CRITICALITY SAFETY STUDIES INVOLVED IN ACTIONS TO IMPROVE CONDITIONS FOR STORING "RA" RESEARCH REACTOR SPENT FUEL

M. V. Mataušek, N. Marinković
VINČA Institute of Nuclear Sciences, 11 001 Belgrade, Yugoslavia

ABSTRACT

A project has recently been initiated by the VINČA Institute of Nuclear Sciences to improve conditions in the spent fuel storage pool at the 6.5 MW research reactor RA, as well as to consider transferring this spent fuel into a new dry storage facility built for the purpose. Since quantity and contents of fissile material in the spent fuel storage at the RA reactor are such that possibility of criticality accident can not be *a priori* excluded, according to standards and regulations for handling fissile material outside a reactor, before any action is undertaken subcriticality should be proven under normal, as well as under credible abnormal conditions. To perform this task, comprehensive nuclear criticality safety studies had to be performed.

KEY WORDS

Nuclear criticality safety, spent fuel storage, handling fissile material outside a reactor, computer codes for calculating reactor core parameters

INTRODUCTION

A project has recently been initiated at the VINČA Institute of Nuclear Sciences to improve conditions in the spent fuel storage pool at the 6.5 MW research reactor RA, as well as to consider transferring this spent fuel into a new dry storage facility built for the purpose [1]. To perform these tasks, comprehensive nuclear criticality safety studies are needed.

A wide variety of computer codes and data libraries have been used for nuclear criticality safety calculations, depending on the complexity of the problem considered and the accuracy required. For standard geometries relatively simple few group or multigroup versions of one- or two-dimensional diffusion theory or neutron transport theory methods may be applied. For nonstandard geometries, as well as for treating certain benchmark configurations, very sophisticated multigroup multidimensional transport theory or Monte Carlo codes, possibly with a larger number of energy groups, are used.

To study nuclear criticality safety problems arising in away-from-reactor handling of the research reactor RA spent fuel, several methodologies, computer codes and data libraries are available and have been used at the VINČA Institute. The well known MCNP Monte Carlo code [2] can treat complicated geometries which may arise in accidental situations and is generally accepted for criticality safety studies. However, an external code [3] is needed if burnup dependent spent fuel isotopic composition has to be taken into account. Two dimensional multigroup transport theory codes have also been developed and applied in the VINČA Institute for reactor core parameters determination and nuclear criticality safety studies [4]. Still, if a large number of cases has to be studied, application of the above codes may become too cumbersome and costly.

Having in mind the existence and availability of computer codes and nuclear data libraries, which have been widely used and experimentally validated for calculating criticality parameters of different reactor core configurations with the same types of nuclear fuel, a possibility has been studied to use this methodology to calculate criticality parameters of certain non-reactor configurations. It is shown that such approach has certain advantages in comparison to the use of more sophisticated nuclear criticality safety codes, which have not been validated earlier for treating systems with the fuel compositions and configurations considered.

DESCRIPTION OF THE PROBLEM

Spent fuel, resulting from 25 years of operating the 6.5 MW heavy water moderated and cooled, graphite reflected, tank type research reactor RA at the VINČA Institute, is presently all stored in the temporary storage pool in the basement of the reactor building. The fuel element is an 11.3 cm long cylinder, with 3.72 cm of outer diameter, consisting of an outer tube with 2 mm thick fissionable material and 1 mm thick inner and outer Al cladding, and an 1 mm thick inner Al tube which serves as cooling intensifier. Fuel elements are inserted into a 2 mm thick Al tube (10 or 11 elements /tube), thus forming a fuel channel. The reactor RA core consists of up to 82 channels in a square lattice with 13 cm pitch. Until 1976 the reactor RA was operated with 2% enriched uranium metal fuel, when new fuel with 80% enriched uranium oxide dispersed in aluminium was bought from USSR. Both fuel types have the same geometry and dimensions and the same initial content of ^{235}U. The RA reactor fuel is also used in the zero power research reactor RB, where it is possible to create mixed cores with variable lattice pitch.

