ASSESSMENT OF THE ACCIDENTAL CRITICALITY FOR CANDU FUEL BUNDLES CONTAINING U-BASED FUELS

CRISTINA ALICE MĂRGEANU[1], ROXANA JOIŢESCU[2], ŞERBAN CONSTANTIN VALECA[2]
[1] Institute for Nuclear Research (RATEN ICN) Piteşti, Romania
[2] University of Piteşti, Romania
e-mail: cristina.margeanu@nuclear.ro

Abstract

Nuclear criticality safety relates to prevention and protection against the consequences of an uncontrolled nuclear fission chain reaction, or a criticality accident. Accidental criticality can result if a sufficient quantity of fissile material (e.g. U^{235}, Pu^{239}) is arranged into a critical configuration. The most adverse and potentially dangerous aspect of a criticality accident is the release of nuclear radiation that affects nearby personnel. The present paper aimed to study the accidental criticality of CANDU fuel bundles containing U-based fuels. The effective multiplication factor (k-eff) was evaluated by KENO VI Monte Carlo code for different potentially dangerous geometrical arrangements of stacked fuel bundles (containing fresh or spent fuel) accidentally immersed in H_2O or D_2O. The arrangements of fuel bundles included 2, 3, 5 or 6 stacked fuel bundles, no any damage of fuel bundles structural integrity being assumed. The criticality analysis has been performed for CANDU fuel bundles with 37 elements containing natural uranium (NU) fuel and for CANDU type fuel bundles with 43 fuel elements containing slightly enriched uranium (SEU) or recovered uranium (RU) fuel. Multiplication factor values were lower than one in all accidental criticality scenarios, except for the accidental immersion of fresh fuel bundles arrangements in D_2O, when the infinite lattice option for neutron leakages treatment was considered. The most dangerous geometrical arrangements seem to be in the case of 6 stacked fuel bundles.

Keywords: *CANDU fuel bundle, U-based fuels, Accidental criticality, Multiplication factor, KENO VI Monte Carlo code*

Introduction

Nuclear energy future depends mainly on its economic benefits, its ability to provide clean and reliable electricity at affordable prices, its contribution to meeting the objectives of the European Union's energy policy including the climate changes and energy security concerns, its impact on the environment and its acceptability by the public. At the same time, availability of energy resources, nuclear safety, decommissioning of nuclear reactors at the end of their useful life, management, transport and final storage of radioactive waste, as well as non-proliferation, are important issues to consider.

A promising option to improve the efficiency in using the natural resources and reducing the nuclear fuel cycle impact on the environment (reducing the radioactive waste amount and their toxicity) is represented by the recycling of reprocessed Uranium and Plutonium.

Recovered Uranium (RU) can be obtained in several ways and its content in U^{235} may vary, depending on the fuel burnup degree on discharge and the reactor type, such as: by enrichment; by mixing with High Enriched Uranium (HEU)/ Low Enriched Uranium (LEU)/ Natural Uranium (NU); by direct recycling; by combining enrichment and mixing with other fissile materials. Some reactors could operate with fuel containing Slightly Enriched Uranium (SEU) instead of NU with SEU, in which case RU becomes very attractive given that it provides fuel with SEU characteristics without incurring the costs of enrichment stage [1].

CANDU reactors operates based on a mature and well-proven technology, characterized by high safety level, evolutionary potential for non-proliferation concerns, and versatility for different nuclear fuel cycles, features creating the premises for a better fuel resource utilization. NU Nuclear Fuel Cycle (NFC) is generally applied in the HWR CANDU type reactors using a simple fuel project. However, CANDU reactors versatility supports alternative fuel cycles utilization without requiring major changes in the reactor basic design. These alternative fuel cycles, also known as advanced fuel cycles, can use various fissile materials, including SEU with up to 1.2 wt% enrichment in U^{235}, RU, Mixed Oxide (MOX) and Thorium-based fuels. The aim is to improve NFC economics by: increasing the fuel burnup (lead to the costs reduction both for the resources consumption and the final storage of spent fuel), increased operating margins (directly connected to the reactor power increasing), reduction by 2-3 times the amount of fuel burned per unit of energy production (due to the relatively low enrichment), reduced fuel cycle costs, and greater flexibility in plant operation [2, 3].

Nuclear criticality safety generally aims to assure the protection against the consequences arising from an uncontroled nuclear chain reaction, preferebly by prevention of potential dangerous accidental conditions occuring if sufficient amount of fissile material (usually U^{235} and Pu^{239}, materials capable of sustaining a nuclear chain reaction) is arranged into a critical configuration. These potential dangerous conditions leading to critical mass formation are refered as criticality accidents and the main consequences are represented by the release of radiations that could be lethal for the personnel in the vicinity, and could damage the environment. Regardless of consequences, the criticality safety objective remains the prevention of a criticality accident.

The paper aimed to study the accidental criticality characterizing CANDU fuel bundles with different U-based fuels. The effective multiplication factor (k-eff) was evaluated for various potentially dangerous geometrical arrangements of stacked fuel bundles (containing fresh or spent fuel) accidentally immersed in H_2O or D_2O. These arrangements included 2, 3, 5 or 6 stacked fuel bundles, assuming no damage of the fuel bundles structural integrity, so we don't have to deal with a potential risks from spreading of the fresh or irradiated nuclear material.

Two CANDU type fuel bundle projects were selected for this study, namely: CANDU standard fuel bundle with 37 fuel elements (C37) containing NU fuel, and CANDU type fuel bundle with 43 fuel elements (C43) developed by the Nuclear Fuel Performance group at RATEN ICN Pitesti containing SEU or RU fuel.

Calculation of the criticality coefficient (k-eff) was performed by using KENO VI Monte Carlo code. For the spent fuel analysis, fuel burnup was simulated by means of ORIGEN-S code assuming that all fuel bundle elements contain the same fuel composition. Both computer codes are included in SCALE6 programmes package developed by Oak Ridge National Laboratory (USA).

In the specific conditions of several defined accidental criticality scenarios, the study included comparisons of estimated k-eff values for C43 fuel bundle with different fuel compositions, and also against the k-eff values characterizing the CANDU standard C37 fuel bundle with NU fuel, considered as reference case.

