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Updated JEF-2.2 Pu data

The effect on several benchmarks

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Abstract

The ^{239}Pu data as given on the JEF-2.2 evaluation appear to be erroneous. The energy distribution for the partial fission cross sections are incorrect.

In this report it is shown, that updated ^{239}Pu data from the JEF-2.2 evaluation yield significantly better results than data from the original JEF-2.2 evaluation.

This is demonstrated in three PNL and three VALDUC plutonium nitrate critical benchmarks. Calculations were performed with the Monte Carlo code MCNP4A, using cross-section data from the EJ2-MCNPlib library.

The values of k_{eff} and relevant leakage and reaction rate data were calculated.

The values of k_{eff} are overpredicted by approx. 480 pcm, compared to approx. 1030 pcm when data from the original JEF-2.2 evaluation are used.

Keywords

Criticality calculations
Benchmark calculations
MCNP4A
EJ2-MCNPlib
JEF-2.2 cross-section data
Thermal Pu data

CONTENTS

1. INTRODUCTION

Core neutronics calculations require the benchmarking of nuclear data and neutron transport codes used in the analyses. Data for the important U isotopes have been extensively benchmarked in the previous years, also in the framework of the JEF-project (see *e.g.* [1,2]). However, recent benchmark calculations for thermal Pu data are scarce. These data play an important role if the neutronics behaviour of MOX fuel in overmoderated assemblies is studied. Therefore, a benchmark exercise was initiated by Nouri [3].

During the JEFF-meeting in July 1996 it appeared, that an error was present in the JEF-2.2 ^{239}Pu evaluation, which may change the results of many calculations.

In this report a comparison is presented of the results from several benchmark calculations with old, incorrect ^{239}Pu data with those in which updated ^{239}Pu data were used. Calculations were performed with the Monte Carlo neutron transport code MCNP4A, using data from the EJ2-MCNPlib [4] library, which is based on the JEF-2.2 evaluation.

In chapter 2 some details are given on the nuclear data used in the analyses. A short description of the benchmarks is given in chapter 3. The model which is used in the calculations is illustrated in chapter 4. The results of the calculations are given in chapter 5. Finally, in chapter 7 conclusions from this work are drawn.

2. NUCLEAR DATA

Recently it was discovered that a serious error is present on the basic JEF-2.2 evaluation for ^{239}Pu. It appears, that the energy distribution as given for the total fission cross section (MT18) is not consistent with the ones given for the partial fission cross sections (MT19, MT20, MT21 and MT38). Whereas the distribution given for the total fission cross section is believed to be correct, the distributions for the partial fission cross sections were used in the production of the EJ2-MCNPlib library.

In [5] results are presented of benchmark calculations in which data were taken from the EJ2-MCNPlib library [4]. It appeared that the experimental value of k_{eff} was severely overpredicted in these calculations.

In this report a comparison is given of results from benchmark calculations in which the original erroneous ^{239}Pu data were used and from calculations in which updated ^{239}Pu data were used. Updated data were obtained by removing the partial fission cross-section data (MT19, MT20, MT21 and MT38) from the evaluation. In this case the total fission cross-section data are used in the production of the MCNP data.

3. THE PNL AND VALDUC BENCHMARKS

Benchmark calculations for six thermal Pu systems were performed. The benchmarks were selected by Nouri in [3]. They can be subdivided in two groups:

1. PNL benchmarks:
   Plutonium nitrate in spherical geometry. Reflection by light water.
2. VALDUC experiments:
   Plutonium nitrate in cubic tank. Reflection by light water.

Plutonium concentrations range from 13.2 through 119 g/l. The amount of ^{240}Pu in plutonium ranges from 3.1 through 18.9%.

For all benchmarks two calculations were performed:

1. bare sphere geometry:
   Simple spherical model; no reflection. Equivalent bare sphere radii are provided by Nouri in [3].
2. simple geometry:
   Simplified geometrical model, taking into account the most prominent features of the actual geometry.

The PNL benchmarks all have a water reflector. In the VALDUC experiments two different geometries are studied:

1. water reflection at six sides of the fuel tank
2. water reflection at five sides of the fuel tank

Figures of the geometrical models are given in figs. 4.1 through 4.4.

4. CALCULATIONS

4.1 Geometrical model

Outlines of the PNL and VALDUC geometries are given in figs. 4.1 through 4.4. Data were taken from [6].

4.2 Calculational procedure

Neutron transport calculations for this benchmark were performed using the Monte Carlo code MCNP4A [7]. Continuous-energy cross section data were used, which allow for a very detailed simulation of the neutron transport.

