program, the United States Department of Energy (DOE) is constructing an intense neutron source at Hanford known as the Fusion Materials Irradiation Test (FMIT) facility. The FMIT facility will generate an intense source of high energy neutrons for the systematic study, evaluation, and development of fusion reactor materials. Since highest intensities can be realized from the Li(d,n) reaction[19], a prototype linear accelerator will provide a high current deuteron beam (~100 mA, 15-35 MeV), which will impinge on a target of flowing liquid lithium. Goal objectives for FMIT are a maximum flux intensity of 10^{15} neutrons/($cm^2 \cdot sec$) with a mean energy of 14 MeV. The unperturbed steady state neutron volume/flux objectives are approximately 10 cm^3 at 10^{15} $n/(cm^2 \cdot sec)$ and 500 cm^3 at 10^{14} $n/(cm^2 \cdot sec)$.

With these capabilities, FMIT will provide entry into a new realm of fusion reactor material testing. No irradiation facility yet built approximates the irradiation environment planned in FMIT, and full exploitation of this unique facility demands characterization of the irradiation environment to a degree consistent with MFER program accuracy requirements.[20] Dosimetry characterization of the FMIT material test assembly volume is complicated by

several factors: (1) large flux component of very high energy neutrons; (2) steep flux and energy spectrum gradients within the test volume; (3) highly directional nature of the neutron flux, as opposed to the essentially isotropic flux in reactors; (4) irregular production of secondary neutrons within material test assemblies; and (5) large sensitivity of the preceding factors to source instabilities.

Both mineral and plastic SSTR will play complimentary roles in quantifying the radiation environment of this unique facility. It is planned that passive dosimeters will be incorporated directly within the FMIT irradiation test assemblies. In-situ passive dosimeters will be mainly radiometric sensors with only an occasional SSTR dosimeter. Since FMIT cycle-to-cycle high energy fluences will be in excess of 10^{19} n/cm^2, mica and quartz crystal SSTR will be used chiefly with threshold fission nuclides. Within the FMIT test cells, SSTR dosimeters of this type will be employed to map the angular and spatial variation of the neutron field as well as to provide absolute time-integrated reaction rate data for input to unfolding codes.

Beyond the ability of mica and quartz crystal SSTR to furnish FMIT dosimetry data, these SSTR will simultaneously serve as high neutron energy damage monitors. In this latter capacity, SSTR may actually play a more important role for the MFER program. Since FMIT will be used for MFER radiation damage studies, it will serve mainly to verify and calibrate correlation models and procedures as well as to produce some test data. However, most of the radiation damage data will continue to come from fission reactor irradiations, where neutron spectra possess much lower energy, by an order of magnitude or more. To calculate and evaluate radiation damage, for example to the first wall of a fusion reactor, fission reactor and FMIT results will have to be incorporated in a correlation methodology. To do this requires more than an accurate knowledge of the fluence and neutron spectra in metallurgical test specimens that are irradiated. Indeed, one must have a proper physical understanding of the neutron-induced radiation damage processes. SSTR damage monitors can furnish data in both fission and fusion reactor environments and thus, in principle, provide insight for attempts to correlate energy-dependent radiation damage effects.

In this regard, effects of different neutron-induced Primary Knock-on Atom (PKA) spectra on radiation damage are not yet fully understood. It is known that PKA spectra induced in fusion reactor environments will be much harder that PKA spectra produced in fission reactor environments. Moreover, PKA spectra play a fundamental role in analytical descriptions of radiation damage. Hence, there is considerable incentive for observation of PKA spectra. In fact, SSTR measurements of PKA spectra have already been advanced.[1,21] SSTR of higher sensitivity such as glass or plastic would be required for such measurements. Low energy-high atomic number PKA detection with SSTR will probably entail special etching techniques as well as higher readout magnifications that are afforded through scanning electron microscopy.

Plastic SSTR will be used at FMIT for both neutron radiography and passive spectrometry. In particular, CR-39 possesses unique attributes for this work as can be surmised from calibration studies with this relatively new SSTR that are reported in a companion paper at this conference.[22] Special emphasis will be placed on neutron radiography of the Li target. The intensity distribution from this source (~ 1 cm by 3 cm) plays a key role in defining

the equilibrium neutron field throughout the test cell. For example, this distribution is essential boundary data in neutron transport calculations. In this respect, the intensity distribution of deuterons impinging on the Li target can also be used for neutron source intensity distribution estimates. However, sufficiently accurate techniques for beam profile measurements in this current regime ($\sim$ 100 mA) have not as yet been developed. Consequently, development of active beam diagnostic techniques are being pursued, wherein it is expected that passive SSTR methods will play a confirmatory role.

