ISSN NO. 2320-5407

Journal homepage: http://www.journalijar.com

INTERNATIONAL JOURNAL OF ADVANCED RESEARCH

RESEARCH ARTICLE

VALIDATION OF ENDF/B-VII.0 NUCLEAR DATA LIBRARY FOR SHIELDING CALCULATIONS USING THE MONTE CARLO METHOD

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Manuscript Info

Manuscript History:

Received: 15 January 2014
Final Accepted: 28 February 2014
Published Online: March 2014

Key words:
ENDF/B-VII.0, shielding benchmark experiments, Pressurized Water Reactor, PCA-REPLICA Benchmark, NESDIP benchmark, MCNPX.

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Abstract

The ENDF/B-VII.0 nuclear data library has been widely adopted by researchers in nuclear science and technology applications and especially for the Integral validation testing for a variety of quantities: For nuclear criticality, shielding applications ...; in this paper we have been validated this library through some benchmarks: the PCA-REPLICA and NESDIP experiments for the Light-Water reactor (LWR) pressure vessel facility benchmark. The principal goal of those experiments is the acquisition of benchmark-quality integral neutron data to validate the shielding transport methods and the nuclear data library used in the calculation of the ex-core fluxes in LWR. For this analysis, a 3-D model of the benchmarks experiments chosen for this study, are established by the mean of the continuous energy Monte Carlo code MCNPX.

The benchmarks analysis was made based on the calculated-to-measured (C/M) dosimetry reaction ratesof $^{103}Rh(n,n')^{103m}Rh$, $^{32}S(n,p)^{32}P$ and $^{115}In(n,n')^{115m}In$ monitors, at different depth in a water/iron shields reproducing the ex-core radial geometry of a PWR. The comparisons between reference Monte Carlo calculations and experimental results of PCA-REPLICA and the NESDIP showed us that the calculation method with the ENDF/B-VII.0 nuclear data library is effective for the shielding study of the REP, and generally, the obtained results show a high degree of agreement with the experiment an especially for PCA-REPLICA benchmark.

Our choice of MCNPX code is based on the main are used routinely for studying radiation shielding analysis and the fact that feature a rich palette of Variance Reduction (VR) techniques, which play a very important role to reduce the computer required to obtain results of sufficient precision. Some of these VR techniques have been used in the frame of this protection study such as the energy cut-offs (CUT card), the Geometry splitting and Russian roulette (IMP card) and the phase space
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Introduction

The increasing interest in the testing of the nuclear data libraries for shielding calculations in nuclear fission reactors, have proliferated dramatically in the last two decades. Thus, this study makes it often to validate the nuclear data library, and is also to establish trends on nuclear data and to recommend improvement for evaluators. This paper does exactly this for ENDF/B-VII.0 nuclear data library (A. D. Carlson et al., 2009; S. C. van der Marck, 2006).

In the present work we aims to contribute validation of the ENDF/B-VII.0 cross section library and the method of calculation, through analysing of some benchmarks: the PCA-REPLICA (Pescarini , et al,1997) and NESDIP (Armishaw, MJ, et al, 1986; Song Hui Zheng, 1993) experiments for the LWR pressure vessel facility benchmark performed at AEA Technology, Winfrith, using the Monte Carlo (Spanier and Gelbard, 1969) radiation transport code MCNPX (MCNPX^{TM} USER'S MANUAL, 2011) which allows simulating all types of interactions and using point-wise cross sections.

Our choice of the ENDF/B-VII.0 nuclear data library which is currently distributed with the MCNP code is based on the main is widely adopted by researchers in nuclear science and technology. It represented the culmination of many years of work by the US nuclear data evaluation, simulation, theory and modelling, and experimental research communities, and benefited from international collaborations, especially for the standards and for the integral data validation testing.