The calculation of k_{eff} and the reaction rates was divided into three subsequent steps:

1. As an initial source S_0 an isotropic point source in the centre of the fuel pin was taken. Using a batch size of 100 neutrons, in 100 cycles a geometrically converged source S_1 was produced, which was used in the second step of the calculation.
2. Starting with S_1 a second calculation was performed, using a batch size of 1000 neutrons. In 100 cycles the initial source S_2 for the final step was produced.
3. In the final run the batch size was extended from 1000 to 10000 neutrons. The values of k_{eff} and the reaction rates were calculated in 100 cycles.

4.3 Cross sections

JEF-2.2 based cross-section data for all isotopes were taken from the EJ2-MCNPlib library [4], processed at ECN Petten.

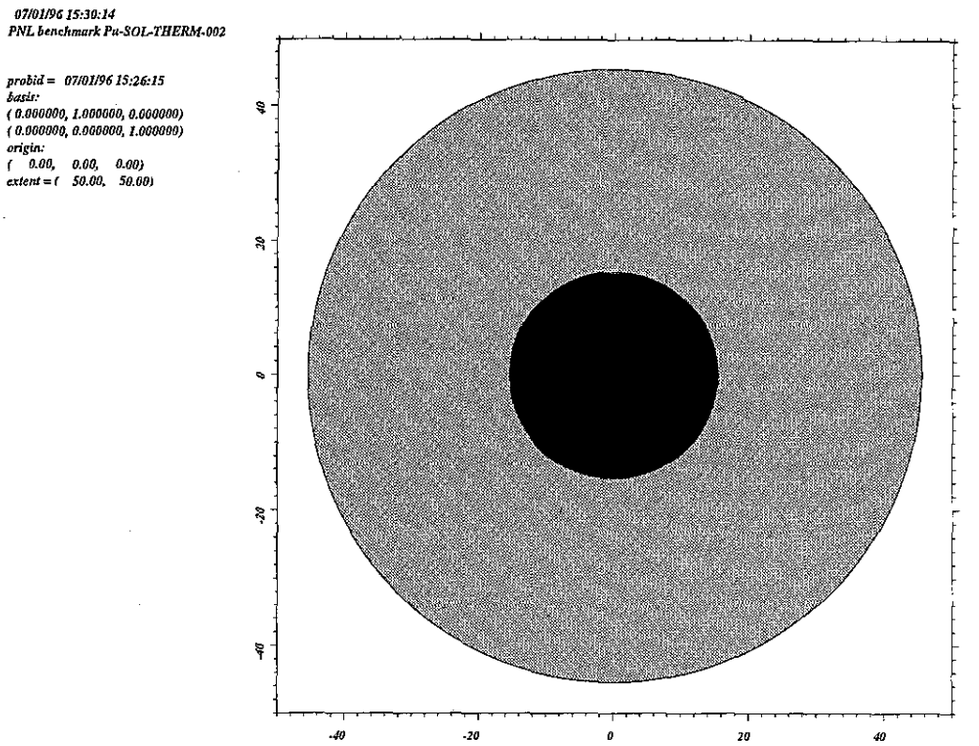

Figure 4.1 *Vertical cross section through the simple MCNP4A model of the PNL geometry (Pu-SOL-THERM-002 case 2). The fuel region is surrounded by a thin shell made of stainless steel and a reflecting water shell.*