Initial data for fuel burnup simulation, computer codes used, assumptions

Fuel bundle projects considered

CANDU type fuel bundle projects selected for this study have cylindrical shape, are composed by fuel elements arranged on concentric rings and one central fuel element,
The 37-element CANDU fuel bundle was developed in Canada, for Bruce NPP's CANDU reactors (750 MWe) and is considered the CANDU standard fuel bundle based on its worldwide utilization in the 600 MWe CANDU reactors operating in Canada (Gentilly), Argentina (Embalse), China (Qinshan), Romania (Cernavoda) and South Korea (Wolsong). C37 fuel bundle contains 37 identical fuel elements arranged on 3 concentric rings (of 6, 12 and 18 elements, respectively) and one central element [4, 5], as shown in Fig. 1. The fuel bundle elements are sealed zirconium alloy tubes of about 1.5 cm in diameter, with a sheath of 0.4 mm thick, and each contains a column of 31 sintered NU fuel pellets. The fuel elements are stiffened in the bundle by two end plates, and the outer ring fuel elements have welded bearing pads in order to assure the bundle easy sliding inside the pressure tube and leaving enough space for the coolant circulation [4, 5]. The fuel bundle is about 50 cm long and 10 cm in diameter, and weighs slightly over 24 kg, the amount of NU fuel being about 22 kg.

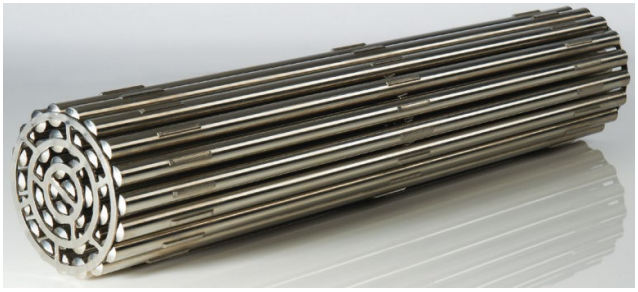


Fig. 1. CANDU standard fuel bundle (C37) view [5]

CANDU type fuel bundle with 43 fuel elements considered in present study was developed, updated and optimized by the Nuclear Fuel Performance Group at RATEN ICN, aiming a perfect compatibility with the Canadian CANFLEX (CANDU Flexible) fuel bundle in terms of geometry and performances.

CANFLEX fuel bundle was developed by Canada to ensure the reactor operation at extended fuel burnups and to increase the operating limits given the aging or to increase the power of the reactor, as well as to ensure the compatibility with the fuel cycles for advanced CANDU reactors (NU, SEU, RU, MOX, Th-based fuels) [6]. Although fuel bundle thermal power increased, the linear power on elements was diminished by increased number of elements, which led to milder operating conditions: reduced fission gases release, lower values for total sheath deformation and central temperature in fuel pellet than those characterizing the standard fuel bundle [6].

As any CANDU type fuel bundle, the C43 fuel bundle is simple structured and contains 43 elements disposed in annular geometry on 3 concentric rings of 7, 14, and 21 elements, respectively, around a central element (Fig. 2). There are 8 thick elements (central element and the inner ring 7 elements) and 35 thin elements (intermediate ring 14 elements and the outer ring 21 elements) in terms of fuel element diameter compared with the CANDU standard fuel element. The thick element is about 0.4 mm larger in diameter than the reference element, the thin element diameter being 0.2 cm smaller compared with the reference one [7, 8]. The sealed zircalloy-4 tubes (0.4 mm thickness) contain a column of 30 fuel pellets (the thick element) and 36 fuel pellets (the thin element), respectively. The length of C43 fuel bundle is the same as that of the C37 fuel bundle, but C43 is lighter weighing about 23.5 kg of which the fuel is 21 kg [7, 8].



Fig. 2. 43-element CANDU type fuel bundle (C43) view [8]

U-based fuel compositions analysed

The proposed analyses were performed considering the following fuel bundle-fuel composition options: C37 fuel bundle with NU fuel [9-11], C43 fuel bundle with SEU fuel (0.9 wt% and 1.0 wt% enriched in U^{235}) [10-12], and C43 fuel bundle with RU fuel (0.9 wt% and 1.0 wt% enriched in U^{235}) [11, 13], respectively.

Table 1 shows the initial inventory of materials for the selected U-based fuel compositions, both for the isotopes of interest (Uranium isotopes and Oxygen), and for the cladding (Zr alloy), UO_2 and the fuel bundle. The working cases identification, according to the fuel composition and fuel bundle project, is also given in Table 1. The cladding is made of Zr alloy, as mentioned above, containing Zirconium, alloying elements (Tin, Iron, Chromium, Nickel) and some impurities (Carbon and Oxygen), as presented in Table 2 for the two considered fuel bundle projects, C37 and C43.

Table 1. Material initial inventory in the selected U-based fuels, [11-13]

Working Case ID	**Enrich. in U^{235} [wt%]**	**U^{234} [kg]**	**U^{235} [kg]**	**U^{236} [kg]**	**U^{238} [kg]**	**O_2 [kg]**	**Zy [kg]**	**UO_2 [kg]**	**Fuel bundle [kg]**
NU37	0.71	0	0.14	0	19.36	2.62	2.14	22.11	24.26
SEU43_wt1	0.9	0	0.17	0	18.48	2.50	2.25	21.15	23.40
SEU43_wt2	1.0	0	0.19	0	18.46	2.50	2.25	21.15	23.40
RU43_wt1	0.9	0.004	0.17	0.037	18.43	2.50	2.25	21.15	23.40
RU43_wt2	1.0	0.004	0.19	0.037	18.42	2.50	2.25	21.15	23.40

Table 2. Fuel bundle projects sheath composition, [11-13]

Fuel bundle	**C37**	**C43**
Element	**Mass [kg]**	**Mass [kg]**
Zirconium (Zr)	2.14	2.25
Alloying elements		
Tin (Sn)	3.11×10^{-2}	3.27×10^{-2}
Iron (Fe)	3.64×10^{-3}	3.83×10^{-3}
Chromium (Cr)	2.14×10^{-3}	2.25×10^{-3}
Nickel (Ni)	1.18×10^{-3}	1.24×10^{-3}
Impurities		
Carbon (C)	4.07×10^{-4}	4.28×10^{-4}
Oxygen (O)	2.57×10^{-3}	2.70×10^{-3}

Simulation of fuel burnup - computer codes used, data and assumptions

The fuel irradiation simulation was performed by means of ORIGEN-S burnup code [14], included in SCALE6 programmes package. The assumption of identical fuel composition in all fuel bundle elements was considered.

For NU37 case, the fuel burnup was simulated at a specific power of 34.644 kW / kgHE (heavy element) up to 8 MWd / kgHE burnup degree. For SEU43 or RU43 cases, the discharge burnup degree of 10 MWd / kgHE was reached, simulating the fuel irradiation at a specific power of 43.24 kW / kgHE.

