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INTERLABORATORY REACTION RATE PROGRAM 12th PROGRESS REPORT

NOVEMBER 1976 - OCTOBER 1979

Hanford Engineering Development Laboratory

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INTERLABORATORY REACTION RATE PROGRAM 12th PROGRESS REPORT

NOVEMBER 1976 - OCTOBER 1979

Hanford Engineering Development Laboratory

Compiled by
E.P. Lippincott
W.N. McElroy
C.C. Preston

September 1980

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12th PROGRESS REPORT

E. P. Lippincott
W. N. McElroy
C. C. Preston

ABSTRACT

This report was compiled at the Hanford Engineering Development Laboratory operated by Westinghouse Hanford Company, a subsidiary of Westinghouse Electric Corporation, for the Department of Energy, Division of Reactor Research and Technology, under Contract No. DE-AC14-76FF02170. It describes technical progress made in the Interlaboratory Reaction Rate (ILRR) program during the reporting period.

ACKNOWLEDGEMENTS

The following organizations are presently participating in the Interlaboratory Reaction Rate (ILRR) program.

Argonne National Laboratory (ANL)
Brookhaven National Laboratory (BNL)
CEN-SCK Laboratory (MOL)
EG&G Idaho, Inc (EGG)
EG&G ORTEC
Exxon Nuclear Idaho Company, Inc (EXXON)
Hanford Engineering Development Laboratory (HEDL)
Los Alamos Scientific Laboratory (LASL)
National Bureau of Standards (NBS)
Oak Ridge National Laboratory (ORNL)
Rockwell International (formerly AI)

FOREWORD

PURPOSES OF THE ILRR PROGRAM

The Interlaboratory Reaction Rate (ILRR) program is establishing the capability to accurately measure neutron-induced reactions and reaction rates for reactor fuels and materials development programs. The goal for the principal fission reactions, ^{235}U , ^{238}U and ^{239}Pu , is an accuracy to within $\pm 5\%$ at the 95% confidence level. Accurate measurement of other fission and nonfission reactions is also required, but to a lesser accuracy, between $\pm 5\%$ and 10% at the 95% confidence level. A secondary program objective is improvement in knowledge of the nuclear parameters involved in the standardization of fuels and materials dosimetry measurements of neutron flux, spectra, fluence and burnup.

These accuracy goals for the ILRR program are severe, but progress in measuring neutron-induced reaction and reaction rates is progressing well towards the desired accuracy. Using a number of techniques in well established neutron environments of interest for reactor development, and critically evaluating the results determines the existing levels of accuracy and indicates the scale of effort required for improvement.

To accomplish the objectives of this program, reliable and documented experimental values of reaction rates and ratios are determined for various well established and/or permanent neutron fields. The "Standard and Reference Benchmark" fields being used are: the Coupled Fast Reactivity Measurement Facility (CFRMF) at EG&G; the 10% enriched ^{235}U critical assembly (BIG-10) at LASL; the Natural Uranium Secondary Standard Neutron Field ($\Sigma\Sigma$) at Mol, Belgium; the near 1/E and Intermediate-Energy Standard Neutron Fields (ISNF) at NBS; the fission spectrum and thermal cavities at NBS and MOL; the Californium-252 fission sources at NBS; and the iron spheres at Mol. The "Controlled Environment Benchmark" fields being used are: FFTF In-Reactor Thimble (IRT) at HEDL; the ORR-PCA and ORR-PSF LWR pressure vessel mockups at ORNL; and a number of accelerator sources now

being used for fusion reactor materials testing. In addition to the benchmark fields, a number of reactor "Test Region" neutron fields are also being used, such as in the Experimental Breeder Reactor (EBR-II) at Idaho and in the Bulk Shielding Reactor (BSR) at ORNL.

The selected neutron fields span the range of spectra encountered in thermal, fast and fusion reactors. These neutron fields are used because of the range and reproducibility of flux levels available for track recorder, fission chamber, radiometric, helium mass spectrometric, and neutron spectrometry measurements. Coupled with active, passive and calculated flux-spectra determinations, results of reaction and reaction rate tests in these facilities are providing the improved and more precise values of nuclear parameters urgently needed for the standardization of dosimetry procedures and data for reactor fuels and materials development programs.

This report is issued to provide not only final results but also preliminary, incomplete experimental data and technical calculations relating to the ILRR program. Since some of the technical data may be unverified or erroneous, the report is issued with the understanding that revision of results is to be expected unless otherwise stated.

PREVIOUS REPORTS IN THIS SERIES

HEDL-TME 71-131	(First Quarterly Progress Report)
HEDL-TME 71-179	(Second Quarterly Progress Report)
HEDL-TME 72-45	(Third Quarterly Progress Report)
HEDL-TME 72-90	(Fourth Quarterly Progress Report)
HEDL-TME 72-129	(Fifth Quarterly Progress Report)
HEDL-TME-72-154	(Sixth Quarterly Progress Report)
HEDL-TME 73-31	(Seventh Quarterly Progress Report)
HEDL-TME 73-54	(Eighth Quarterly Progress Report)
HEDL-TME 74-54	(Ninth Quarterly Progress Report)
HEDL-TME 75-130	(Tenth Quarterly Progress Report)
HEDL-TME 77-34	(Eleventh Quarterly Progress Report)

CONTENTS

	<u>Page</u>
Abstract	iii
Acknowledgements	iv
Foreword	v
Figures	ix
Tables	xi
Summary	S-1
Rockwell International (Formerly Atomics International)	
Integral (n, α) Cross Sections by Helium Measurement	AI-1
Argonne National Laboratory	
Reaction Rate Measurements of Fission and Nonfission Foils Irradiated in the NBS ^{252}Cf Fission Spectrum Irradiation Facility	ANL-1
Brookhaven National Laboratory	
Status of Data Testing of ENDF/V Reactor Dosimetry File	BNL-1
CEN-SCK Laboratory (MOL)	
A. Interlaboratory Niobium Dosimetry Comparison in EBR-II	MOL-1
B. Absolute Measurements of Highly Active Dosimeters, Irradiated in High Flux Nuclear Reactors in Order to Determine Neutron Fluence and Neutron Spectrum	MOL-7
C. Transport Theory Calculations in Support of the Design of the High Flux Reference Neutron Field NIGHTMARE at the NBS Reactor	MOL-25
EG&G Idaho, Incorporated	
A. Reaction Rate Measurements in the ^{252}Cf Neutron Irradiation Facility at NBS	EGG-1
B. Measurement of Relative Reaction Rates in the ^{235}U Cavity Fission Neutron Facility at NBS	EGG-11

CONTENTS (Cont'd)

	<u>Page</u>
EG&G ORTEC	EGG-ORTEC-1
EXXON Nuclear Idaho Company, Inc	EXXON-1
Hanford Engineering Development Laboratory	
A. FTR Measurement Plans for Interlaboratory Calibrations and Fission Product Yields	HEDL-1
B. Reaction Rate Measurements in the NBS ^{252}Cf Fission Neutron Irradiation Facility	HEDL-59
C. Reaction Rate Measurements in the NBS ^{235}U Cavity Neutron Facility	HEDL-65
D. Analysis of SSTR Fission Rate Measurement in BIG-10	HEDL-69
E. Status of In-Situ Reactor Gamma-Ray Spectrometry	HEDL-77
Los Alamos Scientific Laboratory	
Reaction Rate Measurements in Big Ten	LASL-1
National Bureau of Standards	
A. ^{238}U Capture/ ^{235}U Fission Cross Section Ratio in the ISNF and Other Collaborative Tests with ANL (Idaho)	NBS-1
B. The Gold Capture Cross Section in the NBS Intermediate-Energy Standard Neutron Field (ISNF)	NBS-15
C. Fission Cross-Section Ratios for ^{233}U , ^{232}Th , and ^{240}Pu in CFRMF and Preparation for Absolute Fission Counting at 100 kW.	NBS-57
Oak Ridge National Laboratory	
A. Neutron Dosimeter Materials Development and Characterization for ILRR, HEDL, and ANL Dosimetry Programs	ORNL-1

FIGURES

<u>Figure</u>		<u>Page</u>
AI-1	Configuration of HAFMs in Sigma Sigma	AI-15
AI-2	Configuration of HAFMs in the BR1 Fission Cavity	AI-19
MOL A-1	Nb Foil Sample Container	MOL-5
MOL B-1	Measuring Equipment for High Activity Dosimeters	MOL-14
MOL C-1	Thimble Lattice	MOL-31
MOL C-2	NBSR NIGHTMARE Spectrum at the Center of the Facility	MOL-32
MOL C-3	Reactor NBSR - NIGHTMARE Facility in the Glory Hole $^{233}\text{U}(\text{n},\text{f})$ and $^{238}\text{U}(\text{n},\text{f})$ Response Functions at the Center of the Facility	MOL-33
MOL C-4	Reactor NBSR - NIGHTMARE Facility in the Glory Hole Case Without Boron, ^{233}U Response Function at the Center and at the Border of the Inner Void	MOL-34
MOL C-5	Reactor NBSR - NIGHTMARE Facility in the Glory Hole Case Without Boron and Case with a 15 mm Thick Boron Screen, $^{238}\text{U}(\text{n},\text{f})$ Response Functions at the Center of the Facility	MOL-35
MOL C-6	NIGHTMARE With Three Boron Thicknesses (0.0, 10.0, 15.0mm)	MOL-36
MOL C-7	NIGHTMARE With Three Boron Thicknesses (0.0, 10.0, 15.0mm)	MOL-37
HEDL A-1	Characteristics of a Single Period Irradiation (SPI)	HEDL-8
HEDL A-2	Contents of a Typical LP SSTR Dosimetry Capsule	HEDL-17
HEDL A-3	Radiograph of a Typical LP Capsule After Assembly and Welding	HEDL-17
HEDL A-4	Characterizer Pin Locations Available for Wire Wrap Dosimetry in Driver Fuel CA	HEDL-21
HEDL A-5	Radiograph of a Typical Irradiation Capsule for the HP-SSTR Fission Rate Measurements	HEDL-27
HEDL E-1	Detector Probe for Continuous Compton Recoil Gamma-Ray Spectrometry	HEDL-81

FIGURES (Cont'd)

<u>Figure</u>		<u>Page</u>
HEDL E-2	Instrumentation Block Diagram for Compton Recoil Gamma-Ray Spectrometry	HEDL-82
HEDL E-3	Typical Rise Time Spectra Used for Defining the Sensitive Volume of a 2cc Planar Si(Li) Detector	HEDL-83
HEDL E-4a	Background Electron Spectrum	HEDL-89
HEDL E-4b	Foreground Electron Spectrum	HEDL-89
HEDL E-5a	Background Gamma-Ray Spectrum	HEDL-90
HEDL E-5b	Foreground Gamma-Ray Spectrum	HEDL-90
NBS-1	The Intermediate-Energy Standard Neutron Field	NBS-6
NBS-2	Foil Mounting Arrangement	NBS-7
NBS-3	Gradient Corrections for the ISNF Central Test Region	NBS-8
NBS-4	Gold Reaction Rate Profiles Along NBS Cavity Vertical Polar Axis.	NBS-33
NBS-5	Gold Relative Self-Shielding in the ISNF	NBS-34

TABLES

<u>Table</u>		<u>Page</u>
AI-1	Helium Generated in Unencapsulated Single Natural B and ^{10}B Crystals Irradiated in BIG-10 and CFRMF	AI-4
AI-2	Helium Generated in Unencapsulated Single ^6LiF (99.1% ^6Li) Crystals Irradiated in BIG-10 and CFRMF	AI-5
AI-3	Helium Generated in Encapsulated ^{10}B (93.0% ^{10}B) Powder Irradiated in BIG-10 and CFRMF	AI-6
AI-4	Helium Generated in Encapsulated ^6LiF (99.1% ^6Li) Powder Irradiated in BIG-10 and CFRMF	AI-7
AI-5	Helium Concentration Corrections (%) in BIG-10 and CFRMF	AI-8
AI-6	Measured Helium Reaction Rates in BIG-10 and CFRMF	AI-13
AI-7	Helium Generated in Boron and Lithium Irradiated in Sigma Sigma	AI-17
AI-8	Helium Generated in Boron and Lithium Irradiated in BR1 Fission Cavity	AI-21
ANL-1	Reaction Rates from the NBS ^{252}Cf Fission Neutron Irradiation Facility	ANL-2
ANL-2	Physical Constants for Target Atom Determinations	ANL-3
ANL-3	Parameters for $\sigma\phi$ Calculations	ANL-3
ANL-4	Reaction Rates of Stainless Steel (SS) Detector from the NBS ^{252}Cf Fission Neutron Irradiation Facility	ANL-4
ANL-5	Physical Constants for SS Target Atom Determinations	ANL-4
ANL-6	Parameters for SS $\sigma\phi$ Calculations	ANL-4
MOL-B-1	List of Gamma Emitting Detectors Frequently Used to Measure Fluence and Neutron Spectrum	MOL-15
MOL-B-2	Relative Intensities of ^{110}mAg	MOL-16
MOL-B-3	Parameters Used for the Calibration of ^{198}Au and ^{64}Cu Sources	MOL-17
MOL-B-4	Control of the Absolute Level of the ^{110}mAg - ^{60}Co Efficiency Curve with ^{198}Au and ^{64}Cu	MOL-18

TABLES (Cont'd)

<u>Table</u>		<u>Page</u>
MOL-B-5	Comparison of the Shape of the Efficiency Curve at Three Measurement Positions (Arbitrary Units)	MOL-19
MOL-B-6	Characteristics of the Non-Irradiated Eu Sources	MOL-20
MOL-B-7	Intercalibration of ^{152}Eu -Mo1 1 with ^{152}Eu PTB	MOL-21
MOL-B-8	Intercalibration of ^{152}Eu -Mo1 2 with ^{152}Eu -Mo1 1	MOL-22
MOL-B-9	Absolute Activities of the ^{152}Eu -Mo1 Standards	MOL-23
MOL-B-10	Characteristics of the ^{152}Eu Gamma-Rays Used for Calibration of the Measuring Device	MOL-24
MOL-C-1	Flux Spectrum at the Center of the NIGHTMARE Facility	MOL-30
EGG-1	Parameters of Foils in NBS ^{252}Cf Irradiation Package	EGG-3
EGG-2	Gamma-Ray Counting Parameters for NBS ^{252}Cf Irradiation Package	EGG-4
EGG-3	Reaction Rates from Foils Irradiated in NBS ^{252}Cf Irradiation Facility	EGG-5
EGG-4	Irradiation Symmetry for Foil Packages 1 and 4 in the NBS ^{252}Cf Facility	EGG-6
EGG-5	Combined Reaction Rates for Foil Packages 1 and 4 Irradiated in the NBS ^{252}Cf Facility	EGG-7
EGG-6	$^{58}\text{Ni}(n,p)^{58}\text{Co}$ Reaction Rate Corrected for Neutron Scattering and Deduced Integral Microscopic Cross Section for the NBS ^{252}Cf Facility	EGG-8
EGG-7	^{235}U Fission Reaction Rates Corrected for Neutron Scattering and Deduced Fission Yields for the ^{252}Cf Facility	EGG-9
EGG-8	Parameters of NBS SRM-1158 Nickel-Steel Foil Irradiated in the NBS ^{235}U Cavity Fission Neutron Facility	EGG-13
EGG-9	Gamma-Ray Counting Parameters for SRM-1158 Foil Irradiated in NBS ^{235}U Cavity Fission Neutron Facility	EGG-14
EGG-10	Reaction Rates from SRM-1158 Foil Irradiated in NBS ^{235}U Cavity Fission Neutron Facility	EGG-15

TABLES (Cont'd)

<u>Table</u>	<u>Page</u>
EXXON-1 Inventory of FFTF Fission Yield Capsules	EXXON-2
HEDL A-2.1 Locations of IRT Measurements	HEDL-10
HEDL A-2.2 SPI Parameter Assumptions for IRT Experiment 8	HEDL-11
HEDL A-2.3 SPI Parameter Assumptions for IRT Experiment 10	HEDL-11
HEDL A-2.4 SPI Parameter Assumptions for IRT Experiment 12	HEDL-11
HEDL A-2.5 SSTR Loading Diagram for IRT Experiment 12	HEDL-13
HEDL A-3.1 In-Reactor Locations and Isotopes for the LP Irradiation	HEDL-16
HEDL A-3.2 LP-Hollow Wire Wrap Dosimetry Locations	HEDL-22
HEDL A-3.3 LP-Solid Wire Wrap Dosimetry Locations	HEDL-22
HEDL A-4.1 In-Reactor Locations and Isotopes for the SSTR-HP Irradiation	HEDL-24
HEDL A-4.2 HP K-Factor and Nd Experiment Loading Diagram	HEDL-28
HEDL A-4.3 HP-Hollow Wire Wrap Dosimetry Locations	HEDL-29
HEDL A-4.4 HP-Solid Wire Wrap Dosimetry Locations	HEDL-29
HEDL-A1 Summary-Results and Benefits of FTR Supportive ILRR Program Participants' Measurements	HEDL-34
HEDL-B1 Attainment of Highest Accuracy Fission Rate and Burn-Up Estimates for FFTF Fuel Pins	HEDL-45
HEDL-B2 Fission Yield Measurements	HEDL-46
HEDL-C1 CFRMF Measured and Calculated Values of Reaction Rates per Unit Flux	HEDL-53
HEDL-C2 Ratio of Measured to Computed Integral Cross Sections in Dosimetry Benchmark Neutron Fields	HEDL-56
HEDL-C3 ENDF/B-IV Evaluated Threshold Reaction Cross Section Bias Factors	HEDL-57
HEDL B-1 NBS ^{252}Cf Irradiation Foil Parameters	HEDL-61

TABLES (Cont'd)

<u>Table</u>		<u>Page</u>
HEDL B-2	Measured Reaction Rates for Foils Irradiated in the NBS ^{252}Cf Fission Neutron Irradiation Facility	HEDL-62
HEDL B-3	Comparison of Corresponding Foils to Confirm Position Symmetry of Packages 2 and 5 in the NBS ^{252}Cf Facility	HEDL-63
HEDL B-4	Calculated Integral Microscopic Cross Section for the $^{58}\text{Ni}(n,p)^{58}\text{Co}$ Reaction in the NBS ^{252}Cf Facility	HEDL-64
HEDL B-5	Calculated Fission Yields from ^{235}U Reaction Rates in the NBS ^{252}Cf Facility	HEDL-64
HEDL C-1	Parameters of NBS SRM-1158 High-Nickel-Steel Foil Material	HEDL-67
HEDL C-2	Measured Reaction Rates for the NBS ^{235}U Cavity Fission Neutron Facility	HEDL-67
HEDL D-1	Observed Tracks in BIG-10 Irradiated SSTR	HEDL-71
HEDL D-2	SSTR Fission Rate Data for BIG-10 Irradiations	HEDL-72
HEDL D-3	Isotopic Fission Rates for BIG-10	HEDL-74
HEDL E-1	Finite Size Retention Probability	HEDL-86
NBS-1	Fission and Capture Rates for Four ISNF Runs	NBS-9
NBS-2	Summary of U-238/U-235 Cross Section Ratio Measurements for the ISNF Field	NBS-11
NBS-3	Estimated Uncertainties in the Cross Section Ratio Measurements	NBS-12
NBS-4	Summary of Results for ANL/NBS ISNF Tests	NBS-13
NBS-5	ISNF Irradiations: Relevant Parameters	NBS-35
NBS-6	Determination of the $^{115}\text{In}(n,n')^{115\text{m}}\text{In}$ K-Factor for the Transportable 3" x 3" NaI(Tl) Gamma-ray Spectrometer Relative to the CEN-SCK 3" x 3" Reference NaI(Tl) Spectrometer	NBS-36
NBS-7	Details of the Measurement of Calibration Ratios Between the Transportable 3" x 3" NaI(Tl) Gamma-ray Spectrometer and the CEN-SCK 3" x 3" NaI(Tl) Reference Spectrometer for the Reaction $^{115}\text{In}(n,n')^{115\text{m}}\text{In}$	NBS-37

TABLES (Cont'd)

<u>Table</u>		<u>Page</u>
NBS-8	Determination of the $^{197}\text{Au}(n,\gamma)^{198}\text{Au}$ K-factor for the Transportable 3" x 3" NaI(Tl) Gamma-ray Spectrometer Relative to the CEN-SCK 3" x 3" Reference NaI(Tl) Spectrometer	NBS-38
NBS-9	Details of the Measurement of the Calibration Ratio Between the Transportable 3" x 3" NaI(Tl) Gamma-ray Spectrometer and the CEN-SCK 3" x 3" NaI(Tl) Reference Spectrometer for the Reaction $^{197}\text{Au}(n,\gamma)^{198}\text{Au}$	NBS-39
NBS-10	Mass Assay of 5% Au/Al Alloys By Activation Analysis: Raw Data	NBS-41
NBS-11	U-238 Observed Fission Rates and Inferred U-235 Fission Rates	NBS-43
NBS-12	$^{115}\text{In}(n,n')^{115\text{m}}\text{In}$ Saturated Count Rates in ISNF Normalized To Run-to-run Fission Chamber Monitor Response	NBS-44
NBS-13	$^{115}\text{In}(n,n')^{115\text{m}}\text{In}$ Average Saturated Count Rates in ISNF Normalized To Run-to-run Fission Chamber Monitor Response	NBS-46
NBS-14	$^{197}\text{Au}(n,\gamma)^{198}\text{Au}$ Saturated Count Rates in ISNF Normalized to Run-to-Run Fission Chamber Monitor Response for Different Gold Foil Thicknesses	NBS-47
NBS-15	$^{197}\text{Au}(n,\gamma)^{198}\text{Au}$ Reaction Rate Gradient Along the ISNF Vertical Polar Axis	NBS-49
NBS-16	$^{197}\text{Au}(n,\gamma)^{198}\text{Au}$ ISNF Central Saturated Count Rates Corrected for Reactor Leakage Neutron Background and Normalized to Run-to-Run Fission Chamber Monitor Response	NBS-50
NBS-17	$^{197}\text{Au}(n,\gamma)^{198}\text{Au}$ ISNF Central Saturated Specific Disintegration Rates Normalized to Run-to-Run Monitor Response	NBS-52
NBS-18	Gold Gamma-ray Self Absorption Corrections	NBS-56
NBS-19	Fissionable Deposits - Mass Assay Data	NBS-62
NBS-20	Isotopic Composition and Fission Corrections	NBS-63
NBS-21	Observed Fission Cross Section Ratios, Corrections, and Uncertainties	NBS-64
NBS-22	Spectrum-Averaged Fission Cross Section Ratios in CFRMF	NBS-67

INTERLABORATORY REACTION RATE PROGRAM 12th PROGRESS REPORT (WAJQAA)

SUMMARY

PROGRAM HIGHLIGHTS

At a meeting held in May 1979, ILRR Program participants established the final program plan for supporting measurements in the FFTF start-up program. This plan includes irradiations of ILRR foils in the FFTF In-Reactor Thimble (IRT) and concurrent ILRR foil irradiations in the CFRMF to provide fission standards. These foils will be made available to five laboratories (ANL, CEN/SCK, EGG-Idaho, HEDL, and ORTEC) for measurements. Additional measurements of helium monitors will be made at Rockwell International.

ROCKWELL INTERNATIONAL (AI)

All of the ^{10}B and ^6LiF specimens irradiated in four benchmark neutron spectra as a part of the Interlaboratory Reaction Rate (ILRR) program have been analyzed for helium by high sensitivity gas mass spectrometry. Three irradiations took place in the Coupled Fast Reactivity Measurements Facility (CFRMF) and two in the 10% enriched ^{235}U critical assembly, BIG-10. Single irradiations were made in the secondary intermediate energy standard neutron field SIGMA SIGMA ($\Sigma\Sigma$), and in the 1-meter Fission Cavity of the BR1 reactor at Mol, Belgium. The results have provided important integral normalizations of the ^{10}B and ^6Li helium production cross sections. These normalizations have in turn improved the accuracy of using these reactions for neutron dosimetry in breeder reactor environments.

ARGONNE NATIONAL LABORATORY (ANL)

Reaction rates have been measured for two ^{235}U foils, four nickel foils, and one stainless steel foil which were irradiated in the NBS ^{252}Cf Fission Spectrum Irradiation Facility. Pertinent data used in the

calculation of these reaction rates are also included. The quality of these data appears to be good with only the ^{140}Ba (1.596 MeV) reaction rates from both ^{235}U foils having relatively large uncertainties.

BROOKHAVEN NATIONAL LABORATORY (BNL)

The ENDF/V Reactor Dosimetry file was released August 1979, and was immediately followed by start of Phase 2 data testing. Preliminary results from BNL were presented at the Third ASTM-EURATOM Symposium.

CEN/SCK LABORATORY (MOL)

An interlaboratory comparison of niobium dosimetry measurements has been started. Niobium samples were irradiated in the HEDL EBR-II Run 97 dosimetry pin and sent to MOL for recovery and distribution to the various participating laboratories. The study is expected to improve the calibration of the equipment used and increase the confidence in the accuracy of the parameters used for interpretation of niobium measurements.

A brief description of the measurement equipment and analysis methods for determination of reaction rates for highly activated dosimeters at Mol is given. New high-activity ^{152}Eu is being developed and it is expected that precisions of 2% in measurement of dosimeter activities will be obtained within 5 years.

NIGHTMARE (NBSR Intermediate-energy Glory Hole Tailored Mass Assay and Reference Environment) is a high flux benchmark neutron field under design for implementation in the central glory hole of the NBS reactor. It is intended for use for quality assurance and calibration of neutron dosimeters applied in high flux fast breeder reactors, in particular the Fast Flux Test Facility (FFTF). The neutron spectrum will be tailored by a boron-10

cylindrical shell (protected from burnup by a gadolinium sleeve) so as to be similar to the low-flux Intermediate-Energy Standard Neutron Field (ISNF) at NBS.

One-dimensional transport theory calculations of NIGHTMARE have been performed in the S 8 diagonal transport approximation, using a 40-group cross-section library (commonly referred to as "MOL-BR2-40 gr") based on the KEDAK nuclear data file. Two boron thicknesses have been considered, as well as a case without filter.

Reduced cross sections with 16 neutron groups for 13 compositions have been prepared and stored on disk for future two-dimensional R-Z calculations.

EG&G, IDAHO, INCORPORATED

Infinitely dilute reaction rates have been measured in the NBS ^{252}Cf neutron irradiation facility for the $^{58}\text{Ni}(n,p)^{58}\text{Co}$ reaction and for the production of ^{95}Zr , ^{103}Ru , ^{132}Te and ^{140}Ba - ^{140}La following fission in ^{235}U using γ -ray spectrometry. For each reaction the activity levels were sufficiently low that the accuracy of the results were in all cases limited to some extent by counting statistics; in the worst case to 10% for the ^{132}Te production. Combining these reaction rates with the NBS-certified free-field neutron fluences and free-field fission rate for ^{235}U we obtain an integral cross-section value for the $^{58}\text{Ni}(n,p)^{58}\text{Co}$ reaction and cumulative fission yield values for ^{95}Zr , ^{103}Ru , ^{132}Te and ^{140}Ba .

Relative reaction rates from a sample of SRM-1158 high-nickel steel have been measured in the NBS ^{235}U cavity fission facility for the $^{54}\text{Fe}(n,p)^{54}\text{Mn}$, $^{58}\text{Ni}(n,p)^{58}\text{Co}$ and $^{58}\text{Fe}(n,\gamma)^{59}\text{Fe}$ reactions using γ -ray spectrometry. The accuracy of the result for the latter reaction was severely limited by counting statistics. From these data, a value of 1.346 ± 0.007 was obtained for the $^{58}\text{Ni}(n,p)/^{54}\text{Fe}(n,p)$ reaction rate.

EG&G ORTEC

The counting equipment that is routinely available for ILRR radiometric measurements consists of a twenty percent Ge(Li) coaxial detector, a Low Energy Photon Spectrometer (LEPS) detector, and four multichannel analyzer ports in a Gamma Lab 44 System supported by EG&G Ortec 6516-6528FX analytical software. Additional detectors are available as needed. Measurements associated with the LWR PCA pressure vessel mock-up test have been completed and measurements associated with the FTR-Reactor Characterization Program have been planned.

EXXON NUCLEAR IDAHO COMPANY, INC.

Samples of highly enriched ^{233}U , ^{235}U , ^{239}Pu , and ^{241}Pu sealed in nickel capsules have been loaded into five assemblies for irradiation in the FFTF high power characterization test. This irradiation will generate a group of heavy element reference standards for which the number of fissions are well known and will be used to verify the validity of fast reactor fission yields measured in EBR-II for application to FFTF.

HANFORD ENGINEERING DEVELOPMENT LABORATORY (HEDL)

A detailed description of several parts of the FFTF Characterization Program is given in this report to add perspective to planned ILRR experiments in FFTF. These ILRR measurements by ILRR participating laboratories are planned in coordination with the FTR-RCP to provide calibrations to attain the program accuracy goals.

During FFTF acceptance tests, the absolute number of 1) fissions and fission rates and 2) reactions and reaction rates will be determined by active fission chamber (FC), passive solid-state track recorder (SSTR), radiometric monitor measurement (RM), and helium accumulation fluence monitor (HAFM) mapping techniques. High accuracy NBS-FCs and HEDL-SSTRs (using mica and quartz recorders) will be used to make measurements at selected locations during the very low power (VLP) and low power (LP) tests.

Quartz recorders (HEDL SSTRs) and mass spectrometric techniques (Allied Chemical Idaho National Engineering Laboratory (INEL)) will be the only available methods for making direct measurements of the absolute number of fissions at selected locations during the high power (HP) test for the direct establishment of fission yield and counting system calibration factor values.

Interlaboratory radiometric counting will be done on selected sets of VLP, LP and/or HP radiometric monitors to establish in-situ high accuracy (1 to 5%, 1σ) calibrations for the HEDL manual and automated radiometric measurements. Selected individual monitor calibrations using NBS, EG&G, and HEDL "Benchmark Neutron Fields" will also be accomplished by participating laboratories. The interlaboratory-RM and single-laboratory HEDL-RM measurements will have a limiting accuracy of 1 to 3% (1σ) and 2 to 5% (1σ), respectively. HAFM measurements will be made by Rockwell International on dosimeters from the same sets selected for the radiometric intercomparisons. Analyses of these dosimeters will be carried out at Rockwell International with a limiting accuracy of 1 to 3% (1σ).

The NBS, HEDL, and Allied Chemical absolute fission rates and total number of fissions will be combined with Interlaboratory Reaction Rate (ILRR) Program type radiometric (^{140}Ba , ^{95}Zr , ^{103}Ru , ^{131}I , ^{132}Te , and ^{137}Cs) and HEDL and Allied Chemical mass spectrometric (^{143}Nd , ^{145}Nd , ^{146}Nd , ^{148}Nd , and ^{150}Nd) fission product results to measure, validate, and calibrate (at VLP, LP, and HP) the energy dependence of the above identified key fission products. These combined fission yield results will be used to generate FTR in- and out-of-core positional fission yield (r, θ, ϕ) energy dependent mappings which will serve as input to reactor physics and fuel behavior codes as well as for fuel pin burnup and wire-wrap dosimetry measurement techniques. These different techniques will be used to account for variations in point-to-pin, pin-to-assembly, and assembly-to-core and blanket (reflector for FTR) absolute fission rate and burnup results. The goal accuracy for the FTR Reactor Characterization Program (RCP) point-wise measurements is in the 1 to 3% (1σ) range for in-core fuel region experiments and in the 2 to 5% (1σ) range for in- and out-of-core region materials and absorber experiments.

Both Ni and ^{235}U foils were irradiated in the NBS ^{252}Cf fission neutron irradiation facility and the infinitely dilute reaction rates were measured for the ^{58}Co , ^{140}Ba , ^{103}Ru and ^{95}Zr product activities using direct gamma-ray spectrometry. Long counts were required in all cases due to the relatively low activities. Cumulative fission yield values for ^{140}Ba , ^{103}Ru and ^{95}Zr and the integral cross-section value for the $^{58}\text{Ni}(n,p)^{58}\text{Co}$ reaction were obtained by combining the measured reaction rates with the NBS-certified free-field neutron fluences and free-field fission rate.

Relative reaction rates from a sample SRM-1158 high-nickel steel were measured in the NBS ^{235}U cavity fission facility for the $^{54}\text{Fe}(n,p)^{54}\text{Mn}$ and $^{58}\text{Ni}(n,p)^{58}\text{Co}$ reactions using direct gamma-ray counting techniques. A value of 1.360 ± 0.007 was obtained from these data.

Scanning of SSTR specimens from BIG-10 irradiations has been completed. Comparison with absolute NBS fission chamber results reveals agreement within limits of experimental error for absolute ^{235}U and ^{238}U fission rates. Absolute SSTR fission rates observed for ^{239}Pu and ^{237}Np are systematically higher than NBS fission chamber results by 5 and 7 percent, respectively.

Advancements have been made in applying Compton-Recoil gamma-ray spectrometry techniques to reactor environments. In-situ gamma-ray spectrometry efforts at HEDL in U.S. Light Water Reactor (LWR) and Breeder Reactor (BR) programs are described.

LOS ALAMOS SCIENTIFIC LABORATORY (LASL)

Three reports pertaining to the dosimetry data and general neutronic characterization of Big Ten are referenced here.

NATIONAL BUREAU OF STANDARDS (NBS)

A measurement of the ^{238}U capture/ ^{235}U fission cross-section ratio in the Intermediate-Energy Standard Neutron Field (ISNF) is reported and compared with prediction of ENDF/B-IV. This measurement was made in collaboration with the Applied Physics Division of Argonne National Laboratory (Idaho). A comparison of fission rate measurement techniques is also reported.

A measurement of the ^{197}Au capture/ ^{235}U fission cross-section ratio by Fabry of CEN/SCK is reported for the ISNF facility.

Fission cross-section ratios for ^{233}U , ^{232}Th , and ^{240}Pu are reported for measurements made at the EG&G CFRMF field, and preparations for producing fission counting standards at CFRMF are discussed.

OAK RIDGE NATIONAL LABORATORY (ORNL)

Since 1971 IRML has provided more than 5000 vanadium encapsulated dosimeters for use in reactor core characterization, materials compatibility studies, and reactor surveillance operations. Much of the success of the program can be directly attributed to the following:

1. Formation of ceramic oxide wires for many of the pure and dilute dosimeter materials.
2. Refinement of existing analytical techniques to characterize properly the dosimeter materials.
3. Establishment of quality control procedures for all phases of dosimeter fabrication.

AI

ROCKWELL INTERNATIONAL

INTEGRAL (n, α) CROSS SECTIONS BY HELIUM MEASUREMENT

Harry Farrar IV and Brian M. Oliver

Objective

Determine integral (n, α) cross sections of ^6Li and ^{10}B and certain other isotopes in a variety of well-established and permanent neutron fields, including several that cover the neutron energy ranges of interest for breeder, light water, and fusion reactors. The cross sections, which are obtained by mass spectrometric measurement of the amounts of helium generated in miniature capsules containing various isotopes will provide independent, absolute bases for normalization of available energy-dependent (n, α) cross-section data. This will improve the accuracy for using these isotopes as helium accumulation fluence monitors (HAFMs) for neutron fluence dosimetry in breeder, light water and fusion reactors.

Summary

All of the ^{10}B and ^6LiF specimens irradiated in four benchmark neutron spectra as a part of the Interlaboratory Reaction Rate (ILRR) program have been analyzed for helium by high sensitivity gas mass spectrometry. Three irradiations took place in the Coupled Fast Reactivity Measurements Facility (CFRMF) and two in the 10% enriched ^{235}U critical assembly, BIG-10. Single irradiations were made in the secondary intermediate energy standard neutron field SIGMA SIGMA ($\Sigma\Sigma$), and in the 1-meter Fission Cavity of the BR1 reactor at Mol, Belgium. The results have provided important integral normalizations of the ^{10}B and ^6Li helium production cross sections. These normalizations have in turn improved the accuracy of using these reactions for neutron dosimetry in breeder reactor environments.

Accomplishments and Status

A. Introduction

The testing of boron and lithium helium accumulation fluence monitors (HAFMs) in benchmark neutron facilities, in cooperation with other laboratories participating in the Interlaboratory Reaction Rate (ILRR) program, is an integral part of Rockwell International's program to demonstrate and develop HAFMs for use in fast reactor neutron dosimetry. The information from the benchmark tests, when combined with the neutron spectrum data obtained by other laboratories participating in the ILRR program, will provide important insights into the apparent discrepancies between the spectrum integrated and the ENDF/B differential cross sections of ^{10}B and ^6Li . In the meantime, however, the benchmark data so far obtained for ^{10}B and ^6Li are proving very useful in normalizing the HAFM results from the breeder reactors EBR-II and RAPSODIE.

To date, seven separate irradiations of boron and lithium HAFMs in benchmark facilities have been completed. Three of these irradiations took place in the Coupled Fast Reactivity Measurements Facility (CFRMF)¹⁻³ at the Idaho Nuclear Engineering Laboratory (INEL) in April 1972, May 1973, and June 1975. The third and fourth irradiations, conducted in July and September 1974, were in the BIG-10 standard neutron field²⁻⁴ at the Los Alamos Scientific Laboratory (LASL). In the two BIG-10 irradiations and in the third CFRMF irradiation, essentially identical aluminum assemblies⁵ were used to contain various bare and encapsulated crystals of boron and lithium fluoride. The sixth and seventh irradiations were conducted simultaneously in June 1976 in the Sigma Sigma⁶ and Fission Cavity benchmark fields⁵ in the BR1 reactor at the CEN/SCK Laboratories in Mol, Belgium.

Final ^{10}B and ^6Li reaction rate results for the third, fourth, and fifth irradiations in BIG-10 and CFRMF have been obtained. Additional small corrections have also been made to the reaction rate data published

earlier⁷ for the first two CFRMF irradiations because of small changes in recent SAND-II unfolded flux spectra for these facilities, and changes in the ENDF/B cross section file. These changes permit direct comparison of the three CFRMF irradiations.

Helium concentration results for the boron and lithium HAFMs irradiated in the Sigma Sigma and Fission Cavity facilities have also been obtained. These concentration results, however, still require final corrections for such factors as neutron self-shielding and flux gradients in order to obtain final nuclear reaction rates. Preliminary ^{10}B and ^6Li reaction rates in these two benchmark fields are nevertheless reported.

B. BIG-10 and CFRMF Irradiations

1. Experimental Results - Final data analysis has been completed on the helium concentrations generated in unencapsulated (bare) and encapsulated natural boron, enriched boron (93.0% and 95.6% ^{10}B), and enriched ^6LiF (99.1% ^6Li) irradiated in BIG-10 (Runs 1 and 2) and CFRMF (Run 3). The results are given in Tables 1 through 4.

Experimental details,⁵ and preliminary corrected helium generation results on the bare crystals, have been reported earlier.^{3,8} Helium contamination in the vanadium metal used to encapsulate the remaining boron and lithium samples in BIG-10 and CFRMF,⁸ however, resulted in increasing the number of preparation steps required for helium analyses. The final uncorrected results of these encapsulated samples have now been obtained, and the data in Tables 3 and 4 have been corrected for such factors as neutron self-shielding, flux depression, and flux gradients. In addition, the preliminary corrections applied to the earlier reported results^{3,8} for the bare crystals are now subject to slight modification due to recent changes in the SAND-II unfolded flux spectra in both facilities. The new and revised corrections have been applied to the helium concentration data for all the samples irradiated in the three irradiations.

TABLE 1
HELIUM GENERATED IN UNENCAPSULATED SINGLE NATURAL B AND ^{10}B CRYSTALS
IRRADIATED IN BIG-10 AND CFRMF

Irradiation	^{10}B Enrichment	Sample	Mass (mg)	^4He Atoms of ^4He Measured	^4He Concentration (appb)		
					Measured	Corrected*	Average†
BIG-10 (Run 1)	19.8	RNB5	2.944	1.924×10^{11}	5.926	5.955	5.915 ±0.017
	19.8	RNB6	3.844	2.503×10^{11}	5.904 [§]	5.904	
	19.8	RNB10	4.751	3.115×10^{11}	5.944 [§]	5.944	
	19.8	RNB12	4.007	2.610×10^{11}	5.906	5.947	
	95.6	R10B1	3.611	1.201×10^{12}	5.810	5.903	
	93.0	R10B41	2.356	7.612×10^{11}	5.817	5.904	
	93.0	R10B42	1.586	5.176×10^{11}	5.875	5.957	
	93.0	R10B43	1.545	4.639×10^{11}	5.791 [§]	5.808	
BIG-10 (Run 2)	19.8	RNB7	4.436	3.676×10^{11}	7.512	7.542	7.433 ±0.019
	19.8	RNB8	3.982	3.284×10^{11}	7.476 [§]	7.476	
	19.8	RNB9	3.699	3.002×10^{11}	7.358	7.410	
	19.8	RNB14	3.343	2.764×10^{11}	7.497 [§]	7.497	
	95.6	R10B3	3.501	1.459×10^{12}	7.281	7.397	
	93.0	R10B44	1.890	7.675×10^{11}	7.308	7.410	
	93.0	R10B45	1.818	7.342×10^{11}	7.267	7.376	
	93.0	R10B48	1.912	7.799×10^{11}	7.342 [§]	7.386	
CFRMF (Run 3)	93.0	R10B48B	0.356	1.455×10^{11}	7.358 [§]	7.402	
	19.8	RNB11	5.559	9.484×10^{11}	15.47 [§]	15.86	15.88 ±0.03
	19.8	RNB13	4.344	7.411×10^{11}	15.47	15.92	
	19.8	RNB15	4.780	8.120×10^{11}	15.40	15.88	
	19.8	RNB16	5.354	9.183×10^{11}	15.58 [§]	15.95	
	95.6	R10B18	2.733	2.309×10^{12}	14.76	15.98	
	93.0	R10B47	1.451	1.182×10^{12}	14.66	15.68	
	93.0	R10B49	1.907	1.577×10^{12}	14.88 [§]	15.85	
	93.0	R10B50	1.738	1.432×10^{12}	14.83**	15.89	

*Measured helium concentration per ^{10}B , corrected for self-shielding, flux depression, flux gradients, etc. (see text).

†Weighted mean and standard deviation (1σ) of the mean.

§Crystal was etched to remove effect of ^4He lost or gained due to α-recoil.

