ASTM INTERNATIONAL
Selected Technical Papers

Reactor Dosimetry

16th International Symposium

STP 1608

Editors:
Mary Helen Sparks
K. Russell DePriest
David W. Vehar

ASTM INTERNATIONAL

SELECTED TECHNICAL PAPERS
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Reactor Dosimetry: 16th International Symposium

ASTM STOCK #STP1608
DOI: 10.1520/STP1608-EB

ASTM International, 100 Barr Harbor Drive, PO Box C700, West Conshohocken, PA 19428-2959
Printed in the U.S.A.

Library of Congress Cataloging-in-Publication Data

Names: International Symposium on Reactor Dosimetry (16th : 2017 : Santa Fe, N.M.), author. | Sparks, Mary Helen, editor. | DePriest, K. Russell (Kendall Russell), editor. | Vehar, David W., editor. | ASTM International Committee E10 on Nuclear Technology and Applications, sponsoring body.

Title: Reactor dosimetry: 16th international symposium / editors, Mary Helen Sparks, K. Russell DePriest, David W. Vehar.

Description: West Conshohocken, PA : ASTM International, [2018] | Series: Selected technical papers ; STP1608 | "Presented at a symposium held May 7-12, 2017, in Santa Fe, New Mexico, USA. The symposium was jointly sponsored by ASTM International Committee E10 on Nuclear Technology and Applications and Subcommittee E10.05 on Nuclear Radiation Metrology, and the European Working Group on Reactor Dosimetry (EWGRD)"--Foreword. | "ASTM Stock #STP1608." | Includes bibliographical references and index.

Identifiers: LCCN 2018014410 (print) | LCCN 2018014778 (ebook) | ISBN 9780803176621 (ebook) | ISBN 9780803176614 (pbk.)

Subjects: LCSH: Nuclear reactors--Materials--Effect of radiation on--Congresses. | Radiation dosimetry--Congresses. | Nuclear power plants--Safety measures--Congresses.

Classification: LCC TK9185.A1 (ebook) | LCC TK9185.A1 I578 2017 (print) | DDC 621.48/35--dc23

LC record available at https://lccn.loc.gov/2018014410

ISBN: 978-0-8031-7661-4

Peer Review Policy

Each paper published in this volume was evaluated by two peer reviewers and at least one editor. The authors addressed all of the reviewers' comments to the satisfaction of both the technical editor(s) and the ASTM International Committee on Publications.

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Citation of Papers

When citing papers from this publication, the appropriate citation includes the paper authors, "paper title," STP title, STP number, book editor(s), ASTM International, West Conshohocken, PA, year, page range, paper doi, listed in the footnote of the paper. A citation is provided on page one of each paper.

Printed in Mayfield, PA
November, 2018

Foreword

THIS COMPILATION OF Selected Technical Papers, STP1608, *Reactor Dosimetry: 16th International Symposium*, contains peer-reviewed papers that were presented at a symposium held May 7–12, 2017, in Santa Fe, New Mexico, USA. The symposium was sponsored by ASTM International Committee E10 on Nuclear Technology and Applications, ASTM Subcommittee E10.05 on Nuclear Radiation Metrology, and the European Working Group on Reactor Dosimetry.

We thank the additional sponsorship of this symposium by:
American Nuclear Society–Reactor Physics Division,
Bruker Biospin,
Canberra,
Sandia National Laboratories, and
Thermo Scientific.

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Overview

The Sixteenth International Symposium on Reactor Dosimetry (ISRD) is the latest in a series of symposia held approximately every three years. It is a forum for the exchange of information on data bases, benchmark studies, techniques, and standardization of radiation metrology and regulatory information. This is of interest and value to those people involved in reactor dosimetry including researchers, educators, manufacturers, regulators, and the people from industry and utilities.