The results of interest for the U-based CANDU fuel simulation calculations were the inventory of fissile material isotopes (U^{235} and Pu^{239}) and the inventory of isotopes interesting for resistance to proliferation concerns (Pu^{238}, Pu^{240} and Pu^{242}). The values obtained for the actinides of interest (U and Pu isotopes) were further used to prepare the input data required for the criticality calculations of the irradiated fuel.

Fresh and irradiated fuel criticality estimation has been performed by using KENO-VI Monte Carlo code [15], also included in SCALE 6 programmes package. KENO-VI calculations were carried out based on nuclear data from 238GROUPNDF5 library, general nuclear data library used for criticality analyses also known as LAW (Library to Analyze Radioactive Waste). It contains data for more than 300 nuclides, on 148 rapid and 90 thermal neutron energy groups. The nuclides from 238 GROUPNDF5 library use the same averaged spectra, which consist of: a Maxwell spectrum (with a peak at 300 K) from 10^{-5} eV to 0.125 eV; a spectrum ranging from 0.125 eV to 67.4 keV; a fission spectrum from 67.4 keV to 10 MeV; a spectrum evolving as 1/E ranging from 10 MeV to 20 MeV.

In the Monte Carlo simulation, 203 generations of 1,000 particles each have been considered. For the variance reduction, the particle multiplication and Russian roulette procedures were used, with the following values for the weights of interest in the Monte Carlo simulation process: average weight, WTA = 0.5; maximum weight in particle multiplication, WTH = 3; minimum share in Russian roulette, WTL = 0.333.

For each scenario, the standard composition comprising the nuclides of interest is introduced, specifying the mixture of which they belong, the averaged density of the isotope (in g/cc), and the working temperature. The data referring the modelling of the geometrical units (see Figs. 3-5) needed for the criticality calculation are introduced and, at the end, the conditions for treating the neutron leaks on the 3 axes (X, Y and Z) are also given.

Accidental criticality scenarios, modelling, assumptions

In order to perform the criticality calculation and to estimate the accidental criticality (k-eff multiplication factor), 3 scenarios involving 37-element or 43-element CANDU type fuel bundles with fresh or irradiated U-based fuels were considered, as follows:

- SCE1: Accidental immersion of fuel bundles containing fresh fuel in H_2O during the transport to the reactor;
- SCE2: Accidental immersion of fuel bundles containing fresh fuel in D_2O during the transport to the reactor;
- SCE3: Accidental immersion of fuel bundles containing irradiated fuel in H_2O (SCE3H) or D_2O (SCE3D) after the discharge from the reactor.

The accident consists, in fact, in the joining of 2, 3, 5 or 6 fuel bundles stacked close together in a dangerous ″configuration″ at distances that could lead to critical conditions (formation of the critical mass and initiation of the fission reaction), but maintaining their structural integrity (the spreading of radioactive material is avoided and the subsequent radiation exposure issues as well). The stacked of fuel bundles is followed by the immersion in H_2O or D_2O.

In the case of irradiated fuel, the fuel composition has been chosen so as to maximize neutron production, avoiding absorption as much as possible. In this context, fission products have been ignored and, instead, the isotopes of U and Pu are present.

The modeling of the dangerous arrangements of fuel bundles stacked together materialized in the step-by-step modeling of some ″geometrical units″, as follows: fuel element (column of fuel pellets, air gap, sheath), fuel assembly (a central fuel element and 3 concentric rings with 6/12/18 fuel elements in the C37 fuel bundle case, and 7/14/21 fuel elements in the C43 fuel bundle cases, as can be seen in Fig. 3), fuel bundles arrangements formed by stacking together 2, 3, 5 (on 2 rows) or 6 (on 3 rows) fuel assemblies as presented in Figs. 4 and 5.

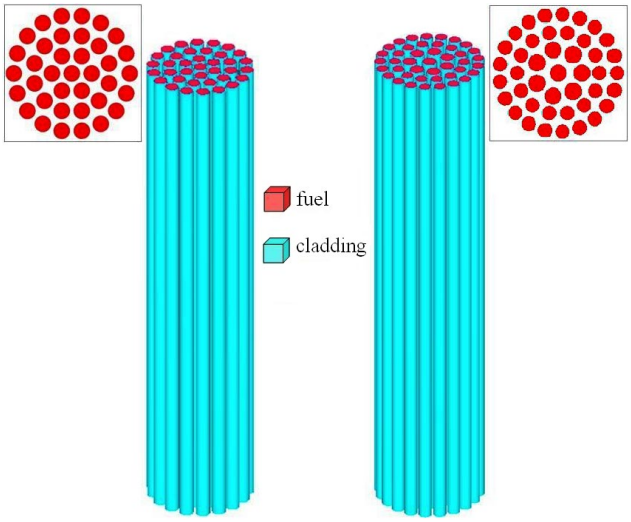


Fig. 3. 3-D modelling of the C37 fuel bundle (left) and C43 fuel bundle (right) using KENO-VI code

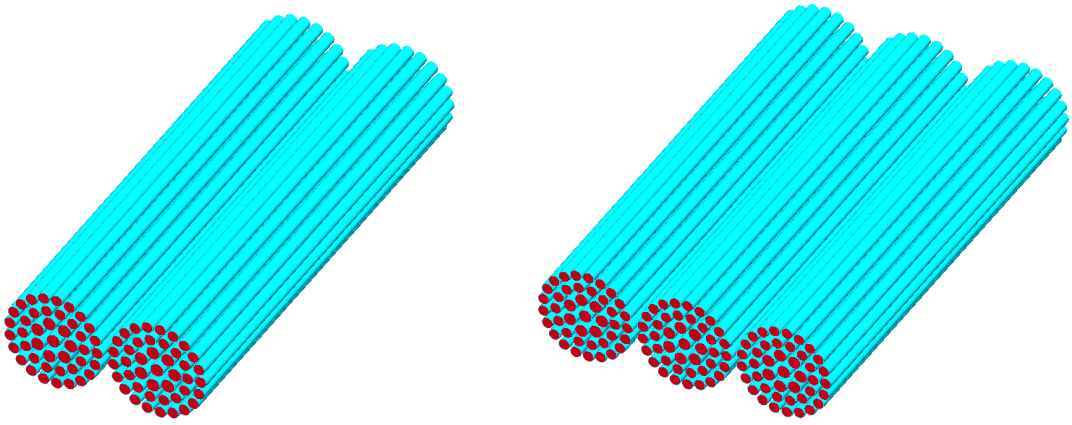


Fig. 4. 3-D modelling of ″dangerous″ arrangements of adjacent fuel bundles using KENO-VI code: 2 fuel bundles (left) and 3 fuel bundles (right)

