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Summary Report

Workshop on Processing of Nuclear Data for Use in Power Reactor Pressure Vessel Lifetime Assessment

IAEA Headquarters, Vienna, Austria
19 - 23 October 1998

Edited by R. Paviotti Corcuera, L.R. Greenwood and D.W. Muir

February 1999

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Abstract

This document summarizes the contents of the Workshop on Processing of Nuclear Data for use in Power Reactor Pressure Vessel Lifetime Assessment. A short description of the main topics of the agenda, the list of participants and comments and recommendations are given.

Edited by R. Paviotti Corcuera, L.R. Greenwood and D.W. Muir

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CONTENTS

Introduction

The International Atomic Energy Agency (IAEA) organized a Workshop on Processing of Nuclear Data for Use in Power Reactor Pressure Vessel Lifetime Assessment. The Nuclear Data Section (NDS) co-ordinated this event, which took place at IAEA Headquarters in Vienna during the week of 19 - 23 October 1998.

Background

The lifetime of a pressurized water reactor (PWR) is often limited by the embrittlement of the pressure vessel due to neutron-induced lattice displacements of the atoms constituting the steel. This decrease in fracture toughness reduces the ability of the pressure vessel to withstand a rapid change in temperature without cracking, as well as to arrest the propagation of flaws.

Estimations of the condition of a reactor pressure vessel (RPV) are based on a combination of chemical/metallurgical analyses, dosimetry measurements, and calculations. Information concerning the neutron spectrum, fluence, and damage at the chosen surveillance locations can be used to judge how long the reactor can be safely operated before decommissioning the reactor or subjecting the RPV to some type of annealing process.

The assessment procedure is complex, involving many steps:

(i) measurement of selected nuclear reaction rates at a surveillance position;

(ii) calculation of the neutron fluence and spectrum at the surveillance position and in the pressure vessel;

(iii) adjustment of the fluence and spectrum from the measured reaction rates;

(iv) extrapolation (in space) to the pressure vessel and (in time) to future exposures; and

(v) determination of the time at which the estimated embrittlement in the RPV will be severe enough to necessitate shutdown.

It is important to reduce as far as possible the uncertainties in each of these steps, because they may lead to the imposition of overly conservative safety margins, and this in turn may lead to the costly premature shutdown.

Nuclear data with reduced uncertainties and computer codes with better mathematical algorithms are now available. Many countries have 10 to 20 year old power plants that may benefit from an updated analysis of pressure vessel conditions.

Objectives

The objective of this workshop was to familiarize the participants with updated nuclear data libraries and associated processing codes for performing calculations of power reactor parameters of importance for pressure vessel lifetime assessment.

The main emphasis was on demonstrations and exercises on subjects related to the evaluation of neutron fluence and the analysis of neutron-induced damage in reactor pressure vessels, employing the latest available data libraries and computer codes.

Internationally known specialists worked together with scientists from developing countries, using the IAEA-NDS online systems to access nuclear data and then performing relevant calculations.

Participation

The IAEA Nuclear Data Section selected 16 participants from different countries. The participants were all senior researchers or engineers directly involved in the nuclear power programme of their respective countries. (See Annex 1). Four IAEA staff members took an active role in the conduct of the workshop.

Main Topics of the Agenda

The workshop agenda is listed in Annex 2. A short description of the main computer codes used in exercises by the participants can be found in Annex 3.

The agenda covered the topics as detailed below:

a) Review of Methods for Determination of Neutron Induced Damage in Reactor Environments

- Overview of the process of determining neutron damage in reactors, including neutronics calculations, neutron dosimetry measurements, spectral adjustments, radiation damage calculations, and applications to material effects.
- Summary of the interaction of neutrons with materials leading to the calculation of displacement damage, gas production, and other transmutation.
- Damage due to gamma interactions.
- Introduction to various computer codes, mainly SPECTER for the calculation of radiation damage.

b) Online retrieval and processing of ENDF formatted Cross Sections and Covariances with NJOY

- Overview of basic data formats and the techniques employed for online retrieval and processing of cross sections and covariances in ENDF format into the formats required in pressure-vessel lifetime. This included a general discussion of the capabilities of the NJOY code, as well as special aspects of handling multigroup covariance matrices.

