Romanian Reports in Physics, Vol. 56, No. 4, P. 721-726, 2004

MONTE CARLO METHODS APPLICATION TO CANDU SPENT FUEL COMPARATIVE ANALYSIS

C. A. MĂRGEANU[1], T. ANGELESCU[2], P. ILIE[1]

[1] *Institute for Nuclear Research Pitesti, Romania, Po Box 78 Pitesti, malice@scn.ro*
[2] *Faculty of Physics, University of Bucharest, Po Box MG-12 Bucharest*

(Received July 22, 2004)

Abstract: The paper follows to apply Monte Carlo methods to CANDU spent fuel analysis, starting from the discharge moment, following spent fuel transport and finally, spent fuel intermediate dry storage. In criticality calculations KENO-VI code is used. Spent fuel inventories are obtained by means of ORIGEN-S code. By using MORSE-SGC code, photon dose rates to the cask/storage basket wall and in air, at different distances from the cask/storage basket have been estimated.

Key words: spent fuel transport, spent fuel intermediate dry storage, natural UO2 fuel, SEU fuel, photon dose rates, shipping cask, storage basket

INTRODUCTION

Last decade, both for operating reactors and future reactor projects, a general trend to rise the discharge fuel burn up degree has been registered. Taking into account for the possible impact on the human and environment, spent fuel characteristics must be well known. According to IAEA data, more than 10 millions packages containing radioactive materials are annually world wide transported. The radioactive materials transport safety must be carefully settled.

Romania has a single NPP with 5 reactors PHWR CANDU 6 type, 705 MW(e) each. In the first 3 years of commercial operation, Unit1 has given ~10% from the total electricity produced in Romania, this percent assuming to increase at 17-20% after 2005 (both Unit1 and Unit2 in operation). By ratifying [1] Romania has shown its willingness to undertake all the necessary steps for achieving the required level in the spent fuel and radioactive waste safe managing.

SPENT FUEL ANALYSIS GENERAL DESCRIPTION

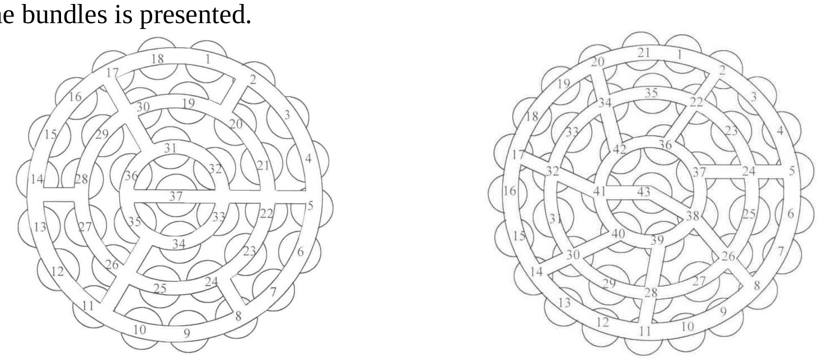
The paper objectives

The paper aims to apply Monte Carlo methods to CANDU spent fuel analysis, starting from the discharge moment, followed by the spent fuel transport and, finally, the spent fuel intermediate dry storage.

3 different “fuel cycle” cases, CANDU type, have been considered: UNAT (37 rods standard fuel bundle, natural UO_2 fuel), SEU37 (37 rods standard fuel bundle, SEU fuel) and SEU43 (43 rods fuel bundle, SEU fuel). After the fuel discharge from the reactor a criticality analysis is done and the criticality coefficient is calculated. CANDU spent fuel will be characterized by concentration, radioactivity, thermal and gamma powers. The shielding calculations follow to estimate the photon dose rates both for spent fuel transport and intermediate dry storage to the shipping cask/storage basket wall and in air, at different distances from the cask/basket. A comparison between CANDU fuel projects is also done.

The shielding problem theoretical model setup

The source of radiation: As source of radiation a single spent fuel bundle, CANDU type (of 37 or 43 zircalloy rods, filled with natural UO_2 and 0.96 wt% ^{235}U enriched SEU pellets) was considered. In Fig. 1 the geometrical arrangement of the bundles is presented.



a) b)

Fig. 1 - Geometrical arrangement for CANDU fuel bundles with: a) 37 rods; b) 43 rods

A fuel bundle residence period inside the reactor core of 235.25 days for natural UO_2 fuel and 312.21 days for SEU fuel, respectively, has been considered, followed by up to 10 years of cooling in the NPP pools.

The cask: All the geometrical and material data related to the shipping cask were considered according to the shipping cask type B model [2]. The geometrical model for the cask consists in right circular cylinders of shielding materials with a central cavity to accommodate the source. The spent fuel storage basket is a cylinder, stainless steel walls, containing 60 spent fuel bundles, in vertical position, arranged on 4 circular rings of 6, 12, 18 and 24 bundles, respectively [3].