The papers in this volume were presented in Santa Fe, New Mexico (USA) from May 7th to 12th, 2017. The Symposium centered on the new work in the following fields:

- Research/Test Reactor and Accelerator Dosimetry
- Cross Sections, Nuclear Data, and Uncertanties
- Reactor Surveillance and Plant Life Extension
- Adjustments, Intercomparisons, and Benchmarks
- Experimental Techniques
- Transport Calculations

Two Keynote addresses opened the Plenary Session of the meeting. Keith Penny, Idaho National Laboratory, presented "The Advanced Test Reactor: A Bright Irradiation Future" which covered the current facility's potential upgrades and new configurations. Dr. Cinzia Da Via, University of Manchester (UK), presented "Radiation Detectors and Imaging Technologies" which reviewed the development of radiation instrumentation. This led to an overview of some of the modern technologies used in radiation imaging.

Dr. Denise Neudecker, Los Alamos National Laboratory, presented a tutorial "Shedding Light on Evaluated Nuclear Data Uncertainties" that explored the process of evaluating nuclear data and discussed how the choices made by the evaluators influence uncertainties. Many participants took the opportunity to discuss questions and share comments with Dr. Neudecker throughout the week.

Attendees presented papers in both poster and oral sessions. Informal, round-table workshops provided an opportunity to further explore issues presented. Summaries of the workshops are included.

The editors wish to thank the authors and organizing committees for making the Sixteenth International Symposium on Reactor Dosimetry a success. We would also like to thank the ASTM staff involved with the preparation of the symposium and those who supported the review process making the publication of this volume possible.

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Past Symposia

1975	Petten, The Netherlands	EUR 5667
1977	Palo Alto, California, USA	NUREG CP 0004
1979	Ispra, Italy	EUR 6813
1982	Gaithersburg, Maryland, USA	NUREG CP 0029
1984	Geestacht, Germany	EUR 9869
1987	Jackson Hole, Wyoming, USA	ASTM STP1001, ISBN 978-0-8031-1184-4
1990	Strasbourg, France	EUR 14356
1993	Vail, Colorado, USA	ASTM STP1228, ISBN 978-0-8031-1899-6
1996	Prague, Czech Republic	World Scientific, ISBN 981-02-3346-9
1999	Osaka, Japan	ASTM STP1398, ISBN 978-0-8031-2884-2
2002	Brussels, Belgium	World Scientific, ISBN 981-238-448-0
2005	Gatlinburg, Tennessee, USA	ASTM STP1490, ISBN 978-0-8031-3412-6
2008	Akersloot, The Netherlands	World Scientific, ISBN 981-4271-10-1
2011	Bretton Woods, New Hampshire, USA	ASTM STP1550, ISBN 978-0-8031-7536-5
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Symposium Organization: ASTM/EWGRD

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Tutorial Summary

"Shedding Light on Evaluated Nuclear Data Uncertainties"
Denise Neudecker (Los Alamos National Laboratory)

The "Shedding Light on Evaluated Nuclear Data Uncertainties" tutorial by Dr. Denise Neudecker concluded the first day of the technical program on May 8, 2017. The tutorial began with an overview and introduction to the evaluation of the nuclear data. In an effort to focus the discussion for participants, Dr. Neudecker chose to examine how aspects of the evaluation and uncertainty quantification of the ^{239}Pu prompt fission neutron spectrum (PFNS) can affect the effective multiplication factor (k_{eff}) of the Jezebel plutonium critical assembly and its associated uncertainty. The overview portion highlighted the fact that an evaluator must make ***subjective*** choices about: 1) the various experimental data and their uncertainties, 2) modeling parameters, parameter and model defect uncertainties, and 3) the evaluation algorithm. As these choices flow through the evaluation process, they impact the final nuclear data evaluation and the uncertainties that are reported for the evaluation.

After providing a base for the statistical methods and techniques used by evaluators, the discussion moved to experimental uncertainties. The EXFOR database was shown to be a primary resource in that it serves as a repository for both experimental data and its uncertainty. Dr. Neudecker demonstrated that simplified estimates of experimental uncertainty and covariance can significantly impact an integral quantity such as the k_{eff} of a critical assembly. After demonstrating the need for detailed uncertainty estimates, the tutorial participants discussed how evaluators can identify and accommodate discrepant data as part of the evaluation process. After concluding the discussion of experimental uncertainty with a caution regarding the need to appreciate how neglected correlations can impact the final evaluations, the subject of model uncertainties was brought to the floor for discussion.

