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Analysis of VENUS-3 Benchmark Experiment

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ABSTRACT - The paper presents the revision and the analysis of VENUS-3 benchmark experiment performed at CEN/SCK, Mol (Belgium). This benchmark was found to be particularly suitable for validation of current calculational tools like 3-D neutron transport codes, and in particular of the 3D sensitivity and uncertainty analysis code developed within the EFF project. The compilation of the integral experiment was integrated into the SINBAD electronic data base for storing and retrieving information about the shielding experiments for nuclear systems. SINBAD now includes 33 reviewed benchmark descriptions and several compilations waiting for the review, among them many benchmarks relevant for pressure vessel dosimetry system validation.

1. Introduction

Computational methodologies used in reactor pressure vessel neutron fluence calculations are usually checked against surveillance capsule and/or cavity dosimetry measurements. These measurements, although being the closest representation of the real pressure vessel neutron environment, have the inconvenience of relatively large uncertainties associated to the calculated neutron fluence. These include, besides generic uncertainties (like nuclear data, radiation source), also plant specific uncertainties (e.g. material compositions, dimensions, operating conditions) which are complex and often difficult to establish precisely [1], [2].

The recommended way to determine the quality and merit of the data files, including cross sections, radiation sources, etc., as well as of calculational procedures is to test them against some well defined benchmark experiments and shield design problems.

The advantage of benchmarks is that the uncertainties, other than those due to the nuclear data, are reduced considerably. They can thus provide indications on needed nuclear data adjustments more accurately, and can serve as test of the computational methods and their ability to meet required standards and safety regulations.

The "lifetime" of well defined benchmark experiments is usually in addition much longer than that of nuclear data libraries, basic data evaluations and computer codes and methods. Although designed to validate some specific computational methods and data at a given time, benchmark experiments do not become obsolete but can very well serve for testing further methods or data that will be developed and produced later. This shows the importance of preserving and making easily accessible the information on these benchmarks.

The SINBAD project [3] (A Shielding Integral Benchmark Archive and Database for PCs) operated by the Oak Ridge National Laboratory's Radiation Safety Information Computational Center (RSICC) and the OECD/NEA Data Bank was started with the objective to develop an electronic data base for storing and retrieving information about the shielding experiments for nuclear systems. The SINBAD data base includes description of experiments, experimental results, and calculational models, as well as stores relevant figures and reports in the digitized page image form.

An integral part of the SINBAD project involves the development of an experimental data base for the reactor pressure vessel dosimetry. The benchmarks relevant for pressure vessel dosimetry system validation are listed in Table 1.

An example of the revision and the analysis of VENUS-3 benchmark [4] [5] is given in this paper. In the mockup, the rods of peripheral assemblies were replaced by partial length shielded assemblies (PLSA) made of half a length of stainless steel and half a length of UO_2 fuel. VENUS-3 was designed essentially to test the refueling patterns reducing the pressure vessel exposure and the ability of the fluence-rate synthesis procedure based on 2-D/1-D calculations to predict adequately 3-D geometry effects. In recent years important progress has been made concerning discrete ordinates and Monte Carlo radiation transport codes. Some powerful 3-D codes were developed, like TORT [6], THREEDANT [7] and were released through RSICC and NEA-DB. Instead of a series of 1-D and 2-D discrete ordinates calculations, a single one should now be sufficient.

2. VENUS-3 Relative Power Distribution

VENUS-3 was revised to be used for validation of current calculational tools. Firstly, a complete 3-D map of the neutron source in the VENUS core was established. The VENUS-3 power distribution was fully measured only at two (out of 14) axial levels [8]: at the axial level corresponding to the mid-plane of the lower-PLSA part of the core loading and at the one corresponding to the mid-plane of the upper part of the core loading. In addition to the two X-Y radial distributions, the full axial power distribution was measured at 374 fuel pin locations, out of the total of 639 fuel pins comprised in the 1/4 of the reactor core.

In order to establish a complete 3-D map of the power distribution in the VENUS core, an extrapolation procedure, based on the RECOG-ORNL code [9] was used. This procedure permitted at the same time to detect some suspicious or faulty values, transcription errors, as well as to give an idea of the accuracy of the neutron source.