The remarkable energy-dependent proton sensitivity of CR-39 will be employed for passive nuetron spectrometry applications at FMIT. Throughout the neutron energy range that will exist at FMIT, i.e., up to 50 MeV, only the hydrogen (n,p) scattering cross section is accurately known. Using appropriate neutron collimators, hydrogenous radiators, and vacuum scattering chambers, the entire FMIT neutron spectrum can be observed with CR-39 placed at suitable (n,p) scattering angles. Hence, the proton energy sensitivity of CR-39, which extends up to at least 12 MeV, will furnish unique passive neutron spectrometry at FMIT on both an absolute and time-integrated basis.

REFERENCES

1. J. H. Roberts and R. Gold, "SSTR and Emulsion Techniques and Their Applications for FBR, LWR, and MFER Programs," Proceedings of the Second International ASTM-EURATOM Symposium on Reactor Dosimetry, Palo Alto, CA, October 3-7, 1977, NUREG/CP-0004, Vol. 2, p. 739.

2. R. Gold, J. H. Roberts, and F. H. Ruddy, "Solid State Track Recorder Materials for Use in Light Water Reactor-Pressure Vessel Exposures," Proceedings of the ASTM Ninth International Symposium on Effects of Radiation on Structural Materials, Richland, WA, July 11-13, 1978, (to be issued).

3. J. H. Roberts, R. Gold, and F. H. Ruddy, "Thermal Annealing of Tracks in Solid State Track Recorders," Tenth International Conference on Solid State Nuclear Track Detectors, Lyons, July 2-7, 1979.

4. J. W. Daughtry, R. A. Bennett, W. L. Bunch, W. N. McElroy, and T. L. King, "FFTF Reactor Characterization Program", Proceedings of the Second International ASTM-EURATOM Symposium on Reactor Dosimetry, Palo Alto, CA, October 3-7, 1977, NUREG/CP-0004, Vol. 1, p. 69.

5. R. Gold, R. J. Armani, and J. H. Roberts, "Absolute Fission Rate Measurements with Solid State Track Recorders," Nucl. Sci. Eng. 34, 13 (1968).

6. J. A. Grundl, D. M. Gilliam, N. D. Dudey, and R. J. Popek, "Measurements of Absolute Fission Rates," Nucl. Tech. 25, 237 (1975).

7. R. Gold, E. P. Lippincott, and W. N. McElroy, "FTR Measurement Plans for Interlaboratory Calibrations and Fission Product Yields," HEDL-TME 79-14 (1979).

8. E. P. Lippincott, R. Gold, F. H. Ruddy, C. L. Long, and J. H. Roberts, "SSTR Dosimetry in Critical Mass Measurements," Am. Nucl. Soc., 1979 Annual Meeting, Atlanta.

9. W. N. McElroy, S. Berg, T. B. Crockett, and R. J. Tuttle, "Measurement of Neutron Flux Spectra by a Multiple Foil Activation Iterative Method and Comparison with Reactor Physics Calculations and Spectrometer Measurements," Nucl. Sci. Eng., 36, 15 (1969).

10. R. Gold, "Mass and Charge Dependence of SSTR Asymptotic Sensitivity," Proceedings of the First International ASTM-EURATOM Symposium on Reactor Dosimetry, Petten, EUR 5667 (1977).

11. L. M. Petrie and N. F. Cross, Keno IV: An Improved Monte Carlo Criticality Program, ORNL-4938, Oak Ridge National Laboratory, Oak Ridge, TN (1975).

12. P. J. Hemmings, "The GEM Code," U.K.A.E.A. Authority Health and Safety Branch Report. (S) R105 (1967).

13. G. E. Whitesides and N. F. Cross, "Keno-A Multigroup Monte Carlo Criticality Program," Report CTC-5, ORNL (1969).

REFERENCES (Cont.)