The shielding calculations: In order to perform the shielding calculations, the fuel bundle was represented by 3 right, circular cylinders, containing an homogenous mixture ("fuel") of fuel, cladding and structure materials, with respect for the volume conservation. Fig. 2 shows the geometrical configurations for the source and the source-shipping cask assembly.

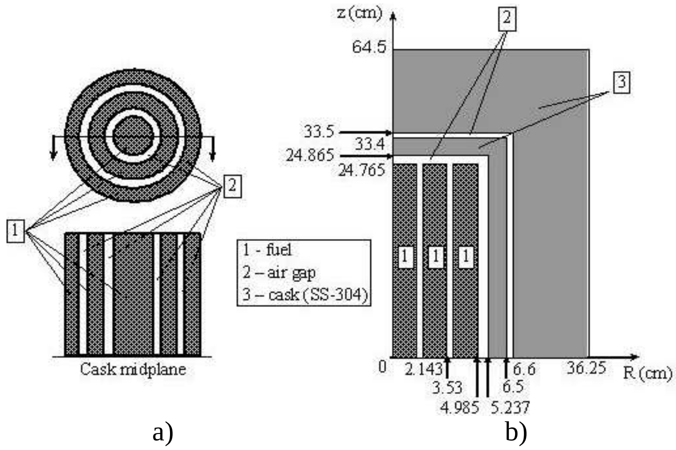


Fig. 2 - 2D geometrical configuration for: a) source; b) source-shipping cask assembly

For the spent fuel storage basket, the previous fuel bundle geometrical model has been used: the central cylinder for first 2 inner rings, the 2nd cylinder for the 18 bundles ring and the 3rd for the 24 bundles outer ring.

General data regarding the spent fuel analysis

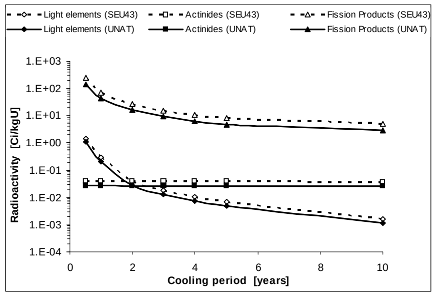
The criticality analysis has been performed by means of KENO-VI Monte Carlo computer code and the radionuclide inventories and source terms have been obtained by using ORIGEN-S code. In order to perform the shielding calculations the MORSE-SGC Monte Carlo code has been used. All the 3 computer codes are included in SCALE4.4.a system, developed by the ORNL, SUA.

RESULTS AND DISCUSSIONS

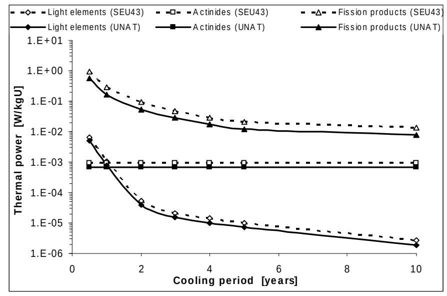
Fuel characteristics and isotopic composition were considered according to [4, 5, 6]. Data regarding the structural materials and nuclides isotopic composition from [7] have been used.

After discharge from the reactor core, the following criticality coefficient values have been obtained: 0.897077 (UNAT), 0.862641 (SEU37) and 0.859966 (SEU43). The relative differences follow as: 3.84% (UNAT-SEU37 comparison), 4.14% (UNAT-SEU43 comparison) and 0.31% (SEU37-SEU43 comparison).

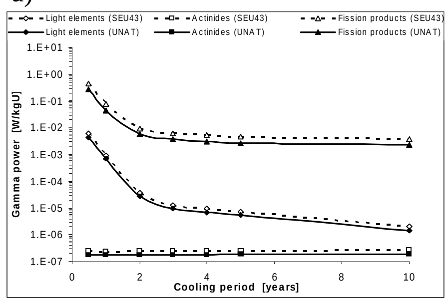
Radionuclide inventory and irradiated fuel characteristics have been obtained by taking into account for all relevant isotopes generation and depletion during both the irradiation and cooling phases of the fuel history. SEU fuels comparison leads to insignificant differences in spent fuel characteristics. Fig. 3 shows the spent fuel characteristics total values evolutions, during the cooling period, for UNAT and SEU43.



a)



b)



c)

Fig. 3. - Total values evolution during cooling period for: a) radioactivity; b) thermal power; c) gamma power

The photon dose rates to the cask wall and in air, at different distances from the shipping cask have been estimated. Their evolution is shown in Fig. 4. UNAT-SEU43 comparison leads to the following relative differences: (8-13)% (first year from discharge), (13-25)% (2-5 years of cooling), 10% (10 years of cooling).