The six meters deep temporary spent fuel storage pool, consists of four connected basins, having thick concrete walls cladded with stainless steel. It is filled with approximately 200 m^3 of stagnant ordinary water. 304 channel-type stainless steel fuel containers, receiving up to 18 spent fuel elements each, are placed vertically in the pool, forming a regular lattice similar to the reactor core.

Initially, it was planned to transfer spent fuel back to the supplier for reprocessing or permanent storage, after 4-5 years of cooling in the temporary storage pool. Since this did not happen, in order to increase the storage capacity, the oldest spent fuel was gradually taken out of the original stainless steel containers and repacked into aluminium barrels, each containing 30 aluminium tubes receiving up to 6 irradiated fuel elements per tube. There are 30 such barrels, placed in two layers in the annex of the basin 4. Cadmium strips were placed in the barrels to assure subcriticality.

The spent fuel inventory in the reactor RA temporary storage pool is the following: 6656 fuel elements with initial enrichment 2% ^{235}U in the channel type stainless steel containers and the aluminium barrels and 884 fuel elements with initial enrichment 80% ^{235}U and relatively low average burnup in the channel type stainless steel containers. Quantity and contents of fissile material in the spent fuel storage at the RA reactor are such that possibility of criticality accident can not be *a priori* excluded. According to standards and regulations for handling fissile material outside a reactor, subcriticality should be proved under normal, as well as under credible abnormal conditions, before any action is undertaken.

DESCRIPTION OF CALCULATIONAL METHOD

To study nuclear criticality safety problems arising in away-from-reactor handling of the research reactor RA spent fuel, several methodologies, computer codes and data libraries are available and have been used at the VINČA Institute. The well known MCNP Monte Carlo code [2] can treat complicated geometries which may arise in accidental situations and is generally accepted for criticality safety studies. However, an external code is needed if burnup dependent spent fuel isotopic composition has to be taken into account. Two dimensional multigroup transport theory codes have also been developed and applied in the VINČA Institute for reactor core parameters determination and nuclear criticality safety studies [3, 4]. Still, if a large number of cases has to be studied, application of the above codes may become too cumbersome and costly. The aim of the present paper is to examine and to illustrate applicability of a standard reactor computational scheme WIMS-TRITON for studying nuclear criticality safety problems arising in away-from-reactor handling of the research reactor RA spent fuel.

WIMS code [5] is a complex modular scheme for calculating detailed space-energy neutron flux and reaction rates and producing few group burnup dependent data needed in diffusion theory calculations. Its main advantages are a very elaborate nuclear data library, several transport theory procedures and geometry options, and the fact that, being generally available, it has been thoroughly tested by a large number of users. Applicability of the reactor code WIMS for calculating criticality parameters of non reactor configurations containing fissile materials has been examined and proved earlier [6], by treating some typical ^{235}U containing systems used for defining the nuclear criticality safety standards. It was concluded that WIMS can be successfully applied for systems with low enriched uranium or diluted highly enriched uranium.

The three dimensional few group diffusion theory code TRITON [7] is used for overall criticality parameters determination. Experimental verification of the WIMS-TRITON calculational scheme has been performed earlier by studying different configurations of the critical facility, i.e. the zero power research reactor RB at the VINČA Institute [8].

Since the configuration of the spent fuel containers and the general arrangement of the existing storage pool are similar to the configuration of the fuel channels and the reactor lattice, it is believed that application of the calculational scheme used for in-core fuel management purposes, to calculate spent fuel storage criticality parameters, is justified. Besides, since the burnup history of each spent fuel element is known, if the same calculational scheme is applied, the actual spent fuel isotopic composition can be taken into account in a straightforward way.

RESULTS

In view of the forthcoming activities aimed to improve conditions for storing the RA reactor spent fuel, a large variety of cases, with different amounts and distribution of fissionable material and different compositions and configurations of structural materials in the present and future spent fuel storage, have been analysed.

Since concrete walls between particular basins of the existing spent fuel storage pool are very thick (80 cm), neutronic interaction between basins can be neglected. Basic data for the particular basins are given in Table 1.

