Summaries of Workshop Meetings

Nine workshop meetings were held at the symposium to foster an informal exchange of information on topics of major interest to the participants. Each workshop lasted over two hours and ran concurrently in groups of three. A survey was conducted before the symposium to determine workshop interests and to produce a schedule to allow each participant to attend as many workshops of primary interest as possible.

Each workshop had two co-chairmen who had the responsibility of organizing the workshop, including defining the scope and relating the workshop to papers presented beforehand at the symposium. The program schedule was designed so that the workshop discussions would build upon the presentations. After the workshops, the co-chairmen produced summaries of the discussions and conclusions. These summaries are presented in this section.

Workshop on LWR Surveillance Dosimetry

S. L. ANDERSON (Westinghouse) AND F. HEGEDÜS (SFIRR)

Discussions among the 35 participants of the workshop centered on the applicability and value of cavity dosimetry for pressure vessel exposure monitoring, the determination and use of lead factors (the ratios of fluence rate at the surveillance capsules to the maximum fluence rate in the pressure vessel) in pressure vessel surveillance evaluations, and the importance of temperature documentation in surveillance capsule analysis.

Cavity Dosimetry

Extensive discussion on the cavity dosimetry issue led to a consensus that accurate exposure determinations require both measurements and reactor physics calculations, and that neither measurement nor calculation alone is sufficient. Several advantages of the cavity dosimetry approach were noted.

In particular, the capability to obtain information on spatial gradients, and the flexibility to establish measurement intervals (cycle by cycle or multiple cycle) were noted. The need to establish uncertainty estimates not only at the measurement locations, but also at positions within the pressure vessel wall, was stressed. The workshop concluded that the use of cavity dosimetry as a method of pressure vessel exposure monitoring is a viable approach, complementary to surveillance capsule dosimetry, and the group recommended that this method continue to be implemented in the future.

Lead Factors

In regard to the use of lead factors for surveillance capsule applications, it was noted that, due to changing fuel management approaches, the lead factor cannot be treated as a constant over plant lifetime. Thus the usefulness of the concept for projections into the future had to be questioned. These temporal projections of vessel exposure must take into account appropriate anticipated changes in fuel management. As a result of these considerations, it was generally agreed that it would be appropriate to drop the concept of lead factor entirely. Also, as a part of this discussion, it was noted that the scraping technique, using small steel samples scraped from the reactor vessel inner radius and other internal structures, can provide valuable information on exposure distributions within the reactor.

Temperature Documentation

Discussions relative to temperature variations in surveillance capsule specimens resulted in the conclusion that any surveillance capsule data base should include documentation of the capsule temperature along with neutron exposure parameters and materials data. Knowledge of the maximum temperature is important for correlating the measured damage with the neutron exposure.

Workshop on Adjustment Methods, Cross-Section Files, and Uncertainties

F. W. STALLMANN (ORNL) AND M. MATZKE (PTB-Braunschweig)

The workshop was attended by 25 participants who discussed the results of the REAL-84 exercise and the REAL-88 follow-up, uncertainties, and covariances for dosimetry cross sections and neutron fluences, and the establishment of standard dosimetry cross-section and nuclear data files for use in adjustment procedures.

REAL-84 and REAL-88

H. J. Nolthenius (ECN-Petten) reported on the results of the REAL-84 exercise organized by the International Atomic Energy Agency (IAEA) in Vienna. The main aim of the exercise was to improve the assessment of accuracies in radiation damage prediction by various laboratories using good quality input data and proper calculational methods. The emphasis was concentrated on radiation damage characterization for reactor pressure vessels and related neutron technology [*1*]. Nolthenius reported that there were 44 sets of adjustment results received from 12 participating laboratories. The results showed in most cases a large and unexpected interlaboratory spread. The consensus reached the previous week at a working group meeting in Jackson, Wyoming, was that the primary reason for this spread could be traced to inconsistencies in the input data sets that were handled differently by different participants. The relatively coarse group structure in both the low and high energy region may also have contributed to the differences, requiring more detailed instructions for interpolation and extrapolation of group fluences.

