19th Symposium on AER for VVER Reactor Physics and Reactor Safety

Burnup Influence on the VVER-1000 Reactor Vessel Neutron Fluence Evaluation

I. Panayotov, N. Mihaylov, K. Ilieva, D. Kirilova, M. Manolova

INRNE-BAS, Sofia, Bulgaria

Abstract

The neutron fluence of the vessels of the reactors is determined regularly accordingly the RPV Surveillance Program of Kozloduy NPP Unit 5 and 6 in order to assess the state of the metal vessel and their radiation damaging. The calculations are carried out by the method of discrete ordinates used in the TORT program for operated reactor cycles. An average reactor spectrum corresponding to fresh U-235 fuel is used as an input neutron source. The impact of the burn up of the fuel on the neutron fluence of VVER-1000 reactor vessel is evaluated.

The calculations of isotopic concentrations of U-235 and Pu-239 corresponding to 4 years burn up were performed by the module SAS2H of the code system SCALE 4.4. Since fresh fuel or 4 years burn up fuel assembly are placed in periphery of reactor core the contribution of Pu-239 of first year burn up and of 4 years burn up is taken in consideration.

Calculations of neutron fluence were performed with neutron spectrum for fresh fuel, for 1 year and for 4 years burn up fuel. Correction factors for neutron fluence at the inner surface of the reactor vessel, in ¼ depth of the vessel and in the air behind the vessel were obtained. The correction coefficient could be used when the neutron fluence is assessed so in verification when the measured activity of ex-vessel detectors is compared with calculated ones.

Introduction

Neutron physical calculations are a key element in the analysis for the safe operation of nuclear facilities. Reliable determination of the neutron flux and its functionality (fluence, activities, dose rates) provides the information needed for engineering calculations for the construction and management of nuclear reactors. Difficulties in solving these problems come as the complexity of the survey (heterogeneous multilayer structures) and their sizes (up to medium free range 15-20). Adequate modeling of neutron transport is limited and from the numerical representation of the complex interaction of neutrons with the substance, which in turn requires a test with different neutron data libraries.

The neutron fluence of the vessels of the reactors is determined regularly accordingly the RPV Surveillance Program of Kozloduy NPP Unit 5 and 6 in order to assess the state of the metal vessel and their radiation damaging. The fluence calculations are carried out by the method of discrete ordinates TORT program […] for the operated fuel cycles.

The neutrons source from the reactor core is determined by the formula:

$$S\,(r,\theta,z,E,B)=\chi(r,E,B\,)\frac{\nu}{E}(r,B\,)k_{\nu}(r,\theta,z,B\,)P\;, \qquad (1)$$

where k_v is the peaking power coefficient for each assembly or fuel pin/node, B- burnup coefficient, MW.d/tU, χ - neutron fission spectrum, ν/E - mean number of neutrons per fission to the mean released energy, P is the power of the reactor core in %.

An average reactor spectrum corresponding to fresh U-235 fuel is used as an input neutron source:

$$\chi_U\left(E_i\right)=2\cdot\left(\frac{E_i}{\pi T_M^3}\right)^{\frac{1}{2}}\cdot e^{\frac{-E_i}{T_M}} \qquad (2)$$

Where E_i , (MeV) is the average value of the energy in the corresponding interval, π = 3.1415, T_M= 1.323 за ^{235}U.

During the reactor operation the burnup is increasing and Pu-239 is accumulating from the reaction:

$$^{238}_{92}U + n = {}^{239}_{92}U$$

$$^{239}_{92}U \xrightarrow{\beta} {}^{239}_{93}Np \xrightarrow{\beta} {}^{239}_{94}Pu$$

Due to this the neutron source spectrum hardens. That is why more correct presentation of the neutron source has to take into account the accumulation ot Pu-239.

The objective of this paper is to obtain correction coefficient for the neutron fluence on the VVER-1000 reactor vessel which takes into account the contribution of Pu-239 in the neutron source spectrum. The aim is to evaluate the neutron fluence change depending on the burnup.