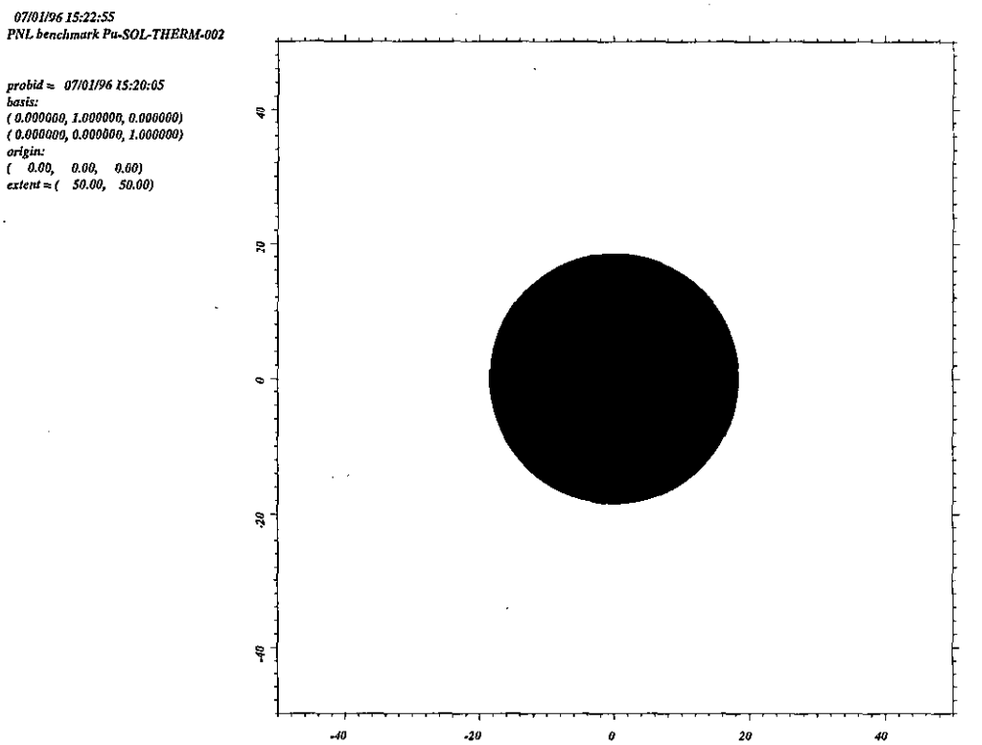

Figure 4.2 *Vertical cross section through the bare sphere MCNP4A model of the PNL geometry (Pu-SOL-THERM-002 case 2). The Pu nitrate solution is contained in a sphere without reflection.*

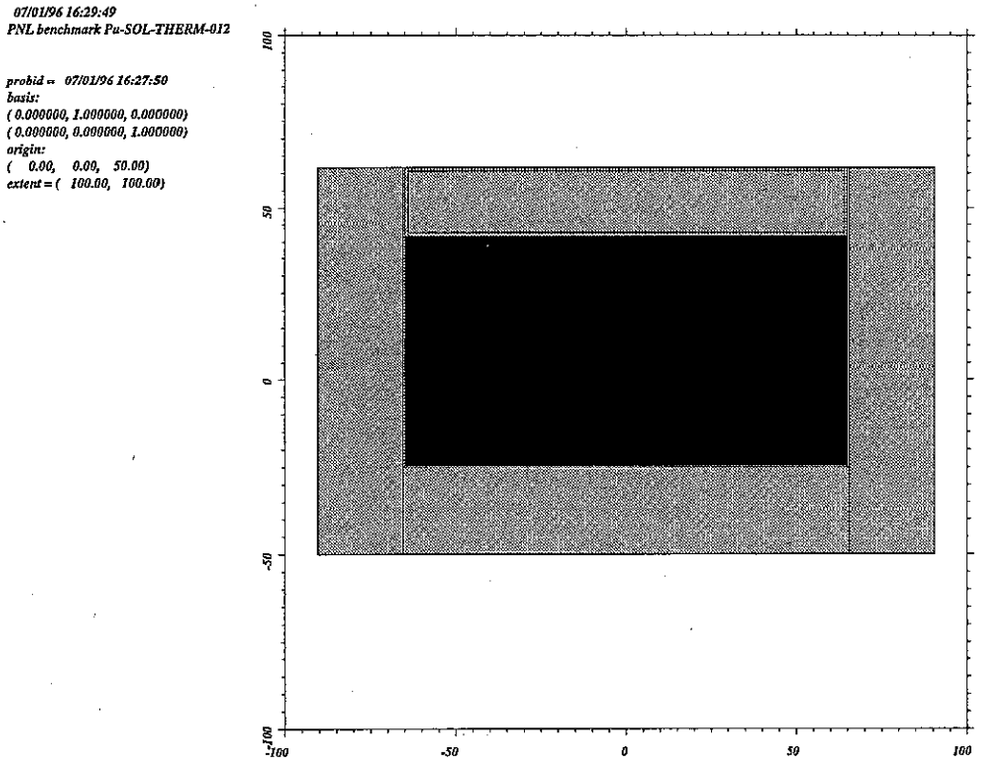

Figure 4.3 *Vertical cross section through the simple MCNP4A model of the VALDUC geometry with reflection on six sides (Pu-SOL-THERM-012 case 5). The fuel region is surrounded by a tank made of stainless steel. On top is a lucoflex box, filled with water. The fuel tank is placed in a pool filled with water.*