The principal results required from the benchmark analysis are the calculated-to-measured C/M ratios of the dosimetry reaction rates, at different depths in a water/iron shield reproducing the ex-core radial geometry of a PWR for the two benchmarks. The detectors used $^{32}S(n,p)^{32}P$, $^{103}Rh(n,n')^{103m}Rh$ and $^{115}In(n,n')^{115m}In$. Comparisons with experiment and results using MCNP code are made for the aim to obtain our objective.

The detectors cross section data were taken from the International Reactor Dosimetry File IRDF-2002 (O. Bersillon et al., 2006), and they have been processed into libraries suitable for use with the MCNP code using the NJOY99 system (MacFarlane, 2002) in conjunction with its most recent updates "up364".

Appropriate Variance Reduction Techniques (VRT) (J.C. Wagner, 2002; T.E. Booth, 1985) available by the MCNP code, were used in the calculations balancing between the statistical errors acceptability and the computer calculation time.

Materials and Methods

2.1 Monte Carlo Methods and MCNP code

Monte Carlo (MC) methods are considered to provide the best calculation engine available today for shielding calculations in nuclear fission reactors. These methods which are stochastic techniques are based on the use of random numbers and probability statistics to investigate problems. Thus MC methods are a collection of different methods that perform the same process: this process involves performing many simulations using random numbers (J.K. Shultis, 2011) and probability distributions to get an approximation of the answer to the problem. One of the most important usages of MC methods concerns the evaluation of difficult integrals. This is especially true for multi-dimensional integrals for which few analytical computation methods are suitable to get their appropriate approximation due to their complexity. It is in these situations that MC approximations become a valuable tool to use; they may be able to give a reasonable approximation with higher accuracy comparatively to other formal techniques.

The Monte Carlo method is, by its nature, very time consuming which constitutes her main disadvantage. In fact many Monte Carlo calculations absolutely require variance reduction methods to achieve practical computation time and improving the statistical results. The MCNPX (Monte Carlo N-Particle) code, which is developed by Los Alamos National Laboratory (LANL) has dramatically used for radiation shielding analysis and dosimetry calculations. This code feature a rich palette of VR techniques and it are attractive for such applications because of their ability to accommodate complex 3-D geometries, inclusion of flexible physics models that provide coupled electron-photon and neutron-photon transport, and the availability of extensive continuous-energy cross section libraries derived from evaluated nuclear data files. It should be noted that these code is general purpose in nature, and are therefore not optimized for any particular application. It is left up to the user to select appropriate variance reduction (VR) tools from a rich palette of such tools bundled with this code. An excellent overview of classic MCNP VR techniques was given by Booth (T.E. Booth, 1985)

The efficiency (ε) of a Monte Carlo simulation may be quantified using the Figure of Merit (FOM), which is defined by the MCNP code developers according to Equation (1) : where: T: is the total calculation time (CPU) for N histories, σ: is the tally relative statistical uncertainty.

$$\varepsilon = FOM = \frac{1}{T.\sigma^2} \qquad (1)$$

For a fixed computing time, the smaller is the variance the larger will be the FOM. It should be noted that σ^2 is proportional to 1/N, where N is the total number of histories and T is proportional to N. Thus, for a well converged simulation, the FOM becomes constant.

2.2. **Shielding Benchmarks Experiments**

The experiments analysed were carried out in the UK (United Kingdom). at the Atomic Energy Establishment of Winfrith in the ASPIS facility of the NESTOR low-flux experimental reactor.

2.2.1. The ASPIS Shielding Facility

The ASPIS shielding facility region (N.D. Carter,I .J . Curl 1986; J. Butler, et al, 1989). is installed on the NESTOR reactor at Winfrith, UK. NESTOR is a light-water cooled, graphite and light-water moderated reactor which operates at powers of up to 30 kW and is used as a source of neutrons for a wide range of applications. The ASPIS shield components are mounted vertically in a mobile tank which has an internal cross-sectional area of 180×190 cm and a length of 370 cm. A fission plate manufactured from 93% enriched uranium/aluminium alloy is located within the experimental shield array. The loaded tank is moved into the cave where the thermal neutrons leaking from the outer graphite reflector of the NESTOR are used to drive the fission plate to provide a well-defined neutron source for penetration measurement.