M. Matzke reported on the results of the IAEA working group meeting in Budapest [*2*] in September 1986 and the May 1987 meeting in Jackson mentioned above. The primary result was the initiation of a follow-up exercise, REAL-88, organized again by IAEA. In the new exercise, the inconsistencies of the REAL-84 data sets will be removed and more detailed spectral information will be provided in order to reduce the interlaboratory spread and pinpoint more precisely the reasons for any remaining discrepancies. It was felt that the revised data sets will serve as benchmarks for testing of adjustment procedures.

Uncertainties and Covariances

Several aspects of covariance matrices for fluence calculations and cross-section data were discussed. Matzke commented on the significance of singular covariance matrices that frequently result when a relatively large number of group fluences or cross sections are determined by a small number of parameters. Several participants commented on problems of creating covariances if no detailed uncertainty information is available. It was pointed out that a Gaussian type correlation matrix is not very realistic and that a pure diagonal matrix may often be preferable.

Cross-Section and Nuclear Data Files

The remaining discussion centered on the availability of cross-section and nuclear data files, with their uncertainties, for use in adjustment procedures. V. Goulo (IAEA-Vienna) reported on an IAEA meeting [*3*] in Rome, November 1986, and pointed out that IAEA's EXFOR system, which includes programs and data files, could play an important role in reactor dosimetry. F. W. Stallmann reported on the LSL-M2 program package [*4*] developed at ORNL, which includes a dosimetry cross-section file and which is available upon request through the Nuclear Regulatory Commission (NRC) and the Radiation Shielding Information Center (RSIC) at ORNL. W. Mannhart (PTB-Braunschweig) handed out a new and updated listing of ^{252}Cf spectrum-averaged cross sections. In a subsequent discussion, problems in the cross-section data were pointed out for several dosimetry reactions including ^{27}Al(n,α), ^{47}Ti(n,p), ^{58}Ni(n,2n), ^{115}In(n,n'), ^{127}I(n,2n), and ^{197}Au(n,2n). It was requested that the IAEA should provide revised versions of their dosimetry cross-section file as soon as feasible.

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Workshop on Detector Activities, Decay Data, and Uncertainties

M. P. MANAHAN (Battelle) AND A. J. FUDGE (Harwell)

There were 30 participants in the workshop; where possible, lead authors were used to start the discussions on each topic.

Nuclear Decay Data

It was agreed that nuclear decay data published in ASTM E 1005 and in Europe in publications by Zijp and Baard [*1*] are reasonably good. However, there were notable exceptions and both data sets need to be updated as soon as possible. It was noted that there also exists a coordinated research program of the IAEA whose purpose is to provide decay data for detector calibration. It was felt that all these efforts should be linked.

Measurement Techniques and Capabilities

Radiometric Dosimetry Comparisons in the ORNL Pool Side Facility

Radiometric dosimetry comparisons were made in the ORNL Pool Side Facility (PSF). Results for non-fissile monitors agreed within ±5%; results for fissile monitors agreed within ±10%. It was noted that these uncertainties are somewhat higher than the routine capabilities of the participating laboratories.

Dosimeter Activity Measurements

All important parameters associated with dosimeter activity measurements were reviewed, including detector specification, calibration, and data analysis. It was generally agreed that provision and use of counting standards of specific nuclides are of great importance and are required internationally.

Solid-State Track Recorders

At present, solid-state track recorders (SSTRs) are used in benchmark fields and, with great care in deposit preparation, their use is capable of being extended up to fluences of 5×10^{18} n/cm^2 ($E > 1$ MeV). Development efforts are needed to extend their use to even higher fluences.

Activity Uncertainty Specification

The importance of reporting all uncertainties associated with measurements of dosimeters was stressed. All spectral adjustment procedures use statistical methods and so require complete statements of covariance information for all quantities involved in the analysis.

Photofission Effects

It was pointed out that photofission effects could contribute as much as 30% of fission events in some neutron fields; thus there is a very real need for their appropriate treatment. Additional work is still required to reduce the uncertainties associated with these effects.

