Annals of Nuclear Energy 47 (2012) 182–191

Contents lists available at SciVerse ScienceDirect

Annals of Nuclear Energy

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

Fuel residence time in BWRs with nitride fuels

Jitka Zakova *, Janne Wallenius

Division of Reactor Physics, Royal Institute of Technology, KTH, Roslagstullsbacken 21, S-10691 Stockholm, Sweden

ARTICLE INFO

Article history:
Received 2 December 2011
Received in revised form 26 March 2012
Accepted 27 March 2012
Available online 6 June 2012

Keywords:
BWR
Uranium nitride
Enrichment
Fuel cycle
Neutronics

ABSTRACT

This paper presents a neutronics study of a BWR core with uranium nitride fuels. Replacing the standard UO_2 fuel with UN or UN-ZrO_2 allows for a higher uranium content, which leads to an increase of the in-core fuel residence time. With the nitride fuels, the total void worth increases and the efficiency of the control rods and burnable poison deteriorates. Taking into account the higher amount of burnable poison needed at the beginning of life, the in-core fuel residence time increases by about 1.4 year comparing to UO_2 fuel with the same enrichment. This implies 1.4% higher availability of the plant and it is therefore of economic interest to the nuclear power plant operators. A similar increase of the fuel in-core lifetime in a UO_2 core could be reached by an increase of the average enrichment of the oxide fuel by roughly 1%.

1. Introduction

Historically, the nitride fuels have been investigated mainly in connection with fast reactors (IAEA, 2009; Matzke, 1986; Rogozkin et al., 2000; Filin, 2000; Adamov, 1997; Kawakita et al., 2002) or ADS systems (Zhang et al., 2011; Wallenius et al., 2001). The stoichiometry of the nitride fuel allows for accommodation of more heavy metal in comparison with oxide fuel, which leads to a harder spectrum. A fast spectrum is convenient for minor actinide transmutation (Youinou and Vasile, 2005; Salvatores, 2002; IAEA, 2009; Kim et al., 2009) and for breeding (Baldev et al., 2002).

Sometimes, nitride fuels have been investigated for minor actinide burning or for breeding in existing or future thermal systems (Feng et al., 2011a,b; Weaver and MacDonald, 2002). The spectral shift associated with the nitride fuels and the possibility to accommodate more heavy metal present a potentially favorable condition for burning of minor actinides. Moreover, the presence of the minor actinides itself causes a further spectrum hardening. The spectrum can be locally hardened even more by use of a convenient cladding material (Wallenius and Westlén, 2008).

As we show in this paper, the main advantage, which the nitride fuels can bring in a thermal light water reactor operating in a standard uranium fuel cycle, is an extension of the in-core residence time at unchanged enrichment. This is a result of the higher heavy metal content and it may be of interest, because it leads to a more economical operation with fewer stops for refueling. However, until recently there has been a problem with deployment of the nitride fuels in the LWR environment, related to the poor stability of the nitrides in water at 300° (Sugihara and Imoto, 1969). This issue could potentially be resolved by addition of a small amount of water resistant oxide such as ZrO_2, into the fuel pellet.

The aim of this study is to offer a comparison of the burnup and safety performance of the traditional UO_2 fueled BWR core and a BWR core fueled with nitride fuels. In this analysis, we employed both UN and UN-ZrO_2 fuels. The first part of the study explains why the replacement of the oxide fuel by nitride fuel brings about a reactivity drop. The second part investigates the somewhat degraded total void worth, which occurs in the nitride fueled BWR. The third part gives a comparison of the oxide and nitride fuels in terms of their burnup performance and the final, fifth, part completes the study with overview of the reactivity coefficients and control rod worth at the beginning and at the end of cycle.

2. Methods and tools

We utilized the stochastic code MCNPX (Pelowitz, 2008) together with the continuous energy cross-section library ENDF B – VII.0 (Carlson et al., 2006) to develop a 3D model of a BWR core. A three-dimensional model is necessary for disclosing effects like the leakage and the capture in the out-of-core materials, which play an important role in the void feedback (Zakova and Wallenius, 2011) and also for a proper depletion calculation. The design of the core in our model corresponds to the Ringhals 1 Swedish nuclear power plant (Eskils Printing Works, 2009; INSC Web Database, Retrieved 2010). For our purposes we chose the Westinghouse SVEA 64 fuel (Lefvert, 1994).

* Corresponding author. Tel.: +46 8 5537 8215.
E-mail addresses: jitka.zakova@neutron.kth.se (J. Zakova), janne@neutron.kth.se (J. Wallenius).

0306-4549/$ - see front matter © 2012 Elsevier Ltd. All rights reserved.
http://dx.doi.org/10.1016/j.anucene.2012.03.033

2.1. Core geometry

Fig. 1 depicts one quarter of the core as we modeled it in MCNPX. We took advantage of the symmetry of the Ringhals 1 core and used a reflective boundary condition to accelerate our calculations. Table 1 summarizes the main dimensions of the core (Eskils Printing Works, 2009; INSC Web Database, Retrieved 2010). In the first part of this work, we assumed that all the Control Rods (CRs) were withdrawn and there was no Burnable Poison (BP) in the fuel. In the later parts, in which we studied the total void worth and the burnup, we introduced both the CR and BP. Fig. 1 shows the three groups of CR that are used for short-term adjustments of reactivity during the cycle. Moreover, in the CR worth calculation, we modeled all the CR that can be inserted into the core in case of an emergency. This we realized by replacing the water in-between the fuel assemblies by the CR material. In the simulation of an emergency shutdown, we conservatively assumed that the most reactive group-G2 remains outside the core.