**Indicates an "unspiked" sample analyzed for ^3He as well as ^4He .

TABLE 2
HELIUM GENERATED IN UNENCAPSULATED SINGLE ${}^6\text{LiF}$ (99.1% ${}^6\text{Li}$)
CRYSTALS IRRADIATED IN BIG-10 AND CFRMF

Irradiation	Sample	Mass (mg)	${}^4\text{He}$ Atoms of Measured	${}^4\text{He}$ ium Concentration (appb)		
				Measured	Corrected*	Average [†]
BIG-10 (Run 1)	R6LiF1A	3.094	2.948×10^{11}	$3.994^{\S}$	4.018	$\left. \begin{array}{c} 4.132 \\ \pm 0.030 \end{array} \right\}$
	R6LiF1B	3.179	3.138×10^{11}	4.138	4.163	
	R6LiF2	4.917	4.904×10^{11}	4.181^{**}	4.167	
	R6LiF4	3.950	3.860×10^{11}	$4.096^{\S}$	4.116	
BIG-10 (Run 2)	R6LiF3A	2.875	3.586×10^{11}	$5.229^{\S}$	5.265	$\left. \begin{array}{c} 5.221 \\ \pm 0.012 \end{array} \right\}$
	R6LiF3B	2.465	3.058×10^{11}	5.202	5.238	
	R6LiF5	4.529	5.660×10^{11}	$5.239^{\S, **}$	5.233	
	R6LiF13A	3.340	4.118×10^{11}	5.168	5.204	
	R6LiF13B	3.209	3.946×10^{11}	5.156	5.192	
CFRMF (Run 3)	R6LiF9A	2.871	5.598×10^{11}	$8.174^{\S}$	8.206	$\left. \begin{array}{c} 8.093 \\ \pm 0.024 \end{array} \right\}$
	R6LiF9B	2.665	5.137×10^{11}	8.082	8.114	
	R6LiF11	4.466	8.573×10^{11}	8.048^{**}	8.047	
	R6LiF14A	3.322	6.338×10^{11}	7.997	8.060	
	R6LiF14B	2.164	4.146×10^{11}	$8.031^{\S}$	8.095	

*Measured helium concentration per ${}^6\text{Li}$, corrected for self-shielding, flux depression, flux gradients, etc. (see text).

†Weighted mean and standard deviation (1σ) of the mean.

§Indicates an "unspiked" sample analyzed for ${}^3\text{He}$ as well as ${}^4\text{He}$.

**Crystal was etched to remove effects of ${}^4\text{He}$ lost or gained due to a α -recoil.

TABLE 3
HELIUM GENERATED IN ENCAPSULATED ^{10}B (93.0% ^{10}B) POWDER
IRRADIATED IN BIG-10 AND CFRMF

Irradiation	Sample	Mass (mg)	^4He Atoms of Measured	^4He Helium Concentration (appb)		
				Measured	Corrected*	Average†
BIG-10 (Run 1)	Rx10B-1	3.916	1.247×10^{12}	5.730	5.828	} 5.859 ±0.030
	Rx10B-2	3.892	1.230×10^{12}	5.688	5.791	
	Rx10B-9	3.870	1.249×10^{12}	5.811	5.916	
	Rx10B-13	3.393	1.095×10^{12}	5.809	5.902	
BIG-10 (Run 2)	Rx10B-3	4.022	1.611×10^{12}	7.209	7.332	} 7.271 ±0.038
	Rx10B-4	3.840	1.482×10^{12}	6.947 [§]	7.072	
	Rx10B-10	3.703	1.461×10^{12}	7.101	7.229	
	Rx10B-11	3.939	1.570×10^{12}	7.173	7.302	
	Rx10B-14	3.762	1.506×10^{12}	7.206	7.322	
CFRMF (Run 3)	Rx10B-6	3.974	3.227×10^{12}	14.61	15.56	} 15.56 ±0.04
	Rx10B-12	3.912	3.161×10^{12}	14.54	15.50	
	Rx10B-15	3.842	3.121×10^{12}	14.62	15.61	

*Measured helium concentration per ^{10}B , corrected for self-shielding, flux depression, flux gradients, etc. (see text).

†Weighted mean and standard deviation (1σ) of the mean.

§Indicates an "unspiked" sample analyzed for ^3He as well as ^4He .

Tables 1 and 2 show the measured and corrected helium concentrations for the bare crystals of boron and ^6LiF . Tables 3 and 4 show similar results for the encapsulated ^{10}B and ^6LiF crystalline powders. In each case, the final column in each of these tables shows the weighted mean and 1σ standard deviation of the mean for each of the different sample types in the three irradiations. A weighting factor of 1/2 was applied to the few unspiked helium analyses (samples analyzed for ^3He as well as ^4He) in order to compensate for the small decrease in overall accuracy in these analyses because of small changes in the mass spectrometer sensitivity.⁹

TABLE 4
HELIUM GENERATED IN ENCAPSULATED ${}^6\text{LiF}$ (99.1% ${}^6\text{Li}$) POWDER
IRRADIATED IN BIG-10 AND CFRMF

Irradiation	Sample	Mass (mg)	Atoms of ${}^4\text{He}$ Measured	${}^4\text{He}$ Helium Concentration (appb)		
				Measured	Corrected*	Average†
BIG-10 (Run 1)	Rx6LiF-1	5.349	5.058×10^{11}	3.964	3.995	4.082 ±0.018
	Rx6LiF-2	4.939	4.746×10^{11}	4.028	4.060	
	Rx6LiF-3	4.890	4.736×10^{11}	4.060	4.100	
	Rx6LiF-10	5.033	4.860×10^{11}	4.048 [§]	4.088	
	Rx6LiF-11	4.945	4.778×10^{11}	4.051 [§]	4.091	
	Rx6LiF-16	5.250	5.107×10^{11}	4.079	4.120	
	Rx6LiF-17	4.364	4.257×10^{11}	4.090	4.127	
BIG-10 (Run 2)	Rx6LiF-4	5.069	6.041×10^{11}	4.996	5.036	5.150 ±0.036
	Rx6LiF-5	2.845	3.512×10^{11}	5.175 [§]	5.216	
	Rx6LiF-6	4.751	5.718×10^{11}	5.045	5.095	
	Rx6LiF-12	4.801	5.914×10^{11}	5.164 [§]	5.216	
	Rx6LiF-13	5.385	6.757×10^{11}	5.261	5.314	
	Rx6LiF-18	5.193	6.291×10^{11}	5.079	5.130	
	Rx6LiF-19	5.128	6.190×10^{11}	5.061	5.107	
CFRMF (Run 3)	Rx6LiF-7	4.750	9.036×10^{11}	7.974 [§]	8.110	8.098 ±0.089
	Rx6LiF-9	4.734	8.967×10^{11}	7.941	8.076	
	Rx6LiF-14	5.220	9.917×10^{11}	7.965	8.077	
	Rx6LiF-15	5.201	1.040×10^{12}	8.387	8.496	
	Rx6LiF-20	5.324	1.001×10^{12}	7.881	7.992	
	Rx6LiF-21	3.499	6.435×10^{11}	7.710	7.841	

*Measured helium concentration per ${}^6\text{Li}$, corrected for self-shielding, flux depression, flux gradients, etc. (see text).

†Weighted mean and standard deviation (1σ) of the mean.

§Indicates an "unspiked" sample analyzed for ${}^3\text{He}$ as well as ${}^4\text{He}$.

2. Neutron Shielding and Scattering Corrections - In order to obtain final comparisons of the ^4He production in the bare crystals and the capsules, and to compare these data with other ILRR dosimetry results, the data in Tables 1 to 4 have been corrected for various perturbations that occurred during, and as a result of, the irradiation. The magnitudes of the corrections are summarized in Table 5. In general, the

TABLE 5
HELIUM CONCENTRATION CORRECTIONS (%) IN BIG-10 AND CFRMF

Sample Type	Self Shielding*	Cross Shielding	Flux Gradient	α -Recoil†	Assembly Scattering	Total Correction**
<u>BIG-10</u>						
Nat. B Crystal	0.3	0.1	-(0.1-0.4)	0.8	-0.3	0.5-0.8
^{10}B Crystal	1.0-1.4	0.1	-0.4	0.8-1.0	-0.3	1.2-1.8
^6LiF Crystal	0.5-0.6	0.1	-(0.0-0.1)	0.8-0.9	-0.3	0.6-0.8 [§]
^{10}B Powder	0.9	0.1-0.2	-(0.1-0.2)	1.1	-0.3	1.6-1.8
^6LiF Powder	0.4	0.1-0.2	-(0.1-0.2)	1.2	-0.3	0.8-1.0 [§]
<u>CFRMF</u>						
Nat. B Crystal	1.8-2.0	0.3-0.8	0.1-0.5	0.6-0.7	-0.3	2.5-3.8
^{10}B Crystal	5.6-7.0	0.3-0.5	0.1-0.5	0.7-0.9	-0.3	6.4-8.6
^6LiF Crystal	0.8	0.1-0.2	0.1-0.2	0.6	-0.3	0.8-1.0 [§]
^{10}B Powder	4.9	0.6-0.8	0.1-0.5	1.0	-0.3	6.3-6.9
^6LiF Powder	0.6	0.2-0.3	0.1-0.5	1.0	-0.3	1.1-1.6 [§]

*Includes correction for surface flux depression.

†Applies only to unetched crystals and crystalline powder from capsules (see text).

§Includes correction for helium generation from ^{19}F (-0.4% in BIG-10 and -0.5% in CFRMF).

**Algebraic sum of individual corrections. Measured helium concentration results are increased by percentage correction.

largest of these corrections is for neutron self-shielding and surface flux depression. The self-shielding and flux depression corrections for both the BIG-10 and the CFRMF irradiations were computed by E. P. Lippincott (HEDL) using the most recent SAND-II unfolded flux spectrum for each facility, and the ENDF/B-IV neutron cross section file. The computations are based on a first flight transport theory approximation.¹⁰ For small self-shielding, the self-shielding factor is very insensitive to the actual surface-to-volume ratio of the sample.¹⁰ Thus, for the bare boron and ^6LiF crystals, the individual factors were computed assuming a spherical sample of equal mass. In the case of the ^{10}B and ^6LiF capsules, the self-shielding factors were computed assuming a cylinder of equal mass whose density and radius were determined from a compromise between the two limiting cases of (1) assuming a fine grain crystalline powder of reduced effective density uniformly distributed over the full inside diameter of the capsule, and (2) assuming a solid cylindrical crystal of ^{10}B or ^6LiF of reduced radius. A compromise was considered appropriate in view of the fact that the actual nature of the capsule contents, as determined from neutron radiographs, was somewhere between these two cases. In addition, the radius determined above was reduced further by the factor $R/(R + L)$, where R and L are the radius and length of the cylinder, in order to correct for end-effects (which tend to reduce the self-shielding).

That this compromise resulted in an improved estimate for the self-shielding was verified by considering several physical models, consisting of infinite cylindrical arrays of spherical crystals, to describe the crystalline powder inside the capsules. Various numbers of spheres per layer were assumed in order to simulate the actual range of packing factors in the BIG-10 and CFRMF capsules.

For each model, self-shielding factors were computed for the individual spheres. These calculations included a correction for shielding caused by adjacent spheres in the array. These factors were then compared with the factors obtained assuming the two limiting cylindrical cases stated above. In each case, the two cylindrical approximations to

the spherical array bracketed the result obtained for the individual spheres. Specifically, the reduced-diameter cylindrical approximation resulted in too large a factor, whereas the reduced-density cylindrical approximation resulted in too small a factor. In view of these results, a compromise approach was chosen. This compromise consisted of assuming an average particle density midway between the actual density of the individual crystals, and the reduced density obtained assuming the mass to be uniformly distributed over the full inside diameter of the capsule. The effective radius was then computed from this average density assuming a cylindrical crystal of equal mass. The resulting self-shielding factors obtained following this approach were very close to those obtained for the individual spheres.

Applying this compromise to the enriched ^{10}B and ^6LiF HAFM capsules resulted in self-shielding factors of 1.009 ± 0.001 and 1.049 ± 0.003 for the ^{10}B capsules and 1.004 ± 0.001 and 1.006 ± 0.001 for the ^6LiF capsules, in BIG-10 and CFRMF respectively. The small uncertainties in the above numbers are estimated from the model calculations stated above, and from a knowledge that the calculations were largely independent of the model chosen.

As indicated in Table 5, the additional smaller corrections for cross-shielding (sample interaction), flux gradients, helium loss from α -recoil, and assembly scattering generally amounted to $\lesssim 1\%$. The cross-shielding corrections, caused by flux depression from neighboring samples, were calculated following an approximate theory by G. E. Hansen (LASL).¹¹ For this calculation, only the nearest neighbors were considered. Corrections for helium loss due to α -recoil were computed from escape probabilities for spheres and cylinders given by Case, *et al.*¹⁰ For these calculations, mean α -particle energies were determined for BIG-10 and CFRMF assuming approximate mean neutron (n, α) reaction energies of 140 keV and 40 keV, respectively. The flux gradients across the volume occupied by the aluminum subassembly for CFRMF and BIG-10 were supplied by JW Rogers (INEL) and G. E. Hansen (LASL), respectively.

For bare HAFM crystals, some loss of helium is to be expected near the surface of the HAFM due to recoil of the α particle. This effect was completely removed in those cases where the surfaces of the crystals were etched prior to helium analysis. In the case of encapsulated HAFMs, however, the encapsulating material provided containment for these recoiling alphas, thus preventing their loss.

A slight increase in the boron and lithium reaction rate can be expected because of elastic scattering of the neutrons in the aluminum subassembly containing the dosimeters. This increased reaction rate is the result of the larger boron and lithium (n, α) cross sections at the slightly lower mean neutron energies caused by the collision process.

The final column in Table 5 indicates the range of total corrections applied to each type of sample. Included in this total, for ^6LiF samples, is a small correction to compensate for expected helium generation from ^{19}F . The results indicate a total correction in the range of 0.5 to 2% for BIG-10 and from 1 to 9% for CFRMF. As stated previously, the larger total correction range in CFRMF is primarily due to the significantly higher amounts of self-shielding in the boron. It is estimated that the combined uncertainty of the corrections themselves is $\approx 20\%$, which translates into an additional uncertainty in the final helium reaction rate results of $\sim 0.3\%$ in BIG-10 and $\sim 1.0\%$ in CFRMF. These uncertainties are within the range of uncertainties in the helium concentration measurements themselves (as indicated in the standard deviations in Tables 1 through 4), and to this extent have not greatly affected the overall accuracy of the final results.

3. Discussion of BIG-10 and CFRMF Results - Comparison of the results of Tables 1 through 4 shows that the agreement between the bare and encapsulated crystal data is good. Indeed, the internal consistency, as reflected in the standard deviations, is very close to 1%. Close examination, however, shows that the corrected helium concentrations in the encapsulated boron and lithium crystals are, in general, slightly lower (1 to 2%) than the corresponding data for the bare crystals. At the

present time, it is believed this is due to a combination of the larger uncertainty in the self-shielding and α -recoil corrections for the more irregular crystalline powder (as compared to the larger single bare crystals), and the increased experimental uncertainty resulting from the additional number of sample preparation steps involved in separating and collecting the capsule contents for analysis in the mass spectrometer. These latter steps to remove the crystals from the capsules were necessary because of inadvertent helium contamination of the vanadium capsule material by the capsule manufacturer, as discussed earlier.⁸ Applying, therefore, an approximate weighting factor of 1/2 to the encapsulated crystal data, yields final helium concentrations (in atomic parts per billion, appb) for the ^{10}B of 5.896 ± 0.032 , 7.379 ± 0.094 , and 15.77 ± 0.18 , for BIG-10 Runs 1, 2, and CFRMF Run 3, respectively. The uncertainties stated are the 1σ standard deviations of the averaged mean values. Similarly, final helium concentrations for ^6Li for the three irradiations, are 4.115 ± 0.029 , 5.197 ± 0.041 , and 8.095 ± 0.003 appb, respectively.

As discussed above, the ^{10}B helium concentration results from the first two CFRMF irradiations in April 1972 and May 1973 have been reported earlier.^{3,8} In view of recent small changes in the SAND-II unfolded spectra for CFRMF and in the ENDF/B cross section file, however, the self-shielding corrections (and additional smaller corrections) applied to these earlier data have now been slightly modified. Applying these modified corrections to the CFRMF Run 1 and 2 data yields helium concentrations in ^{10}B of 12.55 ± 0.08 and 12.46 ± 0.21 appb, respectively. The CFRMF Run 1 results stated above have been normalized to a reactor power of 6 kW in order to correspond to CFRMF Runs 2 and 3. The normalization factor, 0.893, was derived from an average of the fluence ratios for CFRMF Runs 1 and 2, obtained from fission chamber measurements and the SAND-II code.

From the various helium concentration data, the resulting ^{10}B and ^6Li reaction rates for the two runs in BIG-10 and the three runs in

CFRMF are listed in Table 6. The weighted mean values of these data indicate ^{10}B reaction rates (in units of reactions/s · atom × 10^{14}) in BIG-10 and CFRMF of 11.15 ± 0.06 and 15.30 ± 0.07 , respectively. Similarly, for ^6Li , the mean reaction rates are 7.821 ± 0.041 and 7.754 ± 0.003 , respectively. In terms of the ^{10}B -to- ^6Li reaction ratio, these data yield values of 1.426 ± 0.011 for BIG-10 and 1.973 ± 0.009 for CFRMF. A comparison of these values with an empirical curve giving the energy dependence of the ratio of ^{10}B and ^6Li reactions derived from HAFMs irradiated in EBR-II has been published elsewhere.¹²

TABLE 6
MEASURED HELIUM REACTION RATES IN BIG-10 AND CFRMF

Run	Irradiation Time (s)	Reaction Rate (Reactions/s·atom x 10 ¹⁴)		
		¹⁰ B	⁶ Li	¹⁰ B/ ⁶ Li
<u>BIG-10</u>				
Run 1	52,856	11.15 ± 0.06	7.785 ± 0.055	1.432 ± 0.013
Run 2	66,070	11.17 ± 0.14	7.866 ± 0.062	1.420 ± 0.021
	Weighted Means	11.15 ± 0.06	7.821 ± 0.041	1.426 ± 0.011
<u>CFRMF</u>				
Run 1	81,779	15.35 ± 0.08	-	-
Run 2	81,779	15.24 ± 0.26	-	-
Run 3	104,400	15.11 ± 0.17	7.754 ± 0.003	1.949 ± 0.011
	Weighted Means	15.30 ± 0.07	7.754 ± 0.003	1.973 ± 0.009

C. Sigma Sigma Irradiation

Forty five bare and encapsulated crystals of natural boron, enriched boron (93% ^{10}B) and ^6LiF (99.1% ^6Li) have been irradiated for ~108 hours in the Sigma Sigma ($\Sigma\Sigma$) and Fission Cavity dosimetry benchmark neutron facilities of the BR1 reactor at the CEN/SCK Laboratories in Mol, Belgium. These irradiations, which were performed by CEN/SCK as part of the Interlaboratory Reaction Rate program, will provide valuable cross section verification in these two widely characterized fast neutron spectra.

The secondary intermediate energy standard neutron field, Sigma Sigma, is a thermal-fast coupled spherical source assembly located within a 1-meter diameter spherical cavity in the graphite thermal column of the BR1 reactor.⁽⁶⁾ The spherical source consists of a natural uranium shell 24.5 cm OD and 5 cm thick. Inside the uranium, there is a 17.5-cm thick inner shell of natural boron carbide. The 11-cm inner diameter defines the limits of the actual irradiation volume. With this configuration, the outer portion of the uranium shell acts as a thermal-to-fast neutron converter. The resulting fast neutrons are then degraded in energy by inelastic scattering in the interior portion of the uranium shell and by elastic collisions in the boron carbide. The boron preferentially removes the low energy neutrons, resulting in a final "shaped" neutron energy distribution that is particularly useful for fast reactor dosimetry experiments. Flux gradient measurements in $\Sigma\Sigma$ indicate a maximum anisotropy in the central region (± 3 cm) of $\lesssim 4\%$.

Of the total number of boron and lithium HAFMs irradiated in BR1, 21 were irradiated in $\Sigma\Sigma$. The arrangement of the various capsules, bare crystals, and empty capsules, is shown in Figure 1. Empty capsules were included in order to measure any helium production from the capsules themselves, although a negligible amount was expected. This particular loading arrangement was chosen because it approximated a series of infinite cylinders of ^{10}B and ^6Li , and maximized the distance between adjacent dosimeters, thus optimizing the accuracy in the self-shielding, flux depression, and cross shielding corrections.

Configuration of HAFMs in Sigma Sigma

Figure 1.

The HAFM capsules for both the $\Sigma\Sigma$ and the Fission Cavity irradiations were manufactured at Rockwell from thin wall (0.076 mm), 2.36 mm OD, Type 304 stainless steel tubing. The finished length of each capsule (after electron beam welding of each end) was ~17.8 mm. Each capsule was stamped on one end with an identifying letter (A-Z) to aid in post-irradiation recovery. These capsules were considerably larger than any HAFM capsules used previously, in order to compensate for the very low flux levels in the two irradiations. The mass content of ^{10}B and ^6LiF in each capsule ranged from 60 to 70 mg.

Before helium analysis, the bare crystals were first "etched" to correct for alphas that may have been lost (or gained from nearby samples) because of α -recoil. Since the α -recoil distance increases with increasing neutron energy, an upper limit to the α -recoil distance was chosen so as to correspond to that neutron energy (in the source spectrum) above which <1% of the reactions occurred. Etching the outer surface layer of each crystal to a depth equal to this recoil length should, therefore, result in a total α -recoil uncertainty of <<1%. For the neutron spectrum in $\Sigma\Sigma$, the calculated 1% α -recoil lengths for ^{10}B and ^6LiF were 2.6×10^{-3} cm and 3.8×10^{-3} cm, respectively. In addition to etching, the bare ^6LiF crystals were subdivided into two separate samples in order to reduce their size for optimum vaporization in the mass spectrometer tungsten coil furnaces. This also had the advantage of increasing the total number of samples, resulting in improved statistics in the final results.

Preliminary uncorrected results of all the analyses are shown in Table 7. The last column shows the weighted mean helium concentration and associated 1σ standard deviation for each sample type. As with the BIG-10 and CFRMF data, unspiked samples were weighted by a factor of 1/2. From the data it is clear that the ^{10}B capsule and crystal results are in good agreement. As expected, the measured helium concentrations in the natural boron crystals are somewhat higher than for the enriched boron (~1.54 appb as compared to ~1.46 appb) due to their much lower

TABLE 7
HELIUM GENERATED IN BORON AND LITHIUM IRRADIATED IN SIGMA SIGMA

Sample Name	Type	Mass* (mg)	⁴ He Atoms Measured	⁴ He Concentration (appb)	Weighted [†] Average
RB-02	¹⁰ B Capsules	60.129	4.892×10^{12}	1.464	1.454 ±0.008
RB-P2		57.981	4.609×10^{12}	1.432	
RB-R2		62.479	5.091×10^{12}	1.467	
RB-S2		58.716	4.735×10^{12}	1.451	
RB-C2	¹⁰ B Crystals	6.612	5.358×10^{11}	1.459	1.469 ±0.010
RB-C4		6.375	5.149×10^{11}	1.454	
RB-C6		4.141	3.369×10^{11}	1.464	
RB-C8		4.392	3.659×10^{11}	1.499	
RNB-C4	Natural Boron Crystals	6.526	1.112×10^{11}	1.545	1.536 ±0.005
RNB-C5		6.117	1.208×10^{11}	1.523	
RNB-C7		5.910	1.002×10^{11}	1.537	
RNB-C10		5.627	0.955×10^{11}	1.538	
RL-L1	⁶ LiF Capsules	67.202	1.384×10^{12}	0.8631	0.8633 ±0.0008
RL-M1		69.127	1.425×10^{12}	0.8645	
RL-N1		64.969	1.337×10^{12}	0.8627	
RL-P1		66.830	1.372×10^{12}	0.8606	
RL-Q1		68.133	1.407×10^{12}	0.8654	
R6LiF-38A	⁶ LiF Crystals	8.856	1.808×10^{11}	0.8560	0.8559 ±0.0015
R6LiF-38B		8.478	1.719×10^{11}	0.8499	
R6LiF-39A		10.941	2.242×10^{11}	0.8591	
R6LiF-39B		5.256	1.080×10^{11}	0.8616	
R6LiF-40A		8.580	1.758×10^{11}	0.8588	
R6LiF-40B		10.305	2.099×10^{11}	0.8538	
R6LiF-41A		4.884	0.989×10^{11}	0.8487 [§]	
R6LiF-41B		3.492	0.713×10^{11}	0.8558	

*All bare crystals were etched to remove effects of ⁴He atoms lost or gained (from nearby samples) due to α-recoil.

†Weighted mean and standard deviation (1 σ) of the mean. Helium concentrations are calculated with respect to ¹⁰B or ⁶Li.

§Unspiked sample analyzed for ³He as well as ⁴He.

neutron self-shielding. It is expected that the corrections for self-shielding, etc., will bring all the boron results into agreement. The bare lithium crystal data differ by less than 1% from the results obtained from the encapsulated crystals.

The capsules containing boron and lithium were vaporized in the mass spectrometer furnace as a unit together with the contents, thus eliminating the need to correct for α -recoil losses. Irradiated empty versions of these capsules, which were constructed of the same thin-wall stainless steel tubing, were also analyzed for helium in the mass spectrometer, and resulted in negligible amounts of helium being released. Corrections for flux gradients are expected to be significantly higher in Sigma Sigma (up to 6 to 7%) than in the BIG-10 or CFRMF; however, they will be accurately determined and thus are not expected to contribute significant error to the final results.

The uncorrected ^6LiF results, also shown in Table 7, indicate a helium concentration of $\sim 0.859 \pm 0.002$ appb. When this result is combined with the result for the natural boron (the ^{10}B results are excluded at this time because of the higher self-shielding correction), a preliminary $^{10}\text{B}/^6\text{Li}$ reaction ratio of ~ 1.79 is obtained.

Discussions are currently proceeding with A. Fabry (CEN/SCK) and E. P. Lippincott (HEDL) to determine the exact self-shielding, flux gradients, and other corrections which are to be applied to the present data. The results of these corrections will be reported at the Third ASTM-EURATOM Symposium on Reactor Dosimetry, Ispra, Italy, in October 1979.

D. Fission Cavity Irradiation

A separate 1-meter spherical cavity, situated in the vertical graphite column of BR1, has been used to produce a fission neutron spectrum. The experimental geometry employed here, shown in Figure 2, is similar to that used previously by Fabry et al.¹³ Boron and lithium HAFMs were irradiated, along with radiometric foils provided by CEN/SCK, inside

Configuration of HAFMs in the BR1 Fission Cavity

Figure 2.

small cylindrical aluminum boxes (8 mm OD, 46 mm long) at three locations (+30 cm, 0, -30 cm) inside a 1-meter long cadmium tube (10 mm OD, 8 mm ID). The central 6.8 cm section of this tube was wrapped by a 0.1-mm-thick enriched uranium (93% U^{235}) foil, providing the cylindrical "fission cavity." With this arrangement, the HAFMs in the central region of the cadmium tube were exposed to a thermal-neutron-induced fission spectrum adequately described by a Maxwellian distribution having a mean neutron energy of 1.97 MeV.¹³ The cadmium tube provided a shield for the HAFMs (and foil sets) from the incident thermal neutrons from the graphite column. Irradiation of HAFMs near the top and bottom of the cadmium tube provided a means for measuring the additional ^4He generation in the boron and lithium from the approximately epithermal wall return neutron flux that was not absorbed by either the cadmium tube or the uranium foil. Thus, by subtracting the ^4He concentration in the ^{10}B and ^6LiF obtained at the outer locations from their respective values in the central region, the response of ^{10}B and ^6Li in a pure fission spectrum can be obtained.

The helium results from the various samples are listed in Table 8. As for the $\Sigma\Sigma$ irradiation, the bare crystals were first etched to minimize the effects of α -recoil. The appropriate α -recoil lengths determined for the boron and lithium in the fission cavity were 3.8×10^{-3} cm and 5.1×10^{-3} cm, respectively. Because of the heavy neutron self-shielding of the wall return neutrons at the three locations, the most useful helium production data will come from the ^{10}B and ^6LiF HAFM capsules rather than from the bare natural boron and ^6LiF crystals. This is because the self-shielding corrections are largely dependent upon the effective surface area of the sample, which, for small crystals, will vary significantly from one sample to another. Adding to this is the very low helium generation in the bare crystals (10^9 to 10^{10} atoms), with the resulting increased uncertainty ($\sim 10\%$) in the final measurements. The HAFM capsules, on the other hand, generated a much larger amount of helium (10^{11} to 10^{12}) atoms, resulting in significantly increased accuracy ($\sim 1\%$) in the helium determinations. In addition, their more uniform cylindrical geometry will significantly reduce the self-shielding uncertainties of the measurement of the wall return background flux.

TABLE 8
HELIUM GENERATED IN BORON AND LITHIUM
IRRADIATED IN B1 FISSION CAVITY

Sample Name	Type	Location	Mass* (mg)	⁴ He Atoms of Measured	⁴ He Concentration† (appb)
RB-Y2	¹⁰ B Capsule	Center	58.161	8.004×10^{11}	0.2477
RB-Z2	¹⁰ B Capsule		62.551	8.852×10^{11}	0.2547
RNB-C3	Boron Crystal		7.047	2.28×10^{10}	0.293
RNB-C6	Boron Crystal		6.928	2.46×10^{10}	0.322
RB-Q3	¹⁰ B Capsule	+30 cm	52.385	9.33×10^{10}	0.0320
RB-T2	¹⁰ B Capsule		60.137	1.050×10^{11}	0.0314
RNB-C9	Boron Crystal		2.321	2.6×10^9	~0.10
RNB-C1	Boron Crystal		6.881	8.4×10^9	~0.11
RB-I3	¹⁰ B Capsule	-30 cm	54.505	9.43×10^{10}	0.0312
RB-D3	¹⁰ B Capsule		64.280	1.028×10^{11}	0.0297
RNB-C2	Boron Crystal		6.912	4.3×10^9	~0.056§
RNB-C8	Boron Crystal		3.075	2.4×10^9	~0.073
RL-C2	⁶ LiF Capsule	Center	68.092	3.507×10^{11}	0.2159
RL-T1	⁶ LiF Capsule		72.740	3.913×10^{11}	0.2255
R6LiF-45	⁶ LiF Crystal		7.946	3.87×10^{10}	0.204
R6LiF-44	⁶ LiF Crystal		11.138	5.68×10^{10}	0.214
RL-E2	⁶ LiF Capsule	+30 cm	70.662	3.87×10^{10}	0.0230
RL-S1	⁶ LiF Capsule		67.246	3.36×10^{10}	0.0209§
R6LiF-43	⁶ LiF Crystal		11.438	4.9×10^9	~0.018
R6LiF-42	⁶ LiF Crystal		12.814	5.1×10^9	~0.017
RL-K2	⁶ LiF Capsule	-30 cm	69.672	4.22×10^{10}	0.0254
RL-I2	⁶ LiF Capsule		67.402	3.63×10^{10}	0.0226
R6LiF-31	⁶ LiF Crystal		10.724	3.6×10^9	~0.014
R6LiF-30	⁶ LiF Crystal		16.095	6.6×10^9	~0.017

*All bare crystals were etched to remove effects of ⁴He atoms lost or gained (from nearby samples) due to α -recoil.

†Helium concentrations are calculated with respect to ¹⁰B or ⁶Li.

§Unspiked sample analyzed for ³He as well as ⁴He.

From Table 8, preliminary uncorrected helium concentration results for the ^{10}B capsules are 0.251 ± 0.005 appb inside the fission source, and 0.031 ± 0.001 appb at locations at ± 30 cm. Similarly, for the ^6LiF capsules, uncorrected helium concentrations are 0.221 ± 0.007 appb inside the fission source and 0.023 ± 0.002 appb at ± 30 cm. Direct subtraction of the respective wall return background measurements at ± 30 cm yields preliminary helium generation results for a ^{235}U fission neutron spectrum of 0.220 ± 0.006 appb and 0.198 ± 0.009 appb for ^{10}B and ^6Li , respectively. The preliminary $^{10}\text{B}/^6\text{Li}$ reaction ratio obtained from these two results is ~ 1.1 .

Corrections for such factors as neutron self-shielding, particularly in the wall return background spectrum, as well as other smaller corrections such as flux gradients and absorption of wall return neutrons by the uranium, are currently being discussed with A. Fabry (CEN/SCK). Final results for these corrections, including correlation of the helium generation data with other independent flux and spectral measurements performed in the Fission Cavity, will be reported at the Third ASTM-EURATOM Symposium on Reactor Dosimetry, Ispra, Italy, in October 1979.

Expected Achievements

The direction of future Rockwell contributions to the ILRR program will be as follows:

- 1) Compute self-shielding corrections for ^{10}B and ^6LiF in $\Sigma\Sigma$ and the Fission Cavity, and deduce final values for the reaction rates of ^{10}B and ^6Li .
- 2) Correlate the spectrum integrated (n,α) cross sections of ^{10}B and ^6Li with ILRR program neutron spectra for CFRMF, BIG-10, $\Sigma\Sigma$, and the Fission Cavity, and with ENDF/B evaluated cross section data.
- 3) Implement irradiations of ^{10}B and ^6LiF HAFMs in additional benchmark spectra indentified in the ILRR program plans.
- 4) Implement irradiations of the HAFM materials with high (n,α) cross sections in CFRMF while operated at 100 kW.

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ANL

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REACTION RATE MEASUREMENTS OF FISSION AND NONFISSION FOILS
IRRADIATED IN THE NBS ^{252}Cf FISSION SPECTRUM IRRADIATION FACILITY

R. R. Heinrich and R. J. Popek

Objective

To obtain absolute values for infinitely dilute reaction rates from fission and nonfission foils of selected fast-neutron dosimetry materials irradiated in the NBS ^{252}Cf Fission Spectrum Irradiation Facility. Assuming sufficient accuracy will be achieved, these values will permit the testing and evaluation of energy-dependent cross-section sets from the ENDF/B file.

Summary

Reaction rates have been measured for two ^{235}U foils, four nickel foils, and one stainless steel foil which were irradiated in the NBS ^{252}Cf Fission Spectrum Irradiation Facility. Pertinent data used in the calculation of these reaction rates are also included. The quality of these data appears to be good with only the ^{140}Ba (1.596 MeV) reaction rates from both ^{235}U foils having relatively large uncertainties.

Accomplishments and Status

Four nickel foils, two ^{235}U foils, and one stainless steel foil were irradiated in the NBS ^{252}Cf Fission Spectrum Irradiation Facility on July 5, 1977. Reaction rates have been measured for the production of ^{95}Zr , ^{103}Ru , and ^{140}Ba from the ^{235}U foils, and ^{58}Co from both the nickel and stainless steel foils. The production rate for ^{54}Mn is also reported for the stainless steel foil. These data appear to be internally consistent with the exception of the ^{140}Ba reaction rate. The 1.596 MeV peak of this nuclide apparently contained an unknown interference. Half-life data accumulated for this peak over a period of 15-40 days indicated a decay rate of 11.8 days for ^{235}U foil #3, and 12.2 days for foil #6. This behavior was quite unexpected since, in the past, this peak has behaved well and has become somewhat of a standard for monitoring reactor environments.

The data reported in Tables ANL-1 and ANL-4 do not include corrections for neutron scattering which may be necessary to normalize the data from all the reporting laboratories. Also included are pertinent data used in the calculation of the reported reaction rates.

Expected Achievements

None. Work is completed.

TABLE ANL-1
Reaction Rates from the NBS ^{252}Cf
Fission Neutron Irradiation Facility

Foil	Reaction	Decay Range (days)	Number of Spectra	Gamma Energy (KeV)	Reaction Rate (10^{-18} atom/atom-sec)	% Error (1 σ) Relative ^a	% Error (1 σ) Absolute ^b
3Ni-LM	$^{58}\text{Ni}(n,p)^{58}\text{Co}$	16-40	2	810.8	1.66	0.7	2.3
3Ni-JC					1.77	2.2	3.1
6Ni-JG					1.82	1.0	2.4
6Ni-LP					1.63	1.1	2.5
ANL-2 3U	$^{235}\text{U}(n,f)^{95}\text{Zr}$	25-98	4	724.2	1.12	1.7	2.8
	$^{235}\text{U}(n,f)^{103}\text{Ru}$	16-98	5	497.1	0.581	1.2	3.0
	$^{235}\text{U}(n,f)^{140}\text{Ba}$	15-25	2	1596.2	1.02	5.7	6.1
6U	$^{235}\text{U}(n,f)^{95}\text{Zr}$	25-222	4	724.2	1.12	2.2	3.3
	$^{235}\text{U}(n,f)^{103}\text{Ru}$	18-222	5	497.1	0.590	1.5	2.9
	$^{235}\text{U}(n,f)^{140}\text{Ba}$	18-26	2	1596.2	1.03	3.2	3.9

^aStandard deviation of the mean of replica counts.

^bError analysis includes the relative error, the detector calibration error, the ILRR recommended branching ratio error, and the mass determination errors.

TABLE ANL-2
Physical Constants for Target Atom Determinations

Reaction	Sample	Weight, grams		Atomic Weight of Elements (grams)	Isotopic Abundance (atom %)	Chemical Purity (%)	Sample Thickness (mm)
		ANL	RE, Inc.*				
$^{58}\text{Ni}(n,p)^{58}\text{Co}$	3Ni-LM	0.2957	.2959	58.70	68.3	99.6	0.25
	3Ni-JC	0.2942	.2945				
	6Ni-JG	0.2966	.2969				
	6Ni-LP	0.2942	.2946				
$^{235}\text{U}(n,f)\text{FP}$	3U	1.2188		235.04	99.89	100.0	0.51
	6U	1.2503		235.04	99.89	100.0	0.51

*Weights used for calculations.

TABLE ANL-3
Parameters for $\sigma\phi$ Calculations

Isotope	Energy (KeV)	Half-Life (days)	Gamma Intensity	Gamma Self- Absorption Correction	$(1-e^{-\lambda T})$
^{58}Co	810.8	70.85	.9944	.992	.0993
^{95}Zr	724.2	64.1	.441	.945	.1092
^{103}Ru	497.1	39.43	.89	.912	.1713
$^{140}\text{Ba-La}$	1596.2	12.789	1.15097	.975	.440

End of Irradiation - 9:30 EDT, July 5, 1977

Total Elapsed Irradiation Time (T) - 10.69D

TABLE ANL-4

Reaction Rates of Stainless Steel (SS) Detector
from the NBS ^{252}Cf Fission Neutron Irradiation Facility

Reaction	Decay Range (days)	Number of Spectra	Gamma Energy (KeV)	Reaction Rate (10^{-15} atoms/atom-sec)	% Error (1 σ)
$^{54}\text{Fe}(\text{n},\text{p})^{54}\text{Mn}$	18-231	4	834.8	1.37	2.4
$^{58}\text{Ni}(\text{n},\text{p})^{58}\text{Co}$	18-231	4	810.8	1.84	2.2

TABLE ANL-5

Physical Constants for SS Target Atom Determinations

Reaction	Weight, grams		Chemical Purity (%)	Isotopic Abundance (atom %)	Sample Thickness (mm)
	ANL	NBS*			
$^{54}\text{Fe}(\text{n},\text{p})^{54}\text{Mn}$	.7308	.731090	63.15	5.8	0.76
$^{58}\text{Ni}(\text{n},\text{p})^{58}\text{Co}$	.7308	.731090	36.85	68.3	0.76

*Weights used in calculations.

TABLE ANL-6

Parameters for SS $\sigma\phi$ Calculations

Isotope	Energy (KeV)	Half-Life (days)	Gamma Intensity	Gamma Self-Absorption Correction	($1-e^{-\lambda T}$)
^{58}Mn	834.8	312.5	.9997	.980	.002211
^{58}Co	810.8	70.85	.9944	.980	.009715

End of Irradiation - 11:02 EST, June 27, 1977

Total Elapsed Irradiation Time (T) - 0.9979D

ANL-4

BNL

Brookhaven National Laboratory

STATUS OF DATA TESTING OF ENDF/V REACTOR DOSIMETRY FILE
Benjamin A. Magurno

OBJECTIVE

Data Testing of ENDF/V Reactor Dosimetry File

SUMMARY

The ENDF/V Reactor Dosimetry file was released August 1979, and was immediately followed by start of Phase 2 data testing. Preliminary results from BNL were presented at the 3rd ASTM-EURATOM Symposium.

MOL

CEN-SCK LABORATORY

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A. Interlaboratory Niobium Dosimetry Comparison in EBR-II
(Status : June 1979)

H. TOURWE

1. Introduction

The aim of this interlaboratory niobium dosimetry comparison is to

- compare the different measuring techniques
- improve the calibration of the different equipment
- improve the different parameters used for the interpretation of niobium measurements
- improve the knowledge of the ^{93}Nb (n, n') cross-section
- study the influence of impurities in niobium.

The following laboratories are involved:

- (a) Atomic Energy Establishment, Winfrith, United Kingdom (W.H. Taylor)
- (b) C.B.N.M., Geel, Belgium (W. Bambynek, D. Reher)
- (c) C.E.N., Grenoble, France (R. Lloret)
- (d) HEDL, Hanford, U.S.A. (E.P. Lippincott, L.S. Kellogg)
- (e) KFA, Jülich Germany (W. Schneider, L. Weise)
- (f) Max Planck Institute, Stuttgart, Germany (K. Schulze)*
- (g) S.C.K./C.E.N., Mol, Belgium (H. Tourwé)

*as supplier of the high quality Nb

2. Description of the Experiment

The Nb together with multifoil dosimeters were loaded into run 97 of the fast breeder EBR-II (U.S.A.). The experiment was loaded in row II of the reactor. The irradiation time was about 6 weeks. The total flux in the core centre at a nominal power of 62.5 MW is $\sim 3 \times 10^{15}$ n/cm²-sec.

The average neutron energy at this location is ~ 0.8 MeV. The irradiation started in October 1978. The different types of "European" dosimeters were loaded in the following manner:

<u>Distance from Reactor Midplane (inch)</u>	<u>Foils (7)</u>	<u>Wires (8)</u>	<u>Fission⁽¹⁾ Monitors</u>	<u>Type of Container</u>
-23.8, 15.5, 23.4	-	Fe, Ni, Cu, Ti, Nb(9), 1% Co/V	-	Dos(2)
-14.4, 0	-	Fe, Ni, Cu, Ti, Nb(9), 1% Co/V	²³⁸ U, ²³⁵ U ²³⁷ Np	Dos(2)
0, -14.4	10 Nb (MPI)(3) and Ti spacers(5)	-	-	SSTR 1(4)
0, -14.4	2 Nb (France) Ti spacers, Cu, Ti, Ni, Fe	-	-	SSTR 2(4)
0, -14.4	3 Nb (commercial)(6)	-	-	SSTR 3(4)

(1) The fission monitors are being provided by HEDL. They are oxide wires encapsulated in 0.050 inch diameter vanadium.

(2) Dos refers to a type of container used to hold dosimeter wires and capsules i.e. a right circular cylinder with several small holes drilled to contain the dosimeters.