It is supposed that the fuel with 80% enriched UO_2 and very low burnup, which was left in the core since 1984 when the reactor was shut down for refurbishment and reconstruction, is transferred into basin 1. Schematic representation of basins 2 and 3, with 13 material regions, each containing 8 channels containing fuel with different average burnup, is given in Fig. 1. Corresponding representation of the basin 4, together with the annex where aluminium barrels with repacked fuel are placed, is given in Fig. 2.

In order to provide subcriticality of densely packed, initially low enriched metallic uranium spent fuel, which contains a considerable amount of generated plutonium, cadmium strips were placed in the aluminium barrels. Since these strips were uncladded, it can be supposed that during the years they were dissolved in water and that cadmium is now sitting at the bottom of the barrels. This situation is

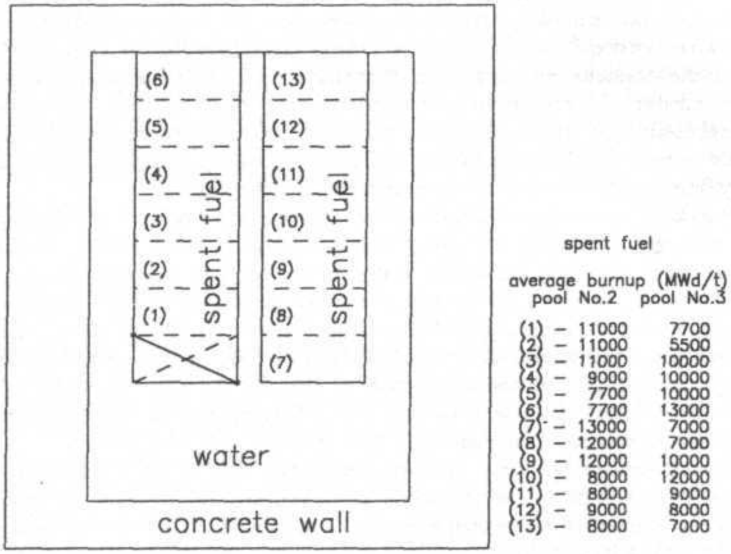

Fig. 1. Scheme of the spent fuel storage basins No. 2 and 3 as applied for TRITON criticality calculations

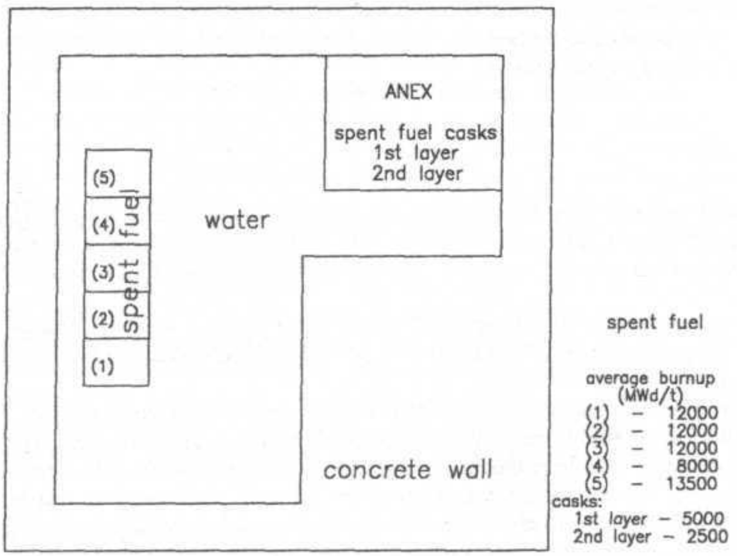

Fig. 2. Scheme of the spent fuel storage basin No. 4 as applied for TRITON criticality calculation

transferred into basin 1. Schematic representation of basins 2 and 3, with 13 material regions, each containing 8 channels containing fuel with different average burnup, is given in Fig. 1. Corresponding representation of the basin 4, together with the annex where aluminium barrels with repacked fuel are placed, is given in Fig. 2.