Quality Control and Traceability

While it was recognized that quality control and traceability are important aspects of detector activity analysis, no agreement was reached concerning their exact definitions or the extent to which these should be implemented. It was the consensus of the group that quality assurance, which consists of treaceability as well as substantive data verification and overcheck, is needed. Nevertheless, the costs of the quality assurance program should be commensurate with the program's cost and should be made to contribute significantly to the technical content.

Benchmark Field Calibration

R. Gold (HEDL) stressed the importance of benchmark calibrations for validating dosimetry techniques, materials, and analytical capabilities, but these calibrations possess limitations and need careful application. It was agreed that the application of each technique and each benchmark needs to be examined in detail for the existence of possible biases. Gold provided details of two instances where such biases can arise, namely from impurities in ultra-low-mass SSTR deposits and from perturbations introduced by active detection systems in low power benchmark fields.

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Workshop on Gamma-Ray Dosimetry

R. GOLD (Metrology Control Corporation) AND M. NAJŽER (J. Stefan Institute)

Gamma-Ray Effects in Power Reactors

P. J. H. Heffer (Berkeley Nuclear Laboratories) reported that an effect of prime importance in U.K. graphite moderated power reactors is the radiolytically induced corrosion of the moderator which arises from interaction with the carbon dioxide coolant ($C + CO_2 \rightarrow 2CO$). Methane is injected into the reactor (along with some CO) to inhibit this corrosion, but at end of life, parts of the core can experience weight losses of 20 to 30%, which may compromise structural integrity. To study this effect, Heffer and co-workers at the Central Electricity Generating Board (CEGB) used a zero-energy test reactor to measure absolute gamma ray dose distributions throughout the core and fuel components. This work was centered on the use of BeO thermoluminescent detectors and specially constructed ionization chambers.

Heffer also reported on a recent example of gamma ray effects that has arisen in a CEGB study of control rod articulation joints, which have welds which are subjected to thermal cycling. Calculations have shown that the steel joints generate considerable self-heating from thermal neutron capture; poor heat transfer to coolant gas can give rise to temperatures in the joint up to 70% above normal. This work has shown that the inaccuracies in gamma transport analyses are dominated by source distribution (thermal capture rate distribution) for this type of problem.

Najžer noted that gamma ray heating of the reactor pressure vessel and surveillance capsules of the 632 MWe Krško nuclear power plant has been investigated using coupled neutron-gamma transport calculations reported in his symposium paper [1]. It has been shown that the gamma flux and heating rate may be 30 to 50% higher at the end than at the beginning of the fuel cycle.

J. B. Sun (Florida Power & Light) led a discussion on possible determination of deviations from symmetry in PWR power plants by gamma scanning of core internals during their in-service inspection. Different detector systems were recommended to perform gamma ray scanning. It was concluded that current problems appear to center on the logistics of conducting in-vessel scanning during the in-service inspection.

Recent Developments in Standard Gamma-Ray Fields

It was generally agreed that gamma sensitivity of neutron detectors represents one of the limiting factors in neutron dosimetry measurements in nuclear reactor environments. T. G. Williamson (University of Virginia) and E. D. McGarry (NBS) reported that cadmium and iron standard neutron capture gamma-ray fields, representing capture gamma-ray fields encountered in power reactor environments, have been recently constructed and calibrated at NBS [*2*]. The fields consist of cadmium and iron cylinders installed in the cavity of the NBS reactor. The gamma ray dose rates in the cadmium and the iron fields are 23 Gy/s and 10 Gy/s, respectively.

Gamma-Ray Sensitivity of Neutron Monitors

Williamson also reported on recent integral measurements of photofission cross sections in ^{238}U, ^{232}Th, and ^{237}Np in the aforementioned standard fields. These measurements range between 20 to 50% lower than calculated values. Measurements of integral inelastic cross sections have been completed for indium and are in progress for niobium.

G. Sandrelli (ENEL-Milan) observed that his symposium paper [*3*] reported discrepancies in the range of 30 to 50% between measured and calculated reaction rates for ^{237}Np, ^{238}U, and Nb for the cavity of the CAORSO BWR reactor. Differences may indicate large (γ,f) and (γ,γ') contributions to the measured reaction rates.