- (3) Max Planck Institute.
- (4) SSTR refers to the dosimeter container system for the foils - see attached figure. The different types of Nb foils will be separated into SSTR 1, 2 and 3 at each axial level. SSTR 1, 2 and 3 will be put together in one box at each axial level.
- (5) Ti spacers: see Figure Mol A-1.
- (6) Commercial niobium: one Nb foil from three different companies
 - Highways International
 - Goodfellow Metals
 - Kawecki Chemical.
- (7) All foils have a diameter of 3 mm.
- (8) All wires have a length of 3 mm.
- (9) Nb needles: commercial niobium of Kawecki Chemical.

Besides the dosimeters mentioned in the table above, different U.S. laboratories loaded dosimeters in the same experiment.

3. Practical Arrangements

All dosimeters mentioned in the table above, were sent to Mol in June 1979. The small containers SSTR 1, 2 and 3 will be opened at Mol. Mol will measure all Nb foils in a non-destructive and relative way before sending them to the different laboratories involved. The laboratories (a), (b), (d), (e) will receive 4 Nb foils (origin Max Planck Institute) i.e. two foils from the level 0" and two foils from the level -14.4". The 4 foils from France will be returned to Grenoble. It is suggested that the results should be sent to the SCK/CEN, Mol, not later than 3 months after the date the Nb was received. In order to compare results, it is strongly advised

that all results be very well documented: description of the measuring technique, calibration procedure of the measuring equipment, applied corrections, used constants, S.C.K./C.E.N. will coordinate and publish the Nb results.

The commercial niobium will be measured at Mol. The aim of these measurements is to study the influence impurities in niobium.

All other dosimeters are intended for neutron spectrum measurements. They have already been counted by HEDL. SCK/CEN will also count the fission and non-fission dosimeters, after which SCK/CEN will ship all fission dosimeters to KFA Jülich. It is suggested that ILRR constants be used for the analysis of these dosimeters. HEDL will coordinate these results.

It was agreed with HEDL that all information obtained with the "European" and the "U.S." dosimeters will be exchanged.

Nb FOIL SAMPLE CONTAINER

FIGURE MOL A-1

Blank

MOL-6

B. Absolute measurements of highly active dosimeters, irradiated
in high flux nuclear reactors in order to determine
neutron fluence and neutron spectrum

H. TOURWE

1. Introduction

The development of high flux reactors, in particular fast reactors, demands a continuous study of fuels and new or improved structural materials. One of the principal aspects of such studies is the behaviour of these materials under irradiation conditions approaching the actual irradiation conditions in fast reactors. For this reason, test irradiations are carried out in high flux reactors such as RAPSODIE (France), KNK II (Germany), EBR II (U.S.), BR2 (Belgium). The irradiation devices used are often equipped with dosimeters in order to establish the correlations between the characteristics of the fuel or of the materials after irradiation, and the parameters which characterize the neutron environment. The parameters which characterize the neutron environment are the neutron flux, instantaneous and integrated, and the neutron spectrum.

Table I shows a list of detectors frequently used to measure fluence and the neutron spectrum. Only detectors emitting gamma rays appear in the list. The activity of the dosimeters varies from a few μCi to about a hundredth of mCi , depending on the type of dosimeter and the neutron fluence which generally exceeds $10^{22} \text{ n cm}^{-2}$. A measuring device has been constructed to measure this activity. The calibration of the system is done in the first place with ^{152}Eu reference sources.

2. Description of the measurement equipment

A device, made up of a "tunnel" which can hold the dosimeters, has been designed to measure very active sources. The tunnel has an interior section of 11 cm by 11 cm; it is shielded with 10 cm of lead. A Ge(Li) crystal of low efficiency

(relative efficiency $1.33 \text{ MeV} \leq 5 \%$) shielded with 6 cm of lead is placed in front of opening of the tunnel, at one of its extremities. The dosimeter to be measured is placed on a movable cart, at the other end of the tunnel. This allows regulating the distance between the detector and the Ge(Li) crystal as a function of the activity. The precision of this movable system is of the order of 1 mm. The maximum distance from the detector to crystal is around 4 m. In order to measure very high activity ($> 5 \text{ mCi}$) a collimator, made up of lead and tungsten, is mounted in the tunnel.

The measurement circuit of the Ge(Li) is connected to a SCORPIO 3000 analyser. This system consists of a PDP-11/04 computer with a central memory of 28K (16 bits). Four ADC (Analog/Digital Converter), capable of collecting the measurement data in parallel, are mounted on the secondary unibus. During the measurements the data are collected in an auxiliary memory of 32K (16 bits). The measured spectra are stored on floppy disks. The software of the analyzer is conceived in such a fashion that the analysis of simple spectra is made in the "background" while the data of the four ADC are collected simultaneously in the "foreground".

3. Analysis of the gamma spectra

All the spectra measured are analyzed with a PDP-11/04 computer. The SPECTRAN software developed by the CANBERRA is used for the analysis of the spectra having a simple structure (a small number of well isolated peaks). This method has the advantage of being applied in the "background" of the system while the measurements are done in the "foreground". In SPECTRAN, the efficiency curve is represented by two parabolas of second degree in $\ln E$: one parabola for the low energies and one for the high energies. The intersection point of the two parabolas is an experimental value. To calculate the surface of the peaks, the code takes an interval equal to the width on $1/10$ of the height increased by 4 channels. The width on $1/10$ of the height is determined as a function of the energy by Gaussian fitting of the peaks of a standard spectrum. The subtraction of the background is linear.

The program SAMPO developed by J. Routti and S. Prussin [1] is used to analyze the more complex spectra (a large number of peaks and doublets). This program is used to analyze the spectra of fissile detectors (^{237}Np , ^{235}U , ^{238}U , ^{239}Pu). The SAMPO program, conceived for large computers, has been adapted for a PDP-11/04 having a memory of 28K. To this end, an overlay structure has been developed for the SAMPO sub-routines so that at any moment no superfluous sub-routines are present in the central memory.

A modification has been applied to the original version of J. Routti and S. Prussin. In the original version each peak is adjusted by a Gaussean and two exponentials (an exponential for each slope). Several years of experience with SAMPO on IBM have nevertheless demonstrated that the descending slope comes much closer to a Gaussean curve than does the ascending slope. For this reason, only an exponential is used in the adapted version. Several options in the original version (for example, the reject criterion for small peaks), have been replaced by fixed values in the version for the PDP-11/04.

The SAMPO program on PDP-11/04 is executed in the "batch" mode under RT11. All the input data are stored in the "batch" files on floppy disks.

The application of SAMPO and of SPECTRAN on the spectra with a small number of well isolated peaks leads to results which differ by 1.5 % at the most. Therefore it is justified to use a simplified software for the analysis of such spectra.

4. Calibration of the measuring device

It is necessary to determine the absolute activity of the dosimeters in order to determine the neutron fluence from the activity measurements. This necessitates knowledge of the efficiency curve as a function of the gamma energy at each measurement position. No standard source with high activity has been available in the trade. This has led the SCK/CEN to develop its proper techniques for calibration of the measurement devices.

The first technique is based on the use of reference sources ^{60}Co , ^{198}Au , ^{64}Cu and of sources $^{110\text{m}}\text{Ag}$. The $^{110\text{m}}\text{Ag}$ sources obtained by irradiation of Ag in the BR2 reactor at Mol, helps to determine the shape of the efficiency curve as a function of the energy. The gamma rays and the corresponding relative intensities are given in Table 2. The energy values and intensity values are taken from reference [2]. The absolute value of the efficiency curve is determined by means of a ^{60}Co standard. To improve the precision in the energy range 400 keV - 600 keV ($^{110\text{m}}\text{Ag}$ only gives one point) isotopes having a short half-life such as ^{198}Au and ^{64}Cu are utilized. These isotopes, activated in the BR1 reactor at Mol, are measured at first where the efficiency curve is determined. Then after sufficient decay, the absolute activity is measured by means of a NaI reference crystal [3]. Table 3 shows the parameters used. In this fashion, not only two supplementary points are obtained but also a verification of the absolute level of the efficiency curve determined by the ^{60}Co sources. In table 4, the measured and calculated energy values for the 411 keV and 511 keV are compared. The calculated values are the interpolated values of the function adjusted across the measurement points of ^{60}Co and $^{110\text{m}}\text{Ag}$. The conclusion of these results is that the error is less than 3 %. Thus, for a given geometry (a crystal and a collimator) the efficiency curve is determined at the center of the tunnel. For determination of the efficiency curve at another measurement point, it is sufficient to measure a ^{60}Co standard at that place, because for a given geometry the shape of the efficiency curve is independent of the position in the tunnel (distance source - collimator $\gg$ collimator diameter). To demonstrate this, some measurements have been made using the same $^{110\text{m}}\text{Ag}$ source, at three points (A, B, C) in the tunnel. The distance between each point was one meter. Position A is located about 95 cm from the Ge(Li) crystal. A function $f(E) = a_1 (\ln E)^2 + a_2 (\ln E) + a_3$ is adjusted across the measurement points at each position. The results are summarized in table 5. These results show that for a given geometry, the efficiency curve is independent of the position. The difference between the maximum and minimum value of the ratio $f_{\text{pos } i}(E)/f_{\text{pos } j}(E)$ is equal to or less than 1 %. At the limits of adjustment interval, the difference sometimes exceeds 1 %. The cause of this is the much lower precision of the adjustment curve at these locations. This technique gives excellent results in the energy range of 600 keV to 1400 keV. The principal disadvantages of this technique are its complexity and the insufficient number of calibration points below 600 keV.

To avoid these disadvantages, a series of ^{152}Eu standards of high activity is developed. These ^{152}Eu standards allow standardizing the system, at a given position by means of a single measurement in an energy range of 100 keV to 1500 keV. Eu-Al alloys, enriched with ^{151}Eu , have been irradiated in BR2. The characteristics of Eu-Al alloys, non-irradiated, are shown in Table 6. The sources are calibrated, one after the other, in order of increasing activity. The weakest Eu source ($^{152}\text{Eu-Mol 1}$) has been standardized by comparing its activity to a PTB* ^{152}Eu reference source (activity : 4.39 μCi the 1/1/79). The ratio of the activity of the two sources has been determined by three measurement installations : a Ge(Li) crystal, a plane NaI and a thin NaI. The results of the measurements are summarized in Table 7. The ratios measured have been deduced from the surfaces of the peaks. The activity ratios between $^{152}\text{Eu-Mol 1}$ and $^{152}\text{Eu-Mol 2}$ have been determined by two installations : the Ge(Li) crystal and the thin NaI (Table 8). The activity of $^{152}\text{Eu-Mol 2}$ was too high to permit measuring with the plane NaI. The $^{152}\text{Eu-Mol 3}$, $^{152}\text{Eu-Mol 4}$... can only be measured with the Ge(Li) because of their high activity. The ratio of activity measured between $^{152}\text{Eu-Mol 3}$ and $^{152}\text{Eu-Mol 2}$ is equal to 3.82 ± 0.01 . The absolute activities are shown in Table 9. This table also gives the thermal fluence received in the BR2 reactor. Determination of the presence of ^{154}Eu in each source is based on the measurement of ray of 1274 keV (absolute intensity : 35 %). The values given for $^{152}\text{Eu-Mol 4}$ are preliminary. The activities shown in Table 9 for the standard sources $^{152}\text{Eu-Mol 2}$ and $^{152}\text{Eu-Mol 3}$ have been checked by remeasuring the activities of the reference sources, as ^{198}Au and ^{60}Co , by means of $^{152}\text{Eu-Mol 2}$ and $^{152}\text{Eu-Mol 3}$.

The activity of the ^{60}Co standard source, measured with the $^{152}\text{Eu-Mol 3}$ was 1 % greater than the value given for the ^{60}Co standard. The activity of the ^{198}Au reference source, measured with the $^{152}\text{Eu-Mol 2}$, was 0.6 % less than the value of the ^{198}Au source measured with the plane NaI [3].

We must emphasize that the maximum activity of ^{152}Eu which one can obtain by irradiating the sources described in Table 6 is of the order of 10 mCi. The activity curve of ^{152}Eu as a function of the neutron fluence presents a maximum due to the large thermal cross section of ^{151}Eu (~ 9200 b) and also to the

*Physikalisch-Technische Bundesanstalt.

large thermal cross section of ^{152}Eu (~ 2300 b). The maximum is located near $n_{\text{v}_0} t \approx 2.10^{20} \text{ n cm}^{-2}$. The presence of ^{154}Eu in the ^{152}Eu standards increases as a function of the neutron dose received. The cause is the relatively small thermal cross section of ^{153}Eu and of ^{154}Eu in comparison with the thermal cross section of ^{151}Eu and of ^{152}Eu . This is confirmed by the experimental results in Table 9.

The ^{152}Eu standards are routinely used to measure the absolute activity of dosimeters of high activity. The peaks of the spectrum of ^{152}Eu used for calibration purposes are shown in Table 10. The same table gives the values of the absolute intensities [4]. For the half-life a value of 13.3 y is used. Apart from their long half-life, the advantage which these sources present is that they permit us to determine, by a single measurement, the efficiency curve in the energy range of 100 keV to 1500 keV.

Conclusion

The techniques described in this document have been used for many years at SCK/CEN at Mol, to measure the absolute activity of dosimeters with high activity. In the near future, new ^{152}Eu standards with high activity will be developed in order to measure dosimeters having activity up to ~ 100 mCi. In general, one estimates that within a delay of 5 years, one should attain a precision better than 5 % on the measurements of fluence and neutron spectrum. This implies a precision of 2 % in the measurements of activity of dosimeters.

References

- [1] J.T. Routti , S.G. Prussin : Photopeak method for the computer analysis of gamma-ray spectra from semiconductor detectors. Nucl. Instr. Meth., 72 (1969) p. 125-142.
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- [3] A. Fabry, F. Cops : Calibration of the SCK/CEN reference NaI gamma ray spectrometer used for reactor dosimetry applications. Interlaboratory reaction rate program, 11th progress report (Nov. 1975 - Oct. 1976), HEDL-TME 77-34.
- [4] K. Debertin : Intercomparison of gamma-ray emission rate measurements by means of Germanium spectrometers and ¹⁵²Eu sources. PTB-Bericht. PTB-Ra-7 (ICRM-S-3), Februar 1978.

FIGURE MOL B-1
MEASURING EQUIPMENT FOR HIGH ACTIVITY DOSIMETERS

TABLE MOL B-1

List of γ emitting detectors frequently used to measure
fluence and neutron spectrum

Nuclear reaction	Observed isotope	Energy (keV)	Half-live	Intensity (%)
$^{59}\text{Co}(n,\gamma)$	^{60}Co	1173,208 1332,464	5,268 y	99,86 99,986
$^{45}\text{Sc}(n,\gamma)$	^{46}Sc	889,258 1120,516	83,85 d	99,984 99,987
$^{58}\text{Ni}(n,p)$	^{58}Co	810,757	71,23 d	99,44
$^{54}\text{Fe}(n,p)$	^{54}Mn	834,827	312,6 d	99,97
$^{46}\text{Ti}(n,p)$	^{46}Sc	889,258 1120,516	83,85 d	99,984 99,987
$^{63}\text{Cu}(n,\alpha)$	^{60}Co	1173,208 1332,464	5,268 y	99,86 99,986
$^{235}\text{U}(n,f)$ $^{238}\text{U}(n,f)$ $^{239}\text{Pu}(n,f)$ $^{237}\text{Np}(n,f)$	^{95}Zr	724,184 756,715	64,4 d	44,2 54,8
	^{103}Ru	497,08	39,43 d	89,3
	^{137}Cs	661,638	30,03 y	85,0
	^{140}Ba	537,35	12,79 d	24,4
	^{140}La	1596,18	40,26 h	95,33
	^{144}Ce	133,53	284,4 d	10,7

TABLE MOL B-2 Relative intensities of ^{110m}Ag

Energy (keV)	Relative intensity (%)
446,79	3,81
657,74	100
677,60	11,26
706,67	17,57
763,93	23,99
884,67	77,87
937,48	37,40
1384,27	26,79
1505,01	14,32

TABLE MOL B-3: Parameters used for the calibration of ^{198}Au and ^{64}Cu sources

	Energy (keV)	Intensity	Half-live	NaI efficiency (contact)
$^{198}\text{Au}^{(1)}$	411,795	0,9547	2,6946 d	0,212
$^{64}\text{Cu}^{(1)}$	511,002	0,368	12,701 h	0,173

(1) dimensions : Au - $\varnothing$ 3 mm x 0,127 mm/Cu - $\varnothing$ 3 mm x 0,5 mm

TABLE MOL B-4

Control of the absolute level of the $^{110m}\text{Ag} - ^{60}\text{Co}$
 efficiency curve with ^{198}Am and ^{64}Cu

Isotope	Energy (keV)	Measured values E (arbitrary units)	Calculated values C (arbitrary units)	$\frac{E - C}{E} \times 100$ (%)
^{198}Au	411,795	13,43	13,83	- 3,0
^{64}Cu	511,002	9,61	9,87	- 2,7

TABLE MOL B-5: Comparison of the shape of the efficiency curve at
3 measurement positions (arbitrary units)

Fitting function $f(E) = a_1(\ln E)^2 + a_2(\ln E) + a_3$ (E in keV)					
	a_1		a_2		a_3
pos. A	0,15351		- 3,3919		16,238
pos. B	0,17452		- 3,6667		15,672
pos. C	0,15465		- 3,4176		14,033
<div> <div>Energy (keV)</div> <div> $\frac{f_{\text{pos. i}}(E)}{f_{\text{pos. j}}(E)}$ </div> </div>	500	700	1000	1300	1500
$\frac{f_{\text{pos. A}}(E)}{f_{\text{pos. B}}(E)}$	4,314	4,334	4,345	4,289	4,243
$\frac{f_{\text{pos. A}}(E)}{f_{\text{pos. C}}(E)}$	10,18	10,24	10,34	10,29	10,23

TABLE MOL B-6: Characteristics of the non-irradiated Eu sources

Alloy	Al - 4,8 wt % Eu
Isotopic analysis	$96,83 \pm 0,05$ at % ^{151}Eu
	$3,17 \pm 0,05$ at % ^{153}Eu
Thickness	0,118 mm
Diameter	3 mm

TABLE MOL B-7: Intercalibration of $^{152}\text{Eu-Mol 1}$ with $^{152}\text{Eu PTB}$

Installation	$\frac{\text{activity } ^{152}\text{Eu-Mol 1}}{\text{activity } ^{152}\text{Eu PTB}}$	σ	Energy (in keV) of the rays taken into account
Ge(Li)	5,35	0,02	122, 245, 344, 444 779, 964, 1112, 1408
Plane NaI	5,35	0,05	344
Thin NaI	5,33	0,02	39,5/40,1 45,4/46,6
Average value	5,340	0,007	

TABLE MOL B-8: Intercalibration of $^{152}\text{Eu-Mol 2}$ with $^{152}\text{Eu-Mol 1}$

Installation	$\frac{\text{activity } ^{152}\text{Eu-Mol 2}}{\text{activity } ^{152}\text{Eu-Mol 1}}$	σ	Energy (in keV) of the rays taken into account
Ge(Li)	3,081	0,007	122, 245, 344, 411, 444, 779, 964, 1112, 1408
Thin NaI	3,06	0,01	39,5/40,1 45,4/46,6
Average value	3,07	0,01	

TABLE MOL B-9: Absolute activities of the ^{152}Eu -Mol standards

Standard	$n \cdot v_{ot}$ (n/cm^2)	Activity of the 1408 keV ray on 1/1/79 (γ/s)	σ (%)	Absolute activity* on 1/1/79	Presence of ^{154}Eu (%)
^{152}Eu -Mol 1	$\sim 2.6 \cdot 10^{17}$	$1,808 \cdot 10^5$	0,7	23,43 μCi	$0,23 \pm 0,05$
^{152}Eu -Mol 2	$\sim 8.0 \cdot 10^{17}$	$5,549 \cdot 10^5$	0,8	71,93 μCi	$0,4 \pm 0,1$
^{152}Eu -Mol 3	$\sim 3,0 \cdot 10^{18}$	$2,119 \cdot 10^6$	0,8	274,6 μCi	$0,37 \pm 0,06$
^{152}Eu -Mol 4	$(1,0 \cdot 10^{19})$	$(7,119 \cdot 10^6)$	-	922,7 μCi	-

*based on an intensity of 0,2085 for the 1408 keV ray.

TABLE MOL B-10: Characteristics of the ^{152}Eu gamma rays
used for calibration of the measuring device

Energy (keV)	Intensity
121,8	0,2837
244,7	0,0751
344,3	0,2658
444,0	0,03121
778,9	0,1296
964,0	0,1462
1112,1	0,1356
1408,1	0,2085

C. Transport theory calculations in support of the design
of the high flux reference neutron field NIGHTMARE at the NBS reactor

L. FOUARGE

Objectives

- Contribute to the design of the NIGHTMARE high flux intermediate energy benchmark neutron field at the NBS reactor by means of preliminary one-dimensional discrete-ordinates transport theory calculations of the radial neutron flux spectra and reaction rates as a function of the boron-10 absorber thickness to be used for shaping of the neutron spectrum.
- Provide reduced group cross sections useful for eventual two-dimensional transport analysis of the as-built facility.

Summary

NIGHTMARE (NBSR Intermediate-energy Glory Hole Tailored Mass Assay and Reference Environment) is a high flux benchmark neutron field under design for implementation in the central glory hole of the NBS reactor. It is intended at quality assurance and calibration of neutron dosimeters applied in high flux fast breeder reactors, in particular the Fast Flux Test Facility (FFTF). The neutron spectrum will be tailored by a boron-10 cylindrical shell (protected from burn up by a gadolinium sleeve) so as to be similar to the low flux Intermediate-Energy Standard Neutron Field (ISNF).

One-dimensional transport theory calculations of NIGHTMARE have been performed in the S 8 diagonal transport approximation, using a 40 gr cross section library (commonly referred to as "MOL-BR2-40 gr") based on the KEDAK nuclear data file. Two boron thicknesses have been considered, as well as a case without filter.

Reduced cross sections with 16 neutron groups for 13 compositions have been prepared and stored on disk for future two-dimensional R-Z calculations.

Accomplishments

1. GEOMETRY USED FOR THE CALCULATION

To make the reactor configuration calculable by the usual R and R-Z codes, the fuel element and thimble lattice (fig. 1) was cylindricalised in the following manner. Adopting as axis the Glory Hole, the fuel elements were divided into three concentric crowns : the first contains six fuel elements, the second six fuel elements and six thimbles, the third eighteen fuel elements. This allows a two-dimensional R-Z calculation; but because of the moderator at the midplane, the usual buckling method is not valid for a one-dimensional calculation in a horizontal plane at the level of the NIGHTMARE facility. To overcome this difficulty, an axial buckling for the fast groups was determined from the fast flux axial curve (NBS estimate). To take into account the presence of the central moderator, a thermal neutron feeding was adopted, in the form of a negative buckling; this neutron feeding is proportional to the flux. In the one-dimensional calculation, the configuration was made critical by iterating on this negative buckling.

The facility height is 10 cm only; however the one-dimensional calculation was performed as if the middle section of this facility stretches over the whole reactor height; regarding the gadolinium and (in some cases) the boron presence, this fact disturbs the flux distribution, mainly the thermal flux and reduces the reactivity. However, as the fast flux spectrum was principally considered, the calculation was continued in this way.

2. CALCULATION SEQUENCE

The first calculation performed was a one-dimensional one, with at the centre the NIGHTMARE facility, with gadolinium, but without boron; this calculation was made with 40 neutron groups; to make the configuration critical, iterations were performed on the thermal neutron feeding.

This calculation was made with the angular coordinate system DS 4, with three meshes in the void at the centre of the facility; it produces the flux spectrum on which the cross-sections were reduced to 16 groups, for the two-dimensional calculation.

To make more reliable the comparison of the different cases, three calculations were performed in parallel, with the same neutron sources in the fuel, viz the fission sources produced by the critical problem mentioned above. To increase the accuracy of the flux spectrum and of the flux distribution in the void, without exaggerated increase of the calculation time, these calculations were made with the angular coordinate system DS 8, with ten meshes in the void at the centre of the facility; but the configuration was limited to the exterior radius of the first fuel crown; this probably modifies the ratio of the flux in the facility to the power level in the fuel crown, but does not seem to modify sensibly the flux spectrum in the facility.

A two-dimensional calculation is in progress, with a diffusion code, in R-Z geometry, with 16 neutron groups and 13 macroscopic cross-section sets (which does not mean 13 different compositions : indeed, one composition, of which the cross-sections are reduced on different spectra gives several reduced cross-section sets).

3. RESULTS

Spectra and response functions

Table 1 gives the spectra at the centre of the facility for the three cases (without boron, with 10 mm and with 15 mm boron). These spectra are drawn on fig. 2.

Figs 3, 4 and 5 give the response functions of ^{235}U and ^{238}U . As can be seen, the percentage fissions above 1 keV are :

	fissions above 1 keV %
without boron	17.6
boron 10 mm	84.9
boron 15 mm	93.0

For the case without boron, an inflexion occurs between 1 and 5 eV; this is due to the presence of gadolinium. The curves for 10 and 15 mm of boron are near each other. The three ^{238}U curves coincide practically. Fig. 5 represents two of these curves with a larger energy scale, which allows the difference between the two extreme curves to appear, hence confirming that this difference is negligible. Fig. 4 gives for the case without boron the ^{238}U response function for the centre and for the border of the central void; it is seen that the difference is small.

Reaction rate radial distribution

Fig. 6 gives the radial distribution of the ^{235}U fission rate, for the three cases (without boron, with 10 and with 15 mm boron).

The fall of the fission rate near the external radius comes from the fact that the calculated configuration is a configuration with neutron sources in the first fuel crown, but confined to the outer border of this crown; actually, there are three fuel crowns.

The presence of gadolinium and (in some cases) of boron leads to a reaction rate fall at the facility neighbourhood.

A reaction rate increase can be seen from the border to the centre of the facility; it is explained by the absorption in the gadolinium (and the boron) which favours the neutrons traversing the screen perpendicularly.

Fig. 7 gives the ^{238}U fission reaction rate versus the radius. There is also an increase in the facility void; this increase exists already in the most energetic group (from 6.5 to 10.5 MeV). It is explained by the moderation in the heavy water; that moderation touches less the neutrons directed to the centre. To be comparable to those of fig. 6, the ordinates of fig. 7 must be divided by 250.0.

The ratios of the centre reaction rates to the border ones are given in the following table :

	^{235}U	^{238}U
without boron	1.06	1.15
boron 10 mm	1.12	1.04
boron 15 mm	1.08	1.03

The ratios of the ^{238}U reaction rates to the ^{235}U ones are given in the following table :

	centre	border
without boron	4.42,-3	4.08,-3
boron 10 mm	3.96,-2	4.26,-2
boron 15 mm	5.34,-2	5.61,-2

It is possible that the one-dimensional R-calculation which stretches the facility over the whole height of the reactor exaggerates the general flux hollowing towards the centre, and therefore the local increase in the axis.

Flux spectrum at the centre of the NIGHTMARE facility *

Upper energy limit eV	without boron	boron 10 mm	boron 15 mm
1.0500E+07	6.3892E-04	1.3897E-03	1.5728E-03
6.5000E+06	3.0232E-03	6.6498E-03	7.5666E-03
4.0000E+06	6.1287E-03	1.3589E-02	1.5534E-02
2.5000E+06	6.4654E-03	1.4586E-02	1.6814E-02
1.4000E+06	4.5326E-03	1.0361E-02	1.2021E-02
8.0000E+05	4.5008E-03	9.5149E-03	1.0657E-02
4.0000E+05	5.7362E-03	1.1383E-02	1.2414E-02
2.0000E+05	7.1591E-03	1.3458E-02	1.4384E-02
1.0000E+05	8.5459E-03	1.4857E-02	1.5405E-02
4.6500E+04	9.4411E-03	1.4639E-02	1.4498E-02
2.1500E+04	9.4307E-03	1.2346E-02	1.1459E-02
1.0000E+04	9.6557E-03	9.8237E-03	8.3323E-03
4.6500E+03	9.6367E-03	7.3017E-03	5.4965E-03
2.1500E+03	9.0519E-03	4.4955E-03	2.8460E-03
1.0000E+03	8.5563E-03	2.3810E-03	1.1760E-03
4.6500E+02	8.0369E-03	9.9602E-04	3.3401E-04
2.1500E+02	6.5824E-03	2.6608E-04	4.6885E-05
1.0000E+02	5.6833E-03	4.1079E-05	2.0025E-06
4.6500E+01	4.7414E-03	2.0564E-06	1.8944E-08
2.1500E+01	3.2043E-03	1.3280E-08	2.0913E-10
1.0000E+01	2.9423E-03	1.3967E-10	2.3589E-12
4.6500E-01	1.4716E-03	1.5619E-12	2.5990E-14
2.1500E-01	1.6322E-03	2.6580E-14	4.3410E-16
1.3000E-01	1.7748E-03	1.9456E-15	3.1236E-17
1.1500E-01	1.5793E-03	2.4816E-16	3.8245E-18
1.0800E-01	1.6006E-03	6.0804E-17	8.6972E-19
1.0300E-01	1.3455E-03	2.3558E-17	3.1024E-19
9.9000E-02	1.5959E-03	1.2034E-17	1.4690E-19
9.4800E-02	1.1750E-02	5.7887E-18	6.6916E-20
5.0000E-02	7.2449E-04	2.5099E-20	2.8746E-22
4.1400E-02	4.2602E-04	3.5039E-21	3.9689E-23
3.4000E-02	1.8986E-04	4.9978E-22	5.5667E-24
2.7800E-02	5.2147E-05	9.7667E-23	1.0737E-24
2.2700E-02	2.3135E-06	1.4144E-23	1.5362E-25
1.5200E-02	1.6829E-07	1.1126E-24	1.1956E-26
9.2400E-03	1.6566E-08	8.2562E-26	8.7896E-28
5.6000E-03	2.1405E-09	7.0011E-27	7.3947E-29
3.4000E-03	3.9199E-10	7.3404E-28	7.6919E-30
2.0600E-03	1.0770E-10	9.7826E-29	1.0157E-30
1.2500E-03	7.5045E-12	3.5095E-30	3.6318E-32
1.0000E-04	0.0	0.0	0.0

TABLE MOL C-1

* normalisation : $\int_0^{\infty} \varphi(u) du = 0.1$

FIGURE MOL C-1. Thimble Lattice.

NBSR NIGHTMARE SPECTRUM AT THE CENTRE OF THE FACILITY

FIGURE MOL C-2

Reactor NBSR - NIGHTMARE facility in the GLORY HOLE
 $^{235}\text{U}(n,f)$ and $^{238}\text{U}(n,f)$ response functions at the
centre of the facility.

FIGURE MOL C-3

Reactor NBSR - NIGHTMARE facility in the GLORY HOLE
 $^{238}\text{U}(n,f)$ response functions at the centre of the facility.
Case without boron, and case with a 15 mm thick boron screen.

FIGURE MOL C-6

NIGHTMARE WITH 3 BORON THICKNESSES, (0.0, 10.0, 15.0 MM)

NIGHTMARE WITH 3 BORON THICKNESSES. (0.0, 10.0, 15.0 MM)

EG&G IDAHO, INCORPORATED

A. REACTION RATE MEASUREMENTS IN THE ^{252}Cf NEUTRON IRRADIATION FACILITY AT NBS

R. C. Greenwood and R. G. Helmer

Objective

Obtain absolute values for infinitely dilute reaction rates for selected fast-neutron dosimetry materials irradiated in the NBS ^{252}Cf neutron irradiation facility with sufficient accuracy to permit testing and evaluation of energy-dependent cross-section sets from ENDF/B.

Summary

Infinitely dilute reaction rates have been measured in the NBS ^{252}Cf neutron irradiation facility for the $^{52}\text{Ni}(n,p)^{58}\text{Co}$ reaction and for the production of ^{95}Zr , ^{103}Ru , ^{132}Te and ^{140}Ba - ^{140}La following fission in ^{235}U using γ -ray spectrometry. For each reaction the activity levels were sufficiently low that the accuracy of the results were in all cases limited to some extent by counting statistics; in the worst case to 10% for the ^{132}Te production. Combining these reaction rates with the NBS certified free field neutron fluences and free field fission rate for ^{235}U we obtain an integral cross section value for the $^{58}\text{Ni}(n,p)^{58}\text{Co}$ reaction and cumulative fission yield values for ^{95}Zr , ^{103}Ru , ^{132}Te and ^{140}Ba .

Accomplishments and Status

Two packages of three foils, each package consisted of a ^{235}U (material type 264CM) foil sandwiched between two Ni foils, were irradiated for 10.69 d (923760s) with the irradiation terminating 0930 EDT on July 5, 1977 in the indoor ^{252}Cf fission neutron irradiation facility at NBS. The two foil packages were at equal distances on opposite sides of the ^{252}Cf source in the 1 and 4 irradiation positions with the separation distance between pairs of foils being: near Ni foils 10.51 ± 0.01 cm; ^{235}U foils 10.70 ± 0.01 cm; and, far Ni foils 10.90 ± 0.01 cm. The physical parameters of the foils are listed in Table EGG-1.

The foils were shipped to the INEL and γ -ray counting began on July 28, 1977 using the ND-5 and OR-5 Ge(Li) detector systems. Two γ -ray spectra, one on each detector, were obtained for each of the Ni foils. A summary of the counting parameters for the Ni and ^{235}U foils is given in Table EGG-2, together with the ranges of various correction factors which had to be applied to the data.

The reaction rates computed from these data are given in Table EGG-3. While the accuracies achieved in these analyses were in all cases limited somewhat by counting statistics, satisfactory results were obtained for all but the $^{235}\text{U}(n,f)^{132}\text{Te}$ reaction. In this latter case the delay before counting resulted in weak 228-keV peaks with decay corrections ranging from 145-345 being required. The consistency of the data set is indicated by the values of the reduced- χ^2 , denoted by ϵ^2 , which are in general close to 1.0.

The reaction rates in Table EGG-3 were actually computed assuming a constant neutron flux throughout the irradiation, whereas in fact the flux level, due to ^{252}Cf decay, decreased 0.77% during this time period. This "computed" reaction rate might reasonably be assumed then to represent the "true irradiation averaged" reaction rate. Separate calculations have demonstrated that for activities with half-lives $>12.7\text{d}$ the "computed" and the "true irradiation averaged" reaction rates differ by $<0.04\%$.

In the actual experimental setup the distances between the corresponding foils in the 1 and 4 irradiation positions were measured quite precisely, to $\pm 0.01\text{ cm}$, with the ^{252}Cf source nominally at the midpoint between the foil packages. It is because of the uncertainty in the ^{252}Cf source location that reaction rates for a specific source-detector distance are most accurately obtained by pairwise irradiation of identical foil packages on opposite sides of the nominally centered ^{252}Cf source: with these data being used to deduce the relative source location, and with the reaction rates from corresponding foils in the two sets being combined remembering that the flux varies as $1/R^2$. In the present experiments the ratios of the reaction rates from the 1-4 foil pairs, shown in Table EGG-4, indicate that to within the precision of the results the ^{252}Cf source is exactly centered between the 1-4 foil packages. Hence, the combined reaction rates, shown in Table EGG-5, were merely obtained by taking $1/\sigma^2$ weighted averages of all of the individual measurements for each specific foil and reaction.

These reaction rates have been corrected to those of infinitely dilute samples located in a free ^{252}Cf neutron field by applying the neutron scattering corrections computed at NBS. These corrections are shown in column 2 of Tables EGG-6 and EGG-7 and the resulting infinitely-dilute free-field reaction rates are shown in column 3 of these tables. Combining these corrected reaction rates with the free field fluences and the free field ^{235}U fission rates certified by NBS we obtain the $^{58}\text{Ni}(n,p)^{58}\text{Co}$ reaction cross section and the ^{235}U fission yields shown in Tables EGG-6 and EGG-7, respectively, for a free field ^{252}Cf fission neutron flux distribution. The excellent agreement between the integral cross section values determined for the inside and the outside Ni foils provides some confidence in the reasonableness of the uncertainty assignments.

Expected Achievements

This work is complete.

TABLE EGG-1
PARAMETERS OF FOILS IN NBS ^{252}Cf IRRADIATION PACKAGE

Set No.	Material- Identification ^a	Foil Mass g	Atomic Weight ^b	Isotopic Abundance ^b (weight fraction)	Elemental Purity ^b (fraction)	Foil Thickness ^b (mil)
1	Ni-JA	0.29524	57.935	0.6743	0.9962	10
	^{235}U -1	1.2253	235.044	0.99886	0.9944	20
	Ni-LK	0.293109				
4	Ni-JE	0.2960				
	^{235}U -4	1.2110				
	Ni-LN	0.2939				

^aFor each foil set the foils are listed in order of increasing distance from the ^{252}Cf source

^bFoils of the same element have identical composition and foil dimensions.

TABLE EGG-2
 γ -RAY COUNTING PARAMETERS FOR NBS ^{252}Cf IRRADIATION PACKAGE

<u>Foil Number</u>	<u>^{235}U-1</u>	<u>^{235}U-4</u>	<u>Ni</u>
Number of Spectra	4	5	2
Decay Times (days)	23-106	23-112	28-35
Source-Detector Distances (cm)	10	10	10
External Absorbers	none	none	none
Range of Correction Factors			
random summing	.994-.997	.994-.997	1.000
real summing	.991-1.000	.991-1.000	1.000
γ -self absorption	.633-.976	.633-.976	.992

TABLE EGG-3
REACTION RATES FROM FOILS IRRADIATED IN NBS ^{252}Cf IRRADIATION FACILITY

Foil	Isotope	Decay Range (Days)	No of Spectra	E_γ (keV)	ϵ^2	Reaction Rate (10^{-19} atoms/atom·sec)		Uncertainty (%)		
						Measured	Average	σ_1^a	σ_2^a	σ_3^a
Ni-JA	^{58}Co	28-30	2	810	0.8	18.48		0.65	0.82	1.71
Ni-JE	^{58}Co	29-31	2	810	3.1	18.40		1.15	1.25	1.95
Ni-LK	^{58}Co	30-35	2	810	2.0	17.13		0.97	1.09	1.85
Ni-LN	^{58}Co	31-34	2	810	0.0	16.82		0.68	0.84	1.72
$^{235}\text{U}-1$	^{95}Zr	36-106	3	724	0.2	11.88	11.87	0.86	0.78 ^b	1.93 ^b
		23-106	4	756		11.86		1.07		
	^{103}Ru	23-106	4	497	0.5	6.258		0.63	0.80	2.08
	^{132}Te	23	1	228		9.76		9.5	9.5	11.1
$^{235}\text{U}-4$	$^{140}\text{Ba-La}$	23-36	2	1596		11.14		0.78	0.93	1.76
	^{95}Zr	37-112	3	724	4.8	12.07	11.97	2.15	0.95 ^b	2.00 ^b
		23-112	5	756	1.5	11.93		0.75		
	^{103}Ru	23-112	5	497	2.0	6.266		0.76	0.91	2.13
	^{132}Te	23-27	2	228	0.4	12.7		5.0	5.0	7.7
	$^{140}\text{Ba-La}$	23-97	4	1596	2.1	10.94		0.94	1.06	1.84

^a σ_1 is relative error excluding terms from foil mass and any external absorber; σ_2 is relative error with foil mass error, taken to be 0.5%, included; and, σ_3 is the total uncertainty.

^b These are the error values associated with the average reaction rate.

TABLE EGG-4

IRRADIATION SYMMETRY FOR FOIL PACKAGES 1 AND 4 IN THE NBS ^{252}Cf FACILITY

Foils in Ratio	Isotope	Reaction Rate Ratio	Uncertainty on Ratio (%)
Ni-JA/Ni-JE	^{58}Co	1.004	1.32
Ni-LK/Ni-LN	^{58}Co	1.018	1.18
^{235}U -1/ ^{235}U -4	$^{95}\text{Zr}(754)$	0.984	2.32
	$^{95}\text{Zr}(756)$	0.994	1.31
	^{103}Ru	0.999	0.99
	^{140}Ba -La	1.019	1.22
	Average ($1/\sigma^2$)	1.005	0.56

TABLE EGG-5

COMBINED REACTION RATES FOR FOIL PACKAGES 1 AND 4 IRRADIATED IN THE NBS ^{252}Cf FACILITY

Foil	Position	Isotope	ϵ^2	Combined Reaction Rate (10^{-19} atoms/atom·sec)	Uncertainty(%)		
					σ_1^a	σ_2^a	σ_3^a
Ni	inside	^{58}Co	1.4	18.44	0.54	0.74	1.67
Ni	outside	^{58}Co	1.9	16.97	0.66	0.83	1.71
^{235}U	middle	^{95}Zr	1.8	11.92	0.50	0.71	1.90
		^{103}Ru	1.2	6.263	0.45	0.67	2.03
		^{132}Te	3.7	11.8	8.5	8.5	10.3
		$^{140}\text{Ba-La}$	1.9	11.02	0.69	0.85	1.65

^a See equivalent footnote to Table EGG-3

EGG-7

TABLE EGG-6

$^{58}\text{Ni}(n,p)^{58}\text{Co}$ REACTION RATE CORRECTED FOR NEUTRON SCATTERING AND DEDUCED INTEGRAL
MICROSCOPIC CROSS SECTION FOR THE NBS ^{252}Cf FACILITY

Ni Foil Position	Perturbed RR Free Field RR	Corrected Reaction Rate ^a (10^{-19} atoms/atom·sec)	Average Free Field Neutron Fluence ^{a,b} (neutrons/cm ² ·sec)	Integral Cross Sections (mb)	
				Individual ^a	Average ^a
inside	1.018	18.11(1.7)	$1.518 \times 10^7(1.6)$	119.3(2.3)	119.4(2.3)
outside	1.002	16.89(1.7)	$1.412 \times 10^7(1.6)$	119.6(2.3)	

^a Assigned total error (1σ) is given in parentheses as a percentage of the value.

^b Determined by taking the free-field neutron fluence values certified by NBS and dividing by 923760 sec.

TABLE EGG-7

^{235}U FISSION REACTION RATES CORRECTED FOR NEUTRON SCATTERING AND DEDUCED
FISSION YIELDS FOR THE NBS ^{252}Cf FACILITY

Fission Product	Perturbed RR Free Field RR	Corrected ^{a,b} Reaction Rate (10^{-19} atoms/atom·sec)	Average Free-Field Specific Fission Rate ^{a,c,d} (10^{-19} fissions/atom·sec)	Cumulative ^{235}U Fission Yield ^{a,e} (%)
^{95}Zr	1.056	11.28(1.9)	175.5(1.6)	6.43(2.5)
^{103}Ru		5.929(2.0)		3.38(2.6)
^{132}Te		11.2(10.3)		6.4(10.4)
$^{140}\text{Ba-La}$		10.47(1.7)		5.97(2.3)

^a Assigned total error (1σ) is given in parentheses as a percentage of the value.

^b No allowance in quoted uncertainties is made for the uncertainty in the neutron scattering correction.

^c Obtained by multiplying "end of irradiation" fission rates certified by NBS by a correction factor of 1.00385.

^d Excluding neutron scattering correction uncertainty this total uncertainty is 1.4%.

