Annals of Nuclear Energy 83 (2015) 137–146

Contents lists available at ScienceDirect

Annals of Nuclear Energy

journal homepage: www.elsevier.com/locate/anucene

Inventory calculation and nuclear data uncertainty propagation on light water reactor fuel using ALEPH-2 and SCALE 6.2

L. Fiorito [a,b,*], D. Piedra [c], O. Cabellos [c,d], C.J. Diez [c]

[a] Institute for Advanced Nuclear Systems, SCK•CEN, Boeretang 200, 2400 Mol, Belgium
[b] ULB, Université Libre de Bruxelles, Avenue Franklin Roosevelt 50, 1050 Bruxelles, Belgium
[c] Dpto. de Ingeniería Nuclear, Escuela Técnica Superior de Ingenieros Industriales, Universidad Politécnica de Madrid UPM, José Guitierrez Abascal 2, 28006 Madrid, Spain
[d] Instituto de Fusión Nuclear, Escuela Técnica Superior de Ingenieros Industriales, Universidad Politécnica de Madrid UPM, José Guitierrez Abascal 2, 28006 Madrid, Spain

ARTICLE INFO

Article history:
Received 7 November 2014
Received in revised form 17 March 2015
Accepted 21 March 2015
Available online 29 April 2015

Keywords:
Nuclear data
Uncertainty
Burnup
Nuclide density
Correlation

ABSTRACT

Two fuel assemblies, one belonging to the Takahama-3 PWR and the other to the Fukushima-Daini-2 BWR, were modelled and the fuel irradiation was simulated with the TRITON module of SCALE 6.2 and with the ALEPH-2 code. Our results were compared to the experimental measurements of four samples: SF95-4 and SF96-4 were taken from the Takahama-3 reactor, while samples SF98-6 and SF99-6 belonged to the Fukushima-Daini-2.

Then, we propagated the uncertainties coming from the nuclear data to the isotopic inventory of sample SF95-4. We used the ALEPH-2 adjoint procedure to propagate the decay constant uncertainties. The impact was inappreciable. The cross-section covariance information was propagated with the SAMPLER module of the *beta3* version of SCALE 6.2. This contribution mostly affected the uncertainties of the actinides. Finally, the uncertainties of the fission yields were propagated both through ALEPH-2 and TRITON with a Monte Carlo sampling approach and appeared to have the largest impact on the uncertainties of the fission products. However, the lack of fission yield correlations results is a serious overestimation of the response uncertainties.

1. Introduction

Burnup calculations are of primary importance for ensuring correct operation and safety of nuclear facilities allowing for the presence of radioactive fuel. The large amount of nuclear data that is fundamental for such calculations is generally provided by evaluated nuclear data libraries in ENDF-6 format (CSEWG, 2013). Amongst them are the most-famous general-purpose libraries ENDF/B (Chadwick et al., 2011) and JEFF (Santamarina et al., 2009). Initially these libraries provided best-estimate data-values; in recent years a further interest in uncertainty evaluation matured and uncertainty and covariance matrices have been added. Several efforts were dedicated to the uncertainty propagation in burnup calculations. In the past, sensitivity analysis and uncertainty quantification studies for material depletion were investigated with methods based on the first order perturbation theory (Williams and Weisbin, 1978; Cacuci, 2003), which were incorporated in dedicated codes like TSUNAMI (Rearden and Mueller, 2011). Recently, the extensive increase of computational power and performances shifted the interest of many expert groups toward random sampling procedures, as implemented in SAMPLER (Williams et al., 2013b), XSUSA (Zwermann et al., 2009), TMC (Rochman et al., 2011) or the Hybrid method developed in ACAB (García-Herranz et al., 2008). These methods are computationally more expensive but cover the uncertainty quantification up to any order. Also the expert group on Uncertainty Analysis in Modelling (UAM) have been working already for several years addressing multi-scale and/or multi-physics aspects of the nuclear data uncertainty analysis. In the framework of burnup or inventory calculation problems in light-water reactors (LWRs) the UAM expert group proposed a series of exercises with the objective to establish a benchmark for uncertainty propagation (Ivanov et al., 2013; Blyth et al., 2014). These systems were given also with experimental results for comparison. Amongst them we studied the Fukushima-Daini-2 and Takahama-3 fuel assemblies in this work.