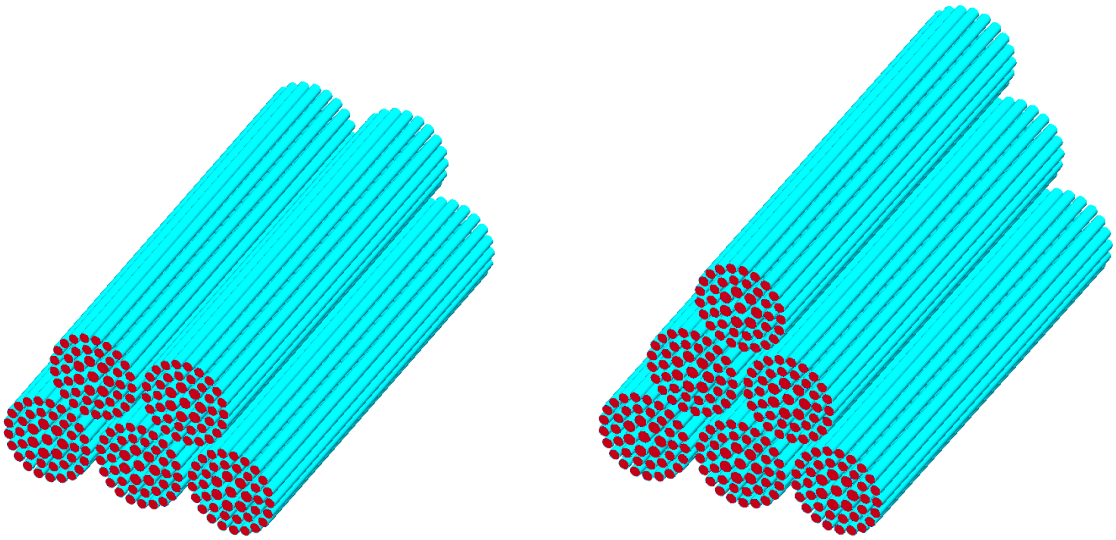


Fig. 5. 3-D modelling of ″dangerous″ arrangements of adjacent fuel bundles using KENO-VI code: 5 fuel bundles on 2 rows (left) and 6 fuel bundles on 3 rows (right)

The input data required to perform the criticality calculations also includes information on how to treat neutron leakage on the 3 axes (X, Y and Z). Three options for neutron leakages treatment were used in KENO-VI calculations, namely:

(a) ±X: REFL, ±Y: REFL, ±Z: REFL – infinite lattice;

(b) ±X: REFL, ±Y: REFL, ± Z: VOID
neutrons ″escaping″ through lattice cell faces +Z and -Z are leaving the system;

(c) ±X: REFL, ±Y: REFL, -Z: CONC4; +Z: VOID
a single ″row″ of fuel bundles placed on a concrete platform and covered by a layer of water; face -Z (bottom) has a specific albedo for a 10 cm thick concrete layer (CONC4), for face +Z keeping VOID (neutrons escaping from the system).

Results and discussion

The paper aims to investigate the potential criticality risks for fresh and spent fuel bundles CANDU type considering some accidents scenarios which involve formation of dangerous arrangements of fuel bundles stacked together and then immersed in H_2O or D_2O, assuming no any damages of their structural material integrity.

Fuel irradiation in CANDU reactor specific conditions was simulated by means of ORIGEN-S burnup code, using the spectral indices provided by the lattice cell neutronic computations [16]. The inventory obtained for the isotopes of interest (U and Pu isotopes) at the End of Irradiation (characterizes the spent fuel cases) is given in Table 3 and it was further used as input data in the criticality calculations with KENO-VI Monte Carlo code.

Table 3. U and Pu isotopes inventory at the EOI [g/assembly] for considered U-based fuels

Working	**Inventory of isotopes of interest [g/assembly]**							
Case ID	U^{234}	U^{235}	U^{236}	U^{238}	Pu^{239}	Pu^{240}	Pu^{241}	Pu^{242}
NU37	2.00	26.26	14.13	16924	24.64	23.71	3.69	1.60
SEU43_wt1	9.90	27.71	71.20	16072	23.47	24.89	4.05	2.05
SEU43_wt2	10.10	34.27	72.72	16064	23.40	23.44	3.71	1.71
RU43_wt1	2.47	27.03	52.67	16100	22.34	25.11	4.00	2.18
RU43_wt2	1.93	20.92	63.19	16094	22.37	26.72	4.36	2.64

The criticality comparative analyses were performed for fresh and spent U-based fuels in CANDU type fuel bundles considering previous described criticality scenarios and the selected dangerous arrangements of 2, 3, 5 or 6 fuel bundles stacked together. The previous described options for the treatment of neutron leakage from the system were also considered. The results are presented in the following figures and tables.

Comparative analysis of k-eff estimated for fresh fuel criticality scenarios

Figs. 6-8 illustrate the criticality factor, k-eff, estimated values in the conditions of SCE1 and SCE2 for the NU37, SEU43_wt2 and RU43_wt2 fuels, respectively. Indices (a), (b), and (c) specify which option for neutron leakage treatment was applied in the criticality calculations. The estimated k-eff values corresponding to SEU43_wt1 and RU43_wt1 were by 3% - 4.5% lower compared with those obtained for SEU and RU fuels enriched 1.0 wt% in U^{235}.