RECOG-ORNL is a general purpose pattern recognition code. Various methods for data analysis, preprocessing and display of data, unsupervised and supervised learning are available in the code. For our purpose basically LEAST and LLOOST supervised learning techniques were used. The techniques use known information about some of the data to generate predicted properties for an unknown set. LEAST calculates the predicted properties from the augmented variables of the pattern set. The relationship between predicted properties and augmented variables is deduced from the training set (measured set in our case), and is then applied to the nontraining set (missing power distributions).

LLOOST performs leave-out statistics on a pattern set. One (or more) patterns can be left out, and the same algorithm as in LEAST is used to predict their properties. This permits to check the validity of the method and the precision of the estimations. This technique gives an insight into the precision of the nuclear source data. It permitted also to detect some transcription errors and inconsistencies.

The source data file was prepared in a format suitable for transport codes. It contains the measured values, where available, and RECOG predicted values elsewhere. Relative nuclear power distribution, normalized to the core averaged power of one fission per second per active pin, is given for each fuel pin position and for 14 axial levels (see Figure 1).

In addition, an information on the accuracy of these values was evaluated in the following way: where the measured values were available the 'uncertainties' were determined as the % difference between the neutron source values calculated by RECOG code and the measured values obtained from SCK/CEN. Elsewhere the uncertainty of the extrapolation procedure in RECOG was estimated to +-5% [5].

3. Sensitivity/Uncertainty Analysis of VENUS-3

The transport methods are combined with sensitivity analysis; the combined application has proved very useful if not necessary in order to establish reliable safety margins for the measured and calculated values, as well as to determine to what extent the benchmark experiment is representative of the real nuclear reactor environment. A procedure which has been extensively used in uncertainty analysis for pressure vessel dosimetry is based on TWODANT, DOT, SUSD, ZOTT codes and the VITAMIN-J/COVA covariance matrix library [10], [11], [12], [2]. This system has been recently updated to include full 3-D sensitivity/uncertainty analysis [13]. The present version of the code can thus use angular flux or flux moment files calculated by discrete ordinates codes ANISN, DOT, TWODANT and TORT. Use of angular moment files instead of bulky angular flux files produced by these codes reduces considerably the size of the files required and represents an acceptable approximation for the problem type analysed here.

The first results of the sensitivity and uncertainty analysis are given in Table 1. The direct and adjoint transport calculations were performed using TORT code [6]. To save computer space and CPU time only very basic S_N approximations were chosen (S_4, P_1). Cross-sections were taken from the BUGLE-96 [14] library (47 neutron energy groups). VENUS-3 was described in rectangular (x-y-z) geometry, using variable space mesh.

Uncertainties due to the ENDF/B-V neutron fission source spectrum used was calculated assuming the uncertainties of 1.2% in the parameter A of the Watt distribution function, and of 5.9% in the parameter B [15]. The sensitivities were therefore expressed as the product of the adjoint fluxes and the sensitivity functions (derivatives) of the Watt spectrum distribution with respect to the parameters A and B [16].

The source space distribution uncertainties depend on the location in the reactor. These amount to about 3 % in the core barrel.

Detector response function covariances used to evaluate the uncertainties in the response functions were taken from IRDF 90.2 evaluation [17].

4. Conclusions

Due to its 3-D complexity VENUS-3 is well suited for validating dosimetry calculational methods using full geometry modelling.

The complete information including the geometry description, material composition, neutron source distribution, as well as examples of the calculations were prepared and will be included into the SINBAD data base to be available worldwide. The data to be included in the VENUS-3 SINBAD compilation was already verified by using it in transport calculations. The sensitivity and uncertainty studies which were started at the same time provide useful supplementary information on the benchmark experiment and measured quantities. Up to now the uncertainties due to the source space and energy distributions, and response functions were established. The sensitivity and uncertainty studies with respect to the transport cross-sections are still under-way.

By comparing the VENUS-3 sensitivity and uncertainty analysis with the similar ones performed in the scope of the reactor pressure vessel surveillance [1], [2] the representativity of VENUS-3 with respect to the real reactor environment can be determined.

Table 1: Results of uncertainty analysis: uncertainties due to the neutron source energy and space distributions, response functions and cross sections were taken into account.