2.2. Water density profiles and temperatures

In order to account for the axial moderator density change we divided the in-core water into 25 axial zones with decreasing density. The shape of the operational density profile used in this study was derived from a representative void profile of an uprated BWR. The water outside the fuel region was assumed to have an axially invariant density, same as the density at the inlet. Fig. 2 shows the density profile employed in the fueled part of the core.

In the present model we assumed constant temperatures for the respective materials across the entire core. In a neutronic calculation a temperature translates into the cross sections. The set of cross sections, available in the evaluated ENDF B – VII.0 library corresponds to the temperatures: 294 K, 600 K, 900 K, 1200 K and 2400 K.

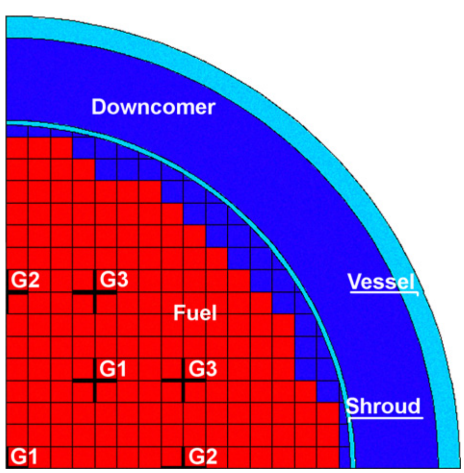

Fig. 1. A quarter of the Ringhals 1 core with control rods.

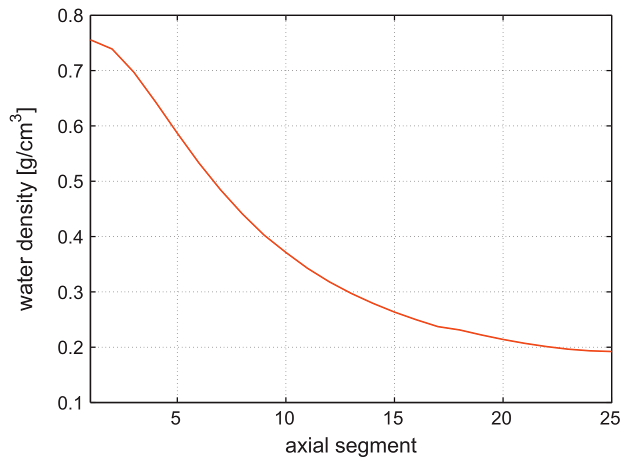

Fig. 2. The axial moderator density profile.

Table 1
Main core parameters.

Total power	2540 MW_{th}/830 MW_e
Active core height	3.68 m
Core diameter	5.95 m
Reactor vessel thickness	15 cm
Reactor shroud thickness	3.5 cm
Fuel inventory UO_2/UN-ZrO_2/UN cores	120/150/166 t_{HM}
Axial reflectors	2 m
Axial water density zones	25
Axially zoned burnable poisons	No
Number of used control rods	4.25
Control rod material	6.7% B-10 mixed with steel
Control rod density	5.52 g cm^{-3}

Table 2
Fuel assembly parameters.

Fuel rod pitch	1.6 cm
Total number of assemblies	648
Fuel assembly pitch	15.275 cm
Assembly box thickness	0.11 cm
Thickness of 1/2 water gap	0.6575
Same gap on top and bottom	Yes
Geometry	8 × 8 SVEA

2.3. Fuel

Table 2 summarizes the geometrical data of the Westinghouse SVEA 64 fuel assembly (INSC Web Database, Retrieved 2010), which is depicted in Fig. 3. In those calculations, where we have not used any BP (Section 3.1) all positions in the assembly lattice were occupied by the same type of fuel rods. Table 3 lists the parameters of the fuel rods.

Table 4 contains the initial compositions of the examined fuel types. Reference calculations were conducted with standard UO_2 fuel. The second type of fuel investigated is UN. This is a hypothetical case because of the instability of UN in water (Sugihara and Imoto, 1969). The third type of fuel is uranium nitride, which contains 10% of ZrO_2. This is expected to stabilize the uranium nitride in the water, making it deployable in a power reactor, while still keeping the advantage of the increased Heavy Metal (HM) content. From the neutronics point of view, this fuel is somewhere between UO_2 fuel and UN fuel.

All the fuels feature 5% porosity. The average enrichment of the fuel is 3.49%. The fuel pins that contain BP feature somewhat decreased enrichment, 2.45% U-235, while all the other pins in the assembly contain 3.56% enriched fuel (Table 4). There are four fuel pins with BP in each fuel assembly (when applicable), as shown in Fig. 3. In the nitride fuels, the nitrogen is enriched to 90% in N-15. This enrichment makes the nitride fuels more expensive, it is however needed for a better neutron economy in the thermal systems. Last but not least, the lower amounts of N-14 in the core also lead to lower production of C-14, which arises in (n, p) reactions.

In the later parts of this study, we made some modifications to these initial fuel compositions. Namely, in Section 3.2 we introduced nitride fuels with 100% N-14 and 100% N-15 to disclose