ASPIS can be used to carry out simple slab geometry benchmarks, for testing nuclear data and method approximations, or large scale engineering benchmark of complicated features of reactor plant, for the further analysis of method approximations and geometric modeling assumptions. The benchmarks experiments chosen are the PCA-REPLICA and NESDIP for a LWR pressure vessel facility benchmark

2.2.2. Description of the PCA-REPLICA benchmark experiment

The analysed benchmark experiment PCA-REPLICA was the preliminary experiment of the NESTOR Shielding and Dosimetry Improvement Programme (NESDIP) and it was carried out in the ASPIS facility of the NESTOR low-flux experimental reactor in the UK Atomic Energy Establishment of Winfrith (P.C. Miller, 1988)

The PCA-REPLICA experimental facility duplicated exactly 12/13 geometrical configuration of the Oak Ridge PCA (Pool Critical Assembly) experiment at ORNL (Oak Ridge National Laboratory) (12 and 13 cm of water respectively between the core and thermal shield and the RPV with the exception that the reactor source was replaced by a rectangular fission-plate to provide a well characterized neutron source, to investigate further the discrepancies between measurements and calculations observed in the PCA studies. During the experiment the rectangular fission-plate was irradiated by the NESTOR reactor (30 kW max power) through a graphite thermal column, of total thickness 43.91 cm, in the ASPIS shielding facility, and it is constructed ^{235}U enriched to 93 % surrounded by an aluminium layer.

The PCA-REPLICA experiment duplicated precisely the Oak Ridge PCA (Pool Critical Assembly)12/13 configuration, with the exception that the reactor source was replaced by a thin fission plate to provide a well-characterised neutron source. The 12/13 configuration implies that 12 and 13 cm of water gaps are placed between the core and the thermal shield and between the thermal shield and the reactor pressure vessel (RPV), respectively. This experiment was performed to determine the neutron spectra and detector reaction rates at different depths in a water/iron shield reproducing the ex-core radial geometry of a PWR. three types of thresholds detectors give the gave the integral measurements (reaction rates) at different depth in a water/iron shield reproducing the ex-core radial geometry of a PWR: $^{103}Rh(n,n')^{103m}Rh$, $^{32}S(n,p)^{32}P$ and $^{115}In(n,n')^{115m}In$. These detectors were located at 10 different positions: in the water gaps 1.91, 7.41, 12.41, 14.01, 19.91, 25.41, 30.41 cm from the fission plate (Rh measurements only) and at 1/4 and 3/4 thickness of the RPV and in the void box (Rh, In, S). The threshold energies are 0.01, 0.3 and 2 MeV, respectively. For the purpose of developing our 3-D model of the benchmark study, we have used the Monte Carlo N-particle transport code MCNPX which allows describing the real geometry and materials composition. This leads to minimize the number of approximations and to highlight the errors linked to nuclear data libraries.

Figs 1 and 2 represent the radial shield and the source plate respectively modelled with MCNP.

2.2.3 NESDIP Benchmark experiment

The NESDIP benchmark experimental array is shown schematically in Figure 3. The shield simulates the radial shield of a PWR and consists of 12.1cm of water, a 5.9cm stainless steel plate simulating the thermal shield, 13.2cm of water, five mild steel plates giving a thickness of 22.8cm to simulate the pressure vessel, a 29.6cm cavity region and a backing shield of aluminium, water and mild steel. The neutron reaction rate were measured in the experimental configuration of the NESDIP shield, used the same detectors as the PCA- REPLICA with the exception of the $^{115}In(n,n')^{115m}In$ detector.

This experiment is modelled in 3-D with the maximum of detail of the geometry and the materials compositions, and with little uncertainty using the Monte Carlo code MCNP. This model guarantees high level of fidelity in the description of the various features for all the components of the shield and the fission plate. Figs 4 and 5 illustrate our MCNP model of the shield the radial shield and the fission plate respectively.