The typical sources of model uncertainty include the parameter uncertainties, model defect uncertainties, numerical uncertainties, and emulation uncertainties. An example was provided to demonstrate that the choice of the physics model and presence of any neglected parameters in the model can significantly affect benchmark simulations for integral quantities such as k_{eff}. The participants learned that correlations between model parameters are usually neglected, and then, Dr. Neudecker demonstrated the potential significant effects of those ignored correlations using as an example the determination of the Jezebel k_{eff} value. In the final discussions about model uncertainty, the participants were shown that ignoring or simplifying model defect uncertainty can result in significantly under-predicted uncertainties in the nuclear data evaluation.

As the tutorial concluded, the participants examined the suite of algorithms that are frequently used to evaluate nuclear data from experimental data, model values, and associated uncertainties. The algorithms include generalized least squares, Unified Monte Carlo (of different flavors), Backward-forward Monte Carlo, and Gaussian processes. The tutorial highlighted those cases where the generalized least squares approach should not be used for nuclear data evaluations and how a few of these choices can impact evaluated data and the Jezebel k_{eff} value. Also, the methodologies enforcing typical nuclear data constraints (such as the constraints that all reaction cross sections sum to the non-elastic cross section, the elastic and non-elastic cross sections sum to the total cross section, etc.) were briefly covered and it was shown how they impact evaluated uncertainties. The tutorial concluded by summarizing the various factors that could impact the nuclear data evaluation and its uncertainties.

The two-hour tutorial was highlighted by interactions between the participants and Dr. Neudecker. Feedback from symposium participants during the week in Santa Fe and after the conclusion of the symposium was extremely positive. The editors believe that symposium participants gained a new appreciation for the effort involved in producing evaluated nuclear data and uncertainties for use by the radiation metrology community.

Workshop Summaries

Igor Remec (Oak Ridge National Laboratory), ASTM Workshop Chair
Dean Thornton (Amec Foster Wheeler), EWGRD Workshop Co-Chair

Six workshops were held during the symposium covering a variety of topics. These were jointly organized by two co-chairs, one ASTM representative and one EWGRD representative. The co-chairs defined the scope of each workshop and led the participants in informal discussions. The summary of each workshop follows.

Adjustment Methods, Cross Sections, and Uncertainty Quantifications
Patrick Griffin (Sandia National Laboratory) and Dean Thornton (Amec Foster Wheeler)

The workshop on Adjustment Methods, Cross Sections, and Uncertainty Quantifications was very well attended with 30 delegates from eight countries and represented a wide spectrum of interests including radiation transport calculations and uncertainty quantification, adjustment methods, nuclear data evaluation, and measurement techniques.

The meeting began with a discussion of the benefits and requirements for performing a spectral adjustment. It was noted that there is often only a few percent change in the evaluated iron DPA metric when using the calculated versus the adjusted spectrum. However, there is usually a significant reduction in the uncertainty. The workshop acknowledged the potential of new state-of-the-art techniques being applied in neutron spectrum adjustment. These techniques, which go beyond the traditional least squares and iterative methods, include the use of Genetic Algorithms, Maximum Entropy techniques, and Total Monte Carlo approaches. Despite a series of spectrum adjustment inter-comparisons conducted in the 1980s, little has been done since to validate the consistency of adjusted spectra used by the community. Roberto Capote (IAEA NDS) indicated that a new neutron spectrum adjustment inter-comparison is scheduled for the 2018 timeframe and that volunteers are being sought to participate in the exercise. The Workshop encouraged the widespread participation by the community in this exercise.

Issues raised as important at the Workshop included:

- The inconsistent availability of details for uncertainty contributions.
- The availability of a complete uncertainty characterization for the *a priori* spectra used in an adjustment.
- The availability of reliable neutron energy-dependent fission yield data.