Najžer reported that photofission contributions of 2% of the total fission rate in ^{237}Np and 7% in ^{238}U located in the surveillance capsule of a PWR have been calculated using the discrete ordinates coupled neutron-gamma transport code, DOT 4.2.

It was generally agreed that further evaluation of photofission and (γ,γ') interferences in fission and (n,n') neutron detectors is needed.

Gamma-Ray Induced Displacements

R. Gold discussed his symposium paper [*4*] where calculations show that the gamma-ray induced displacement rate in iron is less than 0.5% of the neutron induced displacement rate throughout LWR-PV environments. However, results are not indicative of all possible reactor environments, since gamma-to-neutron displacement rate ratios depend sensitively on the gamma-to-neutron intensity ratios, the gamma-ray spectrum, and the material under consideration.

High Level Gamma-Ray Dosimeters

E. D. McGarry reported that the neutron sensitivities of LiF gamma ray dosimeters capable of measuring doses up to and in excess of 10 MGy have been investigated at NBS by Gilliam and co-workers [*5*]. Neutron sensitivity of 7% in typical reactor environments has been found. It was generally agreed that LiF dosimeters require further studies of not only neutron effects but also reproducibility and temperature dependence.

L. Zuppiroli (CEA) reported that a new gamma dosimeter made of an organic crystal has been developed in France [*6*]. The physical principle of the dosimeter is variation of resistivity with absorbed dose. He indicated that this dosimeter is capable of measuring absorbed dose up to 100 MGy. Temperature has to be closely controlled during resistivity measurements to achieve high resolution. Present irradiation temperatures are limited to 50°C. Also, testing of a new organic crystal capable of withstanding temperatures of 100°C is in progress, and extension of the temperature range to above 100°C seems feasible.

Recent Developments in Calorimetry

Heffer reported that recent studies of micro-calorimeters performed with J. Mason and colleagues at Imperial College [7] have shown that heat transfer mechanisms, rather than gamma-electron transport effects, dominate the uncertainty of instrument response. These considerations, in particular the difference between heat generation in electrical calibration and exposure in a radiation field, have been used to develop a new calorimeter design which shows marked improvements over previous devices.

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Workshop on Neutron Dosimetry with Niobium

JW ROGERS (EG&G) AND W. G. ALBERTS (PTB)

There has been considerable effort recently to develop niobium as a neutron monitor for measuring neutron fluences with sufficient accuracy to meet the needs of materials testing and monitoring. The ^{93}Nb$(n,n')^{93m}$Nb reaction is particularly useful as a fluence monitor because its response range extends below 1 MeV and because of its long (16.13 y) half-life. The great interest in this workshop (37 attendees) and the number of papers (9) presented at this symposium on the niobium monitor indicate its wide application in neutron dosimetry. Two papers dealt with applications [1,2], two with materials [3,4], and the others with data development [5–9]. The participation of the attendees in this workshop was excellent and they identified several problem areas and made several recommendations, which are summarized in the following sections.

Availability of High Purity Niobium Material

Two sources of high purity niobium were identified: one is the Central Bureau for Nuclear Measurements (CBNM) [3], Geel, Belgium; the other is Tosoh Corporation in Japan [4]. Both can supply materials (foils and wires) with a tantalum content ranging from below 10 μg/g to less than 1 μg/g. It was pointed out that under some irradiation conditions and with some counting techniques the tantalum content may not be critical. It was also noted that the mechanical stability of very thin, pure niobium foils has been observed to deteriorate during irradiations, and surface oxidation can occur with some conditions.

Nuclear Decay and Cross-Section Data

The half-life of ^{93m}Nb appears to be known to an accuracy of about $\pm 1\%$, which is considered satisfactory for most neutron dosimetry needs, but it was recommended that additional measurements be conducted to further reduce this uncertainty for research and development purposes. The shape of the ^{93}Nb$(n,n')^{93m}$Nb excitation function (cross section versus energy) needs to be better defined. Recent measurements between 1 and 8 MeV in the United Kingdom and Austria indicate the need for a re-evaluation and generation of a new up-to-date data file that will be available to all users. Three measurements of fission spectrum averaged cross sections reported at this symposium [*5–7*] are consistent within 2 or 3% and generally support the point-wise experimental data in the literature, as opposed to the calculated theoretical excitation function [*10*]. The experiences of several participants indicated that by using the cross sections from the recent measurements rather than earlier evaluations, they obtained much more consistent agreement with other radiometric monitors in a variety of reactor and spallation-neutron spectra. It was recommended that a standard set of decay and cross-section data be adopted.