The objective of this study was the calculation of uncertainty propagation in LWR systems using the analysis techniques implemented in different codes. First the inventory evolution was simulated with two different codes: the Monte Carlo burnup code ALEPH-2 (Stankovskiy and den Eynde, 2011) and TRITON (Jessee

* Corresponding author at: Institute for Advanced Nuclear Systems, SCK•CEN, Boeretang 200, 2400 Mol, Belgium.
E-mail address: lfiorito@sckcen.be (L. Fiorito).

http://dx.doi.org/10.1016/j.anucene.2015.03.046
0306-4549/

and DeHart, 2011b), a module of the SCALE code (Bowman, 2011). Computed results were compared to the experimental measurements to prove the accuracy of the model. Then we propagated the uncertainty stemming from three nuclear data sources, i.e. decay constants, neutron cross sections and independent fission yields, to the nuclide densities at several time-steps. Either Monte Carlo sampling or sensitivity analysis procedure were used. For the ALEPH-2 calculations we resorted to the general-purpose ENDF/B-VII.1 (Chadwick et al., 2011) library, while TRITON used the SCALE library. In addition, we generated covariance matrices for ^{235}U and ^{239}Pu thermal fission yields, using a generalised least-square approach and the ENDF/B-VII.1 files as source. The new covariance matrices were also included in the calculations.

2. Inventory calculation

The two benchmarks analysed in this work were a fuel assembly belonging to the Takahama-3 (TK-3) PWR reactor, with a 17 × 17 design, and a 8 × 8 fuel assembly taken from the BWR reactor Fukushima-Daini-2 (FK-2).

2.1. Description of the PWR fuel assembly

The experimental measurements of the PWR spent fuel were performed on the 17 × 17 fuel assembly NT3G23, that was operated between 1990 and 1992 in the Takahama-3 reactor. The post-irradiation examination was performed at the Japan Atomic Research Institute (JAERI) on 0.5 mm thick sections cut from the spent fuel rods, using different analytical measurement techniques (Nakahara et al., 2002). Fig. 1 shows the geometry of the 17 × 17 fuel assembly.

The assembly had 250 standard fuel rods with 4.11 wt.% of ^{235}U enrichment, 25 guide tubes filled with water and 14 fuel rods with gadolinium oxide (Gd_2O_3), with 6.0 wt.% and 2.6 wt.% respectively of gadolinium and ^{235}U enrichment. A cladding of Zircaloy-4 shrouded the fuel pins and the guide tubes. Samples SF95-4 and SF96-4 were taken respectively from the standard fuel rod in the bottom-left corner of the assembly (position 17a in Fig. 1) and from the gadolinium-bearing fuel rod on the outer layer of the assembly (position 13c in Fig. 1). The cutting positions were 1646 and 1671 mm from the bottom of the active length, respectively for SF95-4 and SF96-4. The irradiation history for each measured axial point is reported in Fig. 2. The boron concentration in the reactor cooling water had an initial value of ~1100 ppm at the beginning of the cycle and decreased linearly to a range of ~200 ppm at the end of the cycle. The history of boron concentration is reported by Nakahara et al. (2002) and shown in Table 1.