c) Assessment of Uncertainties for Pressure Vessel Neutron Fluence Calculations

- Main sources of uncertainties, as due to the calculational methods, source, basic nuclear data, data processing, geometry description.
- Sensitivity calculations with respect to radiation source and nuclear cross sections. These analyses can be based either on discrete ordinates methods (direct and adjoint flux calculations) or Monte Carlo methods.
- Determination of uncertainties.
- Methods and codes available for cross-section sensitivity and uncertainty analyses: e.g. SWANLAKE, SENSIT, SUSD.
- Uncertainty reduction by data adjustment (basic notions).

d) NJOY Examples and Exercises

e) The Neutron Metrology File NMF-90 for Reactor Dosimetry Applications

- Reference data files for reactor dosimetry calculations (benchmarks), cross section library IRDF-90, cross section processing codes, neutron spectrum adjustment and radiation damage (exposure) parameter calculation codes, utilities.

f) Nuclear data processing for reactor dosimetry calculations

- The processing codes of NMF-90: processing of ENDF data, multigroup and point cross section data, selfshielding, calculation of cross sections for targets of not 0 K temperatures, dpa and gas production cross sections, handling of cross section covariance matrices.
- Characterisation and use of the cross section library IRDF-90/NMF-G.

g) The program package STAYNL of NMF-90

h) Reactor dosimetry: neutron spectrum adjustment and radiation damage (exposure) parameter calculation

- The generalised least squares (LSQ) method, the LSQ code STAYNL of NMF-90.
- Reactor dosimetry calculations (neutron spectrum adjustment, dpa, gas production calculations with STAYNL in the benchmark reactor dosimetry fields of NMF-90).

i) Practical Applications Examples and Exercises (on the subject above)

j) Visit to the Nuclear Data Section

k) Dosimetry Techniques, problems, solutions

- Discussion of the process of measuring neutron fluences and energy spectra in reactors including nuclear activation reactions and cross sections, uncertainties and covariances, corrections for reactor power history, neutron and gamma self absorption, nuclear burnup and burning of target and reaction products, and influence of gamma radiation on neutron monitors.
- Neutron spectral adjustment with STAY'SL - examples and discussion.
- Discussion of alternate dosimetry techniques such as helium monitors and other stable transmutation.
- Retrospective neutron dosimetry using reactor components.
- Calculation of radiation damage using adjusted neutron spectra.
- Examples with the SPECTER computer code. Radiation damage in compounds using the SPECOMP computer code.
- Influence of gas production and transmutation on material effects.

l) Practical Applications Examples and Exercises (on the subject above)

m) Neutron transport calculations, best-estimate pressure vessel fluence determination and pressure vessel lifetime management

- Methodology used for deterministic transport calculations of the pressure vessel neutron flux. Important blocks of input data: cross-section libraries, geometry and material composition, and core power distribution.
- Typical modelling of a power reactor.
- The need for approximations in the modelling and their effect on the results.
- Examples of the results, e.g., multigroup spectra and flux spatial distributions.
- Determination of the best-estimate pressure vessel fluence.
- Neutron spectrum adjustment method, which combines the measured responses of neutron dosimeters, calculated fluxes, and their covariances to determine the best estimate of the flux with reduced uncertainty.
- Examples.
- The pressure vessel lifetime management through the fluence reduction techniques.
- The use of the low-leakage-fuel-loading schemes and specially built shielding assemblies to reduce the peak vessel flux.

n) International Database on Reactor Pressure Vessel Material VVER Reactor Lifetime Assessment

- Need for the International Database on Reactor Pressure Vessel Materials (IDRPVM).
- Main Database requirements.
- Structure of the IDRPVM.
- Legal framework and participation.
- Release of the Database and point of a contact.
- Future of the Database.

o) VVER Pressure Vessel Benchmarking, Pressure Vessel Data Base of Dosimetry Data

- Review of the results of PV Engineering Benchmarks and recommendations for the use of their results.
- Creation of the Benchmark Library for the PV Material Database.

p) Comments and Panel Discussion by participants on Nuclear Data needs and Evaluation of the Workshop.

- The participants were asked to comment about nuclear data needs in their projects, and to complete an evaluation questionnaire concerning the workshop.