Counting Methods and Corrections

The more commonly used method of using X-ray spectroscopy to measure the activity in niobium deposits was discussed relative to recently developed methods using liquid scintillation spectrometry [*8*] and direct counting of "thick" metal foils [*6*]. It appears that all of these methods can produce correct results when the necessary corrections are known and applied. No consensus as to which counting method is most suitable was reached. For all methods it was stated that standard activity samples of the same type as the samples to be assayed should be made available so that direct relative measurements can be performed. Corrections for self-absorption are required for "thick" foil or heavy deposit counting, and in some cases corrections for fluorescence by ^{182}Ta and ^{94}Nb are required. The ^{93m}Nb activity of monitors can be determined to accuracies of 4 to 5%, depending on the purity of the niobium material and irradiation conditions. It was recommended that standard solutions of ^{93m}Nb, ^{94}Nb, and ^{182}Ta activities also be made available.

Standard Methods

With the wide application of niobium as a neutron monitor it becomes necessary to establish standard procedures as a guide for routine measurements and to transfer the necessary technology to service laboratories. Such standard procedures are being prepared. A DIN draft standard is in print in the Federal Republic of Germany, and ASTM Task Group E10.05.02 is in the process of preparing a standard. The DIN standard covers only the counting of very thin sources by X-ray spectrometry. It is planned that the ASTM standard will also include liquid scintillation spectrometry and "thick" foil counting by X-ray spectrometry.

Photon Contributions

One attendee expressed some concern about photon excitation of niobium being a problem in high-intensity gamma fields. Others noted, however, that in their experiments they could not identify any large effects due to (γ,γ') processes. It was agreed that, since this effect may exist, it would be worthwhile to attempt experiments with varied photon energies and intensities to see if photon excitation can be observed with certainty and if the cross section can be quantified.

Future Joint Publication

The possibility of publishing a number of full papers on niobium-related research in a special issue of a journal was discussed. Those interested in this idea were asked to contact J. G. Williams (University of Illinois, Urbana) and formally indicate the content of the paper(s) they would like to prepare for this purpose.

Conclusions

Based on the presentations at this symposium and this workshop it must be concluded that niobium as a neutron monitor can be confidently used to measure neutron exposures in many conditions and situations. It is anticipated that additional results from ongoing research and routine measurements will be forthcoming.

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Workshop on the NESDIP Benchmark

S. P. GRANT (Carolina Power & Light) AND A. PACKWOOD (AEE Winfrith)

The workshop was attended by 16 participants. It was first of all noted that a review of the NESTOR Shielding and Dosimetry Improvement Programme (NESDIP) is contained in two symposium papers [*1,2*] and in two NESDIP blind-test reports [*3,4*].

It was agreed that as a result of the two-day Light Water Reactor Pressure Vessel Surveillance Dosimetry Improvement Program (LWR-PV-SDIP) review meetings held in Jackson, Wyoming, on May 27 and 28, 1987, the blind test exercise had now been completed, and comparisons of calculations and measurements could be discussed.

R. E. Maerker (ORNL) presented the results of his discrete ordinates calculations for the NESDIP-2 slab geometry benchmark [5]. In addition to performing an *R-Z* calculation, the flux synthesis, adopted in the LEPRICON procedure [6] and commonly used in interpreting pressure vessel surveillance dosimetry, was also applied. The calculations were first run with the ELXSIR cross-section library [7], which is part of the LEPRICON system, and then repeated with the iron data replaced by data from an update of the ENDF/B-V Mod 3 file. Comparisons made on the nuclear axis of the system showed consistency between the *R-Z* and flux synthesis models. The introduction of the modified iron data removed a trend of a falling ratio of the calculated-to-experimental measurements (C/E), versus penetration for the high energy fluxes, as monitored by the $^{32}S(n,p)^{32}P$ detector; but both data files gave similar results for fluxes below 2 MeV, as monitored by $^{115}In(n,n')^{115m}In$ and $^{103}Rh(n,n')$ detectors. The C/E ratios for these two low threshold detectors were generally 0.9 for locations out to the last measuring position in the simulated pressure vessel; then the ratio dropped to 0.70 in the cavity. Comparisons made off-midplane showed good agreement between the two calculational routes within axial limits defined by the source boundary. Comparison of the measured and calculated spectra in the cavity showed an underprediction of the fluxes below 2 MeV, which confirmed the low C/E's observed with the integral detectors.