Results and Discussion

The dosimetry reaction rates for each benchmark experiment have been calculated with the MCNP code for several positions. Tables 1–4 show the calculated-to-measured (C/M) ratios of the dosimetry reaction rates obtained for the threshold detectors used in the PCA-REPLICA, NESDIP respectively.

In the PCA-REPLICA benchmark it observed that the Calculation results are below the experiment at all the $^{103}Rh(n,n')^{103m}Rh$ detector positions with a maximum underprediction of 14% in the last point of the first water gap, while those for the $^{32}S(n,p)^{32}P$ detector tend to increase with the neutron penetration depth with a maximum overprediction of 4% in the cavity.

For the three detectors, the results are well maintained in the simulator RPV within a maximum discrepancy of 4%. However the results are worse in the water gaps for the $^{103}Rh(n,n')^{103m}Rh$ detector, for this detector the reaction rate show a remarkable underestimation in the last point of each water gap, (14% for the first water gap and 6% for the last one). This underprediction is less important (~ 5%) after having crossed a Stainless steel plate (region of the S. Steel in the figure 1). This remark makes that it necessary to qualify the effective section of the water.

The average C/M ratios of the dosimetry reaction rates showed good agreements for $^{115}In(n,n')^{115m}In$ and the $^{32}S(n,p)^{32}P$ detectors, and especially for the higher threshold detector $^{32}S(n,p)^{32}P$ (response domain is superior to 2 MeV), this coherence means that the cross sections of the water at haut energy are just.

In the NESDIP benchmark, the C/M ratios for the two detectors used for this benchmark are underestimating, within a maximum of 21% for the $^{103}Rh(n,n')^{103m}Rh$ detector and 23% for the $^{32}S(n,p)^{32}P$ detector.

In the water gap the C/M ratios for $^{103}Rh(n,n')^{103m}Rh$ and $^{32}S(n,p)^{32}P$ detectors decrease rapidly. However the results of the two detectors tend to increase with the thickness of the steel, and can remark that the results of the $^{103}Rh(n,n')^{103m}Rh$ detector show a similar tendency when compared to the other detector.

The average C/M ratios shows that the $^{103}Rh(n,n')^{103m}Rh$ detector represents well the attenuation of the neutron through the NESDIP shield compared to the $^{32}S(n,p)^{32}P$ detector.