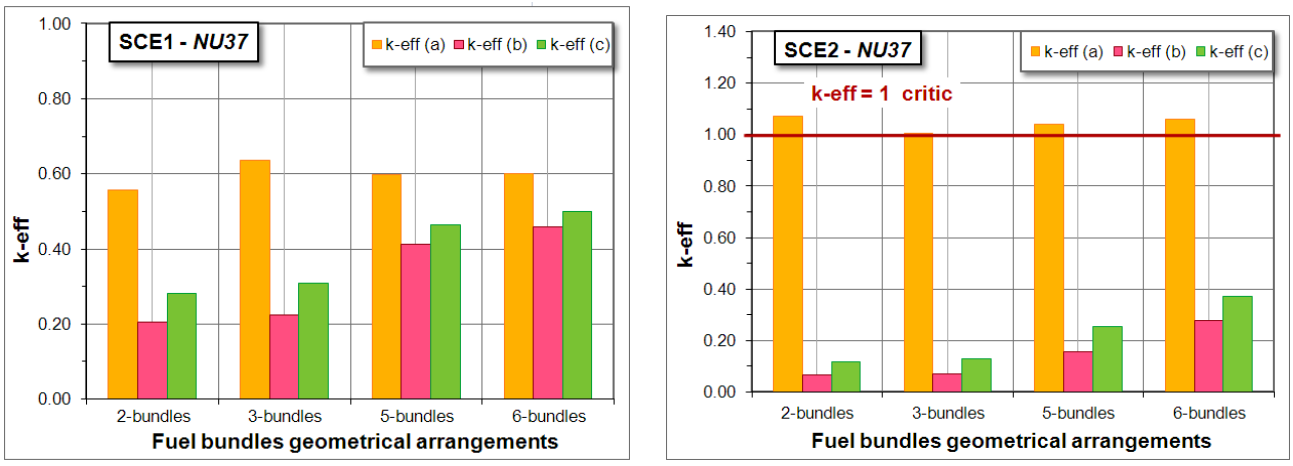


Fig. 6. Estimated k-eff values for NU37 fresh fuel bundles dangerous arrangements
in SCE1 (left) and SCE2 (right) accident scenarios
(a) ±X: REFL, ±Y: REFL, ±Z: REFL (b) ±X: REFL, ±Y: REFL, ± Z: VOID
(c) ±X: REFL, ±Y: REFL, -Z: CONC4; +Z: VOID

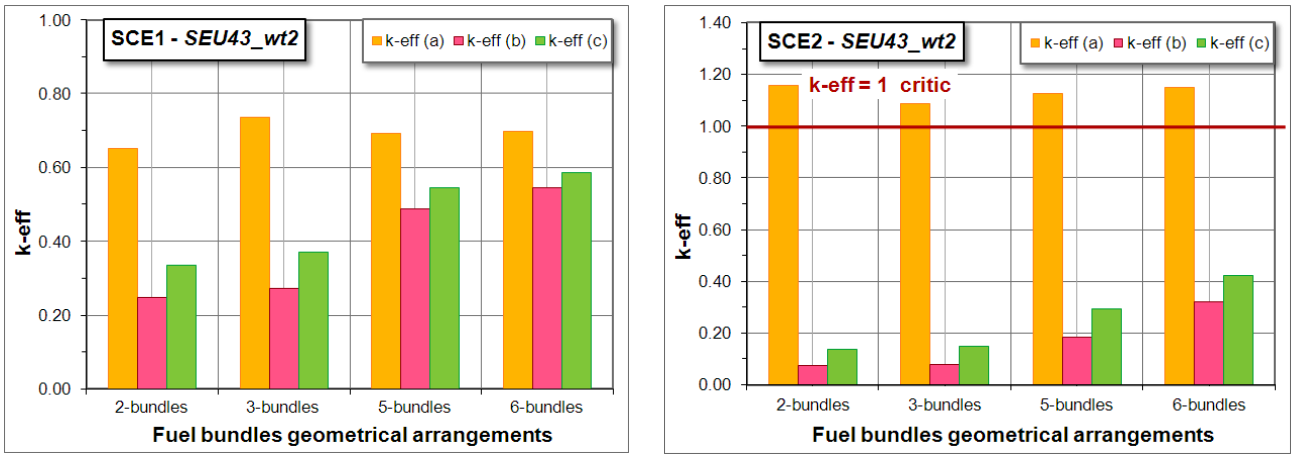


Fig. 7. Estimated k-eff values for SEU43 1.0wt% enriched in U^{235} fresh fuel bundles dangerous arrangements
in SCE1 (left) and SCE2 (right) accident scenarios
(a) ±X: REFL, ±Y: REFL, ±Z: REFL (b) ±X: REFL, ±Y: REFL, ± Z: VOID
(c) ±X: REFL, ±Y: REFL, -Z: CONC4; +Z: VOID

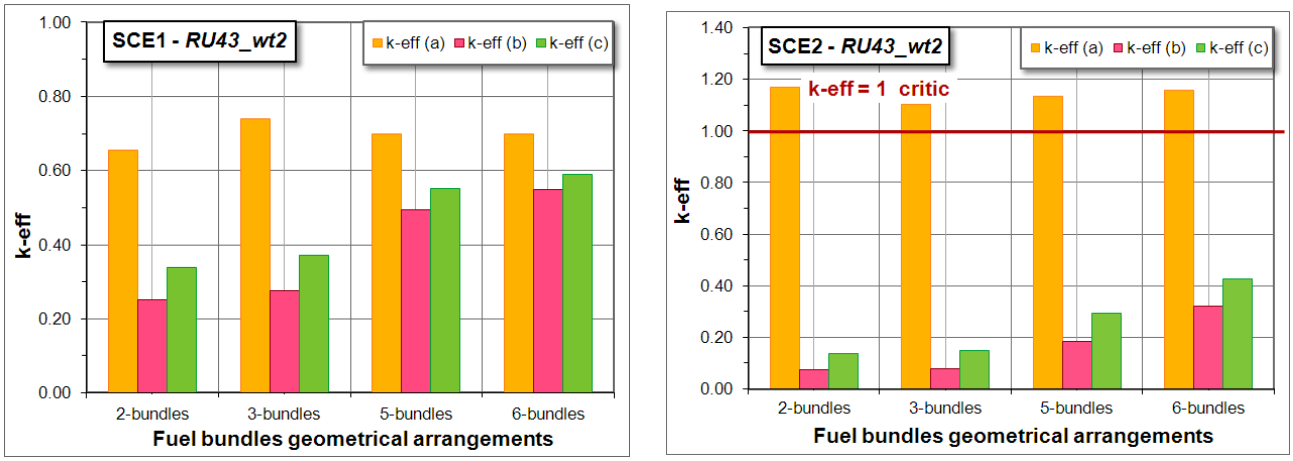


Fig. 8. Estimated k-eff values for RU43 1.0wt% enriched in U^{235} fresh fuel bundles dangerous arrangements
in SCE1 (left) and SCE2 (right) accident scenarios
(a) ±X: REFL, ±Y: REFL, ±Z: REFL (b) ±X: REFL, ±Y: REFL, ± Z: VOID
(c) ±X: REFL, ±Y: REFL, -Z: CONC4; +Z: VOID

In criticality scenario SCE1, the k-eff values estimated for considered U-based fuels were less than 1, regardless the option for neutron leakage treatment. The highest k-eff values were obtained considering option (a) for neutron leakage treatment - the infinite lattice, quite over-conservative option. The relative differences calculated for the comparison between k-eff (a) values and the other 2 options (b and c), regardless the fuel considered, were: ~ 50%-60% for 2 or 3 bundles arrangement, ~ 20%-30% for 5 bundles arrangement, and 15%-20% for 6 bundles arrangement, respectively.