P. C. Miller (AEE Winfrith) presented the results of Monte Carlo calculations performed using the McBEND code [8] with data files from the U.K. Nuclear Data Library. Once again the only significant disagreement between measurement and calculation occurred with the underprediction of the low-energy threshold detectors located in the NESDIP mockup cavity.

In the discussion, A. Fabry (CEN/SCK) suggested that an underprediction in the cavity would occur because the calculations had placed an outer black boundary after just 20 cm of water. The consensus view of the participants, however, was that this water thickness was adequate to preclude any significant input from the boundary.

It was mentioned that blind test calculations for the NESDIP3 slab shield had been carried out by K. Takouchi of the Japanese Ship Research Institute. Takouchi's results were in general agreement with the ORNL and AEEW predictions, although his predicted attenuations were slightly greater. The fact that the Takouchi results also showed larger disagreement between calculations and measurements within the cavity region stimulated a general discussion of possible causes for this effect. No explanation for the poor agreement in the cavity was identified. W. N. McElroy (HEDL) mentioned a future program which would include measurements in two different width cavities, in part to evaluate neutron streaming. He also discussed the decision at the May LWR-PV-SDIP program review meeting to consider changes in the planned HEDL measurement program in NESDIP, which might help resolve the anomaly in cavity results.

Maerker discussed the results he presented at this symposium [9] for the LEPRICON analysis of the reactor H. B. Robinson-2. Maerker expressed the opinion that the underprediction of reaction rates in the cavity, coupled with the conventional use of cavity measurements to determine vessel fluences by extrapolation, *could* result in an overestimate of the fluences at the inner surface of the vessel. This, in turn, could lead to an unwarranted degree of conservatism in the damage estimate.

Miller presented comparisons of calculations and measurements in the NESDIP Phases 4 & 5 21-cm cavity and around the simulated nozzle. The calculations had been performed using albedo techniques in the cavity and Monte Carlo at the nozzle. Very good agreement was found in all regions. Maerker [5] reported that the flux synthesis techniques showed marked departures from the experimental data once the calculation was taken beyond the lateral extent of the source region, and he agreed that some alternative approach would be required to predict these fluxes.

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Workshop on Dosimetry for Fusion Program Irradiations

L. R. GREENWOOD (ANL) AND R. DIERCKX (JRC, Ispra)

The workshop group, consisting of 20 individuals, discussed dosimetry and nuclear data needs for fusion program irradiation facilities. These included *d-t* sources at 14 MeV, Be(d,n), and other accelerator sources (<50 MeV), fission reactors, and spallation sources (<800 MeV). Reviews presented by the chairmen Greenwood and Dierckx were based partly on their symposium presentations [*1,2*]. Large discrepancies were noted for some dosimetry cross sections at 14 MeV; data are needed for short-lived activities and gamma production for diagnostic applications at fusion reactors. It is recommended that further data evaluations be made near 14 MeV. J. H. Roberts (Metrology Control Corp.) presented data on a rotating track recorder device that could prove useful for measuring the time structure of neutron production in fusion reactors.

It was generally agreed that dosimetry data are not well-known for spallation neutron facilities, especially in the energy range of 50 to 800 MeV. Nuclear model calculations and proton-induced yield measurements are needed. Spallation product cross sections would be especially useful for Ti, Fe, Ni, Nb, and Cu, which are now being used routinely for dosimetry at spallation neutron sources. It was recommended that an informal cooperative effort be initiated to compare existing dosimetry cross sections for spallation and to encourage further measurements, calculations, and tests. Greenwood and Dierckx agreed to be contact points for this effort.