Users are warned of potential dangers in selecting individual nuclides from different cross section libraries. Some cross section libraries are tuned for particular applications (e.g., reactor criticality calculations), and individual cross section evaluations may contain compensations designed to preserve integral constraints. This consideration can also apply to reaction specific nuclear decay data. That is, the emission probabilities and decay half-lives should be consistent with the cross section evaluation.

Some recommendations came out of the Workshop, including:

- The IRDFF dosimetry cross section library is considered to be state-of-the-art and is recommended for most dosimetry applications.
- The community welcomes the advent of the TENDL library which is able to provide comprehensive covariances for all of its calculated nuclear data.
- A call was made for experimentalists to better record uncertainty details in EXFOR archives.

The importance of nuclear data, beyond the basic consideration of cross sections, was also emphasized by Workshop participants. The reliability of existing ^{103}Rh(n, n') measurements was discussed. Activation measurements using this dosimeter are useful wherever they are available. This reaction has been used extensively in historic benchmark experiments.

There is much interest in work being carried out on advancing the state of calculational tools to support the 3D characterization of neutron fluences throughout nuclear power plants. The regions of interest included above and below the core, as well as within the reactor cavity. The potential for this work to be incorporated into a Monte Carlo reference standard is welcomed by the community.

Test & Research Reactors

Michael Flanders (White Sands Missile Range) and Pavel Frajtag (Ecole Polytechnique Federale de Lausanne)

The Workshop opened with an introduction of the participants emphasizing background, experience and general interests. Attendees represented a range of experience with considerable insight into the development of nuclear dosimetry, research needs and the history of the research efforts. There was representation by facility operations staff, facility users, and educators. There was a welcome infusion of new energy into the group.

The Swiss are using the CROCUS Zero-Power reactor for several new and unique experiments. The facility recently transformed from a teaching reactor to a research facility. It is particularly suited for physics experiments such as vibrating fuel rods and other novel experiments due to its safety basis. It was also noted that when performing materials or sensor irradiations it can be simpler to instrument a test, and may be less expensive to perform certain tests in a zero-power reactor.

The decline in available research reactors is continuing. The question was raised about what were the responses of the supporting agencies to the decline. DOE-NE has programs to support university research reactors in the area of fuel acquisition, regulatory support, licensing, and inspection assistance. There is also Research Reactor infrastructure support available. In addition to the normal notion of infrastructure such as facilities, equipment and so on, we also interpret infrastructure to include a strong pool of scientific expertise and experience in operations, maintenance and support functions such as dosimetry and reactor physics. Also important is a strong program of inter-laboratory comparison of dosimetry techniques and support for the National Standards Laboratories to which the facilities maintain traceability.

The future of the MYRRHA facility was reviewed. It was mentioned the Belgian government decision regarding continuation and construction is expected near the end of 2017. The phased timeline then continues with the first phase: the construction of a 100 MeV accelerator begun in 2016 and finalized in 2024 with operation in 2025. The accelerator will then be upgraded to 600 MeV and the reactor constructed. This timeline would allow commissioning of MYRRHA to begin in 2030 with operation expected in 2034.

There was a brief mention of the continuing effort to convert facilities to low-enriched fuel. The panel notes that not all facilities and programs are amenable to the transition to LEU. There is the potential for loss of some capabilities and availability of unique features.

It was suggested that an update of a world-wide research reactors database should be undertaken focusing on research activities and capabilities of the existing facilities. It was suggested that perhaps a review paper summarizing the latest research activities at all available facilities could be prepared for the next ISRD to bring the research reactors programs into a more prominent and visible position.

The recent Swiss referendum about nuclear energy strategy was discussed. The positive results of the referendum and public acceptance of the nuclear industry provided a renewed hope for the support of nuclear research. We can hope that other countries will follow this successful example.