General Comments and Recommendations

The workshop was attended by 16 participants from 15 different countries and four IAEA staff scientists (see Annex 1). The participants came from different types of organization; some were from the regulatory bodies of their country, others were from the power industry. Consequently, their needs were also varied. Some were more interested in the regulatory aspects of the pressure vessel lifetime management, while others were more interested in specific data (cross sections) and processing codes. The primary purpose of the workshop was to familiarize the participants with the nuclear data and computer codes used to determine the neutron exposure and resultant radiation damage in the reactor pressure vessels. This included processing of nuclear data for neutron transport calculations, reactor dosimetry and neutron spectral adjustment, and for radiation damage assessments (see attached agenda). In additional to presentations on the above subjects, the participants were given the opportunity to obtain hands-on experience with several computer codes used to process nuclear data, to accomplish neutron spectral adjustment and to perform radiation damage calculations.

At the end of the workshop, the participants were given the opportunity to present their assessment of the workshop and to discuss future needs for the work at their institutes. These comments may be summarized as follows:

1. The attendees generally found that the workshop was very useful in giving them a complete overview of the problem of determining the neutron exposure of reactor pressure vessels as well as providing them access and experience with nuclear data libraries and various computer codes. The participants recommended that more workshops of this type be organized and expressed the desire to have additional training courses in their respective countries (or perhaps with on-line materials) so that more of their colleagues might be presented with similar information.

2. The participants expressed strong interest to stay in contact with the IAEA Nuclear Data Section, lecturers of the workshop, and other participants. The participants pointed out need for training in practically all the areas related to pressure vessel neutron flux calculations, determination of best-estimate fluence, and embrittlement assessment. Besides the needs for the multigroup cross-section libraries and codes to process them, they expressed need for transport codes, spectrum adjustment codes, dosimetry cross sections and covariances, as well as material embrittlement correlations (such as embrittlement versus DPA).

3. The principal suggestions for improvements in the workshop included focusing the workshop more specifically on examples taken from operating nuclear reactors and the inclusion of additional types of reactors. (Many of the attendees use VVER or PHWR type reactors whereas many of the examples shown at the workshop were taken from US PWR reactors.) Some participants also expressed interest in seeing more detailed discussions of particular RPV materials effects, in addition to strictly neutron exposure and radiation damage.

4. It was suggested, in future similar workshops, to provide more complete written materials to the participants before the workshop, so that they could be better prepared for the lectures. This would require that the lecturers deliver such materials to the NDS well in advance of the workshop. The workshop presented a great deal of material that was difficult to assimilate in the few days of the meeting. Some attendees would have liked more time to work with specific computer codes. There were some comments regarding difficulties experienced in retrieving nuclear data and computer codes via the internet. The IAEA is addressing this issue by developing mirror servers in other countries.

5. Presentations at the workshop, as well as the experience of the attendees, clearly show that the accuracy of the reactor pressure vessel neutron fluence and radiation damage calculations and measurements are limited by our present knowledge of several fundamental neutron cross sections. In particular it is necessary to improve the evaluation of the inelastic scattering cross-section for iron because the pressure vessel flux calculation is highly sensitive to the iron inelastic scattering cross-sections. While the current ENDF/B-VI evaluation presents significant improvement over previous versions, the uncertainty needs to be further reduced in order to improve the accuracy of the calculated flux. RPV neutron dosimetry primarily involves neutron activation reactions with long-lived reaction products. The most important reactions are ^{237}Np(n,fission), ^{238}U(n,fission), ^{93}Nb(n,n')^{93m}Nb, and ^{63}Cu(n,α)^{60}Co. Uncertainties in these reactions, especially neptunium fission, strongly influence the uncertainties in the final estimates of neutron exposure. Consequently, the participants recommend that a CRP be established to encourage the improvement of the measurement and re-evaluation of these cross sections.

6. One of the principal difficulties in performing a reliable neutron spectral adjustment at the RPV is the lack of physics-based uncertainty and covariance data for all of the input data, including the calculated neutron spectrum and the neutron cross sections. Uncertainties are difficult to determine in neutron transport calculations and covariances are generally not known. The participants recommended that a CRP be established to study this problem and to encourage the development of more physics-based covariance data both for neutron transport calculations as well as for the neutron activation cross sections used for RPV dosimetry (such as listed in point 5).