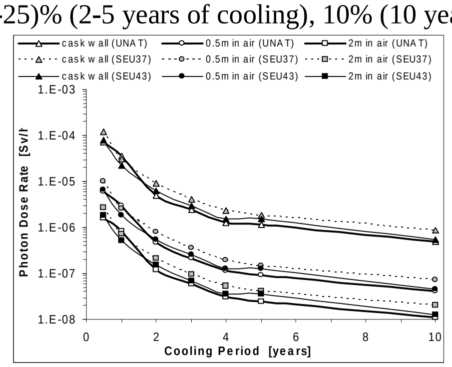


Fig. 4 - Photon dose rates evolution with the cooling time for the analyzed fuels

Finally, photon dose rates to the spent fuel transport/storage basket wall and in air, at different distances from the storage basket, have been estimated. Fig. 5 shows their evolution with the measuring point. Before inserting in the storage basket, spent fuel was kept for 7 and 10 years, in the NPP cooling pools.

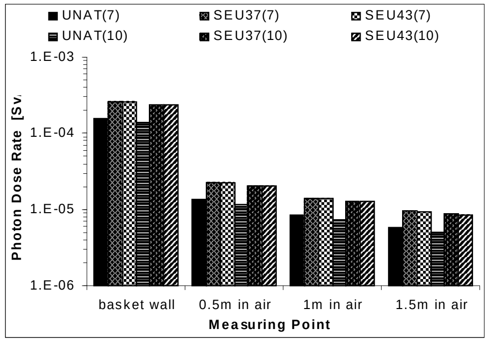


Fig. 5. Photon dose rates evolution with the measuring point for 7 and 10 years of cooling

The cooling period increasing is accompanied by a reduction in photon dose rates values. Cooling period comparison leads to the following relative differences: a) to storage basket wall: 12% (UNAT), 9.7% (SEU37), 9.4% (SEU43); b) at 1.5m distance from storage basket: 14.7% (UNAT), 8.3% (SEU37), 8.0% (SEU43).

CONCLUSIONS

SEU fuels lead to higher burn up degrees than the natural UO_2 one. Meanwhile, for the same amount of electric energy generated, CANDU SEU fuel cycle produces a smaller weight of spent fuel (42.22% relative difference) than the natural UO_2 one.

No significant differences in SEU spent fuel characteristics have been observed. UNAT-SEU43 comparison leads to the following relative differences: a) light elements: (26.4-27.5)% in radioactivity, (26.3-27.7)% in thermal power and (26.4-27.2)% in γ power; b) actinides: (30.9-33.4)% in radioactivity, (25.3-25.4)% in thermal power and (25.8-28.1)% in γ power; c) fission products: (41.5-42.2)% in radioactivity, (41.5-42.3)% in thermal power and (40.2-42.3)% in γ power.

The estimated photon dose rates values both for spent fuel transport and intermediate dry storage are small, allowing a safe manipulation for the spent fuel shipping cask and transport/storage basket, also. The dose rates to the cask wall decrease from 10^{-5} Sv/h (first year from discharge) to 10^{-6} Sv/h (after 2 years of cooling) and reach 10^{-7} Sv/h (after 10 years of cooling). At different distances from the shipping cask, the corresponding values were one degree smaller. For the spent fuel intermediate dry storage the estimated photon dose rate are rapidly decreasing with the distance growth: from 10^{-4} Sv/h to the storage basket wall, reach 10^{-7} Sv/h at 5m distance from the storage basket.

Nuclear energetic is called now to find the optimal solution in solving a series of political and public problems: nuclear accidents risks, radioactive waste disposal, nuclear weapon proliferation and nuclear plant decommissioning.

REFERENCES

1. *Joint Convention on the Safety of Spent Nuclear Fuel Management and on the Safety of Radioactive Waste Management* (1999)
2. L. Tuturici, P. Ilie *Spent fuel shipping cask type B 1-5-35384-1126.1/PE*, INR Protocol 2563 (1998)
3. *Romania National Report*, Joint convention on the safety of spent nuclear fuel management and on the safety of radioactive waste management (2003)
4. *Cernavoda Unit 1 Nuclear Generating Station–core fuel design manual*, 81-37000-DM-000, Rev. 1
5. C. A. Margeanu *Monte Carlo methods application to CANDU fuel cycle*, Ph.D. Dissertation, University Bucharest (2003)
6. G. Horhoianu, D. R. Moscalu, G. Olteanu, D. V. Ionescu *Development of SEU-43 fuel bundle for CANDU type reactors*, in Annals of Nuclear Energy No.16, vol.25, pp.1363-1372 (1998)
7. *Chart of the nuclides–Knolls Atomic Power Laboratory*, Naval Reactors, US. Department of Energy, 15th ed., Rev. 1996