Further specifications on the characteristics of the assemblies, initial isotopic compositions of the fuel rods and operating histories of the TK-3 reactor are given by Nakahara et al. (2002) and Radulescu et al. (2010) and in the UAM document (Blyth et al., 2014).

	a	b	c	d	e	f	g	h	i	j	k	l	m	n	o	p	q
1	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-
2	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-
3	-	-	-	-	G	W	-	-	W	-	-	W	G	-	-	-	-
4	-	-	-	W	-	-	-	-	G	-	-	-	-	W	-	-	-
5	-	-	G	-	-	-	-	-	-	-	-	-	-	-	G	-	-
6	-	-	W	-	-	W	-	-	W	-	-	W	-	-	W	-	-
7	-	-	-	-	-	-	G	-	-	-	G	-	-	-	-	-	-
8	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-
9	-	-	W	G	-	W	-	-	W	-	-	W	-	G	W	-	-
10	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-
11	-	-	-	-	-	-	G	-	-	-	G	-	-	-	-	-	-
12	-	-	W	-	-	W	-	-	W	-	-	W	-	-	W	-	-
13	-	-	G	-	-	-	-	-	-	-	-	-	-	-	G	-	-
14	-	-	-	W	-	-	-	-	G	-	-	-	-	W	-	-	-
15	-	-	-	-	G	W	-	-	W	-	-	W	G	-	-	-	-
16	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-
17	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-

Fig. 1. Geometry of the 17 × 17 PWR fuel assembly NT3G23 belonging to the Takahama-3 reactor. W = control rod filled with water, G = gadolinium-enriched fuel rod, - = uranium oxide fuel rod. Samples SF95-4 and SF96-4 were taken from positions 17a and 13c, respectively.

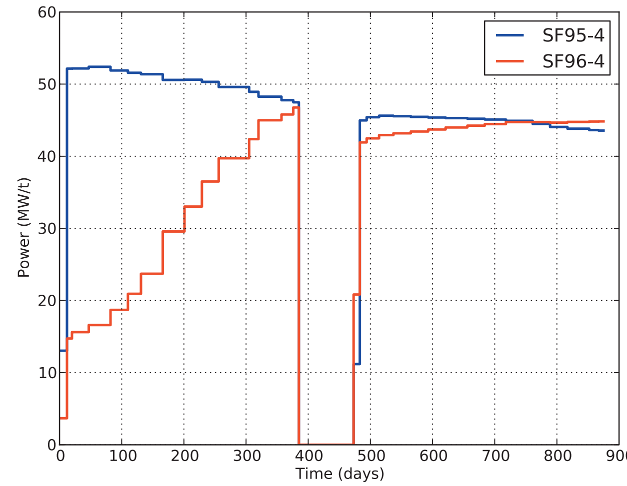

Fig. 2. Power histories of the two rods under study belonging to the fuel assembly of Takahama-3. SF95-4 is a standard uranium oxide rod, while SF96-4 contains burnable poison.

2.2. Description of the BWR fuel assembly

The Fukushima-Daini-2 2F2DN23 fuel assembly was a typical 8 × 8 BWR assembly as shown in Fig. 3. Two central water rods were surrounded by 54 fuel rods with five radial levels of ^{235}U enrichment reported in Table 2 as axial-averaged values. Fuel and water rods were shrouded by a Zircaloy-2 cladding. Eight Gd_2O_3-enriched fuel rods and an external wrapper in Zircaloy-3 completed the fuel assembly. Samples SF98-6 and SF99-6 were taken respectively from a standard fuel rod with the highest level of ^{235}U enrichment (position 2b in Fig. 3) at 692 mm from the bottom of the active length, and from a fuel rod with burnable poison (position 2c in Fig. 3) at 686 mm from the bottom of the active length. Burnup and void ratio axial-distributions were assumed homogeneous, however they are reported by Nakahara et al. (2002). The void fraction value was taken from the corresponding axial location. The assembly under study belonging to the Fukushima-Daini-2 reactor operated between 1989 and 1992 and the operating history is shown in Fig. 4. Further information on the assembly specifications can be retrievable in Nakahara et al. (2002) and Mertyurek et al. (2010) and the UAM document (Blyth et al., 2014).

2.3. Code comparison

We used two different burnup codes to calculate the isotopic inventory of the evaluated fuel assemblies. On one side, the