7. The IRDF-90 cross section file circulated by the Nuclear Data Section needs to be updated with more recent information, especially regarding covariance information. New data from the major national and regional evaluated data libraries needs to be included. This could be done in conjunction with the CRP mentioned in point 5 or as a separate project.

8. The need for VVER specific broad-group cross-section library for transport calculations was pointed out by the participants from Czech Republic and Bulgaria. Support was also expressed for the initiation of a new international nuclear data processing effort for dosimetry applications similar in scope to the FENDL-2 project.

List of Participants

Workshop on Processing of Nuclear Data for Use in Power Reactor Pressure Vessel Lifetime Assessment

Vienna, 19 - 23 October 1998

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Annex 2

Schedule

Workshop on Processing of Nuclear Data for Use in Power Reactor Pressure Vessel Lifetime Assessment

IAEA, Vienna, 19 - 23 October 1998

Day	Morning (9 - 12:30 hr.)*	Afternoon (13:30 -17:30 hr.)
Monday	Review of Methods for Determination of Neutron Induced Damage in Reactor Environments. (1.5 hours) *Larry Greenwood* Online retrieval and processing of ENDF formatted Cross Sections and Covariances with NJOY. (2.0 hours) *Doug Muir*	Assessment of Uncertainties for Pressure Vessel Neutron Fluence Calculations. (40 min) *Ivo Kodeli* NJOY Examples and Exercises. (2.5 hours) *Doug Muir* and *Ivo Kodeli*
Tuesday	• The Neutron Metrology File NMF-90 of IAEA NSD for Reactor Dosimetry applications. • Nuclear data processing for reactor dosimetry calculations. • The program package STAYNL of NMF-90. • Reactor dosimetry: neutron spectrum adjustment and radiation damage (exposure) parameter calculation. (3 hours) *Eva M. Zsolnay*	Examples and Exercises (4 hours) *Eva M. Zsolnay*
Wednesday	Optional: Visit to the Nuclear Data Section. Meeting Time: 8:15 a.m. at Building A, Room 2340 (30 minutes) *Vladimir Pronyaev* Dosimetry Techniques, problems, solutions (3 hours) *Larry Greenwood*	Exercises (4 hours) *Larry Greenwood*

Thursday	Neutron transport calculations, best-estimate pressure vessel fluence determination and pressure vessel lifetime management. (3 hours) *Igor Remec*	International Database on Reactor Pressure Vessel Material VVER Reactor Lifetime Assessment. *Vjatcheslav Lyssakov* VVER Pressure Vessel Benchmarking, Pressure Vessel Data Base of Dosimetry Data. *Bohumil Osmera*
Friday	Comments by participants on Nuclear Data needs in their projects (Oral and written) (1.5 hours) Panel Discussion (1.5 hours)	Individual Exercises (for Participants) Final Report with Summary and Conclusions (for Lecturers)

Annex 3

Description of Main Computer Codes and Cross Section Libraries used in the Exercises

BCF – The BCF (Beam Correction Factor) code performs irradiation history corrections at reactors or accelerator-based neutron sources for a specified list of radioisotopes. Users input the reactor power history and the code determines a correction factor for each specific isotope in the data library. The correction factors are used to properly account for the decay during irradiation and to normalize all activation rates to a common reactor power level.

STAYSL – The STAYSL computer code performs a least-squares neutron spectral adjustment. The code uses pre-processed data libraries for neutron cross sections and uncertainties and neutron self-shielding factors (see SHIELD). Users input the measured activation rates, calculated neutron flux spectrum, and the associated covariances. The output gives an unbiased, adjusted neutron flux spectrum that minimizes the chi-square value. A 100-energy group fine cross section library is normally used to avoid spectral-weighting concerns. At present, the code provided must be used on a larger, workstation class computer (VAX, Alpha, SUN, etc.).

SHIELD - The SHIELD computer code is designed to calculate neutron self-shielding factors for small wires or foils used to measure dosimetry activation rates. The code operates using a 13,768-group fine cross section library and calculates both isotropic and beam neutron flux corrections on a group-by-group basis. The output is then collapsed to a user-specified neutron group structure. However, the output group structure must coincide with the group structure specified in STAYSL. Normally, a 100-group energy grid is used to avoid the necessity of spectral-weighting corrections.