^e Many of the individual neutron scattering correction components are common to the reaction rate and the specific fission rate measurements and this has been taken into account in these uncertainty estimates.

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EGG-10

B. MEASUREMENT OF RELATIVE REACTION RATES IN THE ^{235}U CAVITY FISSION NEUTRON FACILITY AT NBS

R. C. Greenwood and R. G. Helmer

Objective

Obtain relative reaction rates for selected fast-neutron dosimetry materials, irradiated in the NBS ^{235}U cavity fission facility, with sufficient accuracy to permit testing and evaluation of energy-dependent cross-section sets from ENDF/B.

Summary

Relative reaction rates from a sample of SRM-1158 high-nickel steel have been measured in the NBS ^{235}U cavity fission facility for the $^{54}\text{Fe}(\text{n},\text{p})^{54}\text{Mn}$, $^{58}\text{Ni}(\text{n},\text{p})^{58}\text{Co}$ and $^{59}\text{Fe}(\text{n},\gamma)^{60}\text{Fe}$ reactions using γ -ray spectrometry. The accuracy of the result for the latter reaction was severely limited by counting statistics. From these data, a value of 1.346 ± 0.007 was obtained for the $^{58}\text{Ni}(\text{n},\text{p})/^{54}\text{Fe}(\text{n},\text{p})$ reaction rate.

Accomplishments and Status

The NBS SRM-1158 foil was irradiated for 86220 sec. with the irradiation terminating 1102 EDT on June 28, 1977 in the ^{235}U cavity fission neutron facility at NBS. The physical parameters of this foil are listed in Table EGG-8.

The foils were shipped to the INEL and γ -ray counting began on Aug. 12, 1977 using both the ND-5 and OR-5 Ge(Li) detector systems. A total of four γ -ray spectra, two on each detector, were obtained for the SRM-1158 foil. A summary of the counting parameters for the foil is given in Table EGG-9, together with the ranges of various correction factors which had to be applied to the data.

The reaction rates computed from these data are given in Table EGG-10. Only in the case of the $^{58}\text{Fe}(\text{n},\gamma)^{59}\text{Fe}$ reaction is the accuracy achieved limited by counting statistics. The reduced- χ^2 values, ϵ^2 , indicate that the data are consistent.

No correction factors have been applied to reaction rates shown in Table EGG-10 to account for neutron scattering and flux gradients in the sample. For comparison to ENDF/B cross section sets, however, it is convenient with such homogeneous samples to compare cross section ratios, in

which for similar types of reactions these correction terms are essentially identical. Thus, for the two threshold reactions we compute the following ratio of reaction rates, or cross sections, in the NBS ^{235}U cavity fission neutron source facility:

$$^{58}\text{Ni}(n,p)^{58}\text{Co}/^{54}\text{Fe}(n,p)^{54}\text{Mn} = 1.346 \pm 0.007 \quad (0.51\%)$$

This value is identical to that reported earlier by Fleming and Spiegel¹.

Expected Achievements

This work is complete.

References

1. R. Fleming and V. Spiegel, Proc. Second ASTM-EURATOM Sym. on Reactor Dosimetry: Dosimetry Methods for Fuels, Cladding and Structural Materials, NUREG/CP-0004 (1977) Vol. 2, p. 953.

TABLE EGG-8

PARAMETERS OF NBS SRM-1158 HIGH NICKEL-STEEL FOIL IRRADIATED
IN THE NBS ^{235}U CAVITY FISSION NEUTRON FACILITY

Foil Mass g	Thickness (mil)	Elemental Constituents	Elemental Purity (%)	Isotopes Used	Atomic Weight	Isotopic Abundance
0.751689	30	Fe	63.15	^{54}Fe	53.9396	0.322
				^{59}Fe	57.9333	56.05
		Ni	36.03	^{58}Ni	57.9350	67.43

TABLE EGG-9

γ -RAY COUNTING PARAMETERS FOR SRM-1158 FOIL IRRADIATED IN
NBS ^{235}U CAVITY FISSION NEUTRON FACILITY

<u>Foil Type</u>	<u>SRM-1158</u>
Number of Spectra	4
Decay Times (days)	45-66
Source-Detector Distance (cm)	10
External Absorbers	None
Range of Correction Factors	
random summing	1.000
real summing	0.9994-1.0002
γ -ray self absorption	0.9792-0.9822

TABLE EGG-10
REACTION RATES FROM SRM-1158 FOIL IRRADIATED IN NBS ^{235}U CAVITY FISSION NEUTRON FACILITY

Reaction	Decay Range (days)	No. of Spectra	E_γ (keV)	ϵ^2	Reaction Rate (10^{-16} atoms/atom sec)		Uncertainty (%)		
					Measured	Average	σ_1^a	σ_2^a	σ_3^a
$^{58}\text{Ni}(n,p)^{58}\text{Co}$	45-66	4	810	1.5	19.03		0.31	0.59	1.62
$^{54}\text{Fe}(n,p)^{54}\text{Mn}$	45-66	4	834	2.0	14.13		0.41	0.65	1.63
$^{58}\text{Fe}(n,\gamma)^{59}\text{Fe}$	45-66	4	1099	0.1	0.622	0.662	4.8	4.3 ^b	5.1 ^b
			1291	1.4	0.736		5.5		

^a σ_1 is relative error excluding terms from foil mass and any external absorber; σ_2 is relative error with foil mass error, taken to be 0.5%, included; and, σ_3 is the total uncertainty.

^bThese are the error values associated with the average reaction rate.

EG&G ORTEC

EG&G ORTEC
W. H. Zimmer

OBJECTIVE

EG&G Ortec's objective is to perform radiometric measurements of LWR and FBR dosimetry foils as required for the ILRR program. EG&G Ortec's purpose in performing the analyses is to maintain credibility as a measurements laboratory and to provide trial sources against which EG&G Ortec's analytical software can be tested.

SUMMARY

The counting equipment that is routinely available for ILRR radiometric measurements are a twenty percent Ge(Li) coaxial detector, a Low Energy Photon Spectrometer (LEPS) detector, four multi-channel analyzer ports in a Gamma Lab 44 System supported by EG&G Ortec 6516-6528FX analytical software. Additional detectors are available as needed. Measurements associated with the LWR-PCA pressure vessel mock-up test have been completed and measurements associated with the FTR-Reactor Characterization Program have been planned.

ACCOMPLISHMENTS AND STATUS

Measurements associated with the LWR-PCA pressure vessel mock-up test have been completed and measurements associated with the FTR-Reactor Characterization Program have been planned. (See HEDL chapter of this report and HEDL SA-1949).

EXPECTED ACHIEVEMENTS

Documentation of LWR measurements will be provided in NRC Progress Reports and documentation of FBR measurements in ILRR Progress Reports.

EXXON NUCLEAR IDAHO COMPANY, INC.

FAST REACTOR FISSION YIELD MEASUREMENTS IN FFTF

W. J. Maeck, R. L. Tromp

For many years, personnel at the Idaho Chemical Processing Plant (now operated by Exxon Nuclear Idaho Company, Inc.) at the Idaho National Engineering Laboratory, have been conducting a DOE-funded FBR Burnup Program for the purpose of measuring accurate absolute fission yields for use in the determination of burnup in fast reactor fuels. As a continuing effort of this program, an irradiation of heavy element nuclides (^{233}U , ^{235}U , ^{239}Pu , and ^{241}Pu) is planned for the eight-day full-power run associated with the FFTF Reactor Characterization Experiment.

The primary purpose is to generate a group of heavy element reference standards for which the number of fissions and burnup are well known. By combining this experiment with others of a similar nature, these samples will serve as reference and comparison standards. A secondary purpose is to verify the fast reactor fission yields measured in EBR-II and to assess the validity of using EBR-II values for irradiations conducted in FFTF.

For this experiment ten samples each of highly enriched ^{233}U , ^{235}U , ^{239}Pu , and ^{241}Pu (as oxides) sealed in high purity nickel capsules will be irradiated. Eight capsules (two each of ^{233}U , ^{235}U , ^{239}Pu , and ^{241}Pu) will be placed axially in one of the removable pins in each of the following assemblies: 2101, 2201, 2402, 2503, and 2603, which are located in Rows 1, 2, 4, 5, and 6 respectively. Each pin will be located immediately adjacent to an ILRR dosimetry package pin. The amount of material in each sample capsule has been adjusted such that each individual sample will give about 10^{19} fissions. A detailed listing of the individual samples and reactor locations is given in Table EXXON-1. Each column of samples will be centered about the reactor mid-plane.

TABLE EXXON-1
INVENTORY OF FFTF FISSION YIELD CAPSULES

<u>Row</u>	<u>Assembly ID</u>	<u>Pin ID</u>	<u>Top-to-bottom Capsule ID</u>	<u>Actinide</u>	<u>Grams Oxide</u>
1	2101	CW66-527	1-3-T	²³³ U	0.342
			1-5-T	²³⁵ U	0.455
			1-1-T	²⁴¹ Pu	0.375
			1-9-T	²³⁹ Pu	0.495
			1-9-B	²³⁹ Pu	0.498
			1-1-B	²⁴¹ Pu	0.380
			1-5-B	²³⁵ U	0.452
			1-3-B	²³³ U	0.345
2	2201	CW66-528	2-3-T	²³³ U	0.035
			2-5-T	²³⁵ U	0.054
			2-1-T	²⁴¹ Pu	0.044
			2-9-T	²³⁹ Pu	0.052
			2-9-B	²³⁹ Pu	0.051
			2-1-B	²⁴¹ Pu	0.047
			2-5-B	²³⁵ U	0.056
			2-3-B	²³³ U	0.040
4	2402	CW66-532	4-3-T	²³³ U	0.048
			4-5-T	²³⁵ U	0.054
			4-1-T	²⁴¹ Pu	0.049
			4-9-T	²³⁹ Pu	0.063
			4-9-B	²³⁹ Pu	0.061
			4-1-B	²⁴¹ Pu	0.047
			4-5-B	²³⁵ U	0.056
			4-3-B	²³³ U	0.037
5	2503	CW66-586	5-3-T	²³³ U	0.062
			5-5-T	²³⁵ U	0.077
			5-1-T	²⁴¹ Pu	0.067
			5-9-T	²³⁹ Pu	0.079
			5-9-B	²³⁹ Pu	0.077
			5-1-B	²⁴¹ Pu	0.061
			5-5-B	²³⁵ U	0.078
			5-3-B	²³³ U	0.055
6	2603	CW66-603	6-3-T	²³³ U	0.833
			6-5-T	²³⁵ U	0.078
			6-1-T	²⁴¹ Pu	0.823
			6-9-T	²³⁹ Pu	0.895
			6-9-B	²³⁹ Pu	0.891
			6-1-B	²⁴¹ Pu	0.820
			6-5-B	²³⁵ U	0.738
			6-3-B	²³⁵ U	0.822

After completion of the HP irradiation, the samples will be returned to Idaho for analysis. The samples in Rows 1 and 6 will be dissolved and analyzed for the following fission product elements using isotope dilution mass spectrometry: Kr, Rb, Sr, Zr, Mo, Ru, Xe, Cs, Ba, La, Ce, Nd, and Sm. The absolute number of fissions will be established by summing the number of fission product atoms in the heavy mass peak. The samples in Rows 2, 4, and 5 will only be partially analyzed, principally for Nd, Cs, Kr, and Xe to establish relative fission yield values for the intermediate reactor positions. Extensive correlation of the isotopic composition and fission yields of several of the fission product elements as a function of neutron energy will be made.

HEDL

HANFORD ENGINEERING DEVELOPMENT LABORATORY

A. FTR MEASUREMENT PLANS FOR INTERLABORATORY CALIBRATIONS AND FISSION PRODUCT YIELDS

Raymond Gold
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Objective

To formulate and plan ILRR irradiations, calibrations, and benchmark field experiments, including fission product yield measurements, in the Fast Test Reactor-Reactor Characterization Program (FTR-RCP). To subsequently carry out these measurements in the FTR-RCP, attaining goal accuracies of 1 to 3% (1σ) for point-wise in-core fuel region experiments and 2 to 5% (1σ) for in-and-out-of-core material and absorber experiments. To present details of the FTR-RCP related to ILRR plans.

Summary

A detailed description of several parts of the FFTF Characterization Program is given in this report to add perspective to planned ILRR experiments in FFTF. These ILRR measurements by ILRR participating laboratories are planned in coordination with the FTR-RCP to provide calibrations to attain the program accuracy goals.

During FFTF acceptance tests, the absolute number of 1) fissions and fission rates and 2) reactions and reaction rates will be determined by active fission chamber (FC), passive solid state track recorder (SSTR), radiometric monitor measurement (RM), and helium accumulation fluence monitor (HAFM) mapping techniques. High accuracy NBS-FCs and HEDL-SSTRs (using mica and quartz recorders) will be used to make measurements at selected locations during the very low power (VLP) and low power (LP) tests.

HEDL-SSTRs (using quartz recorders) and Allied Chemical, Idaho National Engineering Laboratory (INEL), mass spectrometric techniques will be the only available methods for making direct measurements of the absolute number of fissions at selected locations during the high power (HP) test for the direct establishment of fission yield and counting system calibration factor values.

Manual and/or automated optical track counting will be required for most of the VLP and LP SSTRs. Because of high track densities for some of the LP and HP SSTRs, the track recorders will require manual and/or automated counting with a scanning electron microscope (SEM). Optical and SEM results will have limiting accuracies in the range of 1 to 2% (1σ) and 1 to 5% (1σ) respectively, depending on the irradiation conditions and track recorder deposit mass assay. NBS fission chamber and Allied Chemical isotope dilution mass spectrometric* measurements will have a limiting accuracy of 1 to 2% (1σ).

The NBS, HEDL, and Allied Chemical absolute fission rates and total number of fissions will be combined with Interlaboratory Reaction Rate (ILRR) Program type radiometric (^{140}Ba , ^{95}Zr , ^{103}Ru , ^{131}I , ^{132}Te , and ^{137}Cs) and HEDL and Allied Chemical mass spectrometric (^{143}Nd , ^{145}Nd , ^{146}Nd , ^{148}Nd , and ^{150}Nd) fission product results to measure, validate, and calibrate (at VLP, LP, and HP) the energy dependence of the above identified key fission products. These combined fission yield results will be used to generate FTR in- and out-of-core positional fission yield (r, θ, ϕ) energy dependent mappings which will serve as input to reactor physics and fuel behavior codes as well as for fuel pin burn-up and wire-wrap dosimetry measurement techniques. These different techniques will be used to account for variations in point-to-pin, pin-to-assembly, and assembly-to-core and blanket (reflector for FTR) absolute fission rate and burn-up results. The goal accuracy for the FTR Reactor Characterization Program (RCP) point-wise measurements

*In selected high power encapsulated pellet sized specimens.

is in the 1 to 3% (1σ) range for in-core fuel region experiments and in the 2 to 5% (1σ) range for in- and out-of-core region material and absorber experiments.

ILRR participation consists of radiometric counting on selected sets of VLP, LP and/or HP radiometric monitors to establish in-situ high accuracy (1 to 5%, 1σ) calibrations for the HEDL manual and automated radiometric measurements. Selected individual monitor calibrations using NBS, EG&G, and HEDL "Benchmark Neutron Fields" will also be accomplished by participating laboratories. The Interlaboratory-RM and single-laboratory HEDL-RM measurements will have a limiting accuracy of 1 to 3% (1σ) and 2 to 5% (1σ), respectively. HAFM measurements will be made by RIES** on dosimeters from the same sets selected for the radiometric intercomparisons. Analyses of these dosimeters will be carried out at RIES with a limiting accuracy of 1 to 3% (1σ).

**Rockwell International Energy Systems Group, formerly Atomics International.

ACCOMPLISHMENTS AND STATUS

1.0 INTRODUCTION

Fission product yield data are required as a basis for dosimetry and the routine determination of burn-up (total fissions) through quantitative analysis of the concentrations of fission products in irradiated fission monitors and fuel. An important objective of this experiment is to obtain an accurate measurement of the yields of important fission products, which serve as dosimetry and burn-up monitors for fissionable isotopes, under irradiation conditions spanning the expected range of neutron energy spectra within the FTR core and reflector regions.

While a variety of destructive and nondestructive methods can be used to determine the burn-up of highly irradiated dosimetry monitors and fuel, the most often used techniques rely on an accurate knowledge of the yields of individual fission products.⁽¹⁻¹¹⁾ Since these yields are neutron energy-dependent in general, and a correlation of this dependence adequate to meet long-range accuracy goals of 1 to 3% on burn-up is not yet available, direct measurement of the yields of important fission products in the FTR is necessary. The yield measurements must cover a sufficient range of spectral variation so that accurate burn-up values can be obtained for irradiated driver and experimental fuel based on calculated and/or measured local spectral indices.^(2,5,7) Such measurements are needed for each of the important fissionable isotopes, so that burn-up measurements can be accurately interpreted for FTR fuel and other experiments having a variety of isotopic compositions. Since the fission product absolute yield measurements depend in part on absolute fission rate measurements, it is necessary that the yield and fission measurements both be made during the Reactor Characterization Program (RCP) before significant changes in core configuration are made which would perturb the measured absolute fission distributions.

Fission yield measurements require the determination of total fissions and the amounts of various fission products present in irradiated samples of

fissionable materials. Current FTR-RCP plans for fission product yield measurements call for irradiations at Very Low Power (VLP), Low Power (LP), and High Power (HP).⁽¹²⁾ A separate description is presented in sequel for each experiment related to the fission product yield parameter that is scheduled in the FTR-RCP. The relationship among various aspects of these experiments and irradiations is reviewed with particular stress upon attaining FTR-RCP 2 to 5% (1σ) short-term startup point-wise goal accuracies.

The recommended placement of passive monitor spectral sets and selected non-fission monitors, which is important for the proper interpretation and correlation of the results of these FTR-RCP fission yield experiments with previous EBR-II and ILRR neutron field results, is considered in Appendix A. Appendix A also provides summary information on the expected results and benefits of FTR-RCP supportive ILRR program measurements.

The interrelationship of the FTR-RCP and other EBR-II and ILRR Program yield experiments to ongoing EBR-II/FTR and other fast test reactor fuels and materials experiments is summarized and briefly discussed in Appendix B. The emphasis here is on the attainment of long range point-to-pin, pin-to-assembly and assembly-to-core and blanket (reflector for FTR) absolute reaction rates and burnup results with goal accuracies in the 2 to 5% (1σ) or better range.

Appendix C presents information on the preparation of ILRR fluence standards that will certify the correctness of current calibrations of radiometric counting and data analysis systems used by participating laboratories.

2.0 VERY LOW POWER EXPERIMENTS

Absolute fission rate measurements with SSTR are planned for the VLP, LP, and HP irradiations of the FTR-RCP. Plans for SSTR fission rate measurements in the In-Reactor Thimble (IRT) at VLP include the following experiments:

- 1) Fission Product Yield Irradiation (IRT Experiment 8)
- 2) SSTR and Dosimeter Calibration Irradiation (IRT Experiment 10)
- 3) Passive Dosimeter Irradiation (IRT Experiment 12)

In any of these experiments, the fluence experienced by a passive monitor is, of course, governed by the power-time history of the reactor irradiation. For a simple in-reactor exposure as depicted in Figure HEDL A-1, the time-integrated power P is given by

$$P = p_0 (1 - e^{-t_1/\tau}) + p_0(t_2 - t_1), \quad (2.1)$$

where τ is the period used to attain a reactor power level p_0 , t_1 is the time required to attain p_0 from startup, and $(t_2 - t_1)$ is the ensuing time interval of the irradiation at the constant power level p_0 . Equation (2.1) tacitly implies that the reactor is shut down promptly, i.e. scrammed, at $t = t_2$.

As a consequence of Eq. (2.1), the total number of fissions F deduced with an integrative passive SSTR during such an irradiation is given by

$$F = f (1 - e^{-t_1/\tau}) + f(t_2 - t_1) \quad (2.2)$$

Here, f is the absolute fission rate obtained from an active detector (such as the NBS fission chamber) corresponding to the constant reactor power level p_0 . For $t_1 \gg \tau$, Eqs. (2.1) and (2.2) take the simple form

$$P = p_0 (\tau + t_2 - t_1) \quad (2.3a)$$

and

$$F = f (\tau + t_2 - t_1) \quad (2.3b)$$

respectively.

The simple power-time history depicted in Figure HEDL A-1 together with condition $t_1 > \tau$ shall be designated as a Single Period Irradiation (SPI). Current VLP estimates have been generated under the assumption that SPI conditions hold and Eq. (2.3) is therefore valid. Although it is recognized that the SPI assumption actually represents rather idealized conditions, it is recommended that the VLP irradiation adhere as closely as possible to a SPI power-time history. This should minimize, in principle, any error propagation from uncertainties in power-time history. Clearly, the actual reactor power-time history of the VLP irradiation must be recorded both accurately and continuously at the FTR operating console so that deviations from SPI conditions can be taken into account (simultaneous recording with the FTR-Data Acquisition System [DAS] is also recommended). Actually, the choice of SPI parameters will be defined on the basis of observed NBS chamber fission rates conducted in the In-Reactor Thimble (IRT) Experiment 5, which precedes all of these experiments. Each of these SSTR experiments is discussed separately below.

Comparison between SSTR and NBS fission chamber measurements will be carried out under realistic FBR spectral conditions in the IRT. The NBS chamber will be run in IRT Experiment 5, whereas SSTR will be exposed in IRT Experiments 8 and 10. For each irradiation, a modified NBS double fission chamber using ^{239}Pu deposits will be operated at 38 cm below midplane. This particular absolute NBS chamber will be referred to as the monitor fission chamber, since it will be used to intercompare different IRT experiments.

The monitor fission chamber is scheduled for use only in IRT Experiments 2, 3, 5, 8, and 10. Experiments 2 and 3 are neutron spectrometry observations with proportional counters and emulsions, respectively. For intercomparison

FIGURE HEDL A-1. Characteristics of a Single Period Irradiation (SPI).

with other IRT experiments, the LLFM* can be used. In addition to LLFM normalization, power normalization can also be based on SSTR power monitors included in VLP passive IRT experiments. Recommended SSTR power monitor locations for passive IRT experiments are summarized in Table HEDL A-1.

2.1 Absolute SSTR Fission Rates (IRT Experiments 8 and 10)

In IRT Experiments 8 and 10, SSTR will be irradiated at midplane and at +54 cm (upper axial reflector), with a monitor chamber at -38 cm. The SSTR will be contained in dummy fission chambers and thus the geometry will be as close as possible to that of the absolute NBS modified double fission chamber measurements made in IRT Experiment 5. Operating (SPI) conditions for Experiments 8 and 10 are summarized in Tables HEDL A-2.2 and HEDL A-2.3. The dummy NBS fission chambers will be loaded with SSTR in advance of each exposure to facilitate preparation of the IRT Experimental Insert (EI).

Fission product yield measurements will be conducted in IRT Experiment 8 using SSTR in close proximity to ILRR type radiometric dosimeters.^(2, 3, 6) Exposures will be carried out in dummy NBS fission chambers at midplane and in the upper axial reflector (~54 cm above midplane). The following isotopes will be incorporated: ^{239}Pu , ^{235}U , ^{238}U , ^{237}Np . Absolute yields for ^{140}Ba , ^{103}Ru , ^{95}Zr and ^{137}Cs will be determined.

In order to allow the perturbation due to the foils to be taken into account, a stack of alternating SSTR and ILRR (0.005" thick) foils will be prepared. At midplane and at +54 cm the stack will consist of 5 ^{235}U foils sandwiched between 6 SSTR and ^{235}U fission deposits. Adjacent to

*Low Level Fission Monitor

LOCATIONS OF IRT MEASUREMENTS

Inches Above Core Midplane	One Megawatt Irradiation	Calorimetry	SSTR & Foil Irradiation	Ion Chamber Traverses	Graphite Ion Chambers	Fission Chamber Traverses	TLD Irradiation	Absolute Fission Chambers	Proton Recoil Emulsions	Proton Recoil Chambers	Compton Recoil γ-Spect
	KSS Na										
30											
Upper Axial Reflector	X						X		M	M	
15											
Core Midplane											
0											
-15											
Lower Axial Reflector	X		M				X	M	X		X
-30											
-45											

LEGEND: - Measurement Location X Monitor SSTR
M Monitor Fission Chamber SS Spectral Sets
K Potassium Samples Na Sodium Samples

TABLE HEDL A-2.2

SPI PARAMETER ASSUMPTIONS FOR IRT EXPERIMENT 8

<u>Parameter</u>	<u>Value</u>
p_0	100 kW
τ	1 min*
t_1	10 min
t_2	129 min
P	7.2×10^8 W-sec

TABLE HEDL A-2.3

SPI PARAMETER ASSUMPTIONS FOR IRT EXPERIMENT 10

<u>Parameter</u>	<u>Value</u>
p_0	200 kW
τ	1 min*
t_1	10 min
t_2	249 min
P	2.88×10^9 W-sec

TABLE HEDL A-2.4

SPI PARAMETER ASSUMPTIONS FOR IRT EXPERIMENT 12

<u>Parameter</u>	<u>Value</u>
p_0	1 MW
τ	1 min*
t_1	10 min
t_2	249 min
P	1.44×10^{10} W-sec

*Although a period of 1 min may not be possible, the smallest operationally feasible τ should be used.

these are 5 ^{239}Pu foils and 6 ^{239}Pu fission deposits. These 10 foils and 12 SSTR almost completely fill the dummy chamber and are held in place by Al spacers. A similar stack of ^{237}Np and ^{238}U SSTR fission deposits and ILRR foils is located just above midplane. This latter stack also contains 2 ^{235}U fission deposits and SSTR to act as a normalization.

Interlaboratory exchange of radiometric fission foils afforded by the ILRR program will provide highest absolute counting accuracies.^(2,3,6) Earlier and ongoing ILRR efforts, such as the irradiations in CFRMF, BIG 10, fission spectra and ensuing exchanges, have produced the highest degree of standardization and accuracy.⁽¹³⁾ Comparison of SSTR absolute fission rates with absolute NBS chamber results will be possible using the power monitor data from the experiment and IRT Experiment 5. In this manner, goal accuracy objectives of (2-5%, 1σ) will be attained for fission product yield measurements at VLP.

Further comparison of the absolute NBS fission chamber, SSTR methods, and radiometric foils will be conducted in IRT Experiment 10. Although this is a VLP irradiation, the fluence will be high enough, 800 kW-hr, to permit use of electrodeposited fission sources for SSTR dosimeters. These electrodeposits will be identical in geometry to those used for SSTR measurements in the LP fission product yield irradiation (see Section 4.0). Just as in Experiment 8 this irradiation will be conducted in dummy fission chambers at midplane and in the upper axial reflector (+54 cm).

At each location the following isotopes will be incorporated: ^{232}Th , ^{233}U , ^{235}U , ^{238}U , ^{237}Np , ^{239}Pu , ^{240}Pu , and ^{241}Pu . Both fission deposits and vanadium encapsulated dosimeters similar to those used in the LP test will be included. Measurements will be made of shorter lived fission yields in addition to ^{140}Ba , ^{103}Ru , and ^{95}Zr . Measurements of the latter will allow an intercalibration of the LP results with the absolute fission chamber and VLP SSTR absolute fission measurements.

2.2 Passive Dosimeter Irradiation (IRT Experiment No. 12)

Experiment 12 in the IRT will provide exposure conditions for extrapolating VLP spectral sets and SSTR data to LP. The SPI conditions for this experiment are summarized in Table HEDL A-2.4. However, the EI for Experiment 12 does not permit for inclusion of a monitor fission chamber; consequently, SSTR power monitors are included (See Table HEDL A-2.1). Table HEDL A-2.5 summarizes the SSTR locations and associated isotopes for IRT Experiment 12.

TABLE HEDL A-2.5

SSTR LOADING DIAGRAM FOR IRT EXPERIMENT 12

<u>Axial Position*</u>	<u>Isotopes</u>
0	^{239}Pu , ^{235}U , ^{238}U , ^{237}Np
+54	^{239}Pu , ^{235}U , ^{238}U , ^{237}Np
-80	^{239}Pu , ^{235}U , ^{238}U , ^{237}Np

*Distance above (+) and below (-) midplane in centimeters.

3.0 LOW POWER IRRADIATION

In the LP fission product yield experiments, irradiation capsules containing radiometric dosimeters and SSTR will be exposed in specially designed FTR pins called characterizer pins. These pins are used, in turn, in specially designed FTR Assemblies (A) called Characterizer Assemblies (CA). Exposure of SSTR and radiometric dosimeters in close proximity provides not only fission product yields but conversion factors for the direct use of radiometric dosimeters, i.e., so-called k-factors.^(3,9,10,14) These k-factors are of considerable practical utility, in that they constitute absolute fission rate calibration factors for the given radiometric counting system. SSTR observations provide absolute fission rates, whereas radiometric dosimeters are analyzed off-line by appropriate spectroscopy counting systems for specific fission products. Emphasis will be placed on those fission products most used in dosimetry (^{140}Ba , ^{103}Ru , ^{95}Zr , and ^{137}Cs). Other fission products of interest such as ^{131}I , ^{132}Te , ^{99}Mo , etc. will also be investigated. Since both fission product yields and k-factors can exhibit neutron energy dependence, in-fuel measurements over a wide range of in-core locations will be mandatory to meet FTR-RCP accuracy goals. In this respect, the LP irradiation will provide a unique opportunity for direct in-fuel observations, which are of the highest priority.

The current FTR schedule for the LP irradiation calls for a full day of operation at 1% of full power or 4 MW. The power-time history for the LP irradiation should preferably be a Single Period Irradiation (SPI). Although SPI conditions are ideal and have already been described in Section 3.0, they may be difficult to attain. Consequently, while an attempt should be made to match SPI conditions as closely as possible, the actual power-time history must be accurately and continuously recorded at the FTR operating console. Simultaneous recording with the FTR-Data Acquisition System (DAS) is also recommended.

The LP irradiation includes four related measurements:

- 1) In-Reactor Distribution Measurements
- 2) Perturbation Experiment
- 3) ILRR Exposure
- 4) Wire Wrap Calibrations

Each of these measurements is discussed separately below.

3.1 In-Reactor Distribution Measurements

Locations for fission product yield observations are distributed throughout the FTR core and reflector in the LP irradiation. These locations are summarized in Table HEDL A-3.1, which also specifies the various isotopes that will be used at each location. This plan provides for a limited sampling of radial and axial distributions throughout the FTR core and reflector.

Fission product yield exposures in the LP irradiation will be carried out in capsules especially designed to both minimize perturbations and facilitate post-irradiation recovery. These capsules must fit into characterizer pins and be as short as possible. The actual design of these irradiation capsules is shown in Figures HEDL A-2 and A-3. All dosimeter capsules intended for use in characterizer pins must be sealed by appropriate welding techniques. In addition, the irradiation capsules have been designed to accommodate the standard length radiometric dosimetry wires. Electrodeposited fission sources will be used with mica SSTR for the LP fission product yield irradiation.

In a few selected LP SSTR capsules, at +0, +38, and +54 cm in CA 2101, additional electrodeposited ^{239}Pu sources will be used with quartz crystal SSTR. These LP quartz SSTR will be used to intercompare mica SSTR exposed at LP with quartz SSTR exposed at HP. These comparisons will be helpful in accounting for systematic effects that arise in extrapolating from LP to HP.

TABLE HEDL A-3.1

IN-REACTOR LOCATIONS AND ISOTOPES FOR THE LP IRRADIATION

<u>CA</u>	<u>Axial Location*</u>	<u>Isotopes</u>
2101	0	^{239}Pu , ^{240}Pu , ^{241}Pu , ^{233}U , ^{235}U , ^{238}U , ^{237}Np , ^{232}Th
2101	+38	^{239}Pu , ^{240}Pu , ^{241}Pu , ^{235}U
2101	+54	^{239}Pu , ^{233}U , ^{235}U , ^{238}U , ^{237}Np , ^{232}Th
2101	-150	^{239}Pu , ^{235}U , ^{237}Np
3405	0	^{239}Pu , ^{240}Pu , ^{241}Pu , ^{233}U , ^{235}U , ^{238}U , ^{237}Np
3405	+54	^{239}Pu , ^{235}U , ^{238}U
3608	0	^{239}Pu , ^{240}Pu , ^{241}Pu , ^{233}U , ^{235}U , ^{238}U , ^{237}Np
3608	+54	^{239}Pu , ^{233}U , ^{235}U , ^{238}U , ^{237}Np
3710	0	^{239}Pu , ^{233}U , ^{235}U , ^{238}U , ^{237}Np , ^{232}Th
2909	0	^{239}Pu , ^{235}U , ^{238}U

*Distance above (+) and below (-) midplane in centimeters.

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

UNASSEMBLED CAPSULE AND
FOIL RETAINING SPRING

FISSIONABLE DEPOSITS MICA SSTR

IRON FLUX DISK

QUARTZ SSTR

ASSEMBLED CAPSULE

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

FIGURE HEDL A-2. Contents of a Typical LP SSTR Dosimeter Capsule.

FIGURE HEDL A-3. Radiograph of a Typical LP Capsule After Assembly and Welding.

Neg 7906044-7

3.2 Perturbation Experiment

Goal accuracies of fission product yield measurements (2-5%, 1σ) apply to in-fuel determinations. However, for the LP observations, only empty characterizer pins, i.e. without fuel pellets, will be used. Consequently, some measure of the perturbation created by carrying out measurements in nonfueled pins must be ascertained. To this end, a specific perturbation experiment has been planned for the LP irradiation.

In this experiment, radiometric fission dosimeters and SSTRs will be placed in the fuel column of test pins. Eight identical test pins will be placed around the central pin in assembly 2101; the central pin will not contain fuel, but will contain spectral packages or sensors at axial locations identical to those in the eight test pins. Radiometric dosimeters and SSTR using ^{239}Pu will be used at midplane, and SSTR using ^{238}U will be used at 10 cm below midplane. Except for the nonfueled central pin, all samples must be encapsulated to reduce contamination from the fuel pellets. Except for the sensors and some minor modifications to individual fuel pellets, the internal design will be similar to that of FFTF standard driver fuel pins. The end cap modifications will be the same as those for nonfuel sensor pins.

Comparison of the fueled and nonfueled fission rates observed in this experiment should quantify the perturbation that arises in nonfueled measurements. Nonthreshold and threshold perturbation effects near midplane are treated by using ^{239}Pu and ^{238}U , respectively. Inclusion of radiometric fission dosimeters will provide perturbation effects for both k-factors as well as fission product yields.

3.3 ILRR Exposure

A special ILRR fission product yield experiment has been scheduled for irradiation at axial midplane in CA 2101 near core center and in the radial reflector CA 3710. Dosimetry capsules that will be used are identical to

those already described for the LP irradiation (see Section 4.1). Through extensive interlaboratory counting and calibration efforts with these miniaturized radiometric dosimeters, the highest possible accuracy levels will be attained.^(3,6,12) To this end, five radiometric sets will be irradiated at each of these two locations. Using SSTR based fission rates, a limiting accuracy of 2 to 5% (1σ) is expected for these ILRR fission product yield results. The following nuclides will be used: ^{233}U , ^{235}U , ^{238}U , ^{232}Th , ^{237}Np , ^{239}Pu .

Nonfission reactions that will be used with these fission reactions for 1) benchmarking the neutron environment and 2) ENDF/B data development and testing for this ILRR fission product yield experiment are identified in Appendix A. These include both the standard nonfission radiometric monitors (RM) and ^{10}B and ^6Li helium accumulation fluence monitors (HAFM) used elsewhere in the reactor characterization measurements, but also $\text{Nb}(n,n')$ monitors and quartz damage sensors supplied by CEN/SCK.* Encapsulated melt wires will be included with the quartz sensors to provide required temperature information. ILRR fluence standards from CFRMF that will be available to certify the correctness of counting and data analysis systems for participating laboratories are discussed and identified in Appendix C.

3.4 Wire Wrap Calibrations

In addition to measuring fission product yields, the LP irradiation will provide an opportunity to test and calibrate wire-wrap dosimetry. Either solid wire wrap or hollow wire wrap dosimetry can be performed. In the hollow wire wrap dosimetry technique, which is currently being developed, hollow SS wire is used to replace conventional solid core SS wire wrap. Vanadium encapsulated dosimeters can be loaded into the hollow wire wrap, which is then used to wrap fuel pins in the identical manner to solid wire wrap. With appropriate loading of the hollow wire wrap, it is clear that

*Mol, Belgium

axial dosimetry measurements are possible. On the other hand, solid wire wrap can also be sectioned to obtain axial dosimetry data.

Exposure of wire wrap dosimeters, in close proximity with fission product yield irradiation capsules, will provide calibration factors for extrapolation of observed isotopic fission rates to in-pin values. In the driver fuel CA, wire wrap is available on only the six outermost characterizer pins plus two additional characterizer pins. As a consequence of these constraints, the possible location of wire wrap dosimeters is quite limited. These constraints, together with the already established location of the fission product yield irradiation capsules as shown in Table HEDL A-3.1, restrict even further the possible location of wire wrap dosimeters in the LP irradiation. Table HEDL A-3.2 summarizes the recommended locations for hollow wire wrap dosimeters in the LP irradiation. At all remaining LP yield measurement locations (summarized in Table HEDL A-3.3.) solid wire wrap dosimetry should be performed, i.e., segments of the stainless steel wire will be recovered and analyzed for the several reaction products. The final as-built location of hollow and solid wire wrap dosimetry on a characterizer pin must be carefully certified by pre-irradiation and if possible, post-irradiation x-ray and/or neutron radiography.

W I R E - W R A P P I N S

H O L L O W W I R E - W R A P P I N S

HEDL 7706-255.3

FIGURE HEDL A-4. Characterizer Pin Locations Available for Wire Wrap Dosimetry in Driver Fuel CA.

TABLE HEDL A-3.2

LP-HOLLOW WIRE WRAP DOSIMETRY LOCATIONS

<u>CA</u>	<u>Axial Location*</u>	<u>Isotopes</u>
2101	+53	$^{238}\text{UO}_2$ (2 ea), $^{235}\text{UO}_2$, $^{239}\text{PuO}_2$
	+45	$^{238}\text{UO}_2$ (2 ea), $^{235}\text{UO}_2$ (2 ea), $^{239}\text{PuO}_2$
	0	$^{238}\text{UO}_2$ (2 ea), $^{235}\text{UO}_2$, $^{237}\text{NpO}_2$, $^{239}\text{PuO}_2$, FTR-INNER FUEL
	-46	$^{238}\text{UO}_2$, $^{235}\text{UO}_2$, $^{239}\text{PuO}_2$
	-54	$^{238}\text{UO}_2$, $^{235}\text{UO}_2$, $^{239}\text{PuO}_2$

TABLE HEDL A-3.3

LP-SOLID WIRE WRAP DOSIMETRY LOCATIONS

<u>CA</u>	<u>Axial Location*</u>
2101	+54
3405	0
3608	+54

*Distance above (+) and below (-) midplane in centimeters. These locations are nominal since dosimeters extend over several centimeters.

4.0 HIGH POWER IRRADIATION

The current FTR schedule for the HP irradiation calls for eight full days of operation at full power, i.e. 400 MW. The power-time history for the HP irradiation should preferably be a Single Period Irradiation (SPI). Although SPI conditions are ideal and have already been described in Section 3.0, they may be difficult to attain. Consequently, while an attempt should be made to match SPI conditions as closely as possible, the actual power-time history must be accurately recorded at the FTR operating console. (Simultaneous recording with the FTR-Data Acquisition System [DAS] is also recommended).

Characterizer Assemblies (CA) used at LP will be reused in the HP irradiation. As in the LP irradiation, passive dosimetry in the HP irradiation will be located in special characterizer pins that in turn are loaded into these special CA. Locations and isotopes planned for SSTR-HP absolute fission rate measurements are summarized in Table HEDL A-4.1. Of primary concern are the measurements for three mean neutron energies of about 0.20, 0.35, and 0.50 MeV, which will supplement and complement previous measurements in EBR-II at mean energies of about 0.3, 0.5, and 0.7 MeV.⁽⁷⁾ Emphasis is again placed on attaining radial and axial distribution data throughout the FTR core.

Fission deposits for HP SSTR will be prepared in an identical geometry and with the same stock solutions used for the LP deposits. Most of these deposits lie in the sub-nanogram/cm² region and as such represent the "state of the art" for low mass fission deposits. Hence, special care in preparation, handling and exposure will be mandatory to guard against extraneous track background contributions. In general, SSTR fission deposit specifications are based on attaining a statistical precision of better than 1%, i.e., more than 10⁴ observed tracks.

TABLE HEDL A-4.1

IN-REACTOR LOCATIONS AND ISOTOPES FOR THE SSTR-HP IRRADIATION

<u>CA</u>	<u>Axial Location*</u>	<u>Isotopes</u>
2101	0	^{239}Pu , ^{240}Pu , ^{233}U , ^{235}U , ^{238}U , ^{237}Np
2402	0	^{239}Pu , ^{240}Pu , ^{235}U , ^{238}U , ^{237}Np
2603	0	^{239}Pu , ^{240}Pu , ^{235}U , ^{238}U , ^{237}Np
2704	0	^{239}Pu , ^{233}U , ^{235}U , ^{238}U , ^{237}Np
2804	0	^{239}Pu , ^{235}U , ^{233}U
3808	0	^{239}Pu , ^{235}U , ^{238}U , ^{237}Np

*Distance above (+) and below (-) midplane in centimeters

Track densities generated in the HP irradiation will be on the order of 10^8 tracks/cm², three orders of magnitude higher than track densities anticipated for the LP irradiation. Use of Scanning Electron Microscopy (SEM) will therefore be mandatory for the HP-SSTR, since optical microscopy techniques cannot extend into this high track density regime. As a consequence, current efforts with SEM techniques, manual and/or automated, will provide the requisite capability for HP high track density SSTR.

Whereas a number of track recorder candidates exist for the LP irradiation, such as mica and quartz glass, only one SSTR candidate is applicable for the HP irradiation. Due to track fading at the high temperature of the HP irradiation only natural quartz crystal appears to provide the necessary track retention characteristics.^(11,15) Further track annealing studies for all these major SSTR candidates are underway, so that quantitative evaluation of this effect can be ascertained for both the LP and HP irradiations. On the basis of preliminary results, correction of this effect for natural quartz SSTR should be small for the HP irradiation. It should be recognized, however, that annealing characteristics are not the sole criterion to apply in the selection of the most suitable SSTR for the LP and HP irradiations. For example, trace impurities of heavy elements in SSTR can produce an undesirable background of fission tracks. Consequently, only SSTR of the highest purity can be used in the LP and HP irradiations.