Source of uncertainty		Uncertainty (%)			
		$\Phi > 1MeV$	$^{27}Al(n,\alpha)$	$^{58}Ni(n,p)$	$^{115}In(n,n')$
Fission spectrum		4.4	12	6.5	4.4
Source space distribution		$\sim$ 1.5 - 4			
Absolute power		4			
Response function		0	1.4	2.5	2.2
Cross sections	H	1.9	1.1	1.4	1.6
	O	0.6	1.6	0.7	0.5
	Fe	2	5	2.5	2.1
Total		~ 7	~ 14	~ 9	~ 8

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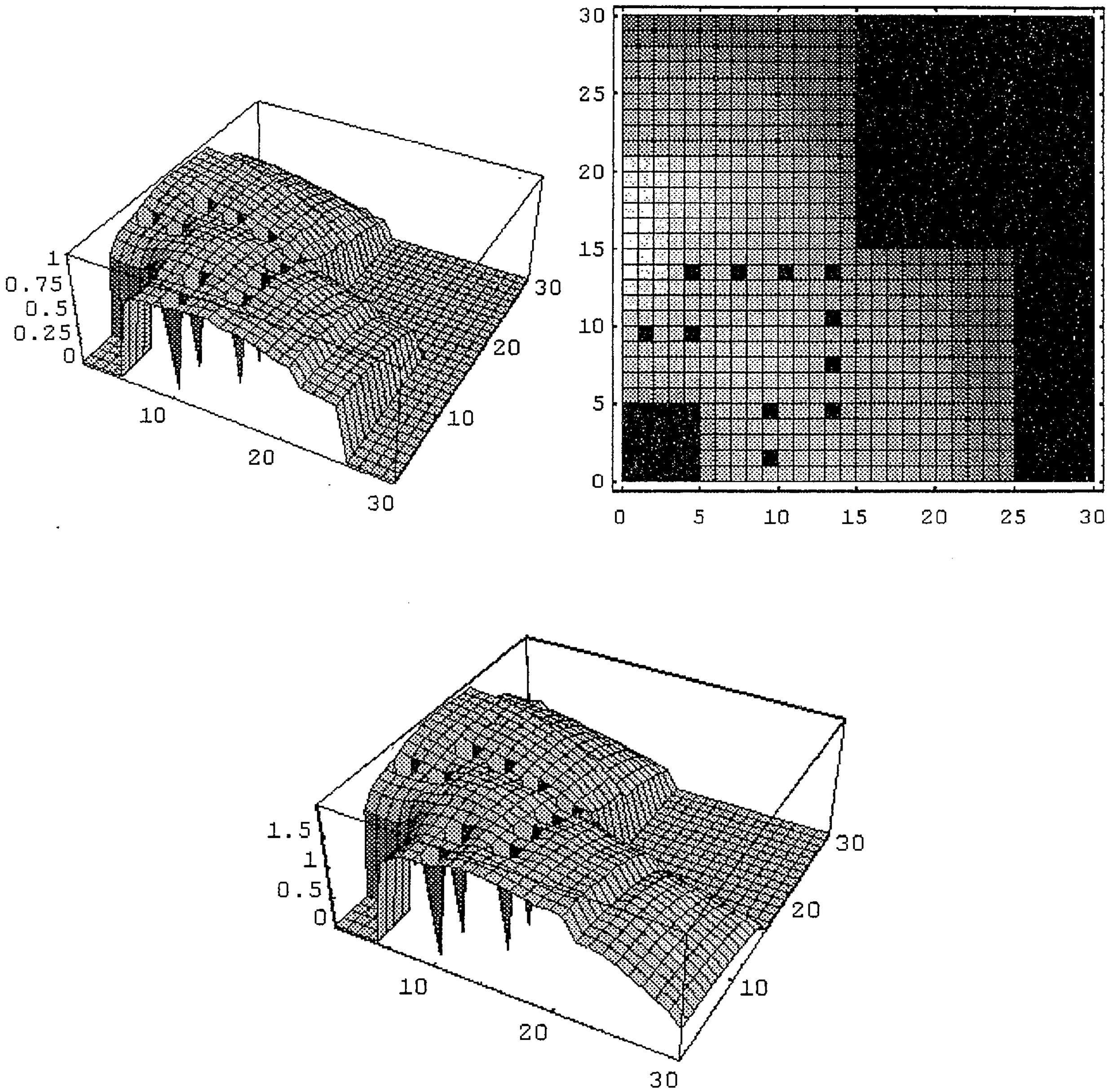

Figure 1. X-Y radial relative power distributions at axial levels 1 (above) and 9 (below).