The neutron spectrum taking into account the Pu-239 accumulation is presented by the formula:

$$\chi(E,B) = \chi_U(E)[1-c(Pu)] + \chi_{Pu}(E)c(Pu) \quad (3)$$

Where c(Pu) is

$$c(Pu) = \frac{[\sigma_f(Pu239) * N(Pu239)] * 100\%}{[\sigma_f(U235) * N(U235) + \sigma_f(Pu239) * N(Pu239)]} \quad (4)$$

Where σ_f is microscopic fission cross-section and N is the nuclear concentration;

Calculations

According the Kozloduy NPP operational practice assemblies with fresh fuel or four years spent fuel are placed in the periphery of the reactor core in azimuth direction of maximum neutron exposure (8 deg in a 30 deg symmetry sector).

Neutron transport calculations were carried out for cycle 10 (with a fresh fuel assembly in the periphery of reactor core) and 11 (four years spent fuel) of Kozloduy NPP Unit 5.

The calculations of isotopic concentrations of U-235 and Pu-239 corresponding to 4 years burn up were performed by the module SAS2H of the code system SCALE 4.4[…]. Since fresh fuel or 4 years burn up fuel assembly are placed in periphery of reactor core the contribution of Pu-239 of first year burn up and of 4 years burn up is taken in consideration.

In the task for calculations with the TORT a 30^0 sector symmetry of the core in the radial-azimuth geometry is considered. The problem oriented neutron cross section library BGL […] (47 neutron groups and 20 gamma) is used. The fluence is calculated for 3 points of the reactor vessel – on the inner side, at debt ¼ of RPV and at the outer side of RPV. All 3 points are in the azimuth direction of maximum exposure (8 deg direction in the 30 deg symmetry sector). For the fluence calculation only neutrons with energy >0.5 MeV are considered. In this case one-dimensional geometric model of the reactor with corresponding radiation history is considered.

The concentration of U-235 and Pu-239, depending on the burn up was calculated by the module SAS2H of the system SCALE 4.4. SAS2H module calculates the mass of the isotopes in the fuel assembly with initial enrichment 4.4% and average burn up level for each fuel cycle. The results are presented in Table 1.

Cycle	Burnup [Mwd/kgU]	^{235}U [$cm^{-3}.10^{24}$]	^{239}Pu [$cm^{-3}.10^{24}$]	c (Pu) [%]
After 1 cycle	1.90E+01	6.34E-04	2.45E-04	26.8
After 2 cycles	3.30E+01	4.64E-04	3.24E-04	39.8
After 3 cycles	4.10E+01	3.89E-04	3.54E-04	46.3
After 4 cycles	4.60E+01	3.48E-04	3.69E-04	50.1

The contribution weight of U-235 and Pu-239 according the formula (4) is given in the last column of Table 1.

The neutron fluence results with and without taking into account the contribution of the Pu-239 are presented in Table 2.

Table 2. Neutron fluence, $1x10^{18} 1cm^{-2}$, results.

Place	F_{01}	F_{b19}	F_1	F_{04}	F_{b41}	F_{b46}	F_4
RPV inner wall	1.18	1.240	1.211	1.246	1.349	1.357	1.353
1/4 RPV depth	0.880	0.924	0.902	0.925	1.002	1.008	1.005
RPV outer wall	0.150	0.157	0.154	0.156	0.169	0.170	0.169

F_{01} - Fluence for first year periphery fuel assembly and fresh fuel neutron spectrum (2)

F_{b19} - Fluence for first year periphery fuel assembly and burnup=19 fuel neutron spectrum (3)

F_1 - Average fluence for first cycle

F_{04} - Fluence for fourth year periphery fuel assembly and fresh fuel neutron spectrum (2)

F_{b41} - Fluence for fourth year periphery fuel assembly and burnup=41 fuel neutron spectrum (3)

F_{b46} - Fluence for first year periphery fuel assembly and burnup=46 fuel neutron spectrum (3)