RU43 fuel compositions conducted to higher k-eff values compared to the SEU43 or NU37 fuel cases. The arrangement of 6 bundles was the most dangerous configuration in majority of cases, except for the infinite lattice treatment of neutron leakage when the most dangerous configuration was the one with 3 bundles stacked together on a single row.

Accidental immersion in D_2O (criticality scenario SCE2) conducted to k-eff exceeding 1 for the infinite lattice of fuel bundles (option (a) for neutron leakage treatment), which is over-conservative. However, 3D calculations performed for realistic situations (options (b) and (c) for neutron leakage treatment) conducted to k-eff values significantly lesser than 1 (k-eff was up to about 0.4).

To be mentioned here that for the realistic situations, k-eff values are significantly smaller (by 40%-50%) than the ones obtained in SCE1, regardless the fuel composition considered. The relative differences calculated for the comparison between k-eff (a) values and the other 2 options (b and c), regardless the fuel considered or the fuel bundle type, were: ~ 90% for 2 or 3 bundles arrangement, ~ 75%-85% for 5 bundles arrangement, and 65%-70% for 6 bundles arrangement, respectively.

RU43 and SEU43 fuel compositions conducted to quite similar values for k-eff (slightly higher for RU43 fuels), and higher than those characterizing NU37 fuel. The stack of 6 bundles on 3 rows was found as the most dangerous configuration, except for the cases of infinite lattice cell for neutron leakage treatment when the most dangerous configuration was the stack of 2 bundles.

Comparative analysis of k-eff estimated for spent fuel criticality scenarios

Figs. 9-11 show the criticality factor, k-eff, estimated values in the conditions of SCE3H and SCE3D for the NU37, SEU43_wt2 and RU43_wt2 fuels, respectively. The option for neutron leakage treatment applied in the criticality calculations was specified by using indices (a), (b), or (c). For SEU43_wt1 and RU43_wt1 fuels, k-eff values were by 2%-3.5% lower compared with those obtained for SEU43_wt2 and RU43_wt2 fuels.

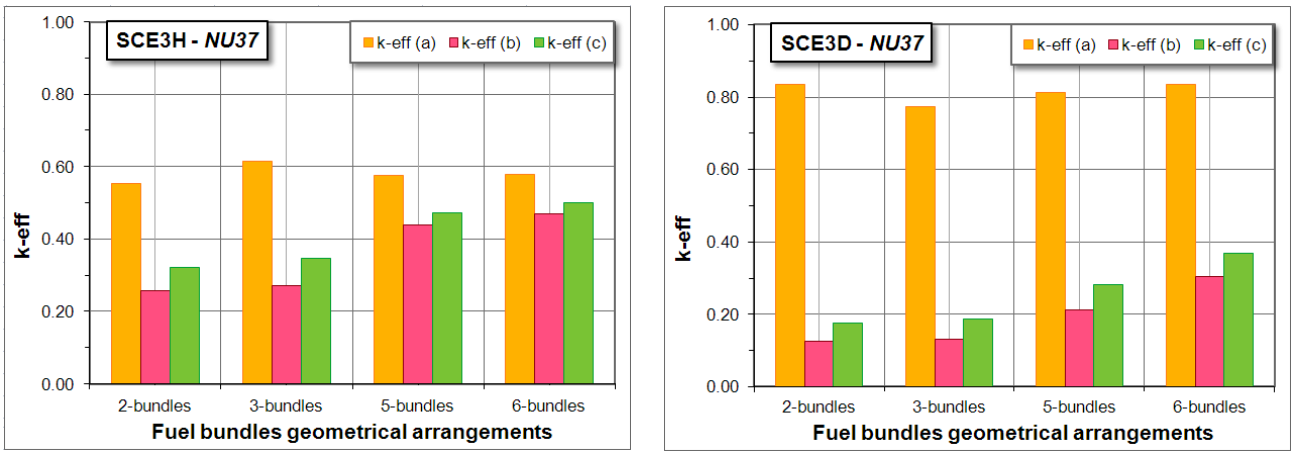


Fig. 9. Estimated k-eff values for NU37 spent fuel bundles dangerous arrangements in SCE3H (left) and SCE3D (right) accident scenarios
(a) ±X: REFL, ±Y: REFL, ±Z: REFL (b) ±X: REFL, ±Y: REFL, ± Z: VOID
(c) ±X: REFL, ±Y: REFL, -Z: CONC4; +Z: VOID

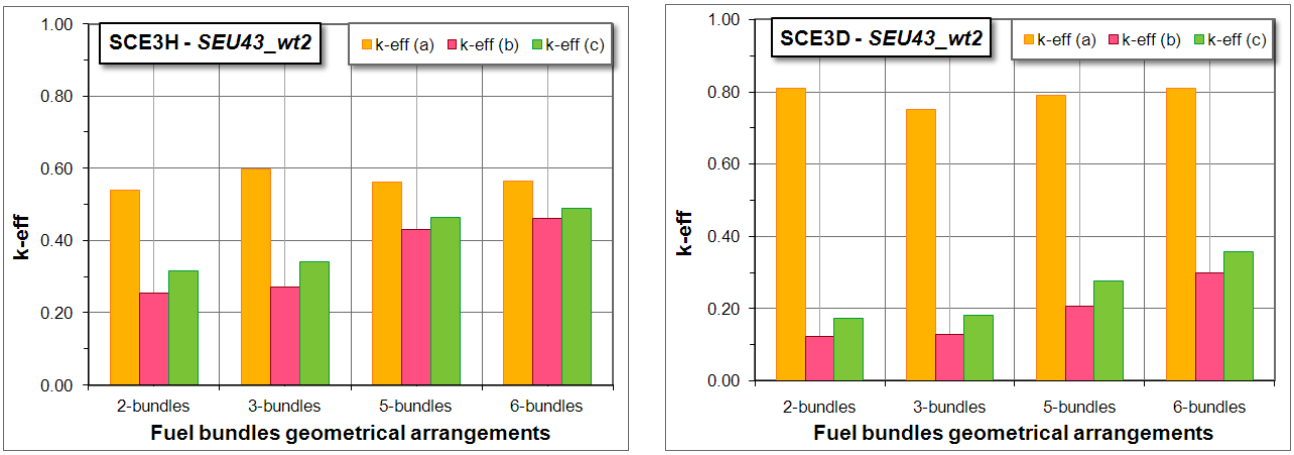


Fig. 10. Estimated k-eff values for SEU43 1.0wt% enriched in U^{235} spent fuel bundles dangerous arrangements in SCE3H (left) and SCE3D (right) accident scenarios
(a) ±X: REFL, ±Y: REFL, ±Z: REFL (b) ±X: REFL, ±Y: REFL, ± Z: VOID
(c) ±X: REFL, ±Y: REFL, -Z: CONC4; +Z: VOID