Absolute fission rates determined with these SSTR techniques in the HP irradiation will be traceable back to the absolute NBS fission chamber (see Section 2.0 for a discussion of VLP fission rate measurements in the IRT). A direct comparison of these SSTR based absolute fission rates will be possible with those absolute fission rate observations carried out by Allied Chemical (INEL). The latter observations employ completely different methods, chiefly mass spectrometry, to effectively sum the entire fission product mass distribution, i.e., either the high or the low mass peak. According to current FTR-RCP plans, such comparisons will be possible for ²³⁹Pu and ²³⁵U in CA 2101, 2402, and 2603. An additional comparison for ²³³U will also be possible in CA 2101.

The HP irradiation affords an opportunity to test the reliability of k-factors that have been generated in the VLP and LP irradiations as well as EBR-II and other ILRR Neutron field standardization work.^{(2,3-7)*} The capsule designed for these SSTR HP irradiations, which is shown in Figure HEDL A-5, provides ample space for miniature radiometric fission wire dosimeters. These capsules will simultaneously provide space for HEDL isotope dilution mass spectrometry encapsulated fission specimens dedicated for Nd measurements. As indicated (implicitly) in the original FTR-RCP plans, a minimum of three vanadium encapsulated hollow-wire wrap type dosimeters of each isotope must be present at each of three locations corresponding to mean energies of about 0.20, 0.35, and 0.50 MeV. Based on previous EBR-II Run 75 D test results, at least three dosimeters are necessary to supply adequate statistical precision.⁽⁷⁾ The three locations as well as the isotopes planned for this combined k-factor-Nd experiment at HP are summarized in Table HEDL A-4.2.

As in the LP irradiation, the HP irradiation provides an opportunity to test and calibrate wire wrap dosimetry. Exposure of wire wrap dosimeters in close proximity with HP-SSTR will provide calibration factors for extrapolating isotopic fission rates to in-pin locations. Just as for the LP irradiation, existing constraints severely restrict possible locations for HP wire wrap dosimeters. Tables HEDL-4.3 and HEDL-4.4 contain current recommendations for hollow wire wrap and solid wire wrap dosimetry, respectively, in the HP irradiation. The final as-built location of hollow and solid wire wrap dosimetry on a characterizer pin must be carefully certified by pre-irradiation, and if possible, post-irradiation x-ray and/or neutron radiography.

Selected fission and nonfission reactions that are expected to receive multi-laboratory counting from foil sets exposed at one core and one axial reflector region are identified in Appendix A. Interlaboratory counting is essential for final certification of the correctness of the HEDL high level counting

*The specific fission products measured will be the same as in the LP test with the addition of the stable isotopes ^{148}Nd , ^{143}Nd , ^{145}Nd , ^{146}Nd , and ^{150}Nd .

**FTR-RCP HIGH POWER TEST
CAPSULE H1 (ASSEMBLY 2101,
PIN #7-11, FP1)**

FIGURE HEDL A-5. Radiograph of a Typical Irradiation Capsule for the HP-SSTR Fission Rate Measurements.

TABLE HEDL A-4.2

HP K-FACTOR AND Nd EXPERIMENT LOADING DIAGRAM

<u>Assembly</u>	<u>Axial Location(a)</u>	<u>$\bar{E}$(b)</u>	<u>Isotopes for Destructive Analysis(c)</u>
2101	0	0.50	^{239}Pu , ^{240}Pu , ^{241}Pu , ^{237}Np , ^{233}U , ^{235}U , ^{238}U , ^{232}Th FTR-0 ^(d)
2704	0	0.35	^{239}Pu , ^{240}Pu , ^{241}Pu , ^{237}Np , ^{233}U , ^{235}U , ^{238}U , FTR-I, FTR-0
2804	0	0.20	^{239}Pu , ^{241}Pu , ^{233}U , ^{235}U FTR-I, FTR-0

(a) Distance above fuel midplane in centimeters

(b) Mean neutron energy

(c) Three radiometric dosimeters of each of the ten isotopes will be placed at each location. Those listed in the table will be destructively analyzed for measurement of Nd fission yields; the remainder will be radiometrically counted only.

(d) FTR-I is a dosimeter containing inner driver fuel; FTR-0 contains outer driver fuel

TABLE HEDL A-4.3
HP-HOLLOW WIRE WRAP DOSIMETRY LOCATIONS

<u>CA</u>	<u>Axial Location*</u>	<u>Isotopes</u>
2101	0	^{239}Pu , ^{240}Pu , ^{241}Pu , ^{233}U , ^{238}U , ^{237}Np , ^{232}Th
	+54	^{239}Pu , ^{233}U , ^{235}U , ^{238}U , ^{237}Np , ^{232}Th
2603	0	^{239}Pu , ^{240}Pu , ^{241}Pu , ^{233}U , ^{235}U , ^{238}U , ^{237}Np

TABLE HEDL A-4.4
HP-SOLID WIRE WRAP DOSIMETRY LOCATIONS

<u>CA</u>	<u>Axial Location*</u>
2402	0

* Distance above (+) and below (-) midplane in centimeters

laboratory equipment and data analyses procedures used for FTR HP irradiated sensors. These HP sensors will have impurity background interferences that are unique to the particular FTR in-and out-of-core flux spectral conditions of the HP irradiation. As such, they cannot be properly reproduced in any other "Benchmark Neutron Field," including EBR-II.

EXPECTED ACHIEVEMENTS

The formulation, planning and fabrication of dosimetry for all ILRR benchmark field experiments in support of the FTR-RCP are expected to be completed before the end of the first quarter of FY 1980. IRT VLP tests are expected to start in late February and early March 1980. The LP and HP test irradiations are expected to be completed by 1980. CFRMF fission and non-fission calibration and irradiations will be scheduled and accomplished during these same time periods. The counting of radiometric, SSTR, and HAFM monitors will be accomplished in the last two quarters of FY 1980 and the first half of FY 1981. Final analysis and reporting of results is expected in the last half of FY 1981.

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A P P E N D I X A

PRELIMINARY RECOMMENDATIONS BY ILRR PROGRAM PARTICIPANTS

The block diagram, Table HEDL-A1, provides summary information on the expected results and benefits of FTR-RCP supportive ILRR Program Participants measurements which have been discussed or considered in this report. More detailed and updated summary type information is provided in Sections A-I, A-II, and A-III.

TABLE HEDL-A1

SUMMARY-RESULTS AND BENEFITS OF FTR-RCP SUPPORTIVE ILRR PROGRAM PARTICIPANTS' MEASUREMENTS

	<u>Very Low Power (VLP) Test</u>	<u>Low Power (LP) Test</u>	<u>High Power (HP) Test</u>
FISSION REACTIONS:	<ul style="list-style-type: none"> Counting Systems Calibrations Highest Accuracy (2-3%, 1σ) Absolute Fission Yields - ¹⁴⁰Ba, ⁹⁵Zr, etc. 	<ul style="list-style-type: none"> Counting Systems Calibrations Lower Accuracy (2-5%, 1σ) Absolute and High Accuracy (1-3%, 1σ) Relative Fission Yields - ¹⁴⁰Ba, ⁹⁵Zr, ¹³⁷Cs, etc. 	<ul style="list-style-type: none"> Lower Accuracy (2-5%, 1σ) Absolute and High Accuracy (1-3%, 1σ) Relative Fission Yields - ¹⁴⁰Ba, ⁹⁵Zr, ¹³⁷Cs, etc., for Direct Wire Wrap Dosimetry Burnup Predictions for FOTAs and Other Fuels Experiments Relative (1-3%, 1σ) and Absolute (2-5%, 1σ) Nd Yields for Direct Fuels and Wire Wrap Dosimetry Burnup Predictions for FOTAs and Other Fuels Experiments
NONFISSION REACTIONS:	<ul style="list-style-type: none"> Counting Systems (1-3%, 1σ) Calibrations and Checks to Previous Dosimeter Benchmark Results 	<ul style="list-style-type: none"> Highest Accuracy (1-5%, 1σ) Absolute Reaction Rates 	<ul style="list-style-type: none"> Lower Accuracy (3-7%, 1σ) Absolute Reaction Rates for MOTAs and Other Fuels and Materials Experiments
QUARTZ DAMAGE SENSORS:		<ul style="list-style-type: none"> Validation/Calibration of Displaced Atom (DPA) and Other Materials Damage Prediction Procedures and Data for Fuel and Materials Experiments, including FOTAs and MOTAs 	

NOTE: The combined results of all of the above tests will also be used to establish a consensus set of integral data for a FTR Dosimetry Benchmark for data testing and development for the ENDF/B "Dosimetry", "Gas Production" and "Fission Yield" files.

A-I. VERY LOW POWER TESTS: FFTF-IRT and CFRMF ILRR TYPE HIGH ACCURACY ABSOLUTE CALIBRATIONS (Exposures $\geq 5 \times 10^{15}$ n/cm², core center total flux) FOR FISSION REACTIONS, SELECTED NON-FISSION REACTIONS, AND SHORTER HALF-LIFE REACTION PRODUCTS.

A. Fission Reactions

- Absolute fissions and fission rates based on consensus NBS chamber and HEDL SSTR results - limiting accuracy ($\sim 1\% 1\sigma$) is expected to be associated with deposits' uniformity and mass assay uncertainties. Consensus established via direct comparison of methods in test exposures of SSTRs at NBS prior to IRT tests.
- Consensus absolute fission yields and counting system calibration factors and checks* based on ILRR type radiometric measurements (¹⁴⁰Ba, ⁹⁵Zr, ¹³⁷Cs**, and others as appropriate)--limiting accuracy (~ 2 to $3\% 1\sigma$) is expected to be associated with counting standards and nuclear data uncertainties.
- Fission Reaction Set: ²³³U(n,f), ²³⁵U(n,f), ²³⁸U(n,f), ²³²Th(n,f), ²³⁷Np(n,f), ²³⁹Pu(n,f).
- Pre- and post-irradiation analysis of assigned calibration and fission yield foil sets is required as follows:

* Against previously established and recommended CFRMF, BIG-10, and Fission Spectra ENDF/B Dosimetry Cross Section and Fission Yield Files Benchmark Field Results.

** High accuracy ¹³⁷Cs results are difficult to obtain, but still will be attempted.

- CFRMF IRRADIATION*: ANL, CEN/SCK, EG&G, HEDL, and ORTEC counting systems calibrations using a complete fission reaction set.
- TWO IRT IRRADIATION POSITIONS (one core and one axial reflector region): EG&G and HEDL using ILRR 1/2" diameter metal foils; EG&G, CEN/SCK, and HEDL using miniaturized vanadium encapsulated oxide foils.

B. Nonfission Reactions

- Absolute counting system calibration checks** based on radiometric and helium mass spectrometry measurements - limiting accuracy (~ 1 to 3% 1σ) is expected to be associated with counting standards and nuclear data uncertainties.
- Nonfission Reaction Set: $^{197}\text{Au}(n,\gamma)^{198}\text{Au}$ (0.1% alloy), $^{58}\text{Ni}(n,p)^{58}\text{Co}$, $^{54}\text{Fe}(n,p)^{54}\text{Mn}$, $^{58}\text{Fe}(n,\gamma)^{59}\text{Fe}$, $^{10}\text{B}(n, \text{total He})$, and $^6\text{Li}(n, \text{total He})$.
- Post-irradiation analysis of assigned nonfission foil sets is required as follows:
 - CFRMF IRRADIATION*: ANL, CEN/SCK, EG&G, HEDL and ORTEC counting systems calibrations using the radiometric foils from a complete nonfission reaction set. All helium mass spectrometry system calibrations using the ^{10}B and ^6Li nonfission foils.

*Each laboratory should schedule the CFRMF irradiation to correspond with their planned counting of IRT and/or Low Power FFTF post-irradiation foil counting requirements. The assigned EG&G and HEDL sets may require a second CFRMF exposure for the counting of both IRT and Low Power tests.

**Against previously established and recommended CFRMF, BIG-10, and Fission Spectra ENDF/B Dosimetry Cross Section and Fission Yield Files Benchmark Field Results.

- IRT IRRADIATION: EG&G and HEDL, complete nonfission reaction set.

C. Irradiation Plan

- CFRMF - Central core position inside duplicate NBS fission chamber.
- IRT - One central core and one axial reflector position inside duplicate NBS fission chamber.

A-II. LOW POWER TESTS: FFTF CORE AND RADIAL REFLECTOR REGION (Miniaturized FBR Type Passive Sensors) HIGH ACCURACY CALIBRATIONS (Exposure $\sim 6.0 \times 10^{18}$ n/cm², core center total fluence)* FOR SELECTED STABLE, SHORT, AND LONG HALF-LIFE FISSION AND NONFISSION REACTIONS.

A. Fission Reactions

- Absolute fission rates based on HEDL SSTR results--limiting accuracy ($\sim 1\%$ 1σ) is expected to be associated with deposit uniformity and mass assay uncertainties.
- Consensus absolute and relative fission yields and counting system calibration factors for miniaturized FBR type passive sensors based on radiometric measurements (¹⁴⁰Ba, ⁹⁵Zr, ¹³⁷Cs, and other as appropriate)--limiting accuracy (~ 2 to 5% 1σ) is expected to be associated with mass assay of vanadium encapsulated sensors. The relative accuracy of fission yields (which is not dependent on mass assay) is expected to be very good and limited (~ 2 to 3% 1σ) by counting standards and nuclear data uncertainties.

*The radial reflector value is expected to be 2×10^{18} total fluence. The core center/radial reflector positions relative ²³⁵U and ²³⁸U fission rates are expected to be ~ 1.4 and 5.5 , respectively.

Passive Dosimeters

- Fission Reaction Set*: 1% $^{233}\text{U/V}$, $^{235}\text{U(n,f)}$, $^{238}\text{U(n,f)}$, $^{232}\text{Th(n,f)}$, $^{237}\text{Np(n,f)}$, $^{239}\text{Pu(n,f)}$.
- Pre- and post-irradiation analysis of assigned fission foil sets is required as follows:
 - FFTF CORE REGION DOSIMETERS: ANL, CEN/SCK, EG&G, HEDL and ORTEC.
 - FFTF RADIAL REFLECTOR REGION DOSIMETERS: ANL, CEN/SCK, EG&G, HEDL and ORTEC.

B. Nonfission Reactions

- Consensus absolute reaction rates based on ANL, CEN/SCK, EG&G, HEDL, and ORTEC radiometric and AI helium mass spectrometry measurements - limiting accuracy (1 to 3% 1σ) is expected to be associated with counting standards, nuclear data, and alloy composition (mass assay) uncertainties.
- Nonfission Reaction Set**: $^{10}\text{B(n, total He)}$, $^6\text{Li(n, total He)}$, $^{45}\text{Sc(n,}\gamma\text{) 0.1\% alloy}$, $^{58}\text{Fe(n,}\gamma\text{)}$, $^{59}\text{Co(n,}\gamma\text{) 0.1\% alloy}$, $^{238}\text{U(n,}\gamma\text{) 0.1\% alloy}$, $^{232}\text{Th(n,}\gamma\text{) 0.1\% alloy}$, $^{54}\text{Fe(n,p)}$, $^{58}\text{Ni(n,p)}$, $^{46}\text{Ti(n,p)}$, $^{47}\text{Ti(n,p)}$, $^{48}\text{Ti(n,p)}$, $^{63}\text{Cu(n,}\gamma\text{)}$; $^{93}\text{Nb(n,n')}$, $^{109}\text{Ag(n,}\gamma\text{) 0.1\% alloy}$, $^{181}\text{Ta(n,}\gamma\text{) 0.1\% alloy}$, $^{60}\text{Ni(n,p)}$, $^{58}\text{Ni(n,}\gamma\text{)}$, $^{54}\text{Fe(n,}\gamma\text{)}$.

Note: Non-HEDL sets do not include ^{10}B , ^6Li , 1% $^{238}\text{U/V}$, 1% $^{233}\text{ThO}_2/\text{MgO}$, and Nb

*Tentative, subject to material availability. Final set will be chosen from this list.

**Separate specimens of each wire type will be cut from the stock used for making the alloy foils for homogeneity and mass assay overchecks. These specimens are expected to be analyzed by HEDL, CEN/SCK and ORNL as required.

- Post-irradiation analysis of assigned nonfission foil sets is required as follows:

- FFTF CORE REGION DOSIMETERS: ANL, CEN/SCK, EG&G, HEDL and ORTEC.
- FFTF RADIAL REFLECTOR REGION DOSIMETERS: ANL, CEN/SCK, EG&G, HEDL and ORTEC.

C. Quartz Damage Sensors

- The CEN/SCK participation will also encompass the preparation of quartz damage sensors, of high purity niobium and iron flux level normalization foils, and of selected quartz or steel encapsulated melt wires for passive temperature measurements for the quartz monitors. These monitors will be placed in a regular (~30 mm long x ~4.5 mm diameter) HEDL dosimetry capsule.

D. Irradiation Location

- FFTF CORE REGION LOCATION: CA 2101, Midplane
- FFTF RADIAL REFLECTOR REGION LOCATION: CA 3710, Midplane

A-III. HIGH POWER TESTS: FFTF CORE AND AXIAL REFLECTOR REGION (Miniaturized FBR Type Passive Sensors) CALIBRATIONS (exposure $\sim 4.8 \times 10^{21}$ n/cm², core center total flux) FOR SELECTED STABLE, SHORT, AND LONG HALF-LIFE FISSION AND NONFISSION REACTIONS.

A. Fission Reactions: Selected Round Robin measurements from existing spectral sets.

- Consensus absolute fission rates based on absolute fission yields/ counting systems calibration factors for ^{140}Ba , ^{95}Zr , ^{137}Cs and others as appropriate (two core region and two axial reflector region spectral sets will be round robin counted) and quartz SSTRs, if feasible - limiting accuracy is to be determined but is expected to be 2-5% (1σ) range.
- Fission Reaction Set*: $^{233}\text{U}(\text{n},\text{f})$, $^{235}\text{U}(\text{n},\text{f})$, $^{238}\text{U}(\text{n},\text{f})$, $^{232}\text{Th}(\text{n},\text{f})$, $^{237}\text{Np}(\text{n},\text{f})$, $^{239}\text{Pu}(\text{n},\text{f})$, $^{240}\text{Pu}(\text{n},\text{f})$, $^{241}\text{Pu}(\text{n},\text{f})$.
- Pre- and post-irradiation analysis of assigned fission foil sets is required as follows:
 - FFTF CORE REGION DOSIMETERS: ANL, CEN/SCK and HEDL**
 - FFTF AXIAL REFLECTOR REGION DOSIMETERS: ANL, CEN/SCK and HEDL**

B. Nonfission Reactions: Selected Round Robin measurements from existing spectral sets.

- Consensus absolute reaction rates based on ANL, CEN/SCK, EG&G and HEDL radiometric and AI helium mass spectrometry measurements - limiting accuracy (1 to 7% 1σ) is expected to be associated with counting standards, nuclear data, and alloy composition (mass assay) uncertainties.

*Tentative, subject to material availability. Final set will be chosen from this list.

**At present, only these laboratories are set up to routinely count sensors from high neutron fluence exposures. HEDL will be the first laboratory to count, ANL second, and CEN/SCK last.

- Nonfission Reaction Set*: $^{10}\text{B}(\text{n}, \text{total He})$, $^6\text{Li}(\text{n}, \text{total He})$, $^{45}\text{Sc}(\text{n}, \gamma)$ 0.1% alloy, $^{58}\text{Fe}(\text{n}, \gamma)$, $^{59}\text{Co}(\text{n}, \gamma)$ 0.1% alloy, $^{238}\text{U}(\text{n}, \gamma)$ 0.1% alloy, $^{232}\text{Th}(\text{n}, \gamma)$ 0.1% alloy, $^{54}\text{Fe}(\text{n}, \text{p})$, $^{58}\text{Ni}(\text{n}, \text{p})$, $^{46}\text{Ti}(\text{n}, \text{p})$, $^{47}\text{Ti}(\text{n}, \text{p})$, $^{48}\text{Ti}(\text{n}, \text{p})$, $^{63}\text{Cu}(\text{n}, \gamma)$; $^{93}\text{Nb}(\text{n}, \text{n}')$, $^{109}\text{Ag}(\text{n}, \gamma)$ 0.1% alloy, $^{181}\text{Ta}(\text{n}, \gamma)$ 0.1% alloy, $^{60}\text{Ni}(\text{n}, \text{p})$, $^{58}\text{Ni}(\text{n}, \gamma)$, $^{54}\text{Fe}(\text{n}, \gamma)$, $^{55}\text{Mn}(\text{n}, 2\text{n})$.
- Post-irradiation round-robin analysis of two nonfission foil sets is required as follows:
 - FFTF CORE REGION DOSIMETERS: In order, HEDL, ANL and CEN/SCK
 - FFTF AXIAL REFLECTOR REGION DOSIMETERS: In order, HEDL, ANL and CEN/SCK

C. Irradiation Plan

- FFTF CORE REGION LOCATION: CA 2101, Midplane
- FFTF RADIAL REFLECTOR REGION LOCATIONS: CA 2704 and CA 2804, Midplanes

*Tentative, subject to material availability. Final set will be chosen from this list.

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HEDL-42

A P P E N D I X B

BASIS OF ABSOLUTE FISSION RATE DETERMINATION FOR FAST REACTORS

Fast breeder reactor fuel development programs have established goal accuracies for determination of fission rates and fuel burnup of 1-3%.^{(2)*} In the near term HEDL has set goal accuracies of 2-5%. This accuracy requirement is set both for destructive assay measurements and for nondestructive fuel and dosimetry measurements.

Destructive fuel assay measurements have relied on measurements of the Nd stable fission products and in particular ^{148}Nd . This isotope is chosen because it is thought not to migrate in the fuel during irradiation and the fission yield is relatively independent of neutron energy.

Measurement of the stable ^{148}Nd gives a good integration over the length of the irradiation, but only provides information on the fission rate if the power time-history is well known. A second limitation is that only few measurements are possible because of the relatively high cost, and thus interpolation and extrapolation to other fuel positions is necessary. The use of a single fission product isotope also adds the uncertainty of the fission yield directly into all burnup determinations.

Nondestructive radiometric dosimetry techniques have been developed, therefore, to provide information on more fission products, at more locations, and at lower cost. These measurement techniques are based on conventional dosimetry techniques with improved dosimeter designs to cause minimal perturbation to fuel assemblies. These techniques require very accurate calibration. Methods to accomplish this calibration to achieve the accuracies listed above (i.e., less than 2% near points of measurement and less than 5% throughout the reactor core) are discussed below. This validation/calibration of the dosimetry techniques involves not only the actual dosimetry measurement but also must include perturbations and extrapolations to give accurate

*Text reference, p. HEDL-31

fission integrals over fuel pins, assemblies, and whole cores. For the latter, the FFTF Reactor Characterization Program (RCP) will provide the necessary experimental information for correlation with reactor physics calculations (diffusion, transport, and Monte Carlo). This is summarized in Tables HEDL-B1 and HEDL-B2.

The Interlaboratory Reaction Rate (ILRR) Program was started to provide a basis for the development of high accuracy dosimetry techniques and accurate assessment of uncertainties. The philosophy of this program is that true evaluation of measurement and calculational uncertainties can only be carried out if a number of techniques and independent laboratories are involved. The ILRR program has already effected a number of irradiations in zero power facilities in order to intercalibrate measurements made by various participating laboratories in several benchmark neutron fields.

Measurements in FFTF will build on these ILRR results. However, it is still necessary, in order to meet the stringent accuracy requirements, that a variety of techniques and laboratories be involved. In addition, tests of a number of dosimetry techniques are necessary, to produce highest accuracy fission measurements under the variety of conditions. The interrelationship of these measurements is explained below.

1. Absolute Fission Rate Measurements

Absolute fission rates will be measured by absolute fission chambers, solid state track recorders (SSTR) and mass spectroscopy. Direct intercomparisons between the fission chambers and SSTR will be made in the In-Reactor Thimble (IRT). SSTR measurements will also be made at Low Power (LP) (4 MW - 24 hour irradiation) and at High Power (HP) (400 MW - 8 day irradiation). The mass spectroscopy measurements will determine absolute fissions by measuring most of the stable isotopes produced and summing. Fission yields of individual isotopes will also be determined.

TABLE HEDL-B1

ATTAINMENT OF HIGHEST ACCURACY FISSION RATE AND BURN-UP ESTIMATES FOR FFTF FUEL PINS

<u>Item</u>	<u>Experimental Data or Other Information</u>
• Fission Yields or k-Factors (^{140}B , ^{95}Zr , etc.)	IRT measurements of fission yields using ILRR type metal foils vs. SSTR and absolute fission chamber
• Fission Yields or k-Factors (^{137}Cs , ^{148}Nd , etc.)	LP Test measurements relative to ^{140}Ba and SSTR
• Accuracy of Measurement Methods	ILRR intercalibrations: Standard fields and FFTF Dosimetry measurements of several fission products
• Dosimetry Perturbations	Wire wrap microdosimetry and stainless steel dosimetry estimates
• Relation of Wire Wrap Dosimetry Measurements to Pin Fission Rate	Correlation of fuel burn-up measurements and wire wrap dosimetry results for FFTF-HP and FOTA
• Characterization of Wire Wrap Dosimetry	Correlation with standard dosimetry in EBR-II Run 97 test and FFTF-LP
• Burn-Up, Burn-In - Burn-Out Effects	Correlation of LP, HP and FOTA measurements
• Fission Product Migration	Correlation of gamma scan, fuel destructive assay and dosimetry measurements
• Time Perturbation Effects Such as Control Rod Movement, Flux Changes, etc.	Long-term fuel dosimetry measurements in FOTA, etc.
• High Cost of Fuel Burn-up Measurements	Increased use of microdosimetry
• Axial Distributions of Fission Rates	FFTF LP and HP test measurements and calculational comparisons
• Characterization of Standard Dosimetry Monitors	Destructive analyses as part of fission yield measurements; intercalibration with SSTR and ILRR foils (FFTF and CFRMF)

TABLE HEDL-B2
FISSION YIELD MEASUREMENTS

<u>VLP Test</u>	<u>LP Test</u>	<u>HP Test</u>
Highest Accuracy	Intermediate Accuracy	High Accuracy
Absolute Fission Yields	K-Factors*	K-Factors*
^{140}Ba , ^{95}Zr and ^{103}Ru	^{140}Ba , ^{95}Zr , ^{103}Ru and ^{137}Cs	Radiometric and Stable Products (Nd)
ILRR Metal Foils vs Absolute Fission Chamber and SSTR	Vanadium Encapsulated Foils vs VLP Measurements and SSTR	Vanadium Encapsulated Foils vs LP Meas., High Power SSTR, and IDMS**

$$* \frac{1}{K} = \frac{\text{Saturated Decay Corrected Count Rate}}{\text{Fission Rate}}$$

**Isotope Dilution Mass Spectrometry

2. Fission Yield Measurement

Comparison of radiometric and mass spectrometric measurements with the absolute fission rates will enable measurements of isotope yields determined by these techniques. Actually, instead of yields, calibration factors will be determined to directly relate measured fission product count rates or measured Nd isotope ratios directly to absolute fission rates. These comparisons will be made at various power levels and counting positions, with several laboratories expected to participate to help uncover and resolve any systematic errors.

In the IRT, irradiations of several fissionable materials will be made using the "ILRR" metal foils which have been mass assayed to high accuracy. In the IRT and LP irradiations only radiometric measurements will be made. In the LP irradiation standard dosimetry sets will be irradiated and also counted at several laboratories. The standard dosimeters consist of oxide material and they cannot be mass assayed to the accuracy of the metal foils. A check of the fissionable material present will be made by destructive analysis.

In the HP irradiation, multiple dosimeter sets will be included in order to have counting done at several laboratories and then provide for several samples to be destructively analyzed for the Nd isotopes. The Nd measurements will provide a direct tie with the fuel burnup assay method.

3. Perturbation Measurements

The perturbation measurements at LP will provide data on relationships of in-fuel fission rates to that measured at nearby dosimetry locations out of the fuel.

4. Other Techniques

Comparisons will be made at LP and HP between wire wrap dosimetry (.035" in. dosimeters), alloy dosimeters (e.g. 1% $^{239}\text{PuO}_2$) and standard dosimeters (.050 in. diameter). In addition, comparisons will be made between measured fission rates and those calculated from other dosimeters. In particular, correlations will be made with fission rates calculated using only reactions measured with stainless steel samples [$^{58}\text{Fe}(n,\gamma)$, $^{50}\text{Cr}(n,\gamma)$, $^{54}\text{Fe}(n,p)$, and $^{58}\text{Ni}(n,p)$], i.e., solid wire wrap dosimetry.

Comparisons will also be made between fission rates measured with dosimeters containing FTR fuel (vanadium capsules containing material from either inner or outer FTR fuel pellets) and those determined from summing measured isotope values for fuel constituents. Comparisons of these will also be made with fuel destructive assay and with fuel gamma scans. Results of radiometric measurements and mass spectroscopy at HP will also be compared.

5. Validation for Longer Irradiations

The RCP will provide monitors irradiated for three exposure levels ranging up to 3200 MWd. Dosimetry has been included in other experiments which are to remain in the reactor for several cycles, most noteworthy being the Fuels Open Test Assemblies (FOTA). These assemblies contain wire wrap dosimetry that will be correlated with fuel gamma scanning and destructive assay. These measurements will provide the required extension of technique validation to very high exposures.

A P P E N D I X C

ILRR FLUENCE STANDARDS FROM CFRMF

Fission rates for ^{235}U , ^{238}U , ^{239}Pu and ^{237}Np and reaction rates for non-fission monitors have been measured in CFRMF at flux levels up to the maximum allowable full power value of $\sim 8.6 \times 10^{11} \text{ n/cm}^2\text{-sec.}$ * The fission ratios are in good agreement at all flux levels and with previous measurements in the ILRR program. These measurements confirmed that no spectral perturbations are associated with recent CFRMF modifications and higher reactor power levels. An NBS standard fission chamber with ^{235}U and ^{238}U deposits is available at CFRMF and can be used at the maximum flux level to produce NBS fluence standards referenced to the NBS certified fission chamber. Two commercial fission chambers are also available** and have been intercompared with the NBS chambers in CFRMF and NBS neutron fields. This makes it possible to use these commercial chambers for monitoring the relative fission rate throughout the irradiation of a set of fission and non-fission fluence standards without subjecting the NBS chamber to excessive neutron exposure and thereby minimizing chamber activation and radiation damage.

During a typical reactor run of about 10 hours, the NBS chamber would be used immediately before and after each irradiation to establish the values of absolute flux level. The relative time history of the commercial fission chamber results would be combined with the NBS chamber's absolute results to

* ^{240}Pu , ^{232}Th and ^{233}U fission rates were also measured in the CFRMF neutron field. However, work on the determination of the final mass assays of the deposits is not finished. ILRR fission yields for these fissionable isotopes for some key fission products will be obtained when high quality fission foils are obtained and irradiated in CFRMF.

**Comparative tests of the NBS and commercial fission chambers revealed, however, that the commercial chambers had relatively poor peak-to-valley ratios in the region where the fission fragment pulse height distribution overlaps the alpha + noise pulse height distribution.

establish an exposure value (about 3.1×10^{16} n/cm²) at approximately the 2% (1 σ) accuracy level for the set of fluence standards irradiated at full power.

To use CFRMF routinely in producing these fluence standards, the following must be accomplished:

- 1) A dummy NBS fission chamber to hold the fluence standard foils must be made available at CFRMF.
- 2) Hardware must be fabricated to accurately and reproducibly position the fission chamber or dummy fission chamber relative to the commercial fission chambers.
- 3) Additional fission rate measurements must be made to establish the relative fission rates between the NBS fission chamber and the commercial fission chambers in their final geometrical arrangements.
- 4) HEDL Reactor Dosimetry Center (RDC) foils that are to be used as fluence standards must be selected and set aside for this purpose for all of the participating laboratories: HEDL, EG&G, ANL, ORTEC, CEN/SCK and RIES.
- 5) As appropriate, the foils must be counted prior to each irradiation to establish the existing levels and/or background values of radioactivity.
- 6) Foil set packaging requirements must be established and the actual packaging must be completed prior to each irradiation.
- 7) Prior to the production of final fluence standards for FTR-RCP foil counting, high quality 0.5 in. diameter ILRR foil sets (as well as a few selected vanadium encapsulated dosimeters) should be irradiated and counted by all participating laboratories to validate the entire procedure, including site-to-site shipment practices.

Fluence standards that presently can be produced by irradiation in CFRMF are listed in Table HEDL-C1. Though they will be referenced to the NBS fission chamber, they cannot be directly certified by NBS without 1) a separate mass assay certification of the elemental and isotopic composition of the foil target material and 2) necessary certification irradiations in selected NBS neutron fields, such as the existing NBS Research Reactor's Fission Cavity, ISNF, and the new Nightmare Facility that is under development.

For the FTR-RCP, each of the participating radiometric counting laboratories will receive a set of fluence standards, selected from Table HEDL-C1, that are to be counted prior to and/or after the actual counting of FTR-RCP foils. This will certify the correctness of the current calibrations of the individual laboratory's radiometric counting and data analysis systems. This same basic procedure will also be required for the stable product HAFMs which will be analyzed by RIES.

The second column in Table HEDL-C1 gives a set of absolute values of measured reaction rates with associated uncertainties for a referenced power level of about one tenth full power. The third column provides the currently recommended values, with assigned uncertainties, of the measured reaction rates per unit flux for each reaction. If any laboratory reports a measured CFRMF reaction rate value for one or more of these reactions that is not in agreement with the Table HEDL-C1 values, within stated uncertainties, then it may be necessary for that laboratory to repeat the entire calibration certification procedure for the reaction or reactions involved.

Calculated values of reaction rates per unit flux are given in the next to the last column of Table HEDL-C1. These values are based on the use of a previously recommended CFRMF Benchmark Spectrum used for Phase II data testing of the ENDF/B-IV* dosimetry cross section file, as submitted to the CSEWG Special Applications Files (SAFE) Subcommittee.

*The ENDF/B-V Dosimetry File is expected to be available by March 1979, for final data analysis for the FTR/RCP.

The last column of Table HEDL-C1 provides measured-to-calculated ratios that should be considered in applying ENDF/B dosimetry cross sections for final FTR-RCP dosimetry data reduction. These CFRMF ratios are compared with those determined for other Benchmark Neutron Fields in Table HEDL-C2. Based on Table HEDL-C2 and other results and considerations, Table HEDL-C3 provides a set of suggested adjustment factors (bias factors) for use with selected ENDF/B-IV threshold type dosimetry cross sections. With the application of the Table HEDL-C3 adjustment factors, the integral consistency of the ENDF/B-IV dosimetry file cross sections for these reactions together with $^{237}\text{Np}(n,f)$, $^{238}\text{U}(n,f)$, $^{58}\text{Ni}(n,p)^{58}\text{Co}$, $^{56}\text{Fe}(n,p)^{56}\text{Mn}$, $^{27}\text{Al}(n,\gamma)^{24}\text{Na}$, $^{63}\text{Cu}(n,2n)^{62}\text{Cu}$, and $^{58}\text{Ni}(n,2n)^{57}\text{Ni}$ (all Category I reactions) is expected to be $\pm 5\%$ or better.

TABLE HEDL-C1

CFRMF MEASURED AND CALCULATED VALUES OF REACTION RATES PER UNIT FLUX*

Reaction	Measured Reaction Rate (Reactions/sec- atom) $\times 10^{15}$	Reaction Rates Per Unit Flux		
		σ_{μ}^i Measured (mb) (a)	$\int_0^{\infty} \sigma_{\mu}^i(E) dE^{(b)}$ Calculated (mb)	Measured/ Calculated
$^{59}\text{Co}(n,\gamma)^{60}\text{Co}$	$7.197 \pm 2.3\%$	$91.6 \pm 3.9\%$	85.3	1.074
$^{58}\text{Fe}(n,\gamma)^{59}\text{Fe}$	$0.481 \pm 1.5\%$	$6.12 \pm 3.6\%$	6.11	1.002
$^{55}\text{Mn}(n,\gamma)^{56}\text{Mn}$	-	-	-	-
$^{63}\text{Cu}(n,\gamma)^{64}\text{Cu}$	$3.56 \pm 4.5\%$	$45.4 \pm 5.7\%$	47.5	0.956
$^{197}\text{Au}(n,\gamma)^{198}\text{Au}$	$33.3 \pm 1.0\%$	$424 \pm 3.3\%$	416	1.019
$^{238}\text{U}(n,\gamma)^{239}\text{U}$	$14.5 \pm 2.4\%$	$223 \pm 4.9\%$ (c)	232	0.961
$^{10}\text{B}(n,\alpha)^7\text{Li}$	$145.8 \pm 1.6\%$	$1814 \pm 3.3\%$ (d)	1694	1.071
$^{45}\text{Sc}(n,\gamma)^{46}\text{Sc}$	$1.845 \pm 1.8\%$	$23.5 \pm 3.8\%$	20.1	1.166
$^{115}\text{In}(n,\gamma)^{116\text{m}}\text{In}$	$22.1 \pm 2.5\%$	$281.5 \pm 3.9\%$	303	0.929

*Necessary documentation and reference material are given by A. Fabry, W. N. McElroy, L. S. Kellogg, E. P. Lippincott, J. A. Grundl, D. M. Gilliam and G. E. Hansen, "Review of Microscopic Integral Cross Section Data in Fundamental Reactor Dosimetry Benchmark Neutron Fields," in Neutron Cross Sections for Reactor Dosimetry, IAEA-208, Vol. I, p. 233; and J. W. Rogers, Y. K. Harker, D. A. Millsap and I. E. Stepan, "Reactor Dosimetry Studies at the CFRMF", *ibid*, Vol. II, p. 117; and in Proceedings of the Second ASTM-Euratom Symposium, NUREG/CP-0004, Volume 3, p. 1237, October 3-7, 1977.

TABLE HEDL-C1 (Cont'd)

Reaction	Measured Reaction Rate (Reactions/sec- atom) $\times 10^{15}$	Reaction Rates Per Unit Flux		
		$\bar{\sigma}_\mu^i$ Measured (mb)(a)	$\int_0^\infty \sigma_\mu^i(E) dE^{(b)}$ Calculated (mb)	Measured/ Calculated
${}^6\text{Li}(n,\alpha){}^3\text{H}$	$76.07 \pm 1.0\%$	$948 \pm 4.1\%$ (d)	988.5	0.959
${}^{235}\text{U}(n,f)(f)$	$122.1 \pm 1.4\%$	$1557 \pm 3.4\%$	1590	0.979
${}^{239}\text{Pu}(n,f)(f)$	$142.3 \pm 1.6\%$	$1783 \pm 3.4\%$	1754	1.016
${}^{237}\text{Np}(n,f)(f)$	$43.5 \pm 1.9\%$	$551 \pm 3.8\%$	547	1.007
${}^{115}\text{In}(n,n'){}^{115\text{m}}\text{In}$	$3.90 \pm 4.4\%$	$51.0 \pm 5.9\%$ (e)	47.5(e)	1.074
${}^{238}\text{U}(n,f)(f)$	$5.96 \pm 1.9\%$	$75.6 \pm 4.0\%$	69.4	1.089
${}^{47}\text{Ti}(n,p){}^{47}\text{Sc}$	$0.328 \pm 4.0\%$	$4.18 \pm 4.8\%$	4.70	0.889
${}^{58}\text{Ni}(n,p){}^{58}\text{Co}$	$1.900 \pm 1.0\%$	$24.0 \pm 3.3\%$	21.9	1.093
${}^{54}\text{Fe}(n,p){}^{54}\text{Mn}$	$1.371 \pm 1.0\%$	$17.5 \pm 3.4\%$	16.5	1.061
$\text{Ti}(n,x){}^{46}\text{Sc}$	$0.205 \pm 2.0\%$	$2.61 \pm 3.8\%$	2.15	1.214
${}^{27}\text{Al}(n,p){}^{27}\text{Mg}$	$0.0687 \pm 2.0\%$	$0.874 \pm 3.8\%$	0.887	0.985
${}^{56}\text{Fe}(n,p){}^{56}\text{Mn}$	-	-	0.238	-
${}^{27}\text{Al}(n,\gamma){}^{24}\text{Na}$	$0.0127 \pm 1.2\%$	$0.161 \pm 3.1\%$	0.162	0.994
${}^{48}\text{Ti}(n,p){}^{48}\text{Sc}$	$0.00541 \pm 2.0\%$	$0.0688 \pm 4.4\%$	0.0385	1.787

HEDL-54

FOOTNOTES

- (a) The absolute flux normalization adopted for the determination of these values involves a flux transfer, using the $^{239}\text{Pu}(n,f)$ reaction and the NBS californium-252 source, along lines outlined by Grundl, et al. The californium source was chosen for this purpose because its strength has been established to $\pm 1.1\%$ (1σ) in the NBS Manganous Sulfate Bath Facility relative to the internationally compared standard Ra-Be photo-neutron source, NBS-1. The absolute flux at the NBS californium-252 facility is derived directly from the source strength and a distance measurement; the uncertainty of the total free-field flux is estimated to be $\pm 1.4\%$ (1σ). The $^{239}\text{Pu}(n,f)$ reaction was used for the transfer because the reaction cross section is among the better known reactions and because of its relatively flat shape in the energy range of interest. The $^{239}\text{Pu}(n,f)$ reaction displays almost a constant average cross section in the benchmark neutron fields: the computed values are 1789, 1781, 1754 and 1735 mb for the X82, X25, CFRMF and $\Sigma\Sigma$ benchmarks, respectively; this represents, in the worst case, a difference of only 3%. Interrelated NBS fissionable deposits of plutonium-239 have been exposed in all these neutron fields in the NBS double absolute ionization fission chamber; therefore, the flux transfer is rather direct and the accuracy of the flux transfer is of the order of $\pm 0.8\%$ or better.
- (b) $\sigma_{\text{u}}^{\text{i}}(E)$: ENDF/B-IV dosimetry file; $\phi(E)$: ENDF/B-III S_N computation, normalized $\int_0^\infty \phi(E) dE = 1$.
- (c) Corrected for spectral shielding effect (21%) in reactor constituents.
- (d) New measurements.
- (e) Based on a ^{115}mIn γ -branching ratio of 45.9%.
- (f) The fission fluence foils will be used primarily to establish K-factors. The NBS and commercial fission chambers will be used to establish the fission rates.