F_4 - Average fluence for fourth cycle

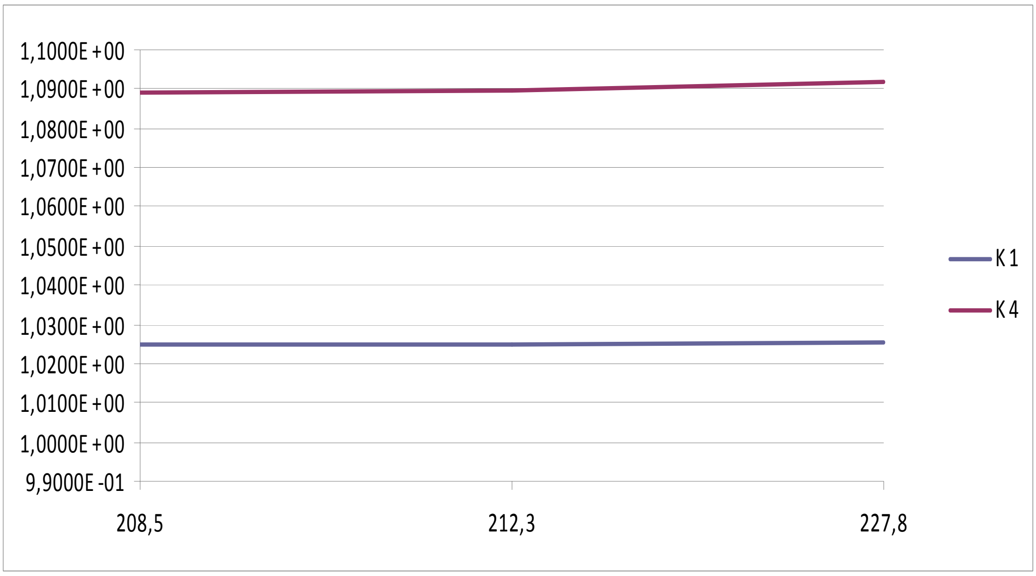
The correction coefficients evaluated by the formula:

$f_n=F_n/F_{0n}$ (5), n= 1 or 4

are given in Table 3.

Campain	Burnup	Correction coef. inner RPV	1/4	Outer RPV
1	0-19	1,0247E+00	1,0249E+00	1,0253E+00
4	41-46	1,0858E+00	1,0864E+00	1,0882E+00

As it is seen from the results the correction coefficient for one year fuel is too small, within the uncertainty of the calculations. This means that the application of the fresh fuel neutron spectrum gives reliable neutron fluence results. When fourth year fuel is used the fluence increases by about 9% and the correction factors can be applied to obtain more precise results.



Graph.1….

The sensitivity analysis carried out shows that the correction coefficient within the burnup interval (41-46) does not vary significantly, within 0.6%. That is why the correction coefficient value obtained for the average neutron fluence F4, could be used for assessment of the neutron fluence.

Conclusion

Calculations of neutron fluence were performed with neutron spectrum for fresh fuel, for 1 year and for 4 years burn up fuel. The calculations of isotopic concentrations of U-235 and Pu-239 corresponding to 4 years burn up were performed by the module SAS2H of the code system SCALE 4.4. Correction coefficient for neutron fluence at the inner surface of the reactor vessel, in ¼ depth of the vessel and in the air behind the vessel were obtained. When fourth year fuel is used the fluence value increases by about 9% and the correction factors can be applied to obtain more precise results. The correction coefficient could be used as when the neutron fluence is assessed so in verification when the measured activity of ex-vessel detectors is compared with calculated ones. The obtained results allow a conservative estimation of neutron fluence in determining the neutron damage of reactor pressure vessel.

Reference:

T. Apostolov, K. Ilieva, S. Belousov, I. Penev. " Methodics for numerical and experimental determination of neutron fluens, activities and doses", INRNE-BAS, July 1998

DOORS2.2, RSICC ORNL, "Code Package CCC-650", 1998

"SCALE: A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation". NUREG/CR-0200, Rev.6, ORNL/NUREG/CSD-2/R6, September 1998.

Bucholz, J.A., Antonov, S.Y. and Belousov, S.I., BGL440 and BGL1000 Broad Group Neutron/Photon Cross-Section Libraries Derived from ENDF/B-VI Nuclear Data, INDC(BUL)-15, Distrib.:G, Nov. 1996