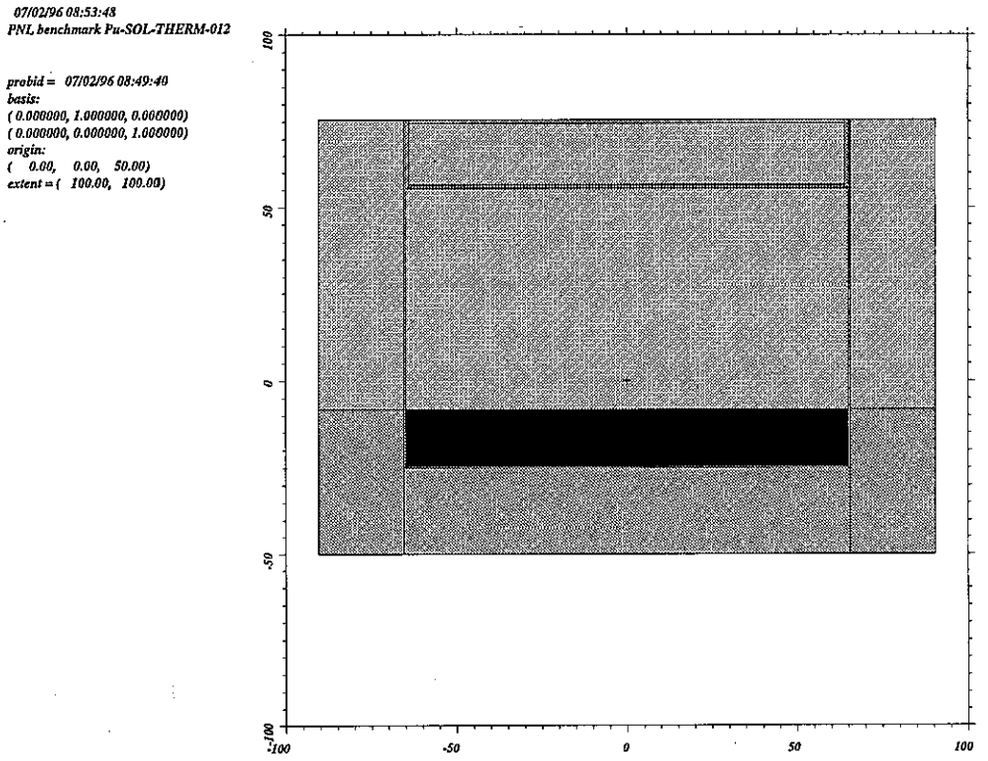

Figure 4.4 *Vertical cross section through the simple MCNP4A model of the VALDUC geometry with reflection on five sides (Pu-SOL-THERM-012 case 6). The fuel region is surrounded by a tank made of stainless steel. On top is a lucoflex box, filled with air. The fuel tank is placed in a pool filled with water.*

5. RESULTS

The neutron flux spectrum calculated in the fuel zone for several benchmarks is given in fig. 5.1. From this fig. it is clear, that in this benchmark exercise a broad range of spectra in thermal Pu systems is probed.