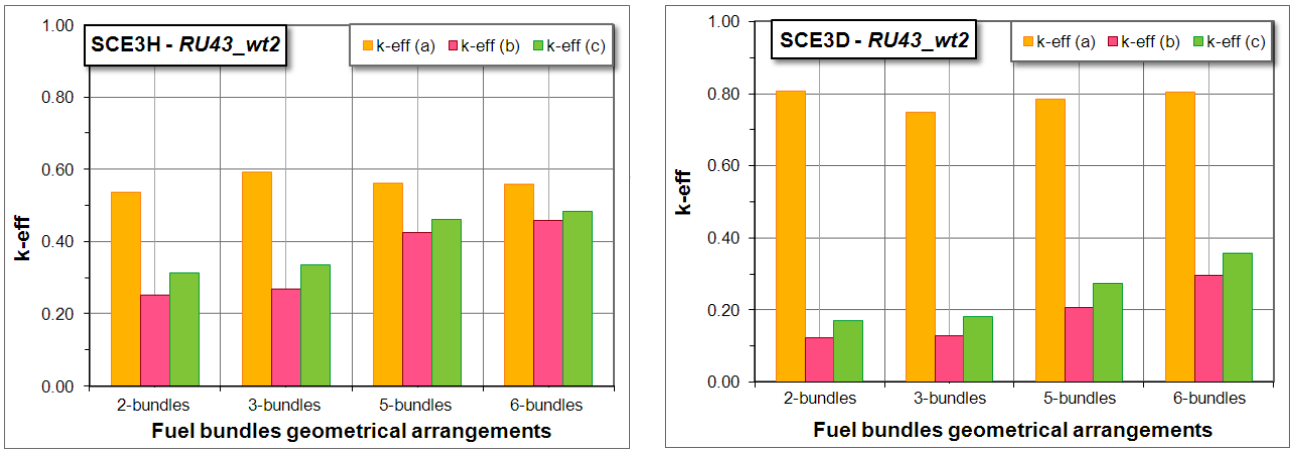


Fig. 11. Estimated k-eff values for RU43 1.0wt% enriched in U^{235} spent fuel bundles dangerous arrangements in SCE3H (left) and SCE3D (right) accident scenarios
(a) ±X: REFL, ±Y: REFL, ±Z: REFL (b) ±X: REFL, ±Y: REFL, ± Z: VOID
(c) ±X: REFL, ±Y: REFL, -Z: CONC4; +Z: VOID

In criticality scenarios for spent fuel, the k-eff values estimated for considered U-based fuels were less than 1, regardless the option for neutron leakage treatment, the fuel bundle type or fuel composition. The highest k-eff values were obtained considering option (a) for neutron leakage treatment. For the other two options, the k-eff values were lower, regardless the fuel bundle type or fuel composition, as follows: SCE3H - ~ 40%-55% for 2 or 3 bundles arrangement, ~ 18%-24% for 5 bundles arrangement, and 13%-18% for 6 bundles arrangement, respectively; SCE3D - ~ 80%-85% for 2 or 3 bundles arrangement, ~ 65%-75% for 5 bundles arrangement, and 55%-65% for 6 bundles arrangement, respectively.

NU37 spent fuel is characterized by slightly higher k-eff values compared to the SEU43 or RU37 fuel cases. The 6 bundles stacked together on 3 rows was the most dangerous configuration in majority of cases, except for the cases with option (a) for neutron leakage treatment when the most dangerous configuration was the 3 bundles arrangement in SCE3H, and 2 bundles and 6 bundles arrangements in SCE3D, respectively. Estimated k-eff values in SCE3D conditions were higher compared with those obtained in SCE3H conditions only considering infinite lattice in neutron leakage treatment. Considering more realistic treatment of neutron leakage, we obtained lower k-eff values in the SCE3D conditions in comparison with the ones characterizing SCE3H.

A comparison of the k-eff values estimated for the analyzed fuel bundles stacked in different arrangements was performed at the criticality scenario level considering option (c) for neutron leakage treatment (the most realistic one). The results are presented in Figs. 12 and 13, for fresh and spent fuel criticality scenarios, respectively.

The relative differences for the comparison between SEU43 or RU43 fuels and NU37 fuel were calculated and Table 4 presents the relative differences for the comparison SEU43_wt2 or RU43_wt2 and NU37, according to stacked bundles configuration and criticality scenario considered.

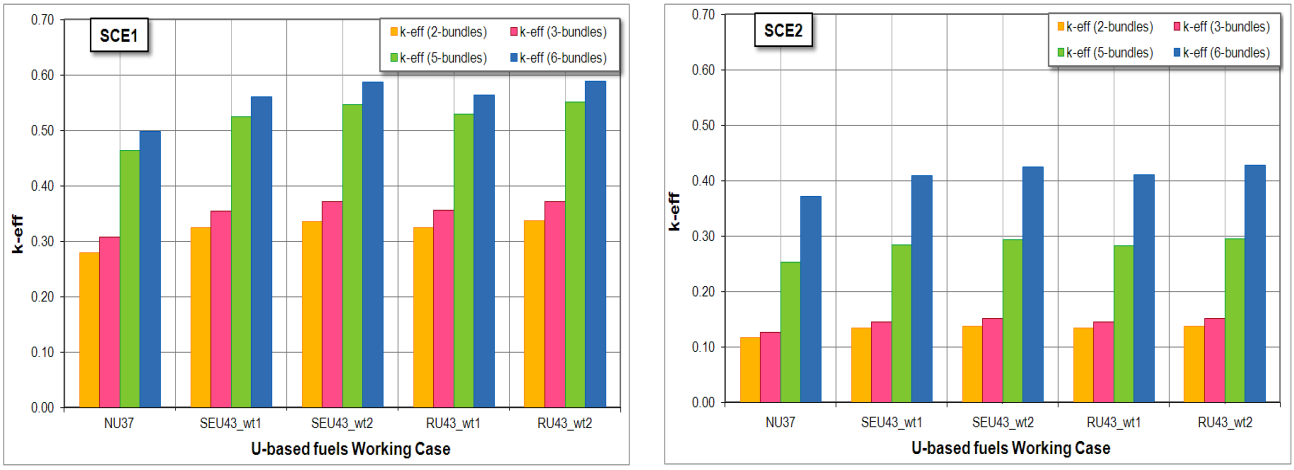


Fig. 12. Estimated k-eff values for NU37, SEU43 and RU43 fuel bundles stacked in different configurations in SCE1 (left) and SCE2 (right) accident scenarios
Neutron leakage treatment: ±X: REFL, ±Y: REFL, -Z: CONC4; +Z: VOID