TABLE HEDL-C2

RATIO OF MEASURED^(a) TO COMPUTED^(b) INTEGRAL CROSS SECTIONS
IN DOSIMETRY BENCHMARK NEUTRON FIELDS

Reaction	Neutron Field				
	χ_{82}	χ_{25}	$\Sigma\Sigma$	CFRMF	BIG-10
$^{59}\text{Co}(n,\gamma)^{60}\text{Co}$	-	-	-	1.074	1.028
$^{58}\text{Fe}(n,\gamma)^{59}\text{Fe}$	-	-	-	1.002	1.382
$^{63}\text{Cu}(n,\gamma)^{64}\text{Cu}$	-	0.846	0.932	0.956	0.932
$^{197}\text{Au}(n,\gamma)^{198}\text{Au}$	1.000	0.987	1.076	1.019	1.041
$^{238}\text{U}(n,\gamma)^{239}\text{U}$	-	-	0.95 ^(c)	0.961	1.002
$^{10}\text{B}(n,\alpha)^7\text{Li}$	-	-	-	1.071	1.141
$^{45}\text{Sc}(n,\gamma)^{46}\text{Sc}$	-	-	-	1.166	1.125
$^{115}\text{In}(n,\gamma)^{116\text{m}}\text{In}$	0.962	0.990	0.842	0.929	-
$^6\text{Li}(n,\alpha)^3\text{H}$	-	-	-	0.959	1.001
$^{235}\text{U}(n,f)$	0.969	0.969	0.991	0.979	0.995
$^{239}\text{Pu}(n,f)$	1.008	1.017	1.017	1.016	1.017
$^{237}\text{Np}(n,f)$	0.986	0.994	0.966	1.007	0.985
$^{115}\text{In}(n,n')^{115\text{m}}\text{In}$	1.036	1.037	1.014	1.074	1.092
$^{238}\text{U}(n,f)$	1.015	1.031	1.044	1.089	1.107
$^{47}\text{Ti}(n,p)^{47}\text{Sc}$	0.793	0.888	-	0.889	0.945
$^{58}\text{Ni}(n,p)^{58}\text{Co}$	1.026	1.068	1.139	1.093	1.159
$^{54}\text{Fe}(n,p)^{54}\text{Mn}$	0.949	1.026	-	1.061	1.120
$^{46}\text{Sc}(n,x)^{47}\text{Ti}$	1.102	1.181	-	1.214	1.290
$^{27}\text{Al}(n,p)^{27}\text{Mg}$	0.992	0.937	1.131	0.985	-
$^{56}\text{Fe}(n,p)^{56}\text{Mn}$	0.983	0.983	1.130	-	-
$^{27}\text{Al}(n,\alpha)^{24}\text{Na}$	0.950	1.017	1.007	0.994	1.078
$^{48}\text{Ti}(n,p)^{48}\text{Sc}$	1.585	1.734	-	1.787	1.993

(a) Normalized by $^{239}\text{Pu}(n,f)$; transfer from californium.

(b) $\int_0^\infty \sigma_i(E) \phi(E) dE$; $\int_0^\infty \sigma_i^j(E)$: ENDF/B-IV file; $\phi(E)$: as recommended, 1975, normalized $\int_0^\infty \phi(E) dE = 1$

(c) Applying the spectral shielding correction computed for CFRMF.

TABLE HEDL-C3

ENDF/B-IV EVALUATED THRESHOLD REACTION CROSS SECTION BIAS FACTORS

<u>Reaction</u>	<u>Bias Factor</u>
$^{115}\text{In}(n,n')^{115\text{m}}\text{In}$	1.017 ± 0.025
$^{232}\text{Th}(n,f)$	1.15
$^{47}\text{Ti}(n,p)^{47}\text{Sc}$	0.825 ± 0.03
$^{32}\text{S}(n,p)^{32}\text{P}$	0.987
$^{54}\text{Fe}(n,p)^{54}\text{Mn}$	0.967 ± 0.018
$\text{Ti}(n,x)^{46}\text{Sc}$	1.128 ± 0.026
$^{59}\text{Co}(n,\alpha)^{56}\text{Mn}$	0.973
$^{127}\text{I}(n,2n)^{126}\text{I}$	0.778
$^{55}\text{Mn}(n,2n)^{126}\text{I}$	0.803
$^{90}\text{Zr}(n,2n)^{89}\text{Zr}$	1.715

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HEDL-58

B. REACTION RATE MEASUREMENTS IN THE NBS ^{252}Cf
FISSION NEUTRON IRRADIATION FACILITY
L. S. Kellogg, E. P. Lippincott and J. M. Ruggles

Objective

Obtain absolute values for infinitely dilute reaction rates for selected fast-neutron dosimetry materials irradiated in the NBS ^{252}Cf neutron irradiation facility with sufficient accuracy to permit testing and evaluation of energy-dependent cross-section sets from ENDF/B.

Summary

Both Ni and ^{235}U foils were irradiated in the NBS ^{252}Cf fission neutron irradiation facility and the infinitely dilute reaction rates were measured for the ^{58}Co , ^{140}Ba , ^{103}Ru and ^{95}Zr product activities using direct gamma-ray spectrometry. Long counts were required in all cases due to the relatively low activities. Cumulative fission yield values for ^{140}Ba , ^{103}Ru and ^{95}Zr and the integral cross-section value for the $^{58}\text{Ni}(n,p)^{58}\text{Co}$ reaction were obtained by combining the measured reaction rates with the NBS certified free field neutron fluences and free field fission rate.

ACCOMPLISHMENTS AND STATUS

Two foil packages consisting of the ILRR ^{235}U (lot 264 CM) sandwiched between two Ni foils (Rx lot) were irradiated for 10.69 days in the NBS ^{252}Cf fission neutron irradiation facility. The foil packages were placed equidistant on opposite sides of the ^{252}Cf source. The separation distances between the corresponding foils were accurately measured by NBS personnel using a Boyce optical coordinate measuring machine. The separation distances were reported as being: near Ni foils 10.55 ± 0.06 cm, ^{235}U foils 10.74 ± 0.06 cm, and far Ni foils 10.93 ± 0.06 cm. The foil physical parameters are listed in Table HEDL B-1.

The irradiation was terminated 0930 EDT, July 5, 1977. Foils were shipped to HEDL and counting was started by July 22, 1977. Two to four measurements in different configurations were made on each foil. Observed activities were converted to reaction rates assuming a steady state irradiation, whereas a 0.77% decrease in flux occurred during the irradiation due to the ^{252}Cf decay. Calculations by NBS indicate a correction of 0.04% would be observed for a half life of 12.79 d(140 Ba). The measured reaction rates and correction factors are tabulated in Table HEDL B-2.

Symmetry in positioning the foil packages on opposite sides of the ^{252}Cf source was confirmed within the accuracy of the measurements by ratioing the corresponding foil reaction rates (Table HEDL B-3). The individual values were averaged and corrected to infinitely dilute samples. The corrected values were then combined with the NBS certified free field fluences and fission rates to obtain the $^{58}\text{Ni}(n,p)^{58}\text{Co}$ integral cross-section and ^{235}U fission rates. These data with the NBS reported corrections and flux-fluence data are tabulated in Tables HEDL B-4 and B-5, respectively.

EXPECTED ACHIEVEMENTS

Additional counting data are being taken in an attempt to determine the ^{137}Cs fission yield.

TABLE HEDL B-1
NBS ^{252}Cf IRRADIATION FOIL PARAMETERS

<u>Set No.</u>	<u>Foil ID^a</u>	<u>Nominal Thickness^b (mil)</u>	<u>Foil Mass (mg)</u>	<u>Element (Wt %)</u>	<u>Target Isotope (Wt %)</u>	<u>Atomic Weight</u>
2	Ni-JB	10	295.266	99.62	67.398	57.935
	^{235}U -2	20	1207.835	99.44	99.887	235.044
	Ni-LL	10	295.319			
5	Ni-JF	10	295.295			
	^{235}U -5	30	1214.465			
	Ni-L0	10	294.182			

^a Each foil set is listed as packaged. The first foil in each listing was positioned nearest the source.

^b Nominal foil dimensions and composition are the same for like foils.

TABLE HEDL B-2
 MEASURED REACTION RATES FOR FOILS IRRADIATED IN THE NBS
²⁵²Cf FISSION NEUTRON IRRADIATION FACILITY

Foil	Reaction	t (Days)	No. of Spectra	E (KeV)	Gamma-Ray SS Factor	Ext Abs Factor	Measured DPS/mg @ TOR	Measured Reaction Rate (10 ⁻¹⁹ atoms/atom-s)	Error Estimate (%)
JB	⁵⁸ Ni(n,p) ⁵⁸ Co	29-64	2	810	0.99923	None	1.292	18.56	1.95
JF		23-112	2				1.305	18.75	2.0
LL		22-102	2				1.202	17.22	2.0
L0		28-109	2				1.168	16.79	2.1
#2	²³⁵ U(n,f) ¹⁴⁰ Ba	17-375	3	1596	0.9729	0.9694	1.145	10.77	1.8
	¹⁰³ Ru		3	497	0.9082	0.9413	0.2720	6.195	2.6
	⁹⁵ Zr		4	724	0.9446	0.9535	0.3361	12.01	2.3
				765	0.9481	0.9549	0.3281	11.73	2.1
#5	²³⁵ U(n,f) ¹⁴⁰ Ba	20-368	4				1.169	11.00	2.0
	¹⁰³ Ru		4				0.2768	6.304	2.5
	⁹⁵ Zr		4				0.3326	11.95	2.5
			4				0.3289	11.76	2.2

TABLE HEDL B-3

COMPARISON OF CORRESPONDING FOILS TO CONFIRM POSITION
 SYMMETRY OF PACKAGES 2 AND 5 IN THE NBS ^{252}Cf FACILITY

<u>Foil Type</u>	<u>Foils Ratioed</u>	<u>Isotope Analyzed</u>	<u>Reaction Rate Ratio</u>	<u>Uncertainty of Ratio (%)</u>
Ni	JB/JF	^{58}Co	0.990	1.76
	LL/LO		1.029	1.67
^{235}U	2/5	^{140}Ba	0.980	1.32
		^{103}Ru	0.983	1.61
		^{95}Zr (724)	1.011	1.63
		^{95}Zr (765)	0.998	2.18

TABLE HEDL B-4

CALCULATED INTEGRAL MICROSCOPIC CROSS SECTION FOR THE
 $^{58}\text{Ni}(n,p)^{58}\text{Co}$ REACTION IN THE NBS ^{252}Cf FACILITY

Ni Foil Position	Perturbed RR/ Free Field RR	Corrected Reaction Rate* (10^{-19} atoms/atom-s)	Average Free Field*,** Neutron Fluence (10^7 neutrons/cm ² -s)	Integral Cross-Section	
				Individual*	Average*
Inside	1.018	18.32 (1.9)	1.506 (1.6)	121.7 (2.5)	121.3 (2.6)
Outside	1.002	16.97 (2.0)	1.404 (1.6)	120.9 (2.6)	

*Assigned total error (1σ) as percentage is given in parentheses.

**Value calculated from NBS certified specific neutron fluence values and total irradiation time.

TABLE HEDL B-5

CALCULATED FISSION YIELDS FROM ^{235}U REACTION RATES
 IN THE NBS ^{252}Cf FACILITY

Fission Product	Perturbed RR/ Free Field RR	Corrected Reaction Rate*,** (10^{-19} atoms/atom-s)	Average Specific Free Field Fission Rate (10^{-19} fissions/atom-s)***	Cumulative ^{235}U Fission Yield*
^{140}Ba	1.056	10.31 (1.9)	173.5(1.6)	5.94 (2.5)
^{103}Ru		5.918 (2.0)		3.41 (2.6)
^{95}Zr		11.23 (1.9)		6.47 (2.5)

*Total error (1σ) as percentage is given in parentheses.

**Uncertainty in the neutron scattering correction is not included.

***No correction was made for ^{252}Cf decay to compensate for square wave reaction rate calculation.

C. REACTION RATE MEASUREMENTS IN THE NBS ^{235}U
CAVITY FISSION NEUTRON FACILITY
L. S. Kellogg and E. P. Lippincott

OBJECTIVE

Obtain relative reaction rates for selected fast-neutron dosimetry materials, irradiated in the NBS ^{235}U cavity fission facility, with sufficient accuracy to permit testing and evaluation of energy-dependent cross-section sets from ENDF/B.

SUMMARY

Relative reaction rates from a sample SRM-1158 high-nickel steel were measured in the NBS ^{235}U cavity fission facility for the $^{54}\text{Fe}(\text{n},\text{p})^{54}\text{Mn}$ and $^{58}\text{Ni}(\text{n},\text{p})^{58}\text{Co}$ reactions using direct gamma-ray counting techniques. A value of $1.360 \pm .007$ was obtained from these data.

ACCOMPLISHMENTS AND STATUS

The NBS SRM-1158 foil was irradiated in the NBS ^{235}U cavity fission neutron facility for a duration of 86,220 seconds. The irradiation was terminated at 11:02 EDT on June 28, 1977. Physical parameters for the foil material can be seen in Table HEDL C-1.

Two x-ray spectra were taken in different configurations approximately eighty days apart. Time history corrections were made assuming a uniform square wave irradiation. Calculated reaction rates and correction factors, other than those listed previously, are found in Table HEDL C-2.

The measured reaction rate (i.e cross section) ratio for the NBS ^{235}U cavity fission neutron facility was found to be:

$$^{58}\text{Ni}(\text{n},\text{p})^{58}\text{Co} / ^{54}\text{Fe}(\text{n},\text{p})^{54}\text{Mn} = 1.360 \pm .007$$

This measured value is 1.04% greater than that previously reported by Fleming and Spiegel but is well within the accuracy of the measurements.

EXPECTED ACCOMPLISHMENTS

No further work is to be done.

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1. R. Fleming and V. Spiegel, "Measurement of Threshold Reaction Cross Section Ratios in Fission Neutron Fields," Proceedings Second ASTM-Euratom Symposium on Reactor Dosimetry: Dosimetry Methods for Fuels, Cladding and Structural Materials, NUREG/CP-0004 (1977), Vol. 2, p. 953.

TABLE HEDL C-1

PARAMETERS OF NBS SRM-1158 HIGH-NICKEL-STEEL FOIL MATERIAL

Foil ID	Foil Mass (mg)	Nominal Thickness (mil)	Target Element	Elemental Composition (Wt %) (1)	Target Isotope	Atomic Wt (g)	Isotopic Abundance (Wt %)
SS #2	745.422	30	Fe	63.15	^{54}Fe	53.9396	5.621
			Ni	36.03	^{58}Ni	57.9353	67.40

(1) Fe composition derived by extrapolation of elemental analysis to 100%

TABLE HEDL C-2

MEASURED REACTION RATES FOR THE NBS ^{235}U CAVITY FISSION NEUTRON FACILITY

Reaction	Δt (Days)	E (Ke)	γ -ray SS Factor	Measured DPS/mg TOR ^a	Measured Reaction Rate (10^{-15} atoms/atom-sec)	Estimated Error (%) ^b
$^{54}\text{Fe}(n,p)^{54}\text{Mn}$	28-121	834	0.9809	1.981	1.942	2.4
$^{58}\text{Ni}(n,p)^{58}\text{Co}$	28-121	810	0.9806	132.2	1.428	2.1

^aCorrections were made for γ self-shielding and decay. No corrections were made for neutron gradients at the sample position, neutron self-shielding, or scatter contributions.

^bThe absolute error estimate includes errors related to items in note "a" as well as weight and half-life.

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HEDL-68

D. ANALYSIS OF SSTR FISSION RATE MEASUREMENTS IN BIG-10

Raymond Gold and James H. Roberts*

OBJECTIVE

The objective of this work is to provide direct comparisons of absolute fission rates as measured by Solid State Track Recorder (SSTR) and fission chamber techniques in order to resolve discrepancies and obtain required measurement accuracies.

SUMMARY

Scanning of SSTR specimens from BIG-10 irradiations has been completed. Comparison with absolute NBS fission chamber results reveals agreement within limits of experimental error for absolute ^{235}U and ^{238}U fission rates. Absolute SSTR fission rates observed for ^{239}Pu and ^{237}Np are systematically higher than NBS fission chamber results by 5 and 7%, respectively.

ACCOMPLISHMENTS AND STATUS

In a comparison between absolute NBS fission chamber and SSTR measurements, a systematic difference exceeding experimental error was reported.⁽¹⁾ A number of irradiations were conducted at the BIG-10 facility some time ago⁽²⁾ to specifically address this issue. Absolute NBS fission chamber measurements for ^{235}U , ^{238}U , ^{237}Np and ^{239}Pu from these BIG-10 irradiations have already been reported.⁽³⁾ Although SSTR irradiations at BIG-10 were conducted in collaboration with Argonne National Laboratory (ANL), track scanning could not be pursued by the ANL group, due to redirection of programmatic efforts.

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Track scanning has now been completed at HEDL for all SSTR specimens from these BIG-10 irradiations. A summary of these scanning results can be found in Table HEDL D-1. Standard manual track scanning techniques were used for these observations.⁽⁴⁾ All tracks on each SSTR specimen were scanned by two independent observers. As can be seen in Table HEDL D-1, Observer 1 was systematically higher than Observer 2. An average of the Observer 1/Observer 2 ratio for all sixteen BIG-10 SSTR yields 1.0058 ± 0.0057 .

This systematic bias between Observer 1 and Observer 2 is considerably higher than anticipated, as based on previous experience and efforts in precision track scanning. Two factors related specifically to the SSTR dosimeters used at BIG-10 have obviously contributed to this systematic observer bias. The mica track recorders for the BIG-10 irradiations were not preselected. In all high precision SSTR applications with mica, pre-selection is mandatory to ensure good clarity and surfaces free from imperfections that could confuse the track scanning task. A difference of only fifty to one hundred or so tracks constitutes the extent of systematic observer bias in these BIG-10 SSTR mica specimens. The second factor relates to the nonuniformity of fission deposits used for these SSTR measurements. Nonuniformity can produce localized regions of very high track density which contain track pile-up events that are especially difficult to resolve. The electrodeposited BIG-10 sources invariably exhibited higher track density out at the circumference of the deposit. While track pile-up could be a contributing factor to the systematic observer bias, it is felt that this bias would have been rendered negligible had the customary mica pre-selection screening process been employed.

Table HEDL D-2 summarizes the absolute SSTR fission rate data obtained from the BIG-10 irradiations. Corrections were made for isotopic fission rate contributions in these sources based on the known isotopic constituents of the batch material used⁽⁵⁾, together with known ^{234}U spikes used for mass assay determinations of the ^{235}U and ^{238}U fission deposits. A final correction was made to determine the unperturbed fission rate in the BIG-10 environment.⁽⁶⁾

TABLE HEDL D-1

OBSERVED TRACKS IN BIG-10 IRRADIATED SSTR

HEDL-71	<u>BIG-10 Irradiation</u>	<u>Isotope</u>	<u>Observer 1</u>	<u>Observer 2</u>	<u>Observer 1/Observer 2</u>
	SSTR 4	^{235}U	13,757	13,673	1.006
	SSTR 4	^{234}U	10,949	10,821	1.012
	SSTR 4	^{238}U	9,238	9,133	1.011
	SSTR 4	^{239}Pu	8,433	8,463	0.9965
	SSTR 5	^{239}Pu	14,435	14,385	1.003
	SSTR 5	^{234}U	9,500	9,499	1.000
	SSTR 5	^{237}Np	14,456	14,427	1.002
	SSTR 5	^{235}U	21,150	21,058	1.004
	SSTR 6	^{238}U	9,522	9,545	0.9980
	SSTR 6	^{239}Pu	14,898	14,705	1.013
	SSTR 6	^{235}U	12,640	12,633	1.001
	SSTR 7	^{235}U	18,062	17,958	1.006
	SSTR 7	^{238}U	9,906	9,865	1.004
	SSTR 7	^{237}Np	13,938	13,770	1.012
	SSTR 8	^{237}Np	13,949	13,819	1.009
	SSTR 8	^{238}U	11,397	11,226	1.015

TABLE HEDL D-2
SSTR FISSION RATE DATA FOR BIG-10 IRRADIATIONS

BIG-10 Irradiation	Isotope	Observed Tracks ^a	Absolute Fission Rates ^b		
			Uncorrected	Corrected for Isotopic Contributions	Corrected to Unperturbed BIG-10 Environment ^c
SSTR 4	²³⁵ U	13,715	0.1641	0.1642	0.1657
SSTR 4	²³⁴ U	10,885	0.0555		
SSTR 4	²³⁸ U	9,186	0.00586	0.00584	0.00600
SSTR 4	²³⁹ Pu	8,448	0.2024	0.2024	0.2042
SSTR 5	²³⁹ Pu	14,410	0.2199	0.2199	0.2219
SSTR 5	²³⁴ U	9,500	0.0583		
SSTR 5	²³⁷ Np	14,442	0.0590	0.0590	0.0606
SSTR 5	²³⁵ U	21,104	0.1762	0.1764	0.1780
SSTR 6	²³⁸ U	9,534	0.00678	0.00678	0.00695
SSTR 6	²³⁹ Pu	14,802	0.2269	0.2269	0.2289
SSTR 6	²³⁵ U	12,637	0.1782	0.1783	0.1800
SSTR 7	²³⁵ U	18,010	0.1809	0.1811	0.1827
SSTR 7	²³⁸ U	9,886	0.00636	0.00635	0.00651
SSTR 7	²³⁷ Np	13,854	0.0601	0.0601	0.0612
SSTR 8	²³⁷ Np	13,884	0.0588	0.0588	0.0599
SSTR 8	²³⁸ U	11,312	0.00650	0.00650	0.00666

^a Average of Observer 1 and Observer 2 scanning results.

^b Fissions per mg x 10⁻⁹; an optical efficiency, $\eta = 0.948 \pm 0.0053$ has been used (see Ref. 4).

^c See Ref. 6.

Comparison of the SSTR absolute fission rates with the NBS fission chamber results^(3,6) for BIG-10 are summarized in Table HEDL D-3. Isotopic fission rate comparisons for both ^{235}U and ^{238}U are generally within the limit of experimental error, whereas similar comparisons for ^{239}Pu and ^{237}Np fall outside experimental error. Average values of the NBS chamber to SSTR results were obtained from the data in Table HEDL D-3. These average ratios are 0.984, 0.992, 0.948 and 0.931 for ^{235}U , ^{238}U , ^{239}Pu and ^{237}Np , respectively.

Consequently, systematic discrepancies exist for the ^{239}Pu and ^{237}Np fission rates in BIG-10, with the SSTR results again being higher than the NBS fission chamber data. In these comparisons the correction factor for flux depression calculated for the NBS fission chamber⁽⁶⁾ was applied to the SSTR data. This could well be an oversimplification, since the SSTR exposures contained considerably more platinum than did the BIG-10 fission chamber irradiations.

SSTR irradiations 5, 6, 7 and 8 in BIG-10 were virtually equivalent, according to fission chamber data.⁽³⁾ NBS fission deposits used in the NBS chamber for all these SSTR exposures were indeed the same. This would suggest that variations in SSTR data arising for irradiations 5, 6, 7 and 8 may be caused by fission deposit mass uncertainties. In particular, natural uranium contamination, if any, in the ^{239}Pu and the ^{237}Np ANL fission deposits would not have been revealed through the mass assay techniques used to characterize these deposits.^(1,3,6)

EXPECTED ACHIEVEMENTS

In order to evaluate the above speculations, it is recommended that:

- Perturbation correction factors be obtained specifically for the SSTR irradiations in BIG-10.

TABLE HEDL D-3
ISOTOPIC FISSION RATES FOR BIG-10

<u>Uncertainties (%)</u>			<u>Fission Rates^c</u>		
<u>Irradiation</u>	<u>Scanning^a</u>	<u>Mass^b</u>	<u>Total</u>	<u>SSTR</u>	<u>NBS Chamber</u>
²³⁵ U					
SSTR 4	1.06	0.62	1.23	0.1657 (20)	0.1620 (30)
SSTR 5	0.91	0.62	1.10	0.1780 (20)	0.1780 (30)
SSTR 6	1.05	0.62	1.22	0.1800 (22)	0.1780 (30)
SSTR 7	0.98	0.62	1.16	0.1827 (21)	0.1780 (30)
²³⁸ U					
SSTR 4	1.31	0.71	1.49	0.00600 (9)	0.00603 (16)
SSTR 6	1.17	0.71	1.37	0.00695 (10)	0.00662 (17)
SSTR 7	1.17	0.71	1.37	0.00651 (9)	0.00662 (17)
SSTR 8	1.33	0.71	1.51	0.00666 (10)	0.00662 (17)
²³⁹ Pu					
SSTR 4	1.24	0.66	1.40	0.2042 (29)	0.1941 (50)
SSTR 5	1.02	0.66	1.22	0.2219 (27)	0.2132 (48)
SSTR 6	1.19	0.66	1.36	0.2289 (31)	0.2132 (48)
²³⁷ Np					
SSTR 5	1.01	1.30	1.65	0.0606 (10)	0.0564 (16)
SSTR 7	1.18	1.30	1.76	0.0612 (11)	0.0564 (16)
SSTR 8	1.02	1.30	1.65	0.0599 (10)	0.0564 (16)

^a Scanning uncertainties include counting statistics, error in optical efficiency for mica (0.56%) and systematic observer bias, assumed to be the standard deviation of the two observations for each SSTR.

^b Mass uncertainties were assumed to be identical with those cited in Ref. 1.

^c Fissions per mg x 10⁻⁹; the number in parentheses is the 1 σ error in the least significant digits.

- ANL fission deposits, i.e., those used in BIG-10, be shipped to HEDL for analysis with SSTR techniques.

The advantages of characterizing fission deposits with SSTR techniques will be reported in sequel.

Additional experiments and intercomparisons between the NBS chamber and SSTR methods of absolute fission rate measurement are clearly warranted and have already been collaboratively planned by NBS and HEDL personnel. These tests are in progress and findings will be disclosed in future ILRR reports when results become available.

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E. STATUS OF IN-SITU REACTOR GAMMA-RAY SPECTROMETRY

Raymond Gold and Bruce J. Kaiser

Objective

Improved characterization of gamma-ray heating and energy deposition in reactor environments.

Summary

The current status of Compton Recoil Gamma-Ray Spectrometry is summarized. Special attention is given to advances in experimental technique for measurements in reactor environments. In-situ gamma-ray spectrometry efforts at HEDL in U.S. Light Water Reactor (LWR) and Breeder Reactor (BR) programs are described.

ACCOMPLISHMENTS AND STATUS

1.0 INTRODUCTION

In reactor environments, the radiation field is comprised of two principal components, neutrons and gamma-rays. The history of reactor development reveals an initial concern for the neutron component of the radiation field that was virtually overriding. Such initial emphasis is completely understandable, what with fission reactors being neutron chain multiplying assemblies. Only after considerable time were significant effects due to the gamma-ray component in reactor design, shielding, and safety actually recognized. Recognition of these effects has provided the impetus for improved characterization of reactor gamma-ray deposition and spectra. These general motivations have been reviewed in depth^(1,2) and were also summarized in the 11th ILRR progress report.⁽³⁾

In contrast to these general motivations, more specialized needs often arise for reactor gamma-ray spectral data. For example, specific needs exist for radiation damage specialists using high power radiation test facilities to test, develop, and improve reactor fuels and materials. For these specialists, the temperature of a given radiation damage experiment is a crucial variable. Such radiation damage experiments can neither be properly designed nor analyzed without an adequate knowledge of the irradiation temperature history. The temperature history can ultimately depend, in turn, upon the reactor gamma-ray component, since the source of reactor heat generation can principally arise through gamma-ray interactions.

The most fundamental quantity underlying the description of the reactor gamma-ray component is the absolute gamma-ray energy spectrum. Radiation effects arising from the gamma-component are induced by the interaction of the absolute gamma-ray energy spectrum in the reactor environment. Consequently, accurate definition of this absolute spectrum is the goal of both theory and experiment. Urgent and pragmatic need now exists for reactor benchmark gamma-ray spectrometry data. Requirements have therefore been

defined for in-core gamma-ray spectrometry in U.S. Light Water Reactor (LWR) and Breeder Reactor (BR) programs.

In reactor environments, gamma-ray spectra are continuous, so the absolute magnitude as well as the general shape of the gamma continuum is of paramount importance. Conventional methods of gamma-ray detection are thus not suitable for in-core gamma-ray spectrometry. To meet such specific needs, a method of continuous gamma-ray spectrometry, namely Compton Recoil Gamma-Ray Spectrometry, was developed for in-situ observations in reactor environments.⁽⁴⁻⁶⁾ In addition to applications in reactor science,⁽⁷⁻¹⁰⁾ it has been used to measure gamma continua which arise in such applied disciplines as shielding, dosimetry,⁽¹¹⁾ health physics,⁽¹²⁾ radiobiology and environmental science.^(13,14) A brief summary of earlier efforts can be found in these reviews.^(1,2)

The current status of the method is described in the next two sections, dealing with gamma spectrometry probe design and response characteristics, respectively. In the remaining two sections, emphasis is given to gamma spectrometry work in U.S. LWR and BR programs. Gamma spectrometry in BR environments is outlined by focussing on startup plans for the Fast Test Reactor (FTR).⁽¹⁵⁾ In the last section, gamma spectrometry results are presented for a LWR pressure vessel mockup in the Poolside Critical Assembly (PCA) at Oak Ridge National Laboratory (ORNL).⁽¹⁶⁾

2.0 SPECTROMETER DESIGN

The basic elements that comprise the current gamma probe are displayed in Figure HEDL E-1. Different lithium drifted silicon solid state detectors (Si(Li)) can be housed in the same vacuum chamber with little or no change in mounting hardware. Three configurations have been used to date: a 1 cm³ planar detector, a 2 cm³ planar detector, and a 12 cm³ coaxial detector.

2.1 Summary Status of the Current Spectrometer Design

2.1.1 The inclusion of a miniaturized vac-ion pump (0.1 l/s), which is capable of maintaining a probe pressure of 10⁻⁶ Torr at ~43°C ambient, offers three improvements:

- Increased thermal insulation due to a reduction in vacuum chamber pressure by a factor of 10³ or more
- Reduced overall probe size
- Extended probe flexibility.

2.1.2 A smaller more efficient Peltier junction thermoelectric cooler (TEC) is used, which increases cooling capacity, reduces the heat load, and is capable of cooling the sensor and FET first input stage to 50°C below the aluminum dissipator temperature.

2.1.3 A slotted aluminum radiator, which acts as a heat dissipator, can maintain the hot side of the TEC at 20°C in small reactor access ports of up to 40°C ambient. If necessary this slotted aluminum radiator can be cooled by forced gas flow.

2.1.4 The net result of 2.1.1 through 2.1.3 acting in consort is to improve probe temperature stability, thus eliminating temperature dependent effects for most in-core gamma-ray spectrometry.

FIGURE HEDL E-1. Detector Probe for Continuous Compton Recoil Gamma-Ray Spectrometry.

2.1.5 The preamplifier, which is a slightly modified and extensively reconfigured version of the ORTEC 142A design, possesses near-ideal characteristics for Compton Recoil Gamma-Ray Spectrometry with a rise time adjusted to ~ 50 nsec and a combined preamp plus cooled sensor noise level of ~ 50 μ V (RMS). As a consequence, electron energy resolution of ~ 5 keV (FWHM) at 0.661 meV (^{137}Cs photo-peak energy) has been achieved.

2.1.6 Overall pulse processing instrumentation has significantly evolved over earlier electronic circuitry (see Figure HEDL E-2.)

2.1.7 The net result of 2.1.5 and 2.1.6 acting in consort produces:

- Excellent rise time resolution (see Figure HEDL E-3)

- Rise time observations that are essentially independent of pulse-height (energy), aside from noise broadening
- The capability of accurately measuring Si(Li) detector sensitive volume (see Section III below)
- Accurate characterization of overall energy-angular response.

HEDL 7908-095.1

FIGURE HEDL E-2. Instrumentation Block Diagram for Compton Recoil Gamma-Ray Spectrometry.

FIGURE HEDL E-3. Typical Rise Time Spectra Used for Defining the Sensitive Volume of a 2 cc Planar Si(Li) Detector. The electron energy window chosen for the collimated beam of ^{54}Mn gamma rays was 0.32 to 0.64 MeV. (a), (b), and (c) are the 0.00", 0.05", and 0.10" side traverse spectra, respectively. The 0.00" position corresponds to the edge of the detector. The total of integrated counts for each step was 20,000.

3.0 RESPONSE CHARACTERISTICS

For absolute measurements, the extent of the sensitive region of the Si(Li) detectors used in Compton Recoil Gamma-Ray Spectrometry is of critical importance.⁽⁹⁾ Current spectrometry system capabilities now permit quantitative measurement of this sensitive volume by resolving differences in pulse rise time between the sensitive and semi-sensitive regions. To date the planar 1 and 2 cc Si(Li) detectors have been investigated by stepping a 0.060 in. diameter beam of ^{54}Mn gamma-rays across their surfaces.

Typical response rise time spectra (RTS) for a 2 cc detector are displayed in Figure HEDL E-3. Note the shift from predominantly slow to fast rise time events as one steps away from the detector face down its side. The (c) type spectrum is observed as the traverse continues until the last two .050 in. steps, where the (b) then (a) type spectra are repeated. Traverses across the face and back of the detector follow the same general pattern. Thus, the two important regions of the detector are dimensionally defined: the semi-sensitive outer shell with its slower rise time pulses (due to trapping and E-field reduction resulting from under or over compensation of lattice impurities) and the sensitive volume with its faster rising pulses.

Of perhaps greater significance is the sensitivity of these RTS observations that enables measurement of finite-size retention probability of recoil electrons in these Si(Li) detectors. The retention probability, P , of electrons in the finite sensitive volume of a detector depends on a number of variables. In general, P depends upon the gamma-ray energy ϵ_0 , the recoil electron energy, E , and the angle of incidence of the gamma-ray θ . Hence the retention probability is generally denoted by $P(\epsilon_0, E, \theta)$.

RTS measurements of such sensitivity were not possible before, so certain simplifying assumptions were made in the unfolding analysis of observed electron spectra.⁽⁴⁻⁶⁾ In particular, P was assumed independent of both ϵ_0 and θ . With the present capabilities of observing RTS, one no longer

need rely on such assumptions and more accurate data analysis can thereby be performed. On the other hand, these very capabilities permit investigation of the validity of these earlier assumptions.

Response measurements are made with a point source of gamma-rays, rotated about the detector from the face (0° incidence) to the back (180° incidence) in 45° steps. Rise time and energy data are accumulated at each position in a two-parameter, 64×64 channel mode. Figure HEDL E-3 displays typical response RTS. The retention probability, $P(\epsilon_0, E, \theta)$ is obtained by first fitting the faster rise time peak with a Gaussian distribution. Then the area under this Gaussian distribution is divided by the total number of counts in the RTS to obtain $P(\epsilon_0, E, \theta)$. Calculations are performed for all electron energy bins, E_i , at each angle of incidence.

Table HEDL E-1 presents the results of such response determinations of $P(\epsilon_0, E, \theta)$ for the 1 cc planar Si(Li) detector using a point source of ^{54}Mn ($\epsilon_0 = 0.835$ MeV). This table reveals that the retention probability is a very slowly varying function of θ . Similar response measurements with point sources of different gamma-ray energy also reveal that P is a slowly varying function of ϵ_0 . Such measurements show, in fact, that P is independent of ϵ_0 within the present limits of experimental uncertainty.

Consequently, these preliminary data demonstrate that to a good approximation $P(\epsilon_0, E, \theta)$ reduces to $P(E)$, i.e., the retention probability depends only upon recoil electron energy. This result supports the assumptions invoked in earlier work which were based upon a Markovian formulation of the retention probability. In such a Markovian approximation, the probability of electron escape from the finite sensitive volume of the detector depends solely upon the recoil electron energy E and is independent of the mode (ϵ_0, θ) which produced the recoil electron.

TABLE HEDL E-1

FINITE SIZE RETENTION PROBABILITY
 $P(\epsilon_0, E, \theta)$ for $\epsilon_0 = 0.835$ MeV (^{54}Mn)

Energy (MeV)	Angle of Incidence					Average Over All Angles
	<u>0°</u>	<u>45°</u>	<u>90°</u>	<u>135°</u>	<u>165°</u>	
0.10	0.15	0.14	0.13	0.14	0.15	0.14
0.14	0.17	0.18	0.17	0.18	0.19	0.18
0.18	0.20	0.20	0.20	0.21	0.20	0.20
0.22	0.21	0.22	0.21	0.21	0.22	0.22
0.26	0.22	0.24	0.23	0.22	0.23	0.23
0.30	0.24	0.24	0.23	0.22	0.22	0.23
0.34	0.24	0.25	0.23	0.22	0.22	0.23
0.38	0.23	0.24	0.23	0.22	0.22	0.23
0.42	0.28	0.25	0.23	0.22	0.22	0.24
0.46	0.24	0.28	0.24	0.23	0.22	0.24
0.50	0.25	0.26	0.26	0.22	0.22	0.24
0.54	0.25	0.24	0.23	0.22	0.20	0.23
0.58	0.23	0.23	0.22	0.21	0.21	0.22

4.0 GAMMA SPECTROMETRY PLANS FOR THE FTR

A reactor characterization program (RCP), which consists of Very Low Power (VLP), Low Power (LP), and High Power (HP) irradiations⁽¹⁵⁾, has been planned and scheduled for the Fast Test Reactor (FTR) at startup. In-core gamma-ray spectrometry will be carried out at VLP in a specially designed FTR insert called the In-Reactor Thimble (IRT). The IRT insert replaces a centrally located fuel assembly (No. 2201) in the FTR core for VLP measurements and provides an adequate environment for the operation of the in-core fission chamber, ionization chambers and spectrometry probes.

In-core continuous gamma-ray spectrometry is planned as the first experiment to be conducted in the IRT. As such, it will be carried out concurrently with fuel loading in the third trisector of FTR. Axial locations at mid-plane and in the lower axial reflector, about 80 cm below mid-plane, have been assigned for these measurements.

Two Si(Li) detectors, the planar 2 cm³ and the coaxial 12 cm³, will be used in the IRT. Background rates for these experiments are expected to be very high. Estimated background rates at mid-plane with all rods in ($k_{eff} = 0.90$) are roughly 5×10^3 and 10^5 count/s for the smaller and larger detector, respectively. These high background count rates stem from the spontaneous fission rate of FTR core 1 fuel, which is 22 wt % plutonium.

Total VLP gamma count rates, with rods adjusted for $k_{eff} = 0.95$ to 0.98, will be roughly 2 to 5 times higher. Even for these very high rates, it may still be possible to collect data in a two-parameter mode for the smaller 2 cm³ planar detector. However, data collection for the larger 12 cm³ coaxial detector will probably be restricted to the one-parameter mode. In this manner, spectral sensitivity will be extended up to roughly 4 MeV and perhaps higher.

5.0 GAMMA SPECTROMETRY IN LWR ENVIRONMENTS

Gamma continua were observed in the low power pressure vessel (PV) mockup at the Pool Critical Assembly (PCA). This mockup represents a controlled irradiation field set up at ORNL to study and quantify the complex radiation field that arises in LWR-PV Irradiation Surveillance Dosimetry Program. There is need for improved gamma heating data to aid in the design of high power LWR-PV irradiation experiments. In addition, fission threshold dosimeters used in LWR-PV environs can possess nonnegligible contributions induced by photo-fission. Consequently, these photo-fission contributions must be more accurately assessed.^(17,18,19)

Gamma measurements were restricted to the interior of the PV block in this work. Observations were carried out at midplane in the T/4, T/2, and 3T/4 locations* using a 12/13 configuration (i.e., water gaps of 12 and 13 cm for the distances between the core face and thermal shield and between the thermal shield and PV face, respectively). At each location foreground data were collected with the reactor at a power level of a few watts and background data was collected with the reactor shut down. As a representative case, the foreground and background electron spectra observed at the T/4 location with a 2 cm³ planar Si(Li) detector are presented in Figure HEDL E-4. The corresponding unfolded gamma spectra are displayed in Figure HEDL E-5. These results can only be regarded as preliminary because analyses for both finite size effects and experimental error have not as yet been performed.

EXPECTED ACHIEVEMENTS

Analysis of finite-size effects for in-situ gamma probes will be completed. Experimental error estimates will be generated for reactor gamma spectrometry. These analyses will be incorporated in computer codes to produce spectral measurements which account for systematic effects and experimental error. Startup gamma spectrometry measurements will be carried out in the FTR.

*The T/4, T/2 and 3T/4 designations represent distances from the front face of a PV block whose total thickness is T.

FIGURE HEDL E-4a. Background Electron Spectrum.

FIGURE HEDL E-4b. Foreground Electron Spectrum.

FIGURE HEDL E-5a. Background Gamma Ray Spectrum.

FIGURE HEDL E-5b. Foreground Gamma Ray Spectrum.

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LASL

LOS ALAMOS SCIENTIFIC LABORATORY

LOS ALAMOS SCIENTIFIC LABORATORY (LASL)
REACTION RATE MEASUREMENTS IN BIG TEN

G. E. Hansen

OBJECTIVE

To measure absolute reaction rates in selected materials irradiated at the center of the Big Ten critical assembly and to compare these data with predictions based on current ENDF/B dosimetry files.

SUMMARY

Three reports pertaining to the dosimetry data and general neutronic characterization of Big Ten are referenced here.

ACCOMPLISHMENTS AND STATUS

D. M. Gilliam, J. A. Grundl, and G. E. Hansen, "Dosimetry Results for Big Ten and Related Benchmarks," Proc. Third ASTM-Euratom Symposium on Reactor Dosimetry, October 1-5, 1979, Ispra (in process of publication).

G. De Leeuw-Gierts, S. De Leeuw, G. E. Hansen, and H. H. Helmic, "Measurements of the Neutron Spectrum of the Big Ten Critical Assembly by Lithium-6 Spectrometry," Proc. Third ASTM-Euratom Symposium on Reactor Dosimetry, October 1-5, 1979, Ispra (in process of publication).

G. E. Hansen and H. C. Paxton, "A Critical Assembly of Uranium Enriched to 10% in Uranium-235," Nucl. Sci. Eng. 72, 230-236 (1979).

EXPECTED ACHIEVEMENTS

The Big Ten measurements, documented in the above references, contribute to the general data base required for successful interpretation of neutron dosimetry. These results will be used in Phase 2 data testing of the ENDF/V Dosimetry File.

NBS

National Bureau of Standards

NATIONAL BUREAU OF STANDARDS

A. U-238 CAPTURE/U-235 FISSION CROSS SECTION RATIO IN THE ISNF AND OTHER COLLABORATIVE TESTS WITH ANL (IDAHO)

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Objectives

To provide U-238 capture cross section validation for the ENDF/B V file.

To compare two independent absolute fission rate measurement techniques: the NBS double fission chamber counting and the ANL (Idaho) Applied Physics Division fission activation foil counting.

To check for perturbations from the instrument penetration of the boron shell of the ISNF.

To obtain preliminary capture and fission cross section results for Th-232 in the ISNF spectrum.

Summary

The U-238 capture to U-235 fission cross section ratio for the Intermediate-Energy Standard Neutron Field (ISNF) was determined experimentally to be

$$\sigma_{\gamma}(\text{U-238})/\sigma_{\text{f}}(\text{U-235}) = 0.1412 \pm 1.7\%$$

in good agreement with the calculated value of 0.1410 which was derived from ENDF/B IV cross sections and a 240 group discrete ordinates calculation. In the experimental determination, both the U-238 capture rate and the U-235 fission rate were measured by post-irradiation gamma counting of activated foils. The experimental result has been corrected in the amount of 3.7% for self-shielding of the U-238 capture reaction. The calculation of this correction was verified by experiment.