In order to provide subcriticality of densely packed, initially low enriched metallic uranium spent fuel, which contains a considerable amount of generated plutonium, cadmium strips were placed in the aluminium barrels. Since these strips were uncladded, it can be supposed that during the years they were dissolved in water and that cadmium is now sitting at the bottom of the barrels. This situation is considered as the first case of the structural changes influencing the criticality parameters of the pool.

Table 1. Basic data for the research reactor RA spent fuel storage pool.

Parameter	Basin 1	Basin 2	Basin 3	Basin 4	Annex
Length (cm)	400	380	380	380	170
Width (cm)	125	160	160	160	160
No. of containers	63	104	104	36	30
No. of fuel elements initial enrichment 2%	40	848	609	230	4929
No. of fuel elements initial enrichment 80%	392	136	265	101	0
Average burnup (%)	5	44	50	54	34

Table 2. WIMS results: k_∞ for barrels with low enriched metallic uranium spent fuel in the annex of basin 4.

Barrel	Without Cd		With Cd			
			Model: layer		Model tubes	
Average burnup	29 %	43 %	29 %	43 %	29 %	43 %
Hexagonal lattice						
In water	.925540	.864445	.580181	.539193	.663173	.616991
In air	.962351	.899570	.657989	.611307	.686088	.638716
Square lattice						
In water	.859653	.801999	.494738	.459993	.623509	.579643
In air	.962229	.899454	.657750	.611092	.685863	.638493

The second case of structural changes considered is change of the kind and the thickness of the upper reflector. Namely, new equipment and/or new materials in the room where the spent fuel storage pool is situated, or even the increased number of workers present during the planned operations, behave as an additional upper reflector when considering the criticality parameters of the system. On the other hand, taking out a certain amount of the pool water for filtering and purification means decrease of the upper reflector when considering criticality parameters of the system.

As an extreme case important for the safety analyses, as well as for the criticality safety analyses, a possibility of losing the water from the pool, for example by an accidental spraying it out when a high pressure spray system is used for underwater cleaning of the pool walls and the outer surfaces, is considered.

Table 3. WIMS-TRITON results: k_{eff} for the annex of basin 4.

Case considered	Upper reflector	Without Cd	With Cd		Cd at the bottom of barrels
			Model: layer	Model: tubes	
Water in the basin	Air	.873410	.539294	.615897	.849341
	Water	.894680	.549250	.606956	.812509
	Concrete	.858792	.530799	.601216	.799186
Air in the basin	Air	.935496	.654752	.745262	.820984
	Concrete	.940090	.656240	.748330	.824653

Table 4. WIMS-TRITON results: k_{eff} for all basins and different kinds of upper reflector

Description of the basin		Upper reflector		
		Air	Water	Concret e
1. 63 x 11 fresh fuel elements initial enrichment 80%		.113841	.125415	.106561
2. 104 x 11 elements initial enrichment 2%, average burnup 44%		.095374	.106034	.077738
3. 104 x 11 elements initial enrichment 2%, average burnup 50%		.088477	.096183	.073451
4. 36 x 11 elements initial enrichment 2%, average burnup 54%, Annex containing spent fuel with initial enrichment 2%, layer 1: 3125 elements in 19 barrels, average burnup 29% layer 2: 1805 elements in 11 barrels, average burnup 43%	No Cd	.948722	.949099	.942085
	With Cd	.597680	.612460	.592212
	Cd at the bottom	.889593	.898140	.882470

The "cluster" option of the WIMS code was used to calculate the infinite multiplication factor for the aluminium barrels containing the repacked spent fuel, for the cases with and without cadmium strips and for the burnup values shown in Fig. 2. Cadmium strips were modelled in two ways, as a continual layer or as vertical rods. The second way underestimates the effect of the absorber, i. e. overestimates k_{∞} which is better from the criticality safety point of view. A part of the barrels is arranged so to resemble square lattice, the others so to resemble hexagonal lattice. In this way the actual arrangement of the barrels is taken into account. The results presented in Table 2. show that cadmium strips, if not dissolved in water, provide sufficient subcriticality even in the accidental case of leak out of the pool water, when the space between the barrels would be occupied by air.