14. W. N. McElroy, D. G. Doran, R. Gold, E. P. Lippincott, J. O. Schiffgens, R. L. Simons, W. C. Morgan, J. A. Grundl, E. D. McGarry, F. B. K. Kam, J. H. Swanks, and G. R. Odette, "Standardization of Dosimetry and Damage Analysis Work for U. S. LWR, FBR, and MFR Development Program," Proceedings of the Second International ASTM-EURATOM Symposium on Reactor Dosimetry, Palo Alto, CA, October 3-7, 1977, NUREG/CP-0004, Vol. 1, p. 17.

15. W. N. McElroy, R. Gold, G. L. Guthrie, L. S. Kellogg, E. P. Lippincott, J. A. Grundl, C. M. Eisenhauer, E. D. McGarry, F. B. K. Kam, L. F. Miller, F. W. Stallmann, C. Z. Serpan, A. Fabry, J. Debrue, G. DeLeeuw, G. Minsart, H. Tourwe, D. Pachur, W. Schneider, L. Weise, M. Austin, P. Burch, G. R. Odette, R. A. Wullaert, and S. L. Anderson, "Development and Testing of Standardized Procedures and Reference Data for LWR Surveillance, "I.A.E.A. Specialists' Meeting on "Irradiation Embrittlement, Thermal Annealing and Surveillance of Reactor Pressure Vessels," February 26-March 1, 1979, Vienna, Austria.

16. R. Gold, F. H. Ruddy, E. P. Lippincott, W. N. McElroy, and J. H. Roberts, "Neutron Emission Observed from Spent Thermal Reactor Fuel Assemblies," First Symposium on Safeguards and Nuclear Material Management, ESARDA, Brussels, April 25-27, 1979.

17. R. Gold, F. H. Ruddy, E. P. Lippincott, W. N. McElroy, and J. H. Roberts, "Neutron Dosimetry Characterization of Spent Thermal Reactor Fuel Assemblies," Am. Nucl. Soc., 1979 Annual Meeting, Atlanta.

18. R. Gold, F. H. Ruddy, E. P. Lippincott, W. N. McElroy, and J. H. Roberts, "Spent Thermal Reactor Fuel Assembly Characterization with Solid State Track Recorders," HEDL-TME 78-89.

19. H. H. Barschall, "Intense Sources of Fast Neutrons", in Annual Review of Nuclear and Particle Science, Annual Reviews Inc., Palo Alto, CA, 28, 207, (1978).

20. J. O. Stiegler and T. C. Reuther, "The Program for Alloy Development and for Irradiation Performance in Fusion Reactors-Review," Symposium on Neutron Cross-Section From 10 to 40 MeV," Brookhaven National Laboratory May 3-5, 1977, BNL-NCS-50681, (1977), p. 51.

21. R. Gold, "PKA Induced Equilibrium Spectra," Bull. Am. Phys. Soc. 22, 25 (1977).

22. E. V. Benton, C. C. Preston, F. H. Ruddy, R. Gold, and J. H. Roberts, "Proton and Alpha Particle Response Characteristics of CR-39 Polymer for Reactor and Dosimetry Applications," Tenth International Conference on Solid State Nuclear Track Detectors, Lyons, July 2-7, 1979.

FAST TEST REACTOR - REACTOR CHARACTERIZATION PROGRAM (FTR-RCP) SOLID STATE TRACK RECORDER (SSTR) CAPSULE ASSEMBLY LOW POWER TEST

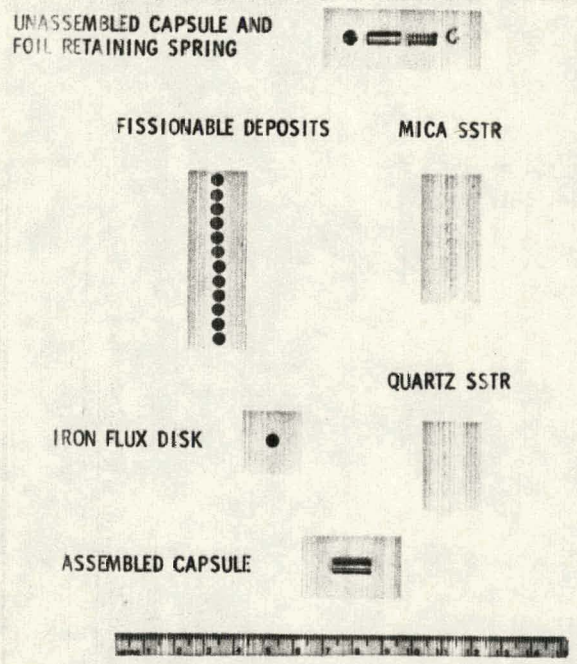

FIGURE 1a Contents of a typical LP SSTR dosimetry capsule.