D. W. Kneff (Rockwell International) reviewed the status of helium production cross sections, including tests in fission reactors, 14 MeV, Be(d,n), and other neutron environments. Fission reactor measurements show serious discrepancies with existing data files for some elements [*3*]. Thermal neutron transmutation and multiple stage reactions in fission reactors produce extra helium in Ni, Cu, and Fe that can be used to great advantage to simulate fusion reactor conditions [*3*]. The Ni and Cu cases are well understood; however, the Fe effect requires further study. For spallation sources, concern was raised that some recoiling helium atoms may be lost due to the large recoil range of high energy helium atoms. It is recommended that this issue be studied in more detail.

Helium production from ($n,x\alpha$) reactions has been combined with radiometric measurements to adjust neutron spectra. This has advantages at higher neutron energies where the helium production cross sections are significant and for long irradiations, since helium is a stable product. However, helium production cross sections need improvement both for fission reactors and spallation neutron sources.

Greenwood presented some new calculations on radiation damage cross sections for compound materials [4]; however, he indicated that further effort is needed to determine suitable displacement threshold energies. L. Zuppiroli (CEA) presented data from X-ray irradiations of $LiAlO_2$ which show that electronic energy losses can produce more defects than expected in insulator materials. It was agreed that such effects require further study. Greenwood noted that attempts to calculate radiation damage (dpa) for spallation sources have encountered large discrepancies between existing codes for energies between 20 and 100 MeV. These differences are not understood, and it is recommended that more study be undertaken to produce adequate damage cross sections for spallation sources.

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Workshop on Radiation Damage Correlations

A. L. LOWE, JR. (B&W) AND A. ALBERMAN (CEA)

This workshop was attended by 35 participants. The discussion was devoted mainly to radiation damage on pressure vessel steels and to the relevance of metallurgical data versus neutron exposure parameters such as $\phi > 1$ MeV or displacements per atom (dpa).

Basically, dpa values are calculated from the neutron-energy spectrum. The dpa value represents the amount of damage occurring in the material as a result of the different energetic particles. It was stressed that dpa is not intended for radiation damage predictions in complex materials such as steels but rather as a method for comparing and reporting irradiation results on a single material irradiated in different neutron environments.

It was observed that the effects of neutron spectrum may be of importance in certain surveillance programs, especially those where there is intercomparison of data from different irradiation sources. For example, spectrum changes might play a role in the lead factor assessment from the location of the capsule to the inside surface or the interior of the vessel wall. Also, spectral change may be of concern in long term irradiations where exposure of the vessel to strain occurs concurrently with neutron damage. It was concluded that, as of now, it is necessary to report metallurgical results versus both fluence ($E > 1$ MeV) and dpa.

It was generally acknowledged that atomic displacements are also caused by thermal neutron captures (recoils), gamma interactions, boron (n,α) reactions, and so forth. These additional displacements should not be included in total dpa exposure calculations before a basic understanding of these mechanisms is accomplished. This is especially important for understanding experiments conducted in test reactors where these effects can be best defined. Correlation procedures must rely upon well documented records of metallurgical properties and the corresponding neutron environments. Use of damage monitors, where possible, aids in the interpretation, and is an important adjunct to characterizing the neutron environment.

A. L. Lowe stressed that current laboratory procedures in metallurgy as well as in dosimetry should include a record system that will permit further re-evaluations of radiation damage studies at a future time.

In support of this position, it was agreed that if test reactor data are to be seriously considered and used to help quantify damage in LWR pressure vessel and support structures, the following information should be obtained and reported as a minimum:

- Flux, total fluence, fluence > 1 MeV, and neutron spectrum obtained from calculations adjusted by dosimetry measurements.
- dpa and dpa/second.
- Documentation of the time-history of the power, local flux, and temperature variation, to the extent that these data can be determined. This requirement applies for test reactor experiments and surveillance capsule irradiations.
- All basic documented dosimetry data (i.e., measured data as in a typical power plant surveillance capsule). This point is of particular concern if the capsule is irradiated in more than one site within a reactor.