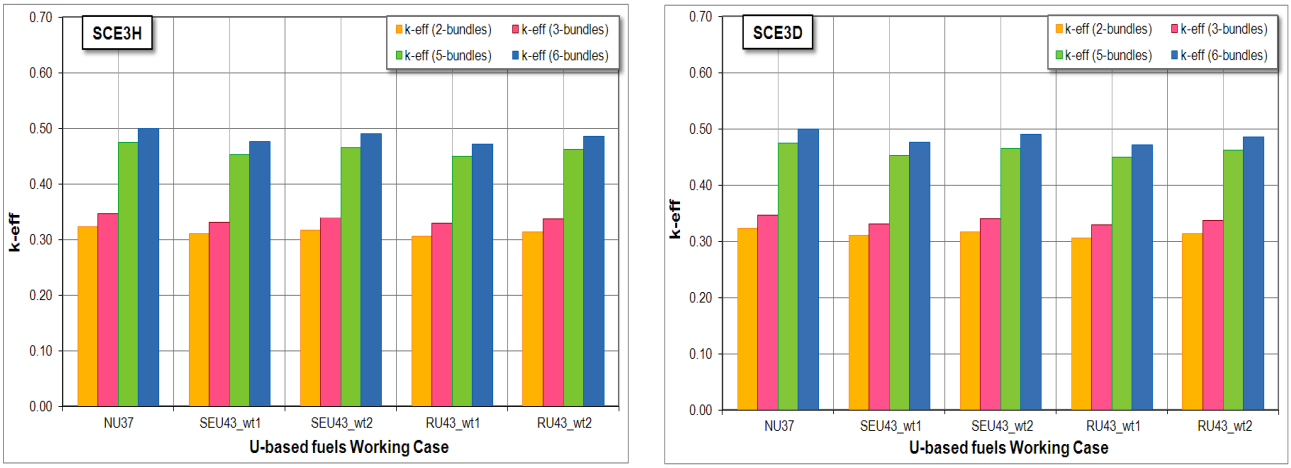


Fig. 13. Estimated k-eff values for NU37, SEU43 and RU43 fuel bundles stacked in different configurations in SCE3H (left) and SCE3D (right) accident scenarios
Neutron leakage treatment: ±X: REFL, ±Y: REFL, -Z: CONC4; +Z: VOID

Table 4. Relative differences (%) for SEU43_wt2, RU43_wt2 fuels vs NUE37 fuel comparison

Stacked bundles configuration	**Criticality scenario**			
	SCE1	**SCE2**	**SCE3H**	**SCE3D**
SEU43_wt2 vs NU37				
2-bundles stack	16.4%	15.2%	-1.9%	-2.3%
3-bundles stack	17.0%	16.0%	-1.6%	-2.1%
5-bundles stack	15.2%	14.0%	-2.0%	-2.1%
6-bundles stack	15.2%	12.5%	-2.0%	-2.4%
RU43_wt2 vs NU37				
2-bundles stack	17.2%	15.2%	-2.8%	-2.8%
3-bundles stack	17.2%	16.0%	-2.6%	-2.1%
5-bundles stack	16.0%	14.2%	-2.5%	-3.5%
6-bundles stack	15.3%	13.3%	-3.0%	-2.7%

Conclusions

Criticality factor k-eff has been estimated for two CANDU fuel bundle projects containing U-based fuel compositions, in several accidental scenarios for fresh and irradiated fuel.

37-element CANDU standard fuel bundle and 43-element CANDU type fuel bundle projects (developed in INR Pitesti) were used in the proposed analyses. Three U-based fuels were analyzed: natural uranium (NU), slightly enriched uranium (SEU) and recycled uranium (RU), with 0.9 wt% and 1.0 wt% enrichment in U^{235} (SEU, RU fuels).

Fuel irradiation was simulated by means of ORIGEN-S burn-up code, using CANDU fuels specific spectral factors, up to 8 MWd/kgHE for NU fuel, and up to 10 MWd/kgHE for SEU and RU fuels. Analyses have been carried out considering that all the fuel bundle elements contain the same U-based fuel composition.

Three accidental scenarios have been defined: SCE1 and SCE2 for fresh fuel (accidental immersion of fuel bundles in H_2O or D_2O), and SCE3 for irradiated fuel (immersion of fuel bundles in H_2O or D_2O). The accident assumes that 2, 3, 5 or 6 fuel bundles become stacked close together at distances that could lead to critical conditions, fuel bundles integrity being preserved.

For fresh fuel accidental criticality scenarios, RU43 and SEU43 fuels were characterized by 10%-16% higher k-eff values comparatively with the NU37 fuel. The relative differences calculated for k-eff estimated for RU43 and SEU43 fresh fuels were less than 1%.

As regarding the spent fuel accidental criticality scenarios, k-eff values obtained for NU37 fuel were 2%-6% higher than those characterizing SEU43 and RU43 fuels; SEU43 spent fuel has 2%-3% higher k-eff values than the RU43 spent fuel.

Fresh fuel accidental criticality scenarios led to subcritical k-eff (less than 1), except for SCE2 (assuming accidental immersion of fresh fuel bundles in D_2O) considering infinite lattice option for neutron leakage treatment. However, 3D calculations performed in SCE2 conditions but assuming realistic neutrons leakage treatment (options b and c) revealed subcritical k-eff values for the considered ″dangerous″ configurations of stacked fuel bundles.

For a single ″row″ of fuel bundles arrangements placed on a concrete platform and covered by a layer of water (the most realistic option for neutron leakage treatment) the 6-bundles stacked together on 3 rows seem to be the most ″dangerous″ configuration, regardless the fuel bundle type or U-based fuel composition.

The use of the innovative CN43 fuel bundle containing U-based fuels in CANDU reactors does not require additional radiation protection measures than those already implemented for the use of standard CANDU fuel bundle with NU fuel. Although at the end of irradiation the fissile material inventory does not exceed that of the C37 standard fuel bundle, the CN43 fuel bundle containing SEU or RU fuel show an increase in the inventory of isotopes interesting for non-proliferation by 30%.

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