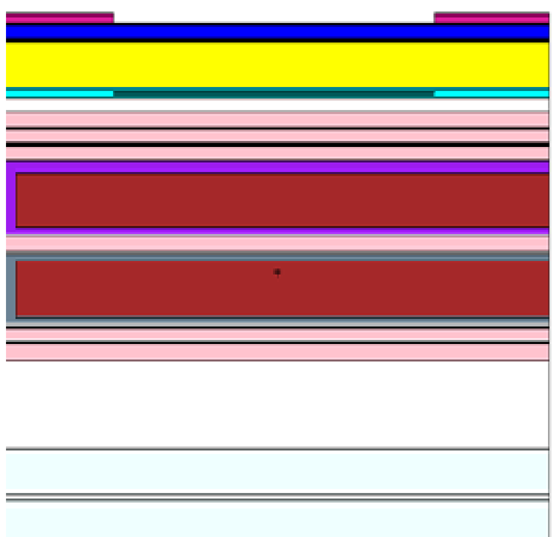

Fig. 4. MCNP model for the radial shield of benchmark NESDIP

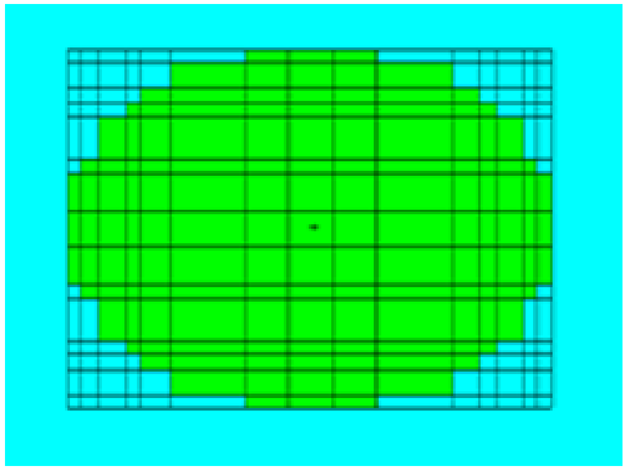

Fig. 5. MCNP model for the source plate geometry

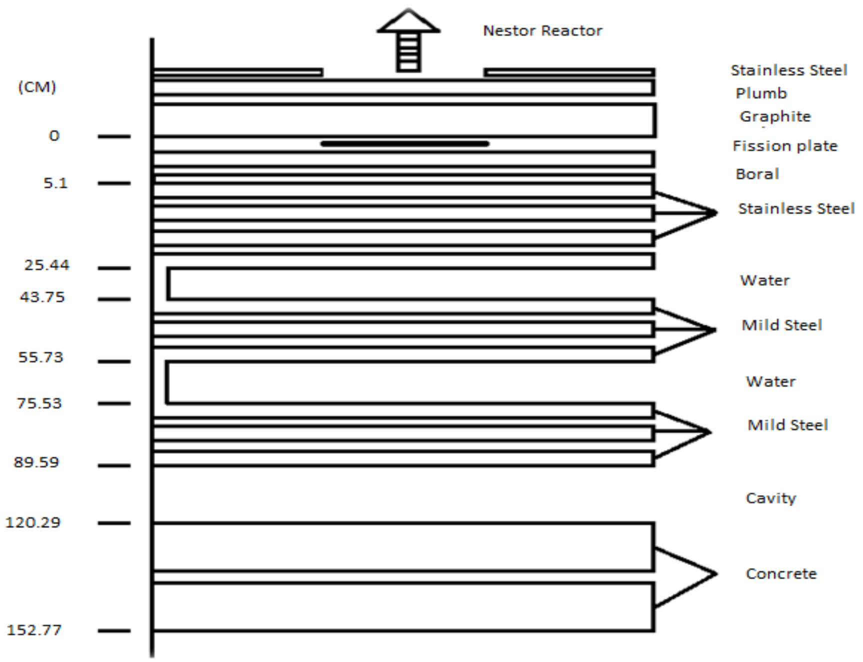

Fig. 3. Radial protection of the NESDIP benchmark experiment

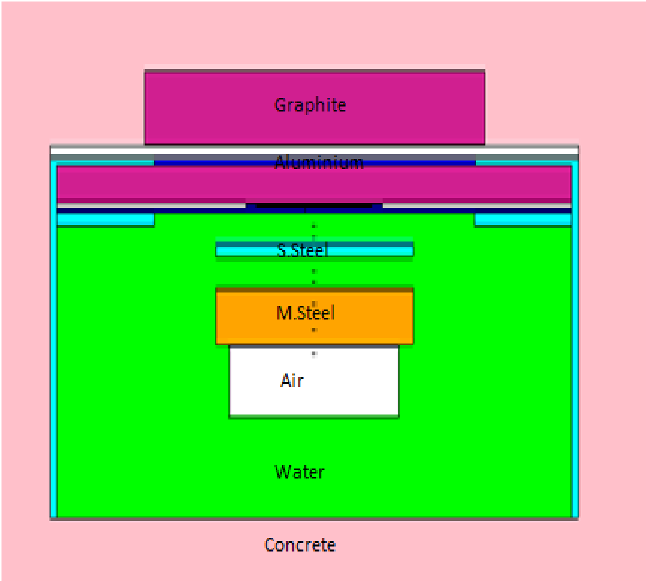

Fig.1. MCNP model for the radial shield of benchmark PCA-REPLICA

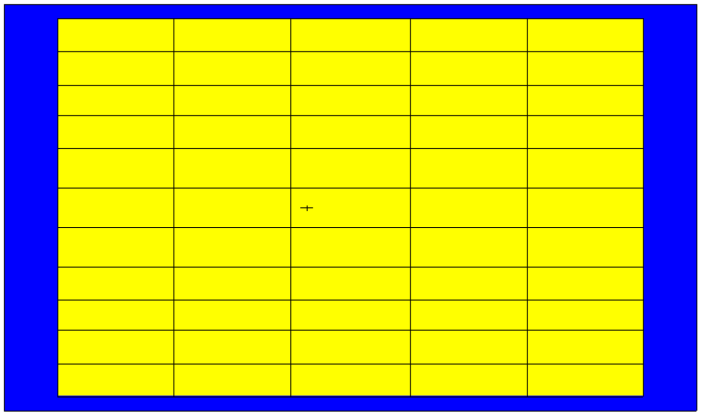

Fig. 2. MCNP model for the source plate geometry

Table 1. Calculated-to-measured ratio of the dosimetry reaction rates in PCA-REPLICA.

Reference	Distance from Fission plate (cm)	$^{103}Rh(n,n')^{103m}Rh$	$^{32}S(n,p)^{32}P$	$^{115}In(n,n')^{115m}In$
12cm water gap	1.91	0.96 (±0.2) $^{(a)}$		
	7.41	0.89 (±0.4)		
	12.41	0.90 (±0.63)		
	14.01	0.86 (±0.75)		
13cm water gap	19.91	0.95 (±1.05)		
	25.41	0.95 (±1.01)		

	30.41	0.94 (±0.88)		
RPV: T/4	39.01	0.98 (±0.31)	0.95 (±0.5)	1.00 (±0.23)
3T/4	49.61	0.99 (±0.32)	0.98 (±0.33)	0.96 (±0.18)
Cavity	58.61	0.94 (±0.46)	1.04 (±0.33)	0.98 (±0.21)
Average		0.94	0.99	0.98

[a] The uncertainty (%) of the measurements

Table 2. Calculated-to-measured ratio of the dosimetry reaction rates in NESDIP.

Reference	Distance from Fission plate (cm)	$^{103}Rh(n,n')^{103m}Rh$	$^{32}S(n,n')^{32}P$
Stainless Steel	5.1	0.87 (±0.4) [a]	0.77 (±0.5)
	10.18	0.95 (±0.5)	0.82 (±0.5)
	15.96	0.93 (±0.7)	0.80 (±0.7)