A compilation of calculated values for k_{eff} is given in table 5.1

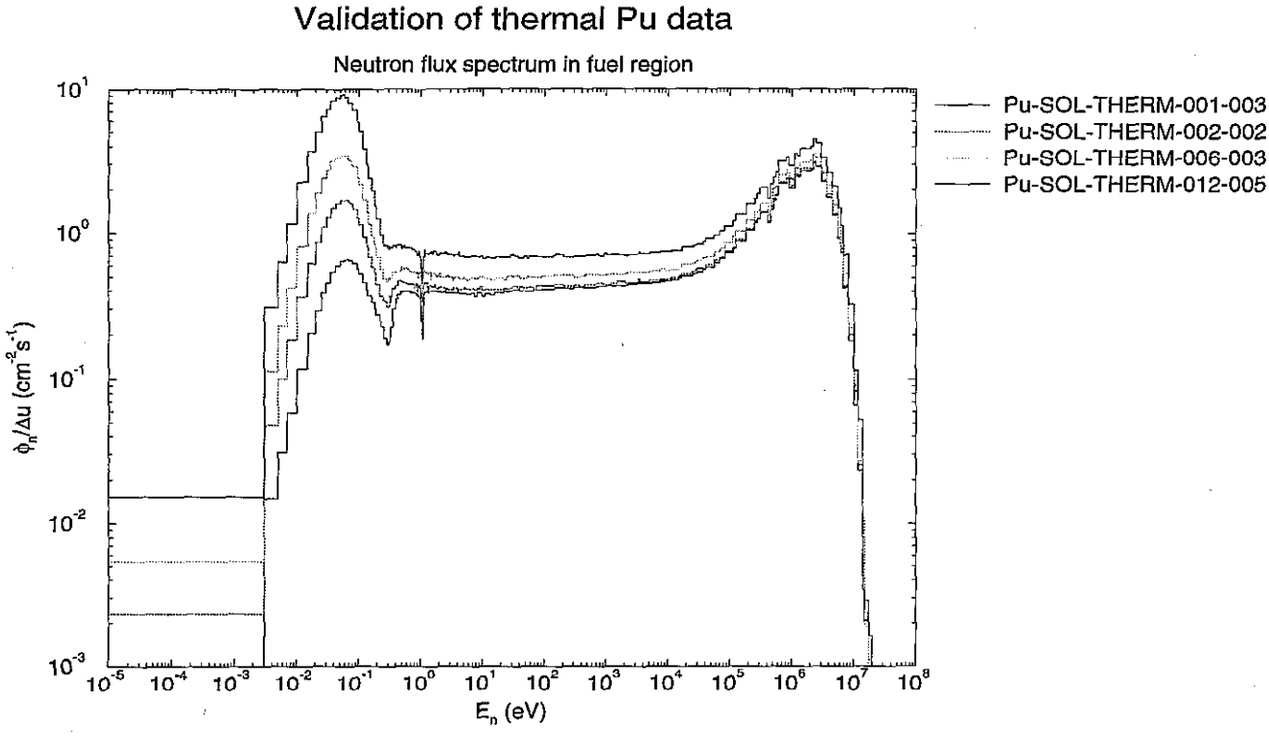

Figure 5.1 *Neutron flux spectrum calculated in the fuel zone in several benchmark geometries.*

Table 5.1 *Results from thermal Pu benchmark calculations. The bare sphere results refer to simplified equivalent bare sphere calculations. For some results the relative uncertainties are high. More accurate results are being obtained.*

identification Pu-SOL-THERM-	case #	C[Pu] (g/l)	geometry	bare sphere radius (cm)	$k_{eff} \pm\sigma$ [%]	$k_{eff}^{NEW} \pm\sigma$ [%]
001	case 3	119	bare sphere	17.91	1.0077 ± 0.07	1.0014 ± 0.30
001	case 3	119	simple geom.	-	1.0143 ± 0.06	1.0060 ± 0.07
002	case 2	51.42	bare sphere	18.47	1.0032 ± 0.10	0.9956 ± 0.10
002	case 2	51.42	simple geom.	-	1.0106 ± 0.10	1.0024 ± 0.09
006	case 3	26.97	bare sphere	22.09	1.0038 ± 0.09	0.9976 ± 0.08
006	case 3	26.97	simple geom.	-	1.0060 ± 0.07	0.9998 ± 0.08
012	case 5	13.2	bare sphere	53.52	1.0006 ± 0.12	0.9967 ± 0.08
012	case 5	13.2	simple geom.	-	1.0099 ± 0.16	1.0091 ± 0.05
012	case 6	105	bare sphere	21.41	1.0076 ± 0.10	1.0029 ± 0.32
012	case 6	105	simple geom.	-	1.0097 ± 0.10	1.0072 ± 0.10
012	case 8	52.7	bare sphere	22.25	1.0061 ± 0.08	0.9997 ± 0.08
012	case 8	52.7	simple geom.	-	1.0112 ± 0.09	1.0042 ± 0.28

The average value of k_{eff} obtained in a simple geometry model using updated

^{239}Pu data is overpredicted by 478 pcm. This value should be compared with an overprediction of 1028 pcm when original ^{239}Pu data are used.

In the bare sphere model the value of k_{eff} is underpredicted by 102 pcm when ^{239}Pu data are used. Using original ^{239}Pu data the vale of k_{eff} is overpredicted by 483 pcm.

6. DISCUSSION

From the results presented in chapter 5 it is clear, that with updated data from the JEF-2.2 evaluation the value of k_{eff} in thermal Pu systems is only slightly overpredicted in analyses with MCNP4A. A much better agreement is observed compared to previous calculations in which original JEF-2.2 data were used [5].

The results are now in good agreement with results obtained with French code systems (APOLLO-1 + MORET, APOLLO-2 + TRIMARAN-2, TRIPOLI-4) [8].

7. CONCLUSIONS

The comparison performed in this report shows, that updated ^{239}Pu data from the JEF-2.2 evaluation provide a much better agreement with experimental k_{eff} data in thermal Pu systems than ^{239}Pu data from the original JEF-2.2 evaluation.

The value of k_{eff} is only slightly overpredicted in simple realistic models (approx. 480 pcm) and slightly underpredicted in bare sphere models (approx. 100 pcm).

The results obtained agree with results from calculations with several French code systems.

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