SPECTER - The SPECTER computer code calculates radiation damage parameters including dpa (displacements per atom), pka (primary knock-on atom) energy recoil spectra, gas production, total energy deposition (KERMA), as well as listing spectral-averaged reaction cross sections. Users only need to specify the neutron flux spectrum and length of irradiation time. The code then provides the above spectral-averaged data for over 40 elements and compounds.

NMF-90 - The Neutron Metrology File is a modular set of computer codes, reference data on benchmark neutron fields and dosimetry cross section libraries. Each module provides a complete neutron metrology calculation for radiation damage purposes: they start with cross section data processing, perform neutron spectrum adjustment by the generalized least squares method and terminate in radiation damage exposure parameter calculation. Each module is designed for use on PC-s. (In frame of the Workshop the use of the module STAYNL was demonstrated, including complete processing of the cross section library IRDF–90/NMF-G. The programs in the module are transferable to other platforms than PC-s, as well.)

IRDF–90/NMF–G - The library is the topical version of the International Reactor Dosimetry File IRDF–90. It includes data on 53 reactions of 37 detector materials, furthermore 9 cross section sets without covariance information (cover materials, dpa calculation). The cross sections are available in a 640 groups (extended SAND II) histogram format.

X333 - The code X333 is the cross section processing code of the module STAYNL. It calculates group cross section values and their covariance matrices from the data given in files MF=3 and 33 of pre-processed cross section libraries in ENDF–6 format like IRDF–90. The cross section data in file MF=3 shall be given in histogram form, in the 640 group SAND II energy group structure. The code performs the self- and cover-shielding calculation, as well. The user's input consists of the list of the reactions to be processed, the optional detector and cover geometry and material data, the required output energy structure, and a fine weighting spectrum. The output of the code is a problem-dependent cross section and cross section covariance library.

STAYNL - The code solves the neutron spectrum adjustment problem by the generalized least squares method (GLSQM). The spectrum normalization before adjustment is performed also by the GLSQM, and the cross–covariance terms deriving from the calculation of the covariance matrix of the reaction rates are also taken into account. The user's input consists of the measured reaction rates with their covariances and a calculated neutron spectrum with its covariance matrix in an arbitrary energy grid. The problem-dependent cross section library can be generated using the code X333. The output of the code is the adjusted neutron spectrum that minimizes the χ^2 value.

XSL&OUT - This utility program calculates the radiation damage exposure parameters based on the outputs of the codes X333 and STAYNL.

The NJOY Nuclear Data Processing System - Participants in the Workshop were introduced to NJOY97, which is the most recent release of the NJOY Nuclear Data Processing System. NJOY is a modular computer code used for converting evaluated nuclear data in the ENDF (Evaluated Nuclear Data File) format into libraries useful for applied calculations. Because the ENDF format is used in major evaluation projects around the world (including ENDF/B-VI in the US, JEF-2.2 in Europe, JENDL-3.2 in Japan and BROND-2.2 in Russia), NJOY gives users access to a very wide variety of up-to-date nuclear data. NJOY provides comprehensive capabilities for processing evaluated data for neutron and photon transport processes and important nuclear effects (including KERMA and DPA), as well as covariance files describing the corresponding data uncertainties. Output includes printed listings, special library files for discrete-ordinates and Monte Carlo transport codes and other important applications, and color graphics. Relative to the previous version, NJOY97 features compatibility with a wider variety of compilers and machines, explicit double precisioning for 32-bit systems and a larger test-problem suite. For the purpose of processing nuclear data for reactor dosimetry cross sections and covariances, the most important changes are a fully supported PC version for Windows95 and NT, produced using the DOS-based Lahey LF90 compiler, and the introduction of color graphics to display fine details of multigroup covariance matrices. The ability to produce a compressed library in BOXR format is retained in NJOY97, and this option has been tested with NJOY97 with a variety of group structures, up to and including the 640-group SAND-IIA structure. The Windows version of NJOY97 with arrays dimensioned for such fine-group covariance processing is large (23 Megabytes of RAM), but still manageable.

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