	21.74	0.92 (±0.8)	0.85 (±0.7)
Water gap	25.44	0.91 (±1.01)	-
	29.9	-	0.81(±1.01)
	32.94	0.88 (±1.01)	-
	34.44	-	0.80 (±1.01)
	40.04	0.79(±1.01)	-
Stainless Steel	46.75	0.83 (±1.03)	0.86(±1.02)
	52.53	0.82 (±1.3)	0.91 (±1.02)
Average		0.88	0.83

Conclusion

The results obtained through the analysis of the PCA-REPLICA and NESDIP experiments performed in the ASPIS shielding facility of the UK ensures that the library has been validated, and considering the uncertainties of the measurements, the library could be applied to the shielding calculations. The results showed also us that calculation method (Monte Carlo) is effective for the protection study of the REP. Generally in the PCA-REPLICA benchmark, the average C/M ratios obtained for the three detectors are reasonably good when the uncertainties of the measurements are taken into account, and generally the average C/M ratios for each detector in the NESDIP benchmark are worse than in the PCA-REPLICA.

Acknowledgements

The authors would like to thank the team of Radiation and Nuclear System in the Abdelmalek Essaadi University especially the professor El Bardouni Tarek and Boukhal Hamid for their great scientific advices.

References

Armishaw, MJ, et al," NESTOR Shielding and Dosimetry Improvement Programme," Cavity Nozzle and Coolant Duct Benchmark Blind Test Edition, AEEW-M2334, Reactor Physics Division, AEE Winfrith, July 1986.