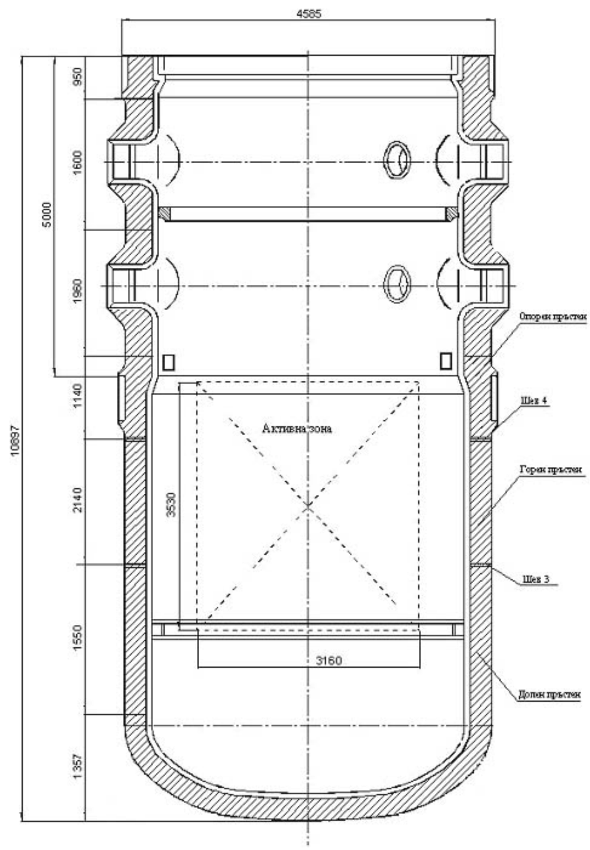
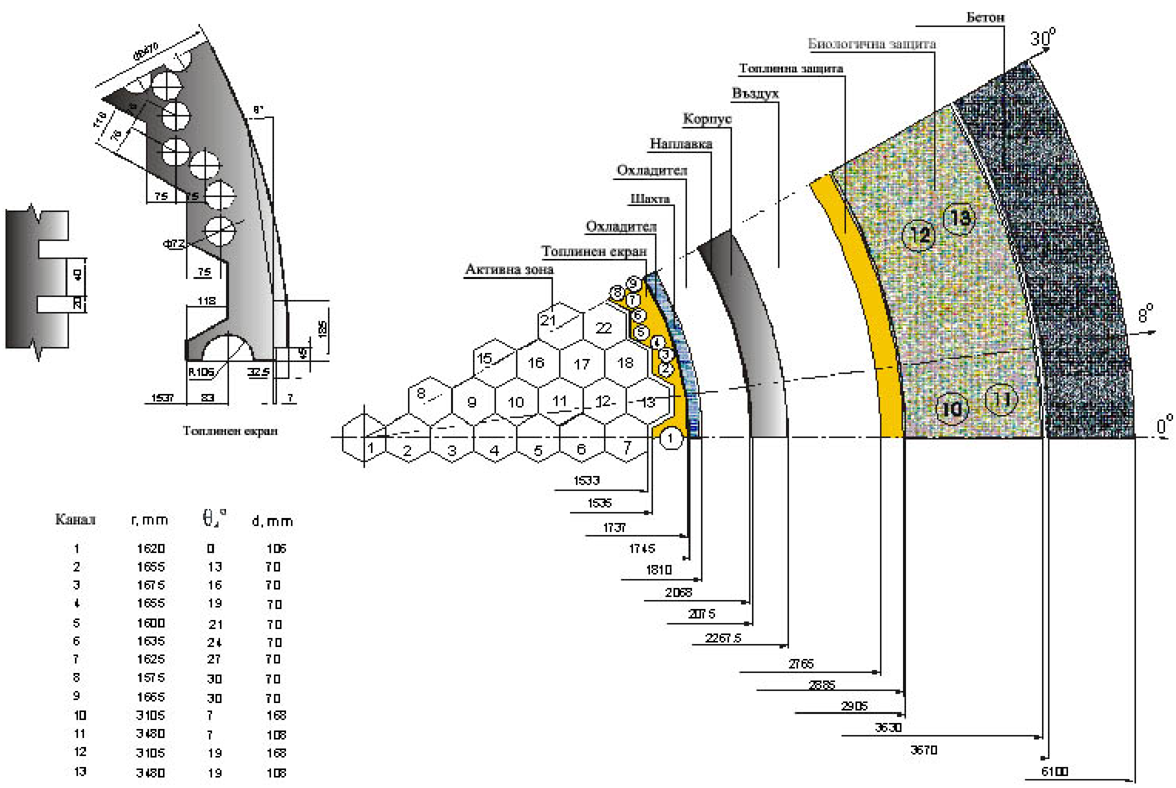


Fig. 1 VVER-1000 reactor vessel



Канал	r, mm	θ, °	d, mm
1	1620	0	106
2	1655	13	70
3	1675	16	70
4	1655	19	70
5	1600	21	70
6	1635	24	70
7	1625	27	70
8	1575	30	70
9	1665	30	70
10	3105	7	168
11	3480	7	108
12	3105	19	168
13	3480	19	108