Transport Calculations, Benchmarks, and Intercomparisons
Igor Remec (Oak Ridge National Laboratory) and Vladimir Smutney (SKODA)

The workshop on Transport Calculations, Benchmarks and Intercomparisons was attended by 31 very active participants from eight countries. The participants had expertise and interests in different areas including radiation transport (neutron and gamma rays), measurements, sensitivity and uncertainly analyses, spectrum adjustment, methods development, and nuclear regulatory policy. Lively discussion addressed numerous topics which are summarized below.

Transport codes status: 3D codes are gaining dominance; both based on the deterministic and the Monte Carlo (MC) methods. This is observed in not only research institutions but also in industry. There is considerable continuing effort in code development for widely used codes such as MCNP6, SERPENT, MCBEND, and newer and improved versions are becoming available on regular basis. Most of the transport codes in widespread use allow parallel processing, and this is becoming a prevalent mode of analysis. Computing resources are generally not a serious constraint anymore. Hybrid methods combining deterministic and MC simulations, such as ADVANTG, are gaining popularity. Some newer parallel deterministic codes do not support cylindrical coordinates which is a very desirable feature for pressure vessel (PV) fluence calculations. However, there are some that do, for example PARTISAN and RAPTOR-M3G. Another desirable feature in transport codes is variable mesh. The variable mesh feature facilitates modeling of complex geometries while preserving memory requirements.

Advancements in the 3D transport codes are particularly needed and welcome since nuclear power plants life extensions to 60 years, and in near future to 80 years, require analysis of the areas other than the typical belt-line regions, such as regions above and below the active fuel region, and PV support structures and nozzles. The US NRC is supporting exploratory work on suitable radiation transport methods, issues of interest, and regulatory requirements related to these regions.

It was a consensus opinion of the workshop participants that radiation transport codes should be designed for affordable clusters, not supercomputers, at least for the user community represented and attending the workshop. It was also suggested that radiation transport codes should be made open source; however, discussion showed that it was a consensus among the participants that the current distribution (through RSICC in US and NEA Data Bank in Europe) is adequate.

Thanks to the ever-improving computer resources, sensitivity and uncertainty analyses are becoming more affordable. For example, a Total MC technique was used to determine a neutron spectrum covariance matrix for an irradiation experiment at the High Flux Reactor in Petten. The preliminary results were presented and further work is underway. However, it was also felt that the current regulatory approach (in the US), which attributes fixed uncertainties to the reactor pressure vessel fluences for PV surveillance programs may need to be modified to encourage efforts to reduce uncertainties.

Significant progress was reported in the automatic conversion of CAD models into models suitable for analysis with radiation transport codes; however, further improvements are necessary.

It was pointed out that web-based resources and community forums are underutilized. It was suggested that RSICC should maintain user forums for popular codes, and that the Reactor Dosimetry website should host a reactor dosimetry community forum for information exchange.

Mixed Field Dosimetry

Jianwei Chen (Westinghouse) and Vit Klupák (Research Center Rez)

A total of 18 people attended this Workshop from national labs, universities, research reactor facilities, standards laboratories, nuclear power plants, and nuclear power vendors. In the reactor dosimetry community, the most examined mixed field is generally the neutron and gamma mixed field. Many times, a dosimetrist wants to perform passive integrated measurements, active real-time measurements, and energy spectral measurements for both neutrons and gammas.

In the non-reactor test environment or test environments outside of a reactor (e.g., a test reactor beam port), it is much easier to perform spectroscopy measurement for both neutron and gammas. Historically, neutron time of flight and neutron slowing-down spectrometry are used to measure the neutron energy spectrum. It is relatively easy to discriminate neutrons from gammas due to the high reaction energy for neutron reactions in most neutron detectors. In a relatively low intensity neutron field, traditional gamma spectrometers such as HPGe, NaI, and CZT detectors are used to measure the gamma energy spectrum. Active detectors include PIN diodes and scintillator fiber detectors. Passive integral detectors for gammas include TLDs (CaF_2 and LiF) and alanine.