A.D.Carlson,V. G. Pronyaev, D. L. Smith, N. M.Larson, Z.-P. Chen, G. M. Hale, F. J. Hambsch, E. V. Gai, S.-

Y. Oh, S. A. Badikov, T. Kawano, H. M. Hofmann, H. Vonach, and S. Tagesen,“International Evaluation of Neutron Cross Section Standards,” Nuclear Data Sheets 110, 3215 (2009).

Chadwick et al., 2006 Chadwick, M.B., et al., 2006. ENDF/B-VII.0.0: Next Generation Evaluated Nuclear Data Library For Nuclear Science and Technology, Nuclear data sheets, vol.107, pp.2931–3060.

J. Butler, M.D. Carter, I.J. Curl, P.C. Miller, A. Packwood, S.Power, Review of the NESTOR Shielding and Dosimetry Improvement Programme (NESDIP), Reactor Dosimetry: Methods, Applications, and Standardization, ASTM STP 1001, Harry Farrar IV and E.P. Lippincott Editors, ASTM, Philadelphia 1989, pp. 295-307.

J.C. Wagner, “An Automated Deterministic Variance Reduction Generator for Monte Carlo Shielding Applications,”Proc. American Nuclear Society, Rad. Prot. Division, 12th Topical Meeting, Santa Fe, NM USA, Apr.14-18,2002.

J. K. Shultis and R. E. Faw, “An MCNP Primer,” Kansas State University, Manhattan, 2011.

J. S. Hendricks, G. W. McKinney, L. S. Waters, et al., MCNPX, Version 2.5.e, LA-UR-04-0569, Los Alamos National Laboratory, Los Alamos, NM (2004).

MacFarlane, R.E., 2002. NJOY-99 Nuclear Data Processing System [online]. On line available from http://t2.lanl.gov/codes/njoy99/index.html (update web site).

MCNPX™ USER’S MANUAL”, Version 2.7.0, LA-CP-11-00438, Manuel/ Los Alamos National Laboratory, April 2011.

N.D. Carter,I .J . Curl, NESTOR Shielding and Dosimetry Improvement Programme. The ASPIS-PCA Slab Geometry Benchmarks. Blind Test Edition. AEEW-W 2329, UKAEA, Winfrith, June 1986 (Corrigenda).

O. Bersillon, L.R. Greenwood, P.J. Griffin, W. Mannhart, H.J.Nolthenius, R. Paviotti-Corcuera, K.I. Zolotarev, E.M. Zsolnay, P.K. McLaughlin, A. Trkov, International Reactor Dosimetry File 2002 (IRDF-2002), 2006, International Atomic Energy Agency, Vienna, Austria, Technical Reports Series No. 452, ISBN 92-0-105106-9, December 2006.

P.C. Miller, A REVIEW OF LWR PRESSURE VESSEL DOSIMETRY AND ASSOCIATED SHIELDING STUDIES, Proceedings of the UKAEA/OECD-NEA 7th International Conference on Radiation Shielding, Bournemouth (UK), September 12-16, 1988, VOL. I, pp. 33-47.

Pescarini, M. and Borgia, M. G. JENDL-3.1 iron validation on the PCA-REPLICA (H2O/Fe) shielding benchmark experiment. Report No. RT/ERG/97/1 Bologna: (ENEA) (1997).

S. C. van der Marck, “Benchmarking ENDF/B-VII.0,” Nuclear Data Sheets 107, 3061 (2006).

Song Hui Zheng : « qualification des méthodes de calculs de fluence dans les réacteurs à eau pressurisées. Amélioration du traitement des sections efficaces par la méthode des tables de probabilité ». Thèse de l’Université de Paris-Sud, numéro d’ordre 2887, Note CEA-N-2745, 1993.

Spanier, J., Gelbard, E.M., 1969. Monte Carlo Principles and Neutron Transport Problems. Addison-Wesley Publishing Co., Reading (MA).

Sunny, C.S., et al., 2008. KAMINI reactor benchmark analysis. Annals of Nuclear Energy 35.

T.E. Booth, “A Sample Problem in VR in MCNP,” Los Alamos National Laboratory, LA-10363-MS (1985).