Some subsidiary measurements were also carried out during the same series of irradiations. The first was a very direct comparison of two independent measurements of the U-235 fission rate in the ISNF. A U-235 fission activation foil was mounted back-to-back with a platinum-backed thin deposit of the same isotope inside a double fission chamber. The fission rate of the thin deposit was observed by the fission chamber, and the fission rate for the foil was found from post-irradiation gamma counting of the fission products. The observed specific fission rates were $3440 \pm 1.6\%$ fissions/(sec·mg of U-235) for the activation foil and $3415 \pm 1.7\%$ fissions/(sec·mg of U-235) for the deposit in the active fission chamber. The discrepancy is 0.73%, well within the estimated uncertainties and consistent with the NBS/ANL U-235 mass-assay bias of 0.7% which was reported in the 11th ILRR progress report (Table NBS-36). A second U-235 activation foil mounted on the bottom of the fission chamber during the same irradiation gave $3447 \pm 1.6\%$ fissions/(sec·mg of U-235), in very good agreement with the first. This agreement between the activities of U-235 foils irradiated at the chamber center and at the chamber bottom also demonstrates that there is no significant collimated beam of thermal or epithermal neutrons streaming through the "stem" of the fission chamber (as discussed further below).

The second subsidiary measurement was a comparison between the U-238/U-235 fission cross section ratio inside the completely-closed ISNF and the same cross section ratio within the ISNF fission chamber, which requires a penetration through the boron shell. The penetration is necessary for electrical and gas lines which run through a thin aluminum tube called the "stem" of the chamber. A cadmium sleeve and flange fit tightly over the stem to prevent thermal neutron entry. A U-238 activation foil mounted on

top of the fission chamber was irradiated at the same time as the U-235 foils mentioned above. The resulting cross section ratio was $\sigma_f(\text{U-238})/\sigma_f(\text{U-235}) = 0.0935 \pm 1.5\%$ for the activation foils as compared with $0.0927 \pm 1.1\%$ as determined with the active fission chamber. The discrepancy is 0.86%, well within the estimated errors. For the completely closed ISNF, however, the foils gave a higher value of $0.0955 \pm 1.5\%$. This cross section ratio for the closed shell is more than 2% higher than the same ratio, measured by the same technique, within the fission chamber when installed with its penetration. The 2% discrepancy is about twice as large as the relative error of the two discrepant results. A repeated run confirmed the discrepancy, and a test for epithermal neutron entry through the fission chamber stem penetration proved negative. This test for epithermal streaming was made by adding a thick B-10 sleeve to the Cd sleeve around the stem of the chamber. The scattering corrections for the fission chamber and the foil mounting tube used with the closed ISNF were re-checked carefully and found to be correct within a 0.5%. Thus all suspected causes of the 2% discrepancy have been checked, but no resolution has yet been found.

Lastly, preliminary measurements of Th-232 fission and capture cross sections relative to the U-235 fission cross section were carried out. The results were as follows $\sigma_\gamma(\text{Th-232})/\sigma_f(\text{U-235}) = 0.165 \pm 3\%$ and $\sigma_f(\text{Th-232})/\sigma_f(\text{U-235}) = 0.0249 \pm 25\%$. The large uncertainty for the $\sigma_f(\text{Th-232})$ result is given because of lack of a calibrated thorium deposit, which is necessary for the usual calibration against fission chambers at ZPPR.

Accomplishments and Status

The preparation of the ENDF/B V compilation for U-238 presented inconsistencies in the experimental data which an accurate integral test could help to resolve. A collaborative experiment was planned which would utilize the NBS Intermediate Energy Standard Neutron Field (ISNF) and the activation foil counting and analysis expertise of the ANL (Idaho) Applied Physics Division.

The arrangement of the ISNF is shown in figure NBS-1. The one-dimensional geometry and the simplicity of the composition have permitted the most rigorous kinds of transport calculations to be employed in the characterization of this neutron field. Figure NBS-2 shows the details of the mounting arrangements for the activation foils. The results of four irradiations are given in Table NBS-1 along with details of foil identification, locations, and timing. The data presented in Table NBS-1 have been corrected for gradients within the ISNF test volume according to the curves given in Figure NBS-3. The corrections were less than 2% in every case. The four runs listed in Table NBS-1 are denoted as the main run, the fission chamber run, the background run, and the self-shielding run. In the main run, two U-235 foils, two U-238 foils, and one thorium foil were activated. The background run was a repetition of the main run except that the U-235 fission source disks (shown in Figure NBS-1) were removed from the spherical cavity. The background response, due to epithermal and fast neutrons in the thermal column, is subtracted from the response observed in the main run to get the net ISNF response. The background-corrected results are given in the first column of Table NBS-2. Calculations of neutron self-shielding for the $U-238(n,\gamma)$ reaction were

made by ANL, and these corrections turned out to be large - about 10% for a 5 mil (0.13 mm) gold foil. The "self-shielding" run included a 5 mil foil, a stack of three 1 mil foils, and an isolated 1 mil foil in order to avoid having to make such a large correction and to obtain an experimental check on the calculated self-shielding corrections. The degree of agreement of the corrected results from all thicknesses of foil (second column of Table NBS-2) is a measure of the accuracy of the self-shielding corrections. The spread of the results is just over 2%. The uncertainty in the self-shielding correction for the weighted average of all the results is estimated to be $\pm 1.0\%$. This and other uncertainties are tabulated in Table NBS-3. A summary of the most important results from the first three tables is given in the concluding Table NBS-4.

No further joint measurements are planned.

Expected Achievements

Propagated spectral uncertainties for spectrum-integrated cross sections and ratios will be calculated for the ISNF field. These uncertainties arise from the limits of accuracy and ideality in the construction of the facility: there are uncertainties in the source disk positions, in the graphite density, and in the mass-thickness of the boron shell; and there are perturbations due to supporting structures.

Figure NBS-1. The Intermediate-Energy Standard Neutron Field

Figure NBS-2. Foil Mounting Arrangement

Figure NBS-3. Gradient Corrections for the ISNF Central Test Region

TABLE NBS-1

Fission and Capture Rates for Four ISNF Runs
(not corrected for neutron self-shielding)

Foil or Deposit	Principal Isotope	Irradiation Location	Gradient-Corrected Fission or Capture Rate (f/sec. or c/sec. per mg of principal isotope)	Random Error (%, 1σ)	Systematic Error (%, 1σ)
Main Run 9/6/78 08:11:27 - 12:11:21 EDT					
8BNY	U-238	2 cm above center	331.6 f 437.4 c	0.76 0.70	1.6 1.6
2ANK	Th-232	1 cm above center	86.0 f 574.0 c	2.36 0.82	25.0 2.7
5CXP	U-235	center	3494.0 f	0.62	1.5
8BNT	U-238	1 cm below center	334.7 f 440.9 c	0.76 0.71	1.6 1.6
5CXI	U-235	2 cm below center	3513.5 f	0.63	1.5
Fission Chamber Run 9/5-6-78 21:44:43 - 01:40:37 EDT (data corrected for 0.56% addition irradiation in long cooling period after high flux exposure)					
8BNV	U-238	top of chamber	320.1 f 433.3 c	0.80 0.68	1.6 1.6
5CXL	U-235	center of chamber	3440. f	0.64	1.5
5CXS	U-235	bottom of chamber	3447. f	0.64	1.5
25S-5-2	U-235	center of chamber	3415. f	0.50	1.6

*denotes 1 mil thickness other 5 mils

TABLE NBS-1 (continued)

Fission and Capture Rates for Four ISNF Runs
(not corrected for neutron self-shielding)

Foil or Deposit	Principal Isotope	Irradiation Location	Gradient-Corrected Fission or Capture Rate (f/sec. or c/sec. per mg of principal isotope)	Random Error (%, 1σ)	Systematic Error (%, 1σ)
Background Run 9/5/78 15:13:27 - 19:13:21 EDT					
2ANJ	Th-232	1 cm above center	<1.3 f 2.39 c	100. 10.	— —
5CXQ	U-235	center	29.9 f	5.	—
8BNQ	U-238	1 cm below center	7.47 f 1.54 c	10. 20.	— —
Self-Shielding Run 3/20/79 08:07:27 - 12:07:21 EDT					
O	U-238*	2 cm above center (top of stack)	452.8 c	0.7	1.6
AB	U-238*	2 cm above center (middle of stack)	450.1 c	0.7	1.6
AC	U-238*	2 cm above center (bottom of stack)	451.3 c	0.7	1.6
8BNP	U-238	1 cm above center	332.2 f 436.4 c	0.7 0.6	1.6 1.6
5CXG	U-235	center	3487. f	0.6	1.5
L	U-238*	1 cm below center	464.1 c	0.7	1.5
5CXH	U-235	2 cm below center	3476. f	0.6	1.5

*denotes 1 mil thickness other 5 mils

TABLE NBS-2
SUMMARY OF U-238/U-235 CROSS SECTION
RATIO MEASUREMENTS FOR THE ISNF FIELD

U-238 Foil Type and Position	Capture/Fission Ratio and Random Error	Capture/Fission Ratio Corrected for Neutron Self- Absorption and Self-fission	Fission Cross Section Ratio, Corrected for Scattering. $\pm$ Random Error	U-238 Foil or Deposit Label	U-235 Foil or Deposit Labels
NBS-11	stacked 1 mil foils, 2 cm above center	0.1324 $\pm$ 0.8% 0.1316 $\pm$ 0.8% 0.1320 $\pm$ 0.8%	— — —	O AB AC	5CXG, 5CXH 5CXG, 5CXH 5CXG, 5CXH
	average of stacked foils	0.1320 $\pm$ 0.6%	0.1427	—	5CXG, 5CXH
	1 mil foil, 1 cm below center isolated	0.1357 $\pm$ 0.8%	0.1406	—	5CXG, 5CXH
	5 mil foil 1 cm above center	0.1276 $\pm$ 0.7%	0.1429	0.0957 $\pm$ 0.8%	8BNP
	5 mil foil 1 cm below center	0.1271 $\pm$ 0.8%	0.1424	0.0949 $\pm$ 0.9%	8BNY
	5 mil foil 2 cm above center	0.1281 $\pm$ 0.8%	0.1435	0.0957 $\pm$ 0.9%	8BNT
	average for 5 mil foils in closed ISNF	0.1276 $\pm$ 0.5%	0.1429	0.0955 $\pm$ 0.5%	8BNP, 8BNY, 8BNT
	5 mil foil on top of fission chamber	0.1288 $\pm$ 0.8%	0.1443	0.0935 $\pm$ 0.9%	8BNV
	thin deposit on platinum backing	—	—	0.0927 $\pm$ 0.6% (total error $\pm$ 1.1%)	28S-2-1 28N-2-1 28HD-8-1

TABLE NBS-3

ESTIMATED UNCERTAINTIES IN THE CROSS SECTION RATIO MEASUREMENTS

Source of Error	Estimated 1σ Limit (%)
$\sigma_Y(U-238)/\sigma_f(U-235)$ Activation Foil Results	
Deposit mass and other minor errors in calibration against back-to-back fission chambers at ZPPR and AFSR.	1.2
Self-shielding correction	1.0
Combined Random Errors	0.72
Total Systematic Error	1.56
Total Random Error	0.72
Total Error	1.72
$\sigma_f(U-238)/\sigma_f(U-235)$ Activation Foil Results	
Deposit mass and other minor errors in calibration against back-to-back fission chambers at AFSR.	1.3
Scattering Corrections	0.6
Combined Random Errors	0.5
Total Systematic Error	1.43
Total Random Error	.6
Total Error	1.52

TABLE NBS-4
SUMMARY OF RESULTS FOR ANL/NBS ISNF TESTS

ISNF Spectrum Average Quantity	Measured Value and Estimated Uncertainty	Calculated Value from ENDF/B IV data and NBS-Evaluated U-235 Fission Spectrum
U-238 Capture/U-235 Fission Cross Section Ratio		
$\sigma_{\gamma}(\text{U-238})/\sigma_{\text{f}}(\text{U-235})$ by activation foils	0.1412 $\pm 1.7\%$	0.1410
Absolute Fission Rate Measurement Comparison		
Fission Rate/mg of U-235 at 9.9 MW by activation foil	3440 $\pm 1.6\%$	—
Fission Rate/mg of U-235 at 9.9 MW by NBS Fission Chamber	3415 $\pm 1.7\%$	—
Penetration and Scattering Perturbation Checks		
$\sigma_{\text{f}}(\text{U-238})/\sigma_{\text{f}}(\text{U-235})$ by activation foils mounted within the NBS Fission Chamber	0.0935 $\pm 1.5\%$	0.0836
$\sigma_{\text{f}}(\text{U-238})/\sigma_{\text{f}}(\text{U-235})$ by NBS Double Fission Chamber	0.0927 $\pm 1.1\%$	0.0836
$\sigma_{\text{f}}(\text{U-235})/\sigma_{\text{f}}(\text{U-235})$ by activation foils within completely closed ISNF	0.0955 $\pm 1.5\%$	0.0836
Preliminary Results for Thorium		
$\sigma_{\gamma}(\text{Th-232})/\sigma_{\text{f}}(\text{U-235})$ by activation foils	0.165 $\pm 3\%$	0.171
$\left(\sigma_{\text{f}}(\text{Th-232})/\sigma_{\text{f}}(\text{U-235}) \right)$ by activation foils	$\left(0.0249 \right)$ $\left(\pm 25\% \right)$	$\left(0.0191 \right)$

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NBS-14

B. THE GOLD CAPTURE CROSS SECTION IN THE NBS
INTERMEDIATE-ENERGY STANDARD NEUTRON FIELD (ISNF)

A. Fabry, CEN-SCK

D. M. Gilliam, C. Eisenhauer, W. E. Slater - NBS

Objectives

Measure the spectral average cross section for the $^{197}\text{Au}(n,\gamma)^{198}\text{Au}$ reaction at the ISNF center relative to NBS fission rates in order to confirm gold as a primary dosimetry reaction for ENDF/B cross section file validation in the intermediate energy range.

Probe the ISNF field for neutron flux gradients, self-shielding and backgrounds in the intermediate energy range.

Summary

The NBS absolute fission chamber and a transportable gamma-ray integral analyzer calibrated relative to the CEN-SCK reference NaI(Tl) gamma-ray spectrometer have been applied to the measurement of the ISNF central reaction rate ratios $^{115}\text{In}(n,n')^{115\text{m}}\text{In}/^{238}\text{U}(n,f)$ and $^{197}\text{Au}(n,\gamma)^{198}\text{Au}/^{115}\text{In}(n,n')^{115\text{m}}\text{In}$. A preliminary experimental gold capture to uranium-235 fission cross section ratio of $0.248 (\pm 2.9\%)$ is obtained, to be compared with the calculated ratio of 0.258.

Neutron self-shielding, flux gradients and neutron backgrounds in the ISNF field are reported together with extensive details of the measurements. This is intended to assist ILRR and other experimenters in the preparation of future dosimetry cross section validation work or dosimetry benchmark field reference work in the intermediate energy range.

Accomplishments

1. Introduction

The NBS Intermediate-Energy Standard Neutron Field^[1] (ISNF) has

been designed primarily to provide an accurately known neutron flux spectrum over the energy range most relevant to Fast Breeder Reactor (FBR) physics, e.g. a few keV to approximately 1 MeV. The ISNF is the only existing neutron field today in which the low-energy tail below 10 keV is calculable to better than $\pm 5\%$; uncertainties of at least $\pm 20\%$ are typical in this range, and 100% is not uncommon. This is an energy range most crucial in terms of safety coefficients for large dilute FBR assemblies. Foil dosimetry techniques have recently been recognized as a valuable support to reactivity measurements in addressing the implications of these uncertainties.

The status of the dosimetry cross section file in the intermediate-energy range is unsatisfactory. Adjustments relative to a few primary reactions^[2] are needed much more in this range than in the threshold domain, where most remaining problems are below the 5% level.^[3] However, the only primary dosimetry capture reaction $^{197}\text{Au}(n,\gamma)^{198}\text{Au}$, a standard for differential cross section measurements, has never been submitted to an integral validation in the intermediate-energy range, and this is mandatory, even though the differential-energy cross section is currently evaluated to be known to better than $\pm 5\%$ in this range; indeed the same statement applies to the ISNF neutron spectrum below 1 MeV, especially at the energies where the $^{10}\text{B}(n,\alpha)$ reaction cross section dominates the shaping of the ISNF spectrum and thereby governs the gold reaction rate and gold neutron self-shielding.

It has been the most important purpose of the present work to provide a confrontation between the gold capture to uranium-235 fission cross section ratios as measured by differential and integral techniques, on the one hand, and the ISNF neutron spectrum on the other. In order to meet the requirement of the highest possible accuracy, this goal has been achieved by relating gold capture to indium-115 inelastic scattering, indium to uranium-238 fission and referring to previous NBS work on fission cross section ratios^[4]; the

experimental reasons for such an approach are made clear in the following sections.

A careful measurement of gold neutron self-shielding has also been done, not only because it is needed to guarantee an accurate extrapolation to infinite-dilution conditions, but as further, less direct confrontation between the gold total and capture cross sections and the ISNF neutron spectrum.

2. Irradiation Procedure

Basically two exposure geometries have been adopted to conduct the experiments:

1. All the gold irradiations, runs ISNF/1 to 6 in Table NBS-5, are made inside of the ISNF boron-10 shell fully closed, e.g. without the access hole needed for active measurements; indium foils are simultaneously used to obtain $^{115}\text{In}(n,n')^{115\text{m}}\text{In}$ reaction rates.
2. In a second series of irradiations, runs ISNF/7 to 9, indium foils are placed inside of the NBS absolute fission chamber, back-to-back with an uranium-238 fissionable deposit, as well as outside of the chamber, on its top cover.

Fluctuations in the reactor power level are recorded by means of a NBS fission chamber run-to-run monitor located in a channel of the graphite thermal column adjacent to, but not too close to the ISNF; all reaction rates are normalized to the response of this monitor, last column of Table NBS-5; the monitor also provides accurate definition of the exposure duration for an "effective" rectangular time flux profile, column 3 of the table. Two irradiations, runs ISNF/2 and 6, are aimed at assessing the neutron background responses of gold and indium (previous work^[4] has established these responses for uranium-238 fission):

In run ISNF/2, the boron curtain, which allows decoupling the ISNF from the driving reactor core, has been lowered; these measurements are used to ascertain the generally small corrections for gold and indium activation while in pile with the curtain down (column 4 versus column 3 in Table NBS-5).

Run ISNF/6 is done with the boron curtain in the raised position (open), but after removal of the eight uranium-235 fission disk sources which generate the ISNF field by conversion of the driving thermal neutrons into fission spectrum neutrons.

All activation foils have a diameter of 19 mm. The nominal thickness for indium is always 0.127 mm (5 mils), while the gold foil thickness varies from 0.005 mm to 0.127 mm. The foils are cut from spectrographically pure elemental metal sheets, cleaned, and weighed. The punching press has been carefully optimized to provide very flat, regularly shaped circular disks; the punch and matrix are different for different classes of materials, arranged by hardness, chemical affinity, activation cross section, and type of radionuclide produced by neutron capture. In this way, the only interfering activities, as checked by high-resolution gamma-ray spectroscopy, are limited to isotopic competing nuclear reactions for the element of interest, except in one case: 5% gold-aluminum alloy foils, 0.1 mm thick, prepared by levitation melting^(*), have been used also to supplement the elemental gold sheets at the extremely thin end of the experimental thickness range; in this case, the $^{27}\text{Al}(n,\alpha)^{24}\text{Na}$ reaction is also contributing and must be discriminated against.

The details of foil arrangement during the irradiations were worked out on the basis of previous experience in the Mol-ΣΣ neutron field^[5] and account for the following:

- a. Gold cannot be exposed directly back-to-back inside of the NBS fission chamber because of re-entrant hole neutrons (perturbations in excess of 3%); therefore, a sequence of two types of irradiation was needed, as outlined.

^(*) Commercial company licensed to apply the procedures developed at CBNM, Euratom, Geel.

- b. As the driving thermal neutron flux falls off almost exponentially along the horizontal axis of the thermal column ("macroscopic" flux gradient ^[5]), the foils must be placed in the vertical plane through the ISNF center; the fissionable deposit and activation foils thus always were on the ISNF central, vertical polar axis.
- c. Boron-10 absorption of the cavity wall-return neutrons produces forward-peaked angular distributions at the inner absorber boundary ^[6] and "microscopic" ^[5] flux gradients result for non-threshold sensors within the ISNF central zone.
- d. Neutron self-shielding for non-threshold sensors may be quite significant, space-dependent, and capable of inducing foil-to-foil interference ^[5].

The extent to which items c. and d. above come into play depends on the energy response of the particular sensor in the ISNF. For capture reactions with a gross $1/v$ energy dependence - as opposed to fission reactions, which are more "flat" overall - the effects are important if the first unbound resonances are well isolated and at high energy, as in manganese and copper for example. For indium, these resonances are so low in energy that, due to the boron-10 tailoring, even 5 mil sensors are effectively transparent to neutrons. Gold and uranium-238 capture in ISNF represent an in-between case, for which "microscopic" gradients and neutron self-shielding are sizeable, but space variations of self-shielding are irrelevant and foil-to-foil shadowing can be kept negligible by 1) adequate spacing (irradiating one foil at a time would be perfect, but expensive) 2) adequate sizing and combinations of different foils in each run.

The actual implementation of these considerations and the value of the resulting experimental design can be further and more directly appraised from the extensive data tabulations in section 4.

The foil holder for the closed shell runs (ISNF/1 to 6) is illustrated in Figure NBS-2 (section A). It is a thin-walled aluminum cylinder of the same

inner diameter as the foils; aluminum spacers are used to keep the separation distance between any two foils (1 cm or larger) accurately known and reproducible; circular holes through the wall allow checking the foil positioning after mounting.

3. Methods for Reaction Rate Measurements

3.1 NBS Fission Chamber

A general description of the NBS Double Fission Chamber and a discussion of the systematic uncertainties in the determination of absolute specific fission rates (fission rates per unit mass) are given in Reference 12.

In the present work, a highly depleted U-238 deposit 28HD-5-2 with principal isotope mass of $700.4 \mu\text{g} \pm 1.7\%$ was employed. The mass density of the 0.50 in. diameter deposit is $552.9 \mu\text{g}/\text{cm}^2$. The U-235 isotopic content is less than 4 ppm. The mass of this heavier deposit was determined relative to that of the reference deposit 28S-2-2 by back-to-back fission counting in the BIG-10 fast critical assembly.

(The reference deposit mass reported for 28S-2-2 in Reference 12 has been revised from $252 \mu\text{g} \pm 2\%$ (old value) to $249 \mu\text{g} \pm 1.5\%$ (new value). This change was based on improved alpha activity assay and on fission comparisons with the normal uranium reference deposit 28N-2-2 in a Cf-252 fission neutron field.)

3.2 Radiometric Reaction Rates

Reaction rates $^{115}\text{In}(n,n')^{115\text{m}}\text{In}$ and $^{197}\text{Au}(n,\gamma)^{198}\text{Au}$ have been measured by means of a transportable 3" x 3" NaI(Tl) gamma-ray spectrometer carried by plane from Mol, Belgium, to NBS.

This spectrometer is operated as an integral counter. The linear pulses from the preamplifier and main amplifier are fed into a set of six integral discriminators followed by multiple scalers. The discriminators are set relative to a multi-channel analyzer using a high precision pulser and/or a linear gate. The settings are such as to provide the following signals:

- total count rate above the noise level
- count rate in a window encompassing the 336 keV and 320 keV photopeaks of ^{115m}In and ^{51}Cr
- count rate in a window centered on the 412 keV photopeak of ^{198}Au
- count rate above 662 keV, the ^{137}Cs photopeak centroid

The last signal is aimed at assessing corrections for the competing reactions $^{113}\text{In}(n,\gamma)^{114m}\text{In}$ and, for gold alloys, $^{27}\text{Al}(n,\alpha)^{24}\text{Na}$. This is done easily, in the first case by counting a similar foil from previous irradiations (the correction is found negligibly small for ISNF counting conditions) and, in the second case, by counting spectrographically pure aluminum foils geometrically identical to the alloy foils and sandwiched to them during irradiation (a thin-catcher foil prevents contamination by recoil nucleus). Corrections for the other competing reactions $^{115}\text{In}(n,\gamma)^{116m}\text{In}$, $^{113}\text{In}(n,n')^{113m}\text{In}$, $^{27}\text{Al}(n,\gamma)^{28}\text{Al}$, $^{27}\text{Al}(n,p)^{27}\text{Mg}$ and $^{197}\text{Au}(n,2n)^{196}\text{Au}$ are kept negligible^(a) by appropriate selection of the cooling time, e.g. the time elapsing between the end of irradiation and the counting interval. This is checked in various ways: by inspection of the count rate ratios for the six discriminator levels, by extending observations over one or more radioactive decay periods, and by spot controls with a high resolution Ge(Li) spectrometer.

The stability of the electronic settings over periods of a few months is remarkable; it has been monitored by counting reference sources of ^{51}Cr , ^{137}Cs , ^{60}Co every day and by following the decay of a gold foil over 14 days. Extensive optimization and testing of the setting sensitivities at Mol prior to shipment of the spectrometer had indicated that the gain of the main amplifier may need slight adjustments, while the discriminator biases did not change at all over the period of observation. It has been most gratifying to find that, as soon as the

^(a) The only exception is for the gold foils in the neutron background run ISNF/6, Table NBS-5 where (n,2n) competition is appreciable, pointing out to serious reactor core leakage contribution to the ISNF field at very high neutron energy.

spectrometer was installed in its new lead-shielded cave at NBS, the count rates of the reference sources, as well as the background count rates, agreed perfectly with the prior observations at Mol for all discriminator levels. In fact, no gain adjustment was necessary over the whole duration of the experiment, from the start with calibration measurements at Mol, through the completion of run ISNF/9 at NBS.

The dead time of the spectrometer, measured by the two source method (and some variant of it) is large, 11 ± 2 μ sec., and this feature will have to be improved for future applications. At the low count rates dealt with in the ISNF experiments, the resulting uncertainties are of the order of 0.1% or less.

It may be useful to briefly discuss here why such an integral NaI(Tl) counter has been selected for a standardization experiment as the present one, at a time when high resolution devices of increasing quality are trivially available, and good multi-channel analyzers can be applied routinely with whichever spectrometer is desired. In a sense, the reason is comparable to the rational by which the NBS fission chamber is not a solid-state one operated with a 4000-channel analyzer: robustness, optimal precision, accuracy (encompassing in particular stability and reproducibility), and high efficiency are achieved in very simple and reliable ways for experiments which, in the present case, do not require high-resolution and sophisticated pulse-height analysis.

The precision of the integral spectrometer-counter used in this work exceeds that of any conventional differential gamma-ray spectrometer, Ge(Li) or NaI(Tl), used by the authors, while the accuracy is comparable. In gamma-ray spectrometry, as with many other techniques, the dominating uncertainty is traceable to efficiency determination.

In contrast to Ge(Li) measurements, NaI(Tl) gamma-ray spectrometry could involve efficiencies derived to high accuracy from first principles, whether in multi-channel analyzer counting or for the channel recording the total count rate above the noise level in the present set-up^[8]. This, however, is seldom

done anymore because high accuracy can be reached as readily by calibration with commercially available sources standardized by a variety of absolute methods or by calibration relative to another standardized instrument. This last approach has been adopted here.

The transportable spectrometer has been compared to an identical 3" x 3" NaI(Tl) spectrometer operated with a multi-channel analyzer and used at MoI as a permanent (since 1962) reference instrument for reactor physics and neutron dosimetry applications^[8]. The comparison is straightforward: activation foils are counted in the main geometry used for the transportable spectrometer (source-to-detector distance of ~4 cm, labeled shelf 4) and are also measured in one or two standardized geometries of the reference spectrometer. The resulting "calibration ratios" relate the photopeak efficiency of the reference instrument to the effective efficiency of the transportable one.

Table NBS-6 to NBS-9 gather all relevant details and uncertainties in deriving the so-called "K-factors" for the reactions $^{115}\text{In}(n,n')^{115\text{m}}\text{In}$ and $^{197}\text{Au}(n,\gamma)^{198}\text{Au}$ as observed by means of the transportable spectrometer.

The K factors are defined by the simple relationship

$$r = K S_{\infty}$$

where r is the reaction rate ($\text{sec}^{-1}/\text{nucleus}$), S_{∞} (called the specific saturation signal), is the count rate per gram of the natural element in which the reaction of interest occurs, as observed by means of the transportable spectrometer, and corrected for background, dead-time, impurity interference, decay during counting, decay between the end of irradiation and the time of counting, and decay during irradiation duration (saturation); note that S_{∞} applies to the actual foil and counting geometries so that gamma-ray self-absorption and geometry corrections are to be included in the K-factor, by convention (e.g. they could have been incorporated as well in the definition of S_{∞} in order to derive K-factors independent of foil size, a more logical, but usually less practical methodology).

The dominating systematic uncertainty in the determination of the K-factors stems from the photopeak efficiencies of the reference spectrometer.

These efficiencies were remeasured very carefully in 1974^[8], generally using monoenergetic standard gamma emitters provided by LMRI (Radioisotope Measurement Laboratory, Saclay), by IAEA, by AMERSHAM, or sometimes prepared by MoI Absolute Radioactivity Section.

Recently a new least-squares analysis of these data has been completed^[9]; it distinguishes correlated from uncorrelated uncertainties and simultaneously fits the measurements at the three reference source-to-spectrometer distances. The efficiency values in Tables NBS-6 and NBS-8 result from this fit. For ⁵¹Cr, which is so close in energy to ^{115m}In that the corresponding efficiencies differ by only ~3%^[10], the fitted efficiencies are 0.4% larger than the measured ones for the three geometries; the measured values are derived from LMRI, IAEA, and MoI standard sources and while these three laboratories used independent standardization approaches (MoI, IAEA^(a) entirely uncorrelated with LMRI), agreement is at the $\pm 1\%$ level^(b). The combined systematic uncertainties for ^{115m}In photopeak efficiencies are consequently assigned as $\pm 1\%$ ^(c) in the three geometries.

For ¹⁹⁸Au, the fitted efficiencies, whose accuracies are of the order of $\pm 1.2\%$, exceed the measured ones by 0.5% to 1.35%, depending on the source-to-spectrometer distance (1.2% for shelf 5 in Table NBS-8). The measurements only refer to $4\pi\beta\text{-}\gamma$ sources prepared at MoI, to an accuracy estimated as $\pm 1.5\%$. The fit and its uncertainty is therefore accepted for the geometry of interest

(a) Determination of disintegration rate (dps) uncorrelated, but same branching ratio accepted subsequently.

(b) This may seem surprising in view of the fact that the branching ratio systematic uncertainty for Cr is quoted as $\pm 1.4\%$ and enters the comparison, as LMRI directly derives $\gamma/\text{min. } 4\pi$ rather than dps (μCi).

(c) Accounting for the additional information contained in the least-squares fit.

here. It is most relevant to indicate that this is backed by a comparison between gold and NBS fissionable deposits (^{235}U and ^{239}Pu) exposed back-to-back in the NBS Fission chamber at the center of a Maxwellian-reference thermal neutron field at $\text{MoI}^{[11]}$.

3.3 Mass Assay

The definition of the number of principal isotope nuclei and isotopic analysis for NBS fissionable deposits has been dealt with in previous publications (for ex., [12]) and is not discussed here.

For pure elemental radiometric foils, physical weight measurements provide the necessary information to an accuracy better by orders of magnitude than any other observation or calculational consideration in this work; tiny impurities revealed by chemical analysis (and essentially undetectable in high resolution gamma-ray spectra) play no role in terms of mass assay.

For gold-aluminum alloys, however, the (5.00 ± 0.05) weight percent gold content assessed from weighing operations during the levitation melting is at variance with a figure of (4.885 ± 0.05) weight percent derived by activation analysis, Table NBS-10, using the following straightforward interpretation:

- predicted ratio of specific saturation signals

$$= \frac{0.994}{1} \times \frac{0.9987}{0.9748} \times \frac{0.9985}{0.8725} = 1.165$$

geometry
gamma-ray
thermal neutron
factors
self-absorption^(a)
self-shielding^(b)

- measured ratio of specific saturation signals assuming the nominal 5.00% weight content = 1.138 (Table NBS-10).

A value of (4.94 ± 0.06) weight percent has been adopted for the moment, but further work is planned; indeed, for some alloys prepared by the same

^(a) See section 4.

^(b) See Reference^[11]

technique and involving the admixture in aluminum of sensor elements with a high melting point or large volatility, losses during levitation melting have been evidenced by activation analysis measurements similar to the present ones and have been recognized as most likely by CBNM experts; such has never been the case, however, for gold alloys in numerous previous studies nor does it seem possible; these previous alloys were carefully prepared at CBNM while the present order was executed by a commercial company supervised by CBNM; on another hand, our previous activation analysis work has been in some respects more systematic and detailed than the one reported here.

4. Data Acquisition and Analysis

4.1 Fission Chamber Data Acquisition and Analysis

Table NBS-11 gives the U-238 count rates observed in runs ISNF 7,8, and 9, with corrections and error values listed in the footnotes. The count rate results are expressed also as a specific fission rate (fission rate per nucleus), using the mass assay discussed above (sect. 3.1). The U-238 specific fission rates are then converted to U-235 specific fission rates using a value of $0.0927 \pm 1.1\%$ for the fission cross section ratio from Reference 4 and recent re-evaluation of the scattering corrections. This cross section ratio is corrected for losses due to the scattering of high energy neutrons to energies below the U-238 threshold. This corrected value is used because the In-115 inelastic scattering threshold is at nearly the same energy as the U-238 fission threshold so that the back-to-back foil and deposit should experience very nearly the same scattering perturbation (estimated relative uncertainty, $\pm 0.5\%$). Finally, the specific U-235 fission rates are normalized by the monitor count rate. These three normalized values agree within $\pm 0.5\%$ (Exper. Std. Dev.). The average of these runs is $8.51 \text{ E-18} \pm 2.0\%$ (U-238 fissions/(sec·nucleus))/monitor cps. The error factors leading to the $\pm 2.0\%$ are given in the footnote of the table.

The total error may seem surprisingly small since the U-238 deposit mass uncertainty was almost that large, $\pm 1.7\%$. However, the error in the mass of that deposit is almost entirely cancelled in the conversion to U-235 specific fission rates. Only the U-235/U-238 cross section ratio uncertainty and the U-235 reference deposit mass uncertainty ultimately contribute to the error in the final result.

4.2 Radioactive Measurements: $^{115}\text{In}(n,n')^{115\text{m}}\text{In}$.

Table NBS-12 and its footnotes provide an extensive description of the $^{115}\text{In}(n,n')^{115\text{m}}\text{In}$ ISNF test matrix as generated by the irradiations listed in Table NBS-5. For the main runs the uncertainties of the individual count rates, are consistent with the overall, excellent reproducibility of the observations. For the neutron background runs the uncertainties are of the order of $\pm 3.5\%$.

The information is summarized in Table NBS-13 and its analysis proceeds along the following, straightforward comments:

- There is a slight and expected polar gradient of the fast flux, of the order of $0.15\%/cm$; the central $^{115}\text{In}(n,n')^{115\text{m}}\text{In}$ reaction rate in the unperturbed configuration is

$$74.77 \times 1.006 \times 6.563 \text{ E }^{-21} = 4.937 \text{ E }^{-19} \\ \text{sec}^{-1}/\text{nucleus} \cdot \text{cps monitor}$$

A correction of $+(0.6 \pm 0.5)\%$ is applied for neutron scattering and absorption within the sensors.

- The effects on the $^{115}\text{In}(n,n')^{115\text{m}}\text{In}$ response of neutron field perturbations by the NBS fission chamber are significant: $\sim 1.5\%$ neutron removal close to the platinum backing of the fissionable deposits, $\sim 0.6\%$ on top of the chamber, in agreement with previous observations in the Mol-ΣΣ facility.

As the $^{115}\text{In}(n,n')^{115\text{m}}\text{In}$ energy responses do not differ appreciably, the reaction rate ratio $^{115}\text{In}(n,n')^{115\text{m}}\text{In}/^{238}\text{U}(n,f)$ can be derived by using the perturbed reaction rates at contact of the platinum backing, e.g. for $^{115}\text{In}(n,n')^{115\text{m}}\text{In}$:

$$73.59 \times 1.006 \times 6.563 \text{ E } -21 = 4.859 \text{ E } -19 \\ \text{sec}^{-1}/\text{nucleus} \cdot \text{cps monitor}$$

4.3 Radiometric measurements: $^{197}\text{Au}(n,\gamma)^{198}\text{Au}$.

The ISNF test matrix for this reaction is displayed in Table NBS-14. The random uncertainties are consistent again with the reproducibility of the repeated observations and are therefore not quoted.

The analysis of these data is not as simple as for indium because

- 1) polar reaction rate gradients are more pronounced and are slightly asymmetrical (see Table NBS-15),
- 2) neutron self shielding and gamma-ray self-absorption effects are sizeable, and
- 3) the gold response to core leakage neutron background involves a neutron spectral distribution significantly softer than the ISNF spectrum itself and therefore, steeper gradients, as well as larger neutron self-shielding corrections, see Table NBS-16, footnote^(*); as these background corrections are fortunately quite small, their gradient difference as compared to the main signal gradient can be neglected so that the more important self-shielding difference can be accounted for on the basis of the limited available measurements, run ISNF/6.

The first step in the analysis is then to derive the vertical polar gradient, Table NBS-15. This has been done by combination of the shapes measured with 5% alloys and 0.028 mm thick metallic foils; interpolation was needed only at the +1 cm position and has been made graphically; neutron background corrections

were assumed constant point-wise establishing this gradient function and the small resulting increase in uncertainties has been approximated as random.

Asymmetry of the gold gradient, although small, was of enough concern to call for further scrutiny. It has been inferred from Monte Carlo calculations^[6] that it must be due to an asymmetry of the source intensities. The vertical polar distribution of the thermal neutron flux available in the NBS cavity has consequently been measured and it was indeed found to display also the undesirable asymmetry (Figure NBS-4); this can be understood easily by rough inspection of the engineering drawings of the NBS thermal column, e.g. asymmetry and limited size of the graphite stack. As the boron-10 tailoring effect is very drastic and the corresponding ISNF absorber-reflector coupling is important^[6], a large fraction of neutrons contributing to the gold response have undergone too few cavity crossing and collisions in the graphite walls to "lose memory" of their origin.

It must be emphasized that these subtle ISNF shortcomings only affect the scalar neutron field out-of-center and the angular neutron flux distributions (even at center); the central scalar neutron flux spectrum, however, is perfectly immune to these imperfections.

In the second step of the analysis, the present test matrix, Table NBS-14, has accordingly been transformed, through Table NBS-15, into central reaction rate data, Table NBS-16, appropriately corrected for core leakage neutron background.

The third step is to account for gamma-ray self-absorption. This must be done accurately because the gold thickness dependency of neutron background-corrected central specific saturation signals is proportional to the product of gamma-ray self-absorption and neutron self-shielding, so that uncertainties in the second term are strongly correlated with uncertainties in the first. Details of the gamma-ray self-absorption calculations are given in the appendix.

In order to check this point of the analysis, the gold thermal neutron self-shielding, which is well established^[12], has been remeasured in the Mol Rotating Graphite Plug Thermal Neutron Field, specifically using the ISNF sensors and the transportable spectrometer, prior to its shipment to NBS. The agreement with calculations was largely within the experimental uncertainties.

In Table NBS-17 are gathered the ISNF central gold specific saturated disintegration rates per monitor count.

The fourth and last step of the analysis is to correct all data for neutron self-shielding, in order to derive central unperturbed infinite-dilution gold-capture rates.

At the time of this reporting, this extrapolation to zero thickness has been done graphically, Figure NBS-5, and is significantly affected by the current mass assay uncertainties for gold alloy, see section 3.3. An analytical calculation of the ISNF gold self-shielding is in progress.

The present, preliminary ISNF $^{197}\text{Au}(n,\gamma)^{198}\text{Au}$ central reaction rate relative to the run-to-run monitor is

$$6560 \times \frac{196.967}{6.022 \text{ E } 23} = 2.146 \text{ E } -18$$

which leads to a cross section ratio

$$^{197}\text{Au}(n,\gamma)^{198}\text{Au} / ^{115}\text{In}(n,n')^{115\text{m}}\text{In} = \frac{2.146 \text{ E } -18}{4.937 \text{ E } -19} = 4.346$$

5. Results and Conclusions

Finally now the monitor-normalized specific reaction rates from section 4 may be combined to give the Au capture to U-235 fission cross section ratio.

$$\bar{\sigma}_{\gamma}(\text{Au-197})/\bar{\sigma}_{\text{f}}(\text{U-235}) =$$

$$\frac{2.146 \text{ E } -18 \pm 1.93\%}{4.937 \text{ E } -19 \pm 1.2\%} \frac{4.859 \text{ E } -19 \pm 1.2\%}{8.51 \text{ E } -18 \pm 2.0\%} = 0.248 \pm 2.9\%$$

$$\frac{\text{Au}_{\gamma}/\text{In}_{\text{n,n}'}}{\text{closed ISNF}} \quad \frac{\text{In}_{\text{n,n}'}/^{238}\text{U}_{\text{f}}(^{235}\text{U}_{\text{f}}/^ {238}\text{U}_{\text{f}}))}{\text{fission chamber in ISNF}}$$

The systematic errors in the two In(n,n') reaction rates are entirely correlated and cancel each other out. A random error of $\pm 0.6\%$ from the ratio of the two In(n,n') results is included in the total, however.

The result above for $\sigma_{\gamma}(\text{Au-197})/\sigma_{\text{f}}(\text{U-235})$ differs from the calculated value of 0.258 by 4.0%. The calculated value was derived from a 240 group discrete ordinates computation which used ENDF/B IV cross sections and the NBS evaluated ^{235}U fission neutron spectrum.

The In(n,n') to Au capture ratio as measured in this series of experiments is also interesting to compare with calculation. The measured ratio is just the first term of the equation above

$$\bar{\sigma}_{\gamma}(\text{Au-197})/\bar{\sigma}_{\text{n,n}'}(\text{In-115}) = 4.346 \pm 2.1\%.$$

In estimating the combined error, allowance was made for the correlation of the uncertainties in the photopeak efficiencies at 336 keV and 412 keV. From the same calculation described above, a calculated value for this cross section ratio of 4.76 was obtained, differing with the experimental result by about 9.4%. This large discrepancy is in the same direction as the 11% discrepancy (See Section A. Table NBS-4) in ISNF calculation/experiment (C/E) for $\bar{\sigma}_{\text{f}}(\text{U-235})/\bar{\sigma}_{\text{f}}(\text{U-238})$.