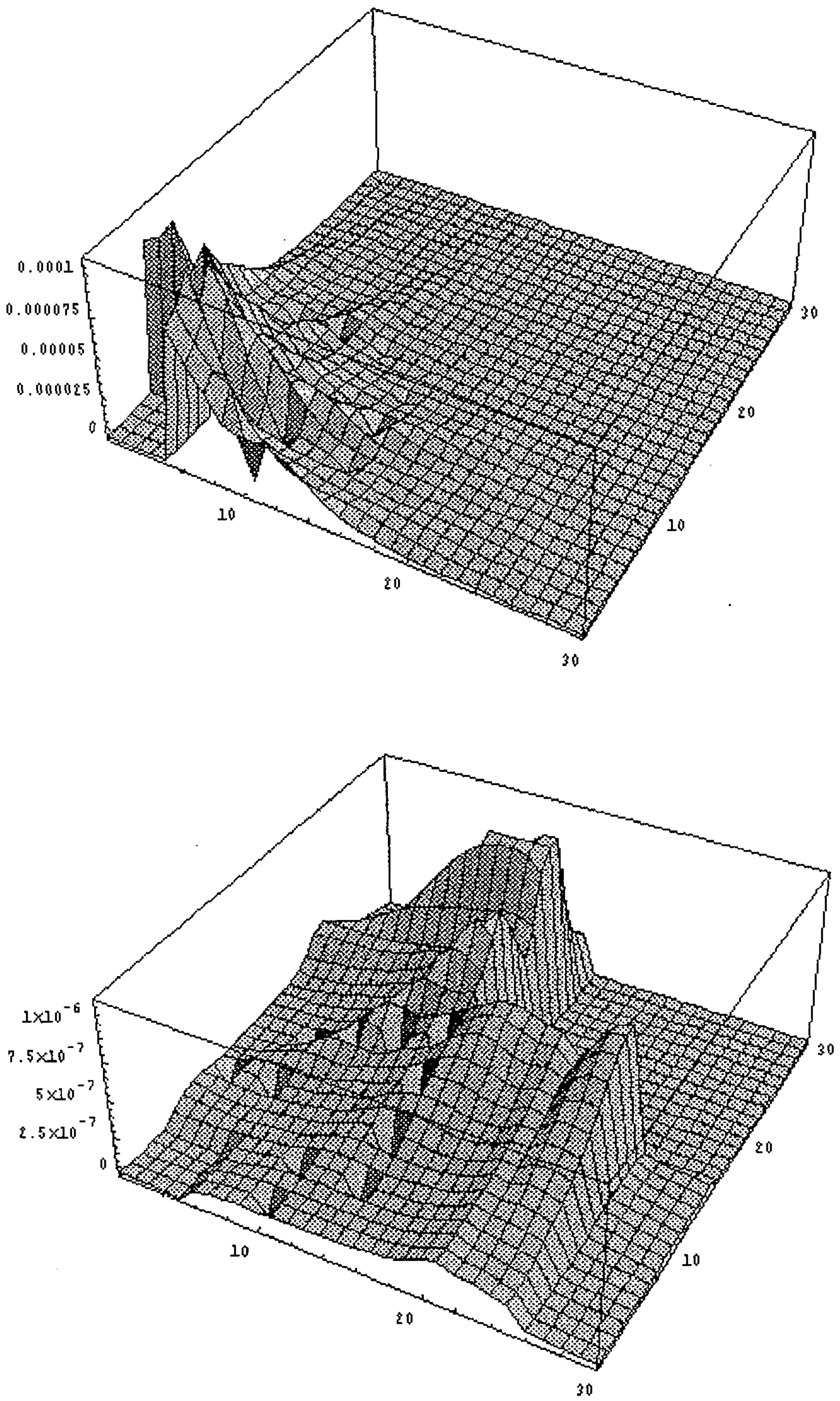

Figure 2. Neutron source distribution in the VENUS-3 core multiplied by the neutron source importance function (adjoint flux). Adjoint source was the Ni-58(n,p) response function in the inner buffle at z=114.5 cm (above) and Al-27(n,α) response function in the core barrel at z=131.5 cm (below). The sum over all axial levels in the core (from z=105 cm to 155 cm) is presented.