Correlation monitor or reference materials should be used in all radiation experiments, where possible, to maximize the understanding and interpretation of irradiation results.

It was agreed that other displacement mechanisms, such as boron (n,α) reactions and γ displacements should be considered, but further experimental studies are needed before including these in damage models.

Workshop on the VENUS Benchmark

M. L. WILLIAMS (LSU) AND A. FABRY (CEN/SCK, Mol)

The two chairmen summarized the VENUS PWR Mockup program which consists of measurements and calculations performed at the VENUS facility at the CEN/SCK Laboratories in Mol, Belgium, in three different configurations called VENUS I, II, and III. The VENUS I analysis is virtually completed, and VENUS II is now in its final phases of analysis. According to A. Fabry, VENUS III loading is scheduled to begin in September 1987 and the program will be completed in mid-1988.

VENUS I

The following comments were made by workshop attendees who were also participants in the program. The VENUS I configuration consists of a "clean" UO_2 core with simulated baffles, barrel, and neutron pad. The measured dosimeter results reported at the previous ASTM-Euratom Meeting at Geesthacht in 1984 have been updated, following a new experimental campaign in the summer of 1986. The discrepancies between the calculated and measured ^{237}Np fission rates observed in the water reflector outside the core have been resolved. Re-evaluation of all data on a consistent basis, however, reveals a discrepancy in the $^{58}Ni(n,p)$ dosimeter results of up to 30% in the core barrel region. These particular measurements are believed to be biased, and Fabry and co-workers will seek a confirmation in VENUS III.

All other dosimeters showed good agreement (within 10%) between the measurements and calculations previously made by Mol and ORNL at most locations.

VENUS II

Much time was devoted among the 15 workshop participants to discussion of the VENUS II results. VENUS II is similar to the VENUS I configuration, except the last eight rows of pins in the core are replaced by mixed oxide (MO_2) pins to simulate a low leakage core. The relative pin power calculations performed with transport theory by Mol and with diffusion theory by ORNL/LSU agree to within 5 to 10% of the measured power distribution. Neutron dosimeter results for in-core and ex-core locations were presented by Mol, and compared with their preliminary transport calculations. The C/E values for most dosimeters are 10 to 15% lower than the corresponding values in VENUS I. It was generally concluded that part of this discrepancy could be due to an inconsistency in the normalization. Fabry indicated the final power normalization for VENUS II is not yet determined.

Extremely poor agreement was observed for the high threshold energy dosimeter reaction $^{27}Al(n,\alpha)$. The calculations are 50% higher than the measurements in the water reflector, and almost a factor of two lower in the neutron pad region which simulates a thermal shield of a PWR. It was suggested that the aluminum discrepancy could be due to a coarse group structure used in Mol's transport calculation and/or to the space-dependent flux spectrum they used in reducing the activation cross section to the coarse group structure. It was concluded that it would be highly desirable for some organization to perform additional transport calculations.

The Mol participants reported on a trend in the C/E values of various dosimeters which showed a decrease with water penetration in the reflector region. A similar behavior had not been observed in the Pool Critical Assembly (PCA) slab configuration at ORNL, indicating a possible geometry effect in VENUS. Several explanations were proposed for this behavior, including treatment of the scattering anisotropy, and treatment of axial leakage; however, no consensus was reached on the cause. It was agreed that a detailed Monte Carlo calculation would be very useful in resolving the discrepancy.

VENUS III

A brief description of the new VENUS III configuration was presented by Fabry. This experiment will model the part-length shield assemblies (PLSA) similar to those used in the H. B. Robinson PWR in South Carolina. As a result of the workshop discussions, the originally proposed design which contained PLSA mockups in all four quadrants was modified to contain PLSAs in only two quadrants. This was done in order to represent the transition region over which the normal core axial power distribution approaches the PLSA distribution. It was pointed out by S. L. Anderson (Westinghouse) that this region will be important in power reactor analysis. It was agreed that this revision to the VENUS III mockup would accurately reflect the important neutronic parameters of H. B. Robinson, and should answer the questions about the calculation methods used in predicting flux reduction factors in PLSA configurations. Calculational analyses will be performed at ORNL and Mol.

Appendices

APPENDIX I

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