In the test reactor environment, it is much more difficult to obtain the detailed energy spectrum for both neutrons and gammas. It is also more difficult to measure prompt gammas than delayed gammas. The following detectors are promising:

1. Diamond probe (EPLF) and detector (EPFL) and photo-conducting detector (PCD) have been used at Sandia National Laboratories (SNL)
2. Calorimeter detectors such as Si, Bi, Zr, Sn, W, and Tl (SNL)
3. Portable HPGe detector can be used to perform Prompt Gamma Neutron Activation Analysis (PGNAA) at test reactor beam ports
4. Delayed gamma spectrum from irradiated fuel can be measured with a collimator after decay
5. Prompt gamma spectrum in reactor can be measured at zero power or very low power

It has also been recognized that even though it is difficult to measure the differential prompt gamma energy spectrum, many times the research purpose of the material or biological damage irradiation can be achieved by placing samples in the test field with passive integral dosimeters.

The state of the art mixed field dosimeters are:

- Diamond detector (both neutrons and gammas—more useful for gammas)
- SiC detector (neutron detection)
- Liquid scintillator and high-pressure gas (Xe) detectors (typically more sensitive to gammas than neutrons)
- Optical stimulated luminescent dosimeter (OSLD), qualification at zero power reactor

The community also expressed interest in exchanging experience in fast response electronics.

- Pulse shaping using fast pre-amp on the order of nanoseconds (Idaho Accelerator Center has custom made some fast response pre-amp with a response time of 2 ns)
- Fluctuation on the current signal
- Yokogawa, potential vendor
- Field Programmable Gate Array (FPGA)

Reactor Surveillance

Greg Fischer (Westinghouse) and Simon Shaw (EDF Energy Nuclear Generation Ltd.)

The Workshop began with introductions and the group included a wide range of experience and perspectives in the reactor surveillance field. The first discussion item concerned whether the continued use of $E_n > 1$ MeV fluence for material damage correlations was still adequate or whether assessments should move to a dpa damage-based parameter. USNRC representatives noted that they receive some submissions on a dpa basis, but most licensees continue to use $E_n > 1$ MeV, as stated in the regulatory guides. As vessels age, a mechanistic damage parameter may become more desirable while also including other components such as nozzles. However, the damage to other components may not be life limiting.

It was noted that construction materials for nozzles may not have been archived, since surveillance capsule contents were developed for reactor pressure vessel (RPV) belt line materials. There is wide spread interest in harvesting material from decommissioned plants, but it is not clear who would be able or willing to provide funding. There is a U.S. Department of Energy program looking to obtain materials from Zion Nuclear Power Plant which has obtained large samples of RPV base metal.

It is now becoming more normal to install ex-vessel neutron dosimetry (EVND) during reactor construction. The main examples of this are South Korea and Russia. EVND is also useful as a mitigation against future needs for dosimetry validation data (e.g., for long term aging investigations and plant life extension). The USNRC opinion is that EVND, while not mandated, strengthens and provides the regulator with more confidence in a surveillance program. The regulator provides some flexibility in the accuracy requirements for EVND comparisons, which are 30% for EVND compared with 20% for in-vessel dosimetry. These requirements may need to be tightened if EVND becomes more important for engineering judgments.

There are certain situations – such as changes to core loading patterns, or changing reactor internals – where it is advisable to make measurements before and after implementing a change. Otherwise, equilibrium operations tend to produce consistent flux values.

Interest was expressed in RPV liner sampling for retrospective dosimetry. This technique produces measurement data directly at the location of interest. There was a discussion of RPV liner (retrospective dosimetry) analyses, published in previous ISRD symposia. The sampling campaign was only performed once for cost reasons. Subsequent RPV inspections have observed the presence of the sample locations.

The NuScale surveillance capsule design and fluence methodology was discussed. Design considerations and historical design decisions made for Westinghouse designs were provided. In the USA, new reactor designs fall under a different regulatory branch than the current operating fleet.