Fig. 2 Radial geometry of VVER-1000

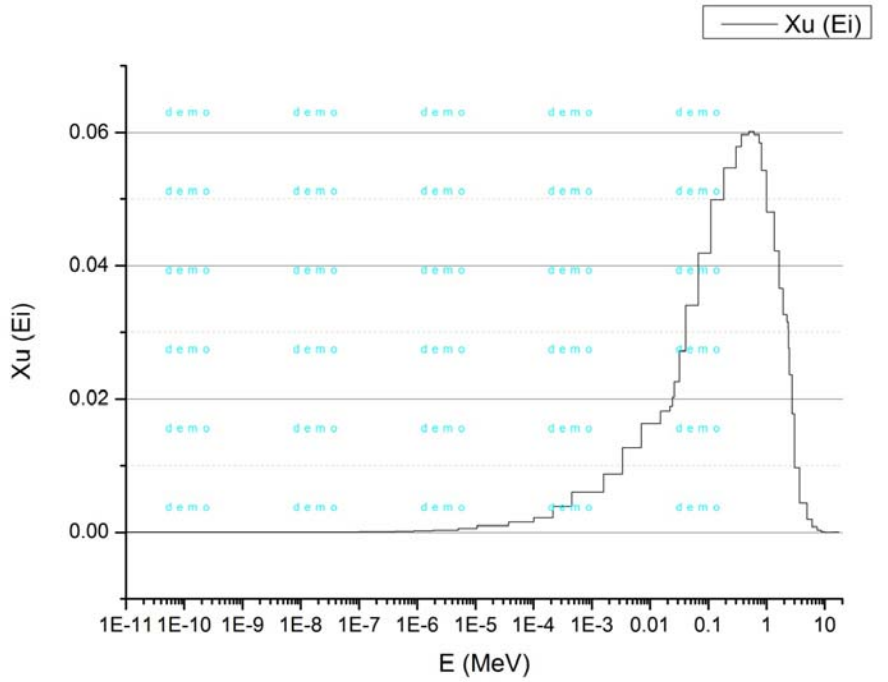


Fig. 3 Neutron spectrum of U-235 for fresh fuel

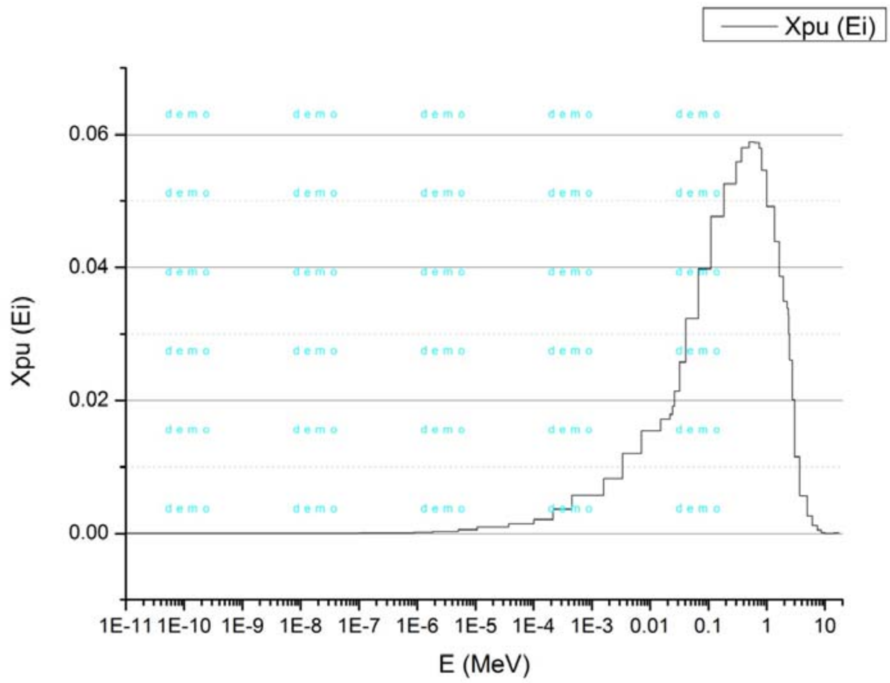


Fig. 4 Neutron spectrum of Pu-239