FTR-RCP LOW POWER CAPSULE 1-6 CA 2101 ocm (D,6)

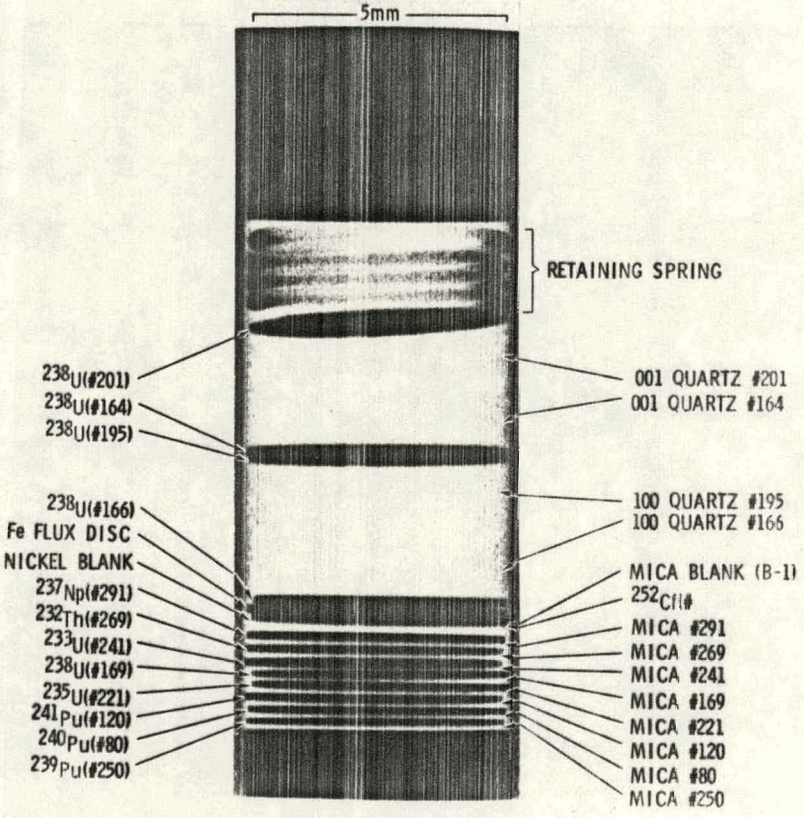

FIGURE 1b Radiograph of a typical LP capsule after assembly and welding.

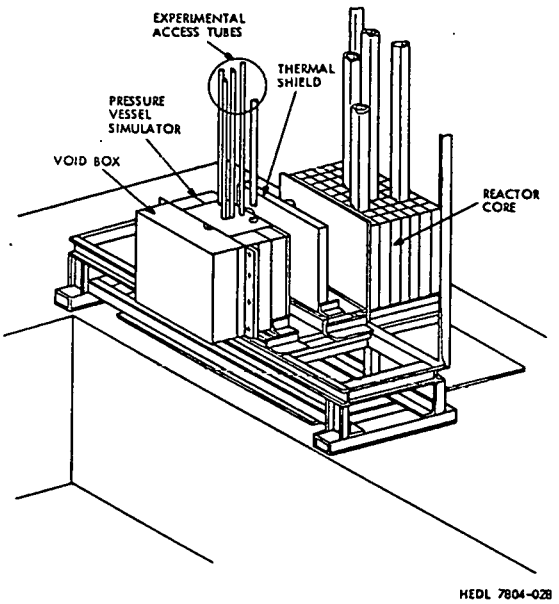

FIGURE 2. Pressure Vessel Wall Mock-up Schematic of Two Equivalent Facilities Under Construction at ORNL. The high-flux version at ORR (PSF) will include damage exposure of metallurgical test specimens; the low-flux version near a low-power critical assembly (PCA) will focus on active and passive dosimetry measurements.