The possibility of influence of gamma rays on non-fissile dosimeters (where photo-fission is unlikely to contribute more than 1% to the fission rate) was discussed. Where pure foils are not used, analysts should be aware of the potential for competing reactions. For example, the ^{55}Mn (gamma, n) reaction may require consideration when analyzing retrospective dosimetry.

Experimental Techniques

Lawrence Greenwood (Pacific Northwest National Laboratory) and Hubert Carcreff (CEA)

This Workshop had an attendance of 24 participants from eight different countries. At the beginning of the meeting, three topics were proposed and were the focus of discussions:

- Nuclear data improvements for qualification & modelling of nuclear instrumentation sensors.
- The state of the art in the field of epithermal neutron detection in the 1 keV – 1 MeV range.
- Special needs for nuclear heating measurements in Nuclear Power Plants.

Preliminary to these topics of discussion, it was mentioned that at the ISRD15 workshop dealing with this same topic, there was a concern about the availability of ^{237}Np fission monitors. This kind of monitor is available again from the National Isotope Development Center at Oak Ridge National Laboratory.

Nuclear Data: JSI in Slovenia noticed significant discrepancies when measuring Cd and BN ratios in their TRIGA reactor: about 20% for the ^{58}Fe (n, γ) reaction and about 100% for the ^{117}Sn (n, n') vs ^{197}Au (n, γ) reaction. Participants recommend improvement in the nuclear data for these two reactions. CEA in France also noticed a discrepancy in the ^{117}Sn (n, n') reaction during their Fluole-2 program. In addition, the half-life of ^{117m}Sn has been pointed out to have some disagreements among the different libraries. SNL in the U.S. specified that the ^{32}Si decay scheme and the half-life need to be improved.

Epithermal Neutron Detection: More sensitive reactions in this energy range are needed for the improvement of neutron dosimetry. The ^{93}Nb(n, n') and ^{237}Np(n, fission) reactions are currently the most reliable reactions in this energy range. IRRM can supply now very pure Nb foils with only 0.3 ppm in Ta impurity leading to less corrections in counting data. Incidentally, some laboratories have difficulties using the ^{93}Nb reaction due to problems with working with hydrofluoric acid. CEA detailed its research program dedicated to use ^{92}Zr(n, γ)^{93}Zr and ^{94}Zr(n, γ)^{95}Zr under a BN filter for the epithermal flux evaluation, using the mass accelerator spectroscopy technique for the ^{93}Zr stable product measurement in dosimeters. CEA mentioned that both ^{92}Zr(n, γ) and ^{94}Zr(n, γ) reactions are not well known and need to be improved to reduce uncertainties measured by this new technique.

Nuclear Heating Measurements: Participants mentioned that the most suitable method in mixed fields and in low power reactors, is the use of CaF_2 TLDs due to their low neutron sensitivity. There was a suggestion to test the Ca sulfate type. However, the issue of the delayed photon contribution after irradiation was mentioned. Optically Stimulated Luminescent Dosimeters (OSLD) seem to be a promising technique to provide instantaneous dose rates and to measure the delayed photon dose after shutdown. For calorimetry measurements at higher powers, there is a suggestion to use Bi, Sn or Zr as samples instead of graphite. Participants agreed that more validation is required by Monte Carlo calculations of mixed fields although some studies have been validated for medical applications. Use of HPGe detectors close to reactors to measure the neutron spectra was suggested although there will be a need to anneal the neutron damage. JSI and PSI discussed the need for more modeling by Monte Carlo techniques to determine the detector efficiency of HPGe in close to reactor geometries.

In concluding the workshop, diamond detectors to measure fluence or radiation damage by X-ray diffraction measurements of the lattice spacing were discussed as a novel technique as was the use of Si, SiC, and sapphire detectors. Older techniques including stable product dosimetry (such as HAFM helium monitors) and etching of fission tracks were suggested. Finally, PSI from Switzerland pointed out that there is a need for a "guide." PSI suggested a publication describing good practices for neutron and gamma dosimetry measurements in reactors and their analysis to include an updated list of available techniques.

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ISBN: 978-0-8031-7661-4
Stock #: STP1608

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