□ In order to analyse the effect of cadmium being dissolved in water and settled at the bottom of the aluminium barrels, WIMS-TRITON calculational scheme is applied for overall calculation of the annex of basin 4. Two group cross sections for cadmium, water, air and concrete were calculated using the PERSEUS option of the WIMS code, while collision probability PIJ option was used to calculate two group data for material regions containing fuel. These data were used as input for the overall criticality search and calculation of the effective multiplication factor in the x-y-z geometry by the few group diffusion code TRITON. Actual arrangement of the barrels was taken into account, i.e. square lattice for the upper layer and mixed square and hexagonal arrangement for the lower layer. The results of these calculations, for different kinds of the upper reflector and for the cases with and without water in the pool, are presented in Table 3.

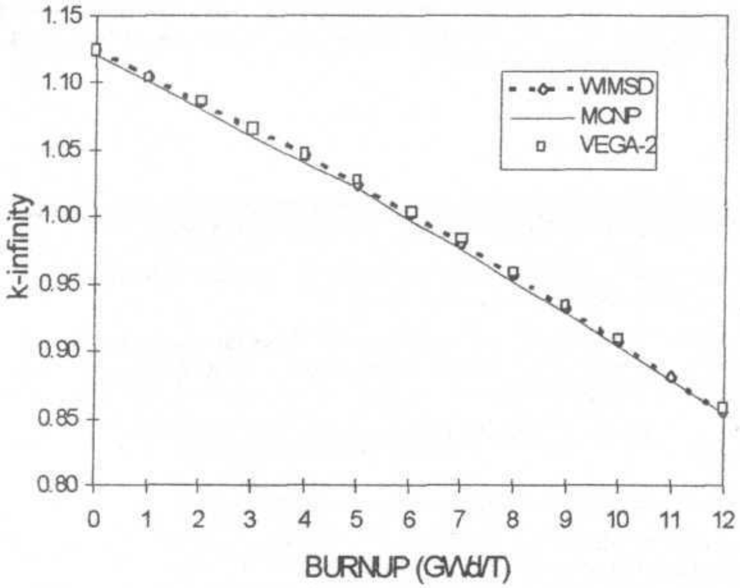

Fig. 3. Comparison of results for single pin cell in the storage barrel

Values of the k_{eff} calculated by WIMS-TRITON scheme for each of the four basins and for different kinds of the upper reflector are shown in Table 4. The basin 4 was treated together with the annex, as presented in Fig. 2.

Direct comparison of the results presented in Tables 2, 3 and 4 with results obtained applying some other methodology was not possible, since such comprehensive study has not been performed by other methods. Comparison could only be performed for some simplified cases. For instance, satisfactory agreement of burnup dependent infinite multiplication factors for pin cells without cadmium, calculated by WIMS-TRITON scheme and by MCNP [2] or VEGA [4] methodologies, was obtained, as shown in Fig. 3.

CONCLUSION

In view of the forthcoming activities aimed to improve conditions for storing the RA reactor spent fuel, a large variety of cases, with different amounts and distribution of fissionable material and different compositions and configurations of structural materials in the present and future spent fuel storage, have been treated. In particular, the following effects are analysed: the influence of cadmium distribution, as well as its representation in the calculational model, the impact of the water quantity present in the pool and the influence of the upper reflector.

Since the configuration of the spent fuel containers and the general arrangement of the existing storage pool are similar to the configuration of the fuel channels and the reactor lattice, it is believed that application of the calculational scheme used for in-core fuel management purposes, to calculate spent fuel storage criticality parameters, is justified. Besides, since the burnup history of each spent fuel element is known, if the same calculational scheme is applied, the actual spent fuel isotopic composition can be taken into account in a straightforward way. It can be concluded that applying methods and computational schemes, normally used for reactor core design and in-core fuel management purposes, to perform nuclear criticality studies of spent fuel storage options, may have certain advantages in comparison to the use of sophisticated nuclear criticality safety codes not validated earlier for treating systems with the fuel compositions and configurations considered.

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