These large C/E discrepancies are consistent with similar discrepancies noted in the case of the U-235 fission neutron field. In this latter case, the

C/E discrepancy^[13] for $\bar{\sigma}_f(\text{U-235})/\bar{\sigma}_f(\text{U-238})$ is $(6 \pm 3) \%$ in the same direction as for the ISNF. There is a consistent tendency in these C/E discrepancies: the reaction rate of the threshold detector relative to a low-energy-sensitive detector is always higher in the measurement than in the calculation. This consistency could be attributable to systematic biases in the representation of the fission neutron energy spectrum and/or systematic biases in the current cross section tabulations.

Expected Achievements.

In order to finalize the data in this progress report, analytical calculations of the gold neutron self-shielding at the ISNF center will be completed in parallel to further mass assay checks of the gold-aluminum alloy sensors used at thicknesses close to infinite dilution.

Fig. NBS-4 Gold Reaction Rate Profiles Along NBS Cavity Vertical Polar Axis.

Fig. NBS-5 Gold Relative Self-Shielding in the ISNF

Table NBS-5

ISNF Irradiations: Relevant Parameters

<u>Run</u>	<u>End of irradiation</u>	<u>Effective duration</u> (sec.)	<u>Total time in-pile</u> (sec.)	<u>Monitor count rate</u> ^(*) (cps)
ISNF/1 ^(a)	6-8-77 17h 50' 20"	9716	64316	143.8 ($\pm$ 0.09%)
2 ^(**) (a)	6-10-77 9h 10'	60300	60300	(**)
3 ^(a)	6-13-77 17h 46' 20"	9176	9800	143.6 ($\pm$ 0.09%)
4 ^(a)	6-15-77 8h 46'	52762	69720	144.4 ($\pm$ 0.04%)
5 ^(a)	6-24-77 16h 34' 30"	24047	29400	122.4 ($\pm$ 0.09%)
6 ^(***) (a)	6-26-77 8h 30'	50725	53700	122.7 ($\pm$ 0.12%)
7 ^(b)	6-29-77 22h 13'	7615	12600	118.1 ($\pm$ 0.12%)
8 ^(b)	6-30-77 17h 15'	18660	19600	126.3 ($\pm$ 0.08%)
9 ^(b)	7-1-77 18h 30'	7175	12000	125.8 ($\pm$ 0.12%)

(*) Gross count rate, extrapolated to zero bias, and dead-time corrected.

(**) Background run with boron curtain closed but fission disc sources in place.

(***) Background run with fission disc sources removed but boron curtain open.

(a) The ISNF boron-10 shell is fully closed.

(b) The ISNF boron-10 shell has access hole to accomodate the NBS fission chamber.

Table NBS-6

Determination of the $^{115}\text{In}(n,n')^{115\text{m}}\text{In}$ K-Factor^(a)
 For the Transportable 3" x 3" NaI(Tl)
 Gamma-ray Spectrometer Relative to the
 CEN-SCK 3" x 3" Reference NaI(Tl) Spectrometer

Type of data	(b) Shelf 0 of reference spectrometer	(b) Shelf 5 of reference spectrometer
Calibration ratio ^{(c)(c')}	0.2673 $\pm$ 0.4% r.	1.419 $\pm$ 0.3% r.
Photopeak efficiency of reference spectrometer	0.2527 $\pm$ 0.3% r. $\pm$ 1.0% s.	0.04720 $\pm$ 0.3% r. $\pm$ 1.0% s.
Geometry correction ^(c)	0.9876 $\pm$ 0.2% s.	0.9940 $\pm$ 0.1% s.
Gamma-ray self-absorption ^{(d)(e)}	0.9905 $\pm$ 0.5% s.	0.9937 $\pm$ 0.1% s.
$^{115\text{m}}\text{In}$ 336 keV branching ratio	0.459 $\pm$ 0.2% s.	0.459 $\pm$ 0.2% s.
Indium mass number	114.82	114.82
^{115}In isotopic abundance	0.9572	0.9572
K-factor ^(c)	6.567E-21 $\pm$ 0.5% r. $\pm$ 1.2% s.	6.559E-21 $\pm$ 0.4% r. $\pm$ 1.0% s.
Best value of K-factor (c)	6.563E-21 $\pm$ 0.4% r. $\pm$ 1.0% s.	

(a) The K-factor is defined by the relation $r = K S_{\infty}$ where r is the reaction rate ($\text{sec}^{-1}/\text{nucleus}$) and S_{∞} is the specific saturation signal of the instrument (cps/g, at end of irradiation of infinite duration). See text.

(b) Foil to spectrometer distance is respectively 0.55 cm for shelf 0; 5.57 cm for shelf 5.

(c) For foils diam. 19 mm, thickness 0.127 mm on shelf 4 of the transportable spectrometer). (c') Measurements documented in Table NBS-7.

(d) Mass attenuation coefficient: $0.1335 \text{ cm}^2/\text{g}$.

(e) Foil thickness 0.127 mm.

Table NBS-7

Details of the Measurement of Calibration Ratios Between the Transportable
3" x 3" NaI(Tl) Gamma-ray Spectrometer and the CEN-SCK 3" x 3" NaI(Tl) Reference
Spectrometer for the Reaction $^{115}\text{In}(n,n')^{115\text{m}}\text{In}$ (for foils diam. 19 mm, thickness 0.127 mm)

Foil (a)	Transportable spectrometer (shelf 4) (b) count rate(*)	Reference spectrometer (shelf 0) (c) count rate(*)	Calibration ratio	Reference spectrometer (shelf 5) (c) count rate(*)	Calibration ratio
Run 1 Position 1	5909	22035	0.2682	4144	1.426
3	5841	21940	0.2662	4141	1.410
5	5924	22186	0.2670	4170	1.421
Run 2 Position 1	5935	22113	0.2684	4169	1.424
3	5860	22070	0.2655	4151	1.412
5	5950	22171	0.2684	4190	1.420
			Mean: 0.2673 $\pm$ 0.4% r.		
				Mean: 1.419 $\pm$ 0.3% r.	

(a) Foils have been irradiated in the Mol Cavity Uranium-235 Fission Spectrum.

(b) Shelf 0 has also been calibrated but was used only on very weakly activated foils from the ISNF background irradiations; the ratio of efficiencies for shelf 0 relative to shelf 4 is $4.000 \pm 0.3\%$ r., uncorrected for differences in gamma-ray self-absorption and geometry.

(c) See footnote (b), Table NBS-6.

(*) These are specific saturation count rates.

Table NBS-8

Determination of the $^{197}\text{Au}(n,\gamma)^{198}\text{Au}$ K-factor^(a)
 For the Transportable 3" x 3" NaI(Tl) Gamma-ray
 Spectrometer Relative to the CEN-SCK 3" x 3"
 Reference NaI(Tl) Spectrometer

Type of data	Data and uncertainties
Calibration ratio ^(b)	1.310 $\pm 0.2\%$ r. $\pm 0.5\%$ s.
Photopeak efficiency of reference spectrometer (shelf 5)	0.0404 $\pm 0.5\%$ r. $\pm 1.2\%$ s.
Geometry correction ^(b)	0.9940 $\pm 0.1\%$ s.
Gamma-ray self-absorption ^(c)	a_γ
^{198}Au 412 keV branching ratio	0.9548 $\pm 0.1\%$ s.
Coincidence summing correction	1.000 $< 0.05\%$ s.
Gold mass number	196.967
K-factor	$6.511\text{E} - 21/a_\gamma$ $\pm 0.6\%$ r. $\pm 1.3\%$ s.

(a) Same definition as in footnote^(a), Table NBS-6 .

(b) For foils diam. 19 mm, thickness ≤ 0.127 mm on shelf 4 of the transportable spectrometer, relative to shelf 5 of the reference spectrometer.

(c) Specified separately in Table NBS-18, as gold foil thickness vary.

Table NBS-9

Details of the Measurement of the Calibration
 Ratio Between the Transportable 3" x 3" NaI(Tl)
 Gamma-ray Spectrometer and the CEN-SCK 3" x 3"
 NaI(Tl) Reference Spectrometer for the Reaction
 $^{197}\text{Au}(n,\gamma)^{198}\text{Au}$

Foil specification ^(a)	Calibration ratio ^{(b) (d)}
5% Au/Al alloy diam. 19 mm, thickness 0.1 mm	1.310 $\pm 0.2\%$ r., typ.
	1.310
	1.307
	1.311
	1.305
	1.325
	1.321
	1.318
	1.319
	1.312
Pure gold diam. 3 mm, thickness 0.127 mm	1.297
	1.306
	1.295
	1.307
Mean	1.310 $\pm 0.5\%$ s. ^(c) $\pm 0.2\%$ r.

Footnotes

- (a) Foils have been irradiated in the Mol Rotating Graphite Plug Thermal Flux Neutron Field.
- (b) Every figure corresponds to a different foil and every block of data to a different irradiation.
- (c) The uncertainty accounts for the slight systematic trend in the ratio, depending on the type of foil used.
- (d) The calibration ratio is the ratio of the count rate observed by means of the transportable spectrometer (shelf 4) to the count rate observed by means of the reference spectrometer (shelf 5).

Table NBS-10

Mass Assay of 5% Au/Al Alloys
By Activation Analysis^(a): Raw Data

Specific count rate ratio ^{(b)(c)} alloy foil/gold foil ^(d)	Deviation from the mean (1.138 $\pm$ 0.2% r.)
1.141 $\pm$ 0.2% r., typ.	+0.26%
1.147	+0.79%
1.142	+0.35%
1.144	+0.53%
1.144	+0.53%
1.135	-0.26%
1.136	-0.18%
1.142	+0.35%
1.141	+0.26%
1.144	+0.53%
1.133	-0.44%
1.130	-0.71%
1.133	-0.44%
1.132	-0.53%
1.141	+0.26%
1.130	-0.71%
1.134	-0.35%
1.138	0.0%
1.135	-0.26%
1.145	+0.61%

Footnotes

- (a) Irradiations in the Mol Rotating Graphite Plug Thermal Neutron Field.
- (b) Ratio of count rates divided by nominal foil weights, accepting a 5.00% weight content of gold in the alloy. Counting has been done at 5 cm from the 3" x 3" NaI(Tl) reference gamma-ray spectrometer.
- (c) Each block of data corresponds to different irradiations.
- (d) Foil sizes are: pure gold-diam. 3mm, thickness 0.127mm.
alloy- diam. 19mm, thickness 0.1mm.

Table NBS-11

U-238 Observed Fission Rates and Inferred U-235 Fission Rates

Run	Observed Fission Rate Corrected for losses and background ^(a)	In-situ U-238 fission rate per nucleus	Inferred U-235 fission rate per nucleus
	(cps)	(sec ⁻¹)	(sec ⁻¹)
ISNF 7	165.8 ± 1.1%	9.357 E -17	10.09 E -16 ± 1.7%
ISNF 8	175.8 ± 1.1%	9.922 E -17	10.70 E -16 ± 1.7%
ISNF 9	176.1 ± 1.1%	9.938 E -17	10.72 E -16 ± 1.7%

NBS-43

Run	U-235 Fission Rate per Nucleus per cps of monitor
ISNF 7	8.54 E -18 ±1.7%
ISNF 8	8.47 E -18 ±1.7%
ISNF 9	8.52 E -18 ±1.7%
Average Runs 7,8,9	8.51 E -18 ±1.96% ^(b)

(a) Losses: Extrapolation to zero (ETZ) pulse-height (2.1 ± 0.5)%
 Complete fragment absorption (3.8 ± 0.9)%
 Dead Time Correction negligible <.04%
 Background: In ISNF with U-235 driver disks removed (2.8 ± .3)%
 Fission in other isotopic constituents <.002%

(b) Error Factors
 Residual ETZ and Fragment Absorption ±0.6%
 Random error ±0.4%
 Background ±0.3%
 U-238/U-235 In Situ σ_f ratio ±1.1%
 U-235 reference deposit mass ±1.0%
 Scattering to E < threshold ±0.6%

Table NBS-12

$^{115}\text{In}(n,n')^{115\text{m}}\text{In}$ Saturated Count Rates^(a) in ISNF Normalized
To Run-to-run Fission Chamber Monitor Response^(b)

Run	Distance to ISNF Center (cm) ^(c)				Position Relative to NBS Fission Chamber	
	-4	-2	0	+2	center ^(d)	top ^(e)
ISNF/1 ^(f)				76.22 ^(h)		
2 ^(f) (*)	0.178	0.170		0.179		
3 ^(f)	75.45		75.22	74.75		
4 ^(f)	75.70			75.46		
5 ^(f)	75.63		75.21	75.59		
6 ^(f) (**)	0.443		0.448	0.454		
7 ^(g)					74.01	74.75
8 ^(g)					73.80	74.65
9 ^(g)					74.31	74.61

Footnotes

- (a) Count rates S_{∞} per gram of indium at end of an irradiation of infinite duration, corrected for decay during measurement, for dead-time losses and for residual activation due to background neutrons present while boron curtain is closed and the foils are still in-pile. These count rates apply to foils diam. 19 x 0.127 mm on shelf 4 of the CEN-SCK transportable NaI(Tl) 3" x 3" spectrometer.
- (b) See last column, Table NBS-5 .
- (c) Measured along the ISNF vertical polar axis on which the foils are exposed.
- (d) Foil back-to-back with NBS uranium-238 fissionable deposit.
- (e) Foil on NBS fission chamber top cover.
- (f) The ISNF boron-10 shell is fully closed.
- (g) The ISNF boron-10 shell has access hole to accomodate the NBS fission chamber.
- (h) Correction due to background neutrons is sizeable, 6.5%, because foil remained a long time in-pile after boron curtain was closed, see table NBS- .
- (*) Run with boron curtain closed but fission disk sources in place.
- (**) Run with fission disk sources removed and boron curtain open.

Table NBS-13

$^{115}\text{In}(n,n')^{115\text{m}}\text{In}$ Average Saturated Count Rates^(a)
 In ISNF Normalized to Run-to-run Fission
 Chamber Monitor Response^(b)

1. Boron-10 shell closed and foils on radiometric target holder

Distance to ISNF center ^(c)	Raw data	Data corrected for background neutrons ^(d)	Relative response
+2	75.50	$75.05 \pm 0.8\% \text{ r}$	1.004
0	75.22	$74.77 \pm 0.5\% \text{ r}$	<u>1.000</u>
-4	75.59	$75.14 \pm 0.3\% \text{ r}$	1.005

2. Boron-10 containing NBS absolute fission chamber

Location in NBS chamber	Raw data	Data corrected for background neutrons ^(d)	Relative response
center (back-to-back with ^{238}U deposit)	74.04	$73.59 \pm 0.4\% \text{ r}$	0.984 ^(e)
on top cover	74.67	$74.22 \pm 0.2\% \text{ r}$	0.993 ^(e)

(a) See footnote^(a), Table NBS-12. The average is taken over the different runs documented in Table NBS-12.

(b) See last column, Table NBS-5 .

(c) See footnote^(c), Table NBS-12.

(d) As detected during run ISNF/6, Table NBS-5 .

(e) The ratio $0.993/0.984 = 1.009$ agrees well with similar observations at the center of the Mol-ΣΣ neutron field.

Table NBS-14

$^{197}\text{Au}(n,\gamma)^{198}\text{Au}$ Saturated Count Rates^(a) in ISNF Normalized
To Run-to-run Fission Chamber Monitor Response^(b) for
Different Gold Foil Thickness^(c)

Run	Distance to ISNF Center (cm) ^(d)						
	-4	-2	-1	0	+1	+2	+4
ISNF/1	284.4 [0.01]	317.2 [0.005]		324.5 [0.01]		280.8 [0.127]	276.9 [0.01]
ISNF/2 ^(e)						1.075 [0.127]	
ISNF/3		317.2 [0.01]	329.0 [0.005]	324.6 [0.01]	323.9 [0.005]	280.6 [0.127]	
ISNF/4	290.0 [5%]	322.0 [5%]	325.1 [0.005]	332.9 [5%]	325.6 [0.005]	318.8 [5%]	282.4 [5%]
ISNF/5	279.3 [0.028]	323.4 321.0 321.2 [5% STACK]	314.6 [0.028]	315.2 [0.028]	300.2 [0.06]	277.3 [0.127]	271.7 [0.028]
ISNF/6 ^(f)	1.39 [0.127]	2.13 [0.005]	2.04 [0.028]	1.735 [0.127]	1.89 [0.06]	2.04 [0.01]	1.36 [0.127]

Footnotes

- (a) Count rates S_{∞} per gram of gold at end of an irradiation of infinite duration, corrected for decay during measurement, for dead-time losses, and for residual activation due to background neutrons present while boron curtain is closed and the foils are still in-pile; not corrected for gamma-ray self-absorption, neutron self-shielding, reactor core streaming neutron background (run ISNF/6) and applicable to foils diam. 19 mm on shelf 4 of the CEN-SCK transportable 3" x 3" NaI(Tl) spectrometer.
- (b) See last column, Table NBS-5 .
- (c) Nominal foil thicknesses are given in brackets under every individual result, in mm, except for 0.1 mm thick 5% gold-aluminum alloys, referred to simply as 5%.
- (d) Measured along the ISNF vertical polar axis on which the foils are exposed.
- (e) Background run with boron curtain closed, but fission disk sources in place.
- (f) Background run with fission disk sources removed and boron curtain open.

Table NBS-15

$^{197}\text{Au}(n,\gamma)^{198}\text{Au}$ Reaction Rate Gradient
Along the ISNF Vertical Polar Axis*

Distance to ISNF Center (cm)	Relative Gold Reaction Rate
-4	$0.878 \pm 0.3\%$ typical
-2	0.977
-1	0.992
0	1.000
+1	0.988
+2	0.965
+4	0.855

* See plot of the data, Fig. NBS-4, p. NBS-32.

Table NBS-16

$^{197}\text{Au}(n, \gamma)^{198}\text{Au}$ ISNF Central Saturated Count
 Rates^(a) Corrected for Reactor Leakage Neutron
 Background^(*) and Normalized to Run-to-run
 Fission Chamber Monitor Response

Nominal gold foil thickness	Actual average foil thickness (mg/cm ²)	Run	Location ^(b) (cm)	Count rate ^(c)
5% Al alloy	1.36	ISNF/4	-4	328.1
		"	-2	327.4
		"	0	330.7
		"	+2	328.2
		"	+4	328.1
Stack of 3, 5% Al	4.05	ISNF/5	-2	327.2
0.005 mm	10.2	ISNF/1	-2	322.5
		ISNF/3	-1	329.4
			+1	325.6
		ISNF/4	-1	325.5
			+1	327.3
0.01 mm	18.0	ISNF/1	-4	321.8
			0	322.4
			+4	321.8
		ISNF/3	-2	322.6
			0	322.5

Table NBS-16 Continued

Nominal gold foil thickness	Actual average foil thickness (mg/cm ²)	Run	Location ^(b)	Count rate ^(c)
0.028 mm	50.4	ISNF/5	-4	316.0
			-1	315.0
			0	313.1
			+4	315.7
0.06 mm	112	ISNF/5	+1	301.9
0.127 mm	258	ISNF/1	+2	289.4
		ISNF/3	+2	289.2
		ISNF/5	+2	285.8

(a) See footnote (a) Table NBS-14. Gradient corrections derived from Table NBS-15.

(b) Distance to ISNF center at which the foil was actually exposed.

(c) Data in this column, after correction for gamma-ray self-absorption will be applied, are proportional to gold neutron self-shielding in the ISNF.

(*) Assuming the gradient data of Table NBS-15 applicable to the background responses, which is only approximately true, as the neutron self-shielding of the background component is more pronounced than for the main responses (lower average energy and thus steeper gradient). The background count rates at ISNF center, normalized to the run-to-run monitor, are, under the above assumption: 1.6 ($\pm$ 7%), 1.9, 2.05, 2.1 and 2.2 cps/gr. cps for the nominal gold thickness of 0.127, 0.06, 0.028, 0.01 and 0.005 mm, respectively. They have been gradient-corrected before subtraction from the total responses.

Table NBS-17. $^{197}\text{Au}(n,\gamma)^{198}\text{Au}$ ISNF CENTRAL SATURATED^(a)
 SPECIFIC DISINTEGRATION RATES NORMALIZED
 TO RUN-TO-RUN MONITOR RESPONSE

Gold thickness (mg/cm ²)	^{198}Au disintegration rate ^(b) (dps/gr. cps monitor)
1.36 $\pm$ 1.2%	6548 $\pm 0.7\%$ r $\pm 1.8\%$ s
4.05 $\pm$ 1.2%	6522 $\pm 0.8\%$ r
10.2 $\pm$ <0.1%	6499 $\pm 1.0\%$ r
18.0 $\pm$ <0.1%	6427 $\pm 0.6\%$ r
50.4 $\pm$ <0.1%	6301 $\pm 0.7\%$ r
112 $\pm$ <0.1%	6078 $\pm 0.8\%$ r
258 $\pm$ <0.1%	5885 $\pm 0.9\%$ r

(a) For irradiation of infinite duration.

(b) Errors listed include contributions due to mass assay and K-factor uncertainties.

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NBS - 54

Appendix

Gold Gamma-Ray Self-Absorption for the ^{198}Au 412 keV Transition

The total photon interaction cross section for atom for 411.794 keV gamma rays with metallic gold ($z=79$) is obtained as follows.

Absorption

Photonuclear reactions and electron-positron pair production are negligible.

The K-shell photoelectric absorption cross section is calculated from an empirical formula given in reference [1] (p. 25); the result is 33.07 barn/atom ($\pm 1-2\%$). Contributions from other shells are derived by assuming energy-independence of the ratio z/z_k between the total and K-shell cross sections; one has in this case $z/z_k = 1.217(\pm 3\%)$. Therefore $z = 40.25$ b/atom ($\pm 1.5-2.5\%$).

Scattering

Elastic and inelastic nuclear scattering are negligible. Inelastic incoherent scattering with atomic electrons, e.g. Compton scattering is computed from the KLEIN-NISHINA theory (electron free and at rest) as 24.74 b/atom; included is a small correction (0.04%) for combined radiative escape and double-Compton effects. Corrections for electron binding energy and electron motion are negligibly small. Elastic coherent scattering with atomic electrons, e.g. Rayleigh scattering, is estimated as having a cross section of 3.6 b/atom but is confined to angles less than 10° .

In the geometry of our measurements - disk sources with diameters of 19mm at 4-5cm from a 3" x 3" NaI(Tl) spectrometer the contribution of Rayleigh scattering to gamma-ray attenuation is significantly damaged and, but not negligible

[1] J. H. Hubbell - "Photon Cross Sections, Attenuation Coefficients and Energy Absorption Coefficients from 10 keV to 100 GeV" NSRDS-NBS-29 (August, 1969).

altogether. Our self-absorption calculations are done in the "narrow-beam" approximation and buildup factors are not accounted for; it is estimated that the bias due to such simplified treatment is more or less compensated for by neglecting Rayleigh scattering as well.

Total Cross Section

Therefore the total photon-interaction cross section for ^{198}Au gamma rays with gold is taken as 65.0 b/atom ($\pm 2.5\%$) and the corresponding mass attenuation coefficient as $\mu/\rho = 0.1988 \text{ cm}^2/\text{g}$ ($\pm 2.5\%$).

The resulting gamma-ray self-absorption corrections are tabulated below (Table NBS-18).

Table NBS-18. GOLD GAMMA-RAY SELF ABSORPTION CORRECTIONS

Nominal foil thickness (mm.)	Thickness ^(a) (mg/cm ²)	Self Absorption (α_γ)	Uncertainty
0.127	258	0.9748 ^(b)	$\pm 0.25\%$
0.06	112	0.9889	$\pm 0.1 \%$
0.028	50.4	0.9950	$\pm 0.05\%$
0.01	18.0	0.9982	$\pm 0.01\%$
0.005	10.2	0.9990	$\leq \pm 0.01\%$
5% Au/Al alloy	(c)	0.9987	$\pm 0.01\%$

(a) Ratio of average foil weight to surface.

(b) A value of 0.9760 is obtained if one accepts the nominal thickness and gold density of 19.3 g/cm^3 .

(c) The mixture rule $\mu/\rho (\text{Au } 5\%/\text{Al}) = 0.05 \mu/\rho (\text{Au}) + 0.95 \mu/\rho (\text{Al})$ has been applied.

C. Fission Cross-Section Ratios for U-233, Th-232, and Pu-240 in CFRMF
and Preparation for Absolute Fission Counting at 100 kW

D. M. Gilliam, NBS

J W. Rogers, EG&G

Objectives

To measure the CFRMF spectrum-averaged integral fission cross sections of U-233, Th-232, and Pu-240 relative to U-235 fission.

To remeasure the integral fission cross sections for U-238, Pu-239, and Np-237 relative to U-235, to see if the present CFRMF configuration gives the same results as previous measurements which were made prior to modifications of the assembly for permitting neutron radiography and 100 kW operation.

To provide an active fission chamber for absolute fission counting at 100 kW and to provide a dummy fission chamber for foil activation experiments with identical perturbations of the neutron field.

Summary

The newly measured spectral indices are as follows:

$$\bar{\sigma}_f(\text{U-233})/\bar{\sigma}_f(\text{U-235}) = 1.56 \pm 3\%$$

$$\bar{\sigma}_f(\text{Th-232})/\bar{\sigma}_f(\text{U-235}) = 0.0130 \pm 3\%$$

$$\bar{\sigma}_f(\text{Pu-240})/\bar{\sigma}_f(\text{U-235}) = 0.368 \pm 3\%$$

The re-measured spectral indices $\bar{\sigma}_f(\text{U-238})/\bar{\sigma}_f(\text{U-235})$, $\bar{\sigma}_f(\text{Pu-239})/\bar{\sigma}_f(\text{U-235})$, and $\bar{\sigma}_f(\text{Np-237})/\bar{\sigma}_f(\text{U-235})$ were all found to agree with previously measured and reported values within better than $\pm 1\%$. The direct measurement of the perturbations introduced by the presence of the neutron radiography facility, an in-core dry tube, and the reactor

source were found to be too small to be detected within the precision of the comparisons ($\pm 0.2\%$). Together these spectral index results and direct perturbation measurements show that the neutron spectrum of CFRMF has not been significantly changed as a result of the modifications to the assembly.

Accomplishments and Status

Fission Cross Section Ratios

Over the course of three days of measurements in May, 1978, NBS fission chambers were loaded with 14 different combinations of fissionable deposits and 27 experimental runs were made.

The masses of the fissionable deposits are given in Table NBS-19. The masses for the new reference deposits for U-233, Th-232, and Pu-240 were all obtained by the same single method: absolute alpha counting using half-life data from the literature (U-233 Ref. 3,4; Th-232 Ref. 5, Pu-240 Ref. 6). On the other hand, the masses of reference deposits for U-235, U-238, Pu-239, and Np-237 as given in Ref. 2 represent consensus values which are an average of two or more independent mass assay measurements with stated accuracies that are consistent with the spread of the multiple analyses. Although the conceivable mass assay errors for the new reference deposits are not much larger than the stated errors of the consensus assay data, the reliability of the new reference deposit masses cannot be regarded as comparable until confirming analyses have been performed.

The isotopic composition of the various fissionable materials is given in Table NBS-20. In only two cases is the fission correction for fission in minor isotopic constituents greater than 1%: for batch 28S (depleted uranium) the correction is 1.25% and for batch 40 (Pu-240) the correction is 4.3%.

Table NBS-21 lists deposit designations, count rates, and corrections for the ten runs from which the cross section ratios of Table NBS-22 were derived.

The two runs involving U-233 give U-233/U-235 cross section ratios differing by 3.7%, which is several times larger than the estimated errors in foil masses and counting statistics. In the first of these runs, high noise was indicated by an anomalously high S_L scaler rate (See Reference 2 for fission counting method and nomenclature). The pulse height extrapolation-to-zero-energy (ETZ) correction was accordingly estimated on the basis of normal performance and the result of this run was given relatively low weight in the average. The error estimate was increased due to the discrepancy.

The two measurements of the Th-232/U-235 fission cross section ratio were completely consistent. The higher mass assay errors and the larger uncertainties in the ETZ and self-absorption corrections again resulted in a combined error estimate of 3.0%.

The second run involving Th-232 and U-235 was repeated with the electronics associated with the respective deposits interchanged. The previous result was reproduced well within expected statistical limits.

Only one run measuring the Pu-240/U-235 fission cross section ratio is listed in Table NBS-21. A second run was actually made, but an error was made in recording the deposit label so that the results could not be interpreted unambiguously. Due to lack of a confirming measurement, the error quoted for the Pu-240/U-235 ratio was increased by about 1% above the actual total of the separate error factors listed.

In addition to the newly measured cross section ratios, Table NBS-22 also lists the results for U-238, Pu-239, and Np-237 which had been measured previously. All the new measurements were made with the recently installed radiography facility in place. The U-238/U-235 ratio was also measured with an in-core dry tube in place, then with the tube removed. These runs were made in immediate succession, without moving the fission chamber within the CRMF

test cell. The results agreed within the $\pm 0.2\%$ statistical precision. In like manner, the perturbation of the reactor start-up source was checked and found to be nil within $\pm 0.2\%$ statistical limits.

The previously (1972) measured cross section ratios in Table NBS-22 agree with the 1978 measurements within $\pm 1\%$ in every case.

Table NBS-22 also compares the experimental results with calculated values. The largest discrepancy is in the Th-232/U-235 cross section ratio; this discrepancy of 14% suggests serious error in either the Th-232 cross section in the ENDF/B IV dosimetry file or the Th-232 mass evaluation in the present experiment. In more recent measurements at the BIG-TEN facility (not yet published), a large calculation/experiment (C/E) discrepancy in the same direction was also observed, confirming the CFRMF measurement, but not giving further insight as to the source of the discrepancy.

High Power Operation

An NBS double fission chamber was supplied to EG&G for absolute fission rate measurements at 100 kW. The deposits supplied with the chamber were 25S- β -1-3 (highly enriched uranium, $1.248\mu\text{g}$ U-235 $\pm 1.3\%$) and 28G-1-1-16 (depleted uranium, nominal mass $10\mu\text{g}$). Seven runs were made at high power levels: five at 30 kW and 2 at 100 kW. The chambers continued to operate normally at the higher power levels. The U-235 count rate at 100 kW was about 4300 cps, and the U-238 count rate was about 1700 cps at that power level. The fraction of the count rate in the valley of the U-235 fission pulse height distribution was stable within $\pm 0.15\%$ over the power range 5 kW - 100 kW. The performance of the U-238 chamber and electronics showed about twice that much variation, but still well within acceptable limits. Only once, during one of the 30 kW runs,

was significant noise encountered; on this run the U-238 ETZ indicated more than 1% noise, and the U-235/U-238 count rate ratio dropped by 4%. In addition to the NBS double fission chambers, commercial sealed fission chambers were operated within the CFRMF test volume during these measurements. The commercial fission chamber pulse height distribution was found to be less stable, with significant filling in of the valley region at higher power levels. Nevertheless, these chambers gave adequate stability for run-to-run normalization purposes over the narrow power variations that are generally encountered in CFRMF runs.

A dummy fission chamber of the same general construction as the active NBS chamber was supplied to CFRMF for mounting of activation foils so that field perturbations closely matching those of the active fission chamber could be realized.

Table NBS-19

Fissionable Deposits - Mass Assay Data

Deposit Label	Principle Isotope Mass (μg)	Mass Assay Method*
23-1-2	114.4 $\pm$ 1.5%	RAC(23-1-1)
23-1-1	117.5 $\pm$ 1.5%	Reference Deposit AAC $t_{1/2} = 1.592 \pm .003 \times 10^5 \text{ y}$
32L-5-2	603.5 $\pm$ 2.0%	Reference Deposit AAC $t_{1/2} = 1.41 \pm .01 \times 10^{10} \text{ y}$
40-2-2	144.8 $\pm$ 1.5%	RAC(40-2-1)
40-2-1	146.4 $\pm$ 1.5%	Reference Deposit AAC $t_{1/2} = 6569 \pm 6 \text{ yr}$
25S-01-1	5.13 $\pm$ 1.1%	TFC(25S-2-2)
25S- β -1-3	1.248 $\pm$ 1.3%	TFC(25S-01-1, 25S-2-2)
25K-1-2	133.0 $\pm$ 1.2%	WA(10th Prog. Rept. p. NBS-8)
28S-2-1	247.5 $\pm$ 1.6%	WA(10th Prog. Rept, p. NBS-8) Revised (11th Prog. Rept., Section G)
49I-02-1	16.02 $\pm$ 1.3%	TFC(49I-1-2)
37S-1-2	116.8 $\pm$ 1.8%	RAC(37S-1-1)

*Mass Assay Notes:

TFC(Reference Deposit) Thermal neutron fission counting comparison to reference deposit listed, via intermediate comparison to other deposit listed (if any).

RAC(Reference Deposit) Relative alpha counting comparison to reference deposit listed.

AAC Absolute alpha counting

WA Weighted average

Table NBS-20

Isotopic Composition and Fission Corrections

Batch label and Isotopic Assay (atom percent)	Total Fission Correction and Relative Fission Rate Contributions in the CRMF Spectrum
23 ²³³ U 97.96 ²³⁴ U 1.37 ²³⁵ U 0.07 ²³⁸ U 0.6	0.9964 ± 0.001 1. 0.0029 0.0005 0.0002
32L ²³² Th 100.0	1.
40 ²⁴⁰ Pu 98.35 ²³⁹ Pu 0.671 ²³⁸ Pu 0.017 ²⁴¹ Pu 0.594 ²⁴² Pu 0.368 ²⁴⁴ Pu 0.0005	0.9570 ± 0.006 1. 0.0204 0.0001 0.0213 0.0031 -
25S ²³⁵ U 99.748 ²³⁸ U 0.1261 ²³⁴ U 0.0608 ²³⁶ U 0.0652	0.99967 ± 0.00015 1. 0.000061 0.000201 0.000071
25K ²³⁵ U 99.89 ²³⁸ U 0.053 ²³⁴ U 0.035 ²³⁶ U 0.025	0.99983 ± 0.00007 1. 0.000026 0.000116 0.000027
28S ²³⁸ U 99.94 ²³⁵ U 0.061 ²³⁴ U < 0.001 ²³⁶ U < 0.005	0.9875 ± 0.0007 1. 0.0125 0.000065 0.000100
49I ²³⁹ Pu 99.10 ²⁴⁰ Pu 0.883 ²⁴¹ Pu 0.010 ²⁴² Pu 0.006 ²³⁸ Pu < 0.005	0.9969 ± 0.0005 1. 0.00297 0.0001 0.000016 0.000014
37S ²³⁷ Np 99.98 ²³⁹ Pu 0.02	0.9994 ± 0.0005 1. 0.0006

Table NBS-21

Observed Fission Cross Section Ratios, Corrections, and Uncertainties

Run Date/No.	Deposit Labels	S_U Ratio ± Statistical and Dead-time errors	ETZ Corrections (to S_U Counts) ± error in ratio	Fragment Absorption Correction ± error in ratio	Correction for Fission in Minor Isotopic Constituents	Cross Section Ratio ± combined errors (including mass errors)
5-25-78 No. 2	23-1-1 25S-01-1	$\frac{2467.3}{71.18}$ ±0.5%	$\frac{1.0124}{1.0026^*}$ ±2.0%*	$\frac{1.0064}{1.0003}$ ±0.2%	$\frac{0.9964}{0.99967}$ ±0.1%	1.519 ±2.8%
5-26-78 No. 1	23-1-2 25K-1-2	$\frac{1842.7}{1347.4}$ ±0.1%	$\frac{1.0103}{1.0072}$ ±0.5%	$\frac{1.0062}{1.0072}$ ±0.2%	$\frac{0.9964}{0.99967}$ ±0.1%	1.575 ±2.0%
5-24-78 No. 7	32L-5-2 25S-01-1	$\frac{1285.1}{896.7}$ ±0.2%	$\frac{1.0444}{1.0024}$ ±1.5%	$\frac{1.0329}{1.0003}$ ±0.9%	$\frac{1.0}{0.99967}$ ±0.02%	0.0129 ±2.9%
5-26-78 No. 3	32L-5-2 25S-β-1-3	$\frac{2317.7}{391.4}$ ±0.2%	$\frac{1.0381}{1.0012}$ ±1.5%	$\frac{1.0329}{1.0}$ ±0.9%	$\frac{1.0}{0.99967}$ ±0.02%	0.0129 ±3.0%

*Nominal value used due to noisy S_L scaler. Experimental $S_L/S_U = 1.0088$ vs. nominal 1.0013.
Large uncertainty assigned due to discrepancy.

Table NBS-21 (continued)

Observed Fission Cross Section Ratios, Corrections, and Uncertainties

Run Date/No.	Deposit Labels	S_U Ratio ± Statistical and Dead-time errors	ETZ Corrections (to S_U Counts) ± error in ratio	Fragment Absorption Correction ± error in ratio	Correction for Fission in Minor Isotopic Constituents	Cross Section Ratio ± combined errors (including mass errors)
5-26-78 No. 4 Repeat of No. 3 w/electronics for deposits interchanged	$\frac{32L-5-2}{25S-\beta-1-3}$	$\frac{2317.3}{392.3}$ ±0.4%	$\frac{1.0366}{1.0012}$ ±1.5%	$\frac{1.0329}{1.0}$ ±0.9%	$\frac{1.0}{0.99967}$ ±0.02%	0.0129 ±3.0%
5-26-78 No. 2	$\frac{40-2-2}{25K-1-2}$	$\frac{832.94}{2041.58}$ ±0.2%	$\frac{1.0052}{1.0068}$ ±0.2%	$\frac{1.0075}{1.0073}$ ±0.2%	$\frac{0.9570}{0.99967}$ ±0.6%	0.366 ±3%*
5-24-78 No. 2	$\frac{28S-2-1}{25S-01-1}$	$\frac{1023.9}{449.0}$ 0.35%	$\frac{1.0202}{1.0028}$ ±0.43%	$\frac{1.0135}{1.0003}$ ±0.35%	$\frac{0.9875}{0.99967}$ ±0.07%	0.0487 ±2.1%
5-24-78 No. 5	$\frac{28S-2-1}{25S-01-1}$	$\frac{1027.6}{449.6}$ ±0.2%	$\frac{1.0196}{1.0024}$ ±0.43%	$\frac{1.0135}{1.0003}$ ±0.35%	$\frac{0.9875}{0.99967}$ ±0.07%	0.0488 ±2.0%

*The error estimate has been increased over the known combined error factors due to the lack of a second, confirming measurement for this initial Pu-240 measurement.

Table NBS-21 (continued)

Observed Fission Cross Section Ratios, Corrections, and Uncertainties

Run Date/No.	Deposit Labels	S_U Ratio ± Statistical and Dead-time errors	ETZ Corrections (to S_U Counts) ± error in ratio	Fragment Absorption Correction ± error in ratio	Correction for Fission in Minor Isotopic Constituents	Cross Section Ratio ± combined errors (including mass errors)
5-24-78 No. 3	49I-02-1 25S-01-1	$\frac{1590.0}{447.8}$ ±0.2%	$\frac{1.0026}{1.0024}$ ±0.1%	$\frac{1.0008}{1.0003}$ ±0.1%	$\frac{0.9969}{0.99967}$ ±0.05%	1.154 ±1.7%
5-24-78 No. 4	37S-1-2 25S-01-1	$\frac{1763.3}{224.3}$ ±0.22%	$\frac{1.0074}{1.0032}$ ±0.2%	$\frac{1.0064}{1.0003}$ ±0.35%	$\frac{0.9994}{0.99967}$ ±0.05%	0.352 ±2.2%

Table NBS-22
Spectrum-Averaged
Fission Cross Section Ratios in CFRMF

Nuclides	Measured Fission Cross Section Ratio**	Calculated Fission Cross Section Ratio*	Calculation/ Experiment
$^{233}\text{U}/^{235}\text{U}$	1.56 $\pm$ 3.0%	1.47	0.94
$^{232}\text{Th}/^{235}\text{U}$	0.0130 $\pm$ 3.0%	0.0110	0.85
$^{240}\text{Pu}/^{235}\text{U}$	0.368 $\pm$ 3.0%	0.373	1.014
$^{238}\text{U}/^{235}\text{U}$ (1978)	0.0491 $\pm$ 2.0%	0.0482	0.982
$^{238}\text{U}/^{235}\text{U}$ (1972)	0.0490 $\pm$ 1.4%		
$^{239}\text{Pu}/^{235}\text{U}$ (1978)	1.154 $\pm$ 1.7%	1.117	0.968
$^{239}\text{Pu}/^{235}\text{U}$ (1972) [†]	1.145 $\pm$ 1.5%		
$^{237}\text{Np}/^{235}\text{U}$ (1978)	0.354 $\pm$ 2.2%	0.363	1.025
$^{237}\text{Np}/^{235}\text{U}$ (1972) [†]	0.354 $\pm$ 2.3%		

* The calculated values were obtained by means of the NBS DETAN code, using the published CFRMF spectrum from Reference 1.

† The 1972 values are from Reference 2; except the ^{238}U value which has been adjusted in accordance with the U-238 mass scale change reported in the 11th Progress Report, Sect. G.

** A correction of 0.6% was applied to the fission rates of the threshold detectors to compensate for scattering below the threshold.

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OAK RIDGE NATIONAL LABORATORY

NEUTRON DOSIMETER MATERIALS DEVELOPMENT AND CHARACTERIZATION
FOR ILRR, HEDL, AND ANL DOSIMETRY PROGRAMS
H. L. Adair

OBJECTIVE

Preparation and accurate characterization of neutron dosimetry samples containing high purity, isotopically enriched fissile and stable materials will be performed by the Isotope Research Materials Laboratory of the ORNL Solid State Division. Archive samples and inventories of each isotopically enriched material will be maintained (as allocated by the Department of Energy Division of Research) and dispensed exclusively in support of these various programs.

SUMMARY

Since 1971 IRML has provided more than 5000 vanadium encapsulated dosimeters for use in reactor core characterization, materials compatibility studies, and reactor surveillance operations. Much of the success of the program can be directly attributed to the following:

1. Formation of ceramic oxide wires for many of the pure and dilute dosimeter materials.
2. Refinement of existing analytical techniques to characterize properly the dosimeter materials.
3. Establishment of quality control procedures for all phases of dosimeter fabrication.

ACCOMPLISHMENTS AND STATUS

The IRML dosimeter program was begun in 1971 to provide samples that could be used to characterize the neutron environment in reactor core and reflector regions. The information obtained (neutron energy spectrum, flux, and fluence) from the use of various sets of neutron dosimeters is essential in evaluating fuel compatibility and fuel burnup. The program was initiated

in support of fast breeder reactor technology, however, in the past several years an increasing number of the dosimeters are being used in light water reactor surveillance operations. In the future, it is anticipated that a large number of the dosimeters will be used in support of fusion technology.

The most significant contribution to the IRML dosimeter program has been the development and production of ceramic oxide wire containing fissile and many nonradioactive materials. This development, along with the types of dosimeter materials provided and the status of the characterization of these materials, is described in a paper presented at the Third ASTM-Euratom Symposium on Reactor Dosimetry held in Ispra, Italy.⁽¹⁾

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