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Evaluation of Loading Pattern Characteristics Influence on VVER 1000 Nuclear Reactor Pressure Vessel Neutron Fluence

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Abstract

The present study deals with computational and analytical evaluation for the first approach of fuel loading, and presents the changes in neutron flux distribution of VVER-1000 reactor with each fuel loading type. The first approach proposed is based on dividing the core into axial cylindrical areas (equal number of batches), with each area having similar fuel (burn-up history). This paper includes a description of the core and fuel assembly lattice. The results allow us to determine the optimal first approach loading pattern, taking into account the maximum burn-up value, best power distribution and minimum vessel neutron fluence. The pressure vessel integrity analysis is carried out by comparing the vessel neutron fluence for each fuel loading scheme. Deterministic and probabilistic methods are employed in this study in order to achieve our goals.

Keywords: *: GETERA 90; MCNP4C2; Linear Reactivity Model; Burnable Absorber; VVER-1000; WWER-1000; Fuel Loading Scheme.*

1. Introduction

The life-time and integrity of a pressurized water reactor directly depends on the neutron irradiation of the reactor pressure vessel [1,2]. The pressure vessel is a fundamental component of light water reactors, since it contains the core and control mechanisms at high temperature and under high pressure. Therefore, pressure vessel integrity is important from the safety point of view, since increasing neutron fluence may significantly alter the behavior of vessel steel [3,4].

The embitterment of the material of the reactor vessel is primarily due to the fast neutron flux. The possible extension of the reactor life time needs neutron physical investigation of how new types of fuel elements, and new loading patterns, influence the neutron flux attaining the reactor vessel and detectors [5,6].

As indicated in the literature [7,8], core management plays an important role in the assessment of nuclear safety, as well as in the associated economics. In LWRs, the reactor is operated for normally one year long after the fuel is loaded into the core. After a period of operation, part of the fuel is replaced. Fuel loading and operation are repeated within a cycle. In-core fuel management implies designing fuel loading schemes over several cycles, such that the core produces the required energy output in an economical way, without violating safety constraints.

There are various strategies used to design the reload pattern. The out-in pattern is one wherein fresh fuel is loaded in the periphery of the core, then moved inward in subsequent cycles. The in-out pattern is the reverse. In the first few decades of PWR operation, the out-in patterns were employed [9]. More recently, the in-out procedure has replaced the out-in method to obtain low leakage cores and conserve ^{235}U. However, low leakage cores require the use of burnable poisons.

Economics and safety are the two primary concerns competing with each other in the in-core fuel management. In order to design all optimal fuel loading schemes, the constraints that the scheme must satisfy should be first identified.

In this study, pressure vessel integrity analysis is carried out for VVER-1000. This is achieved by studying the actual types of fuel loading strategies, and determining suitable schemes such as low leakage fuel-loading.

2. Fuel Management Objectives and Constraints

Most of the constraints are safety-related, except for the fundamental energy production requirements [7,8,9]. Therefore, reactivity and power distribution, in addition to the associated fuel enrichment and burnable absorbers, are described. The objective of fuel management is to design a fuel-loading scheme that is capable of producing the required energy at the minimum cost, while satisfying the safety constraints. More specifically, the objectives are:

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(a) to meet the energy production requirements;
(b) to satisfy all safety-related design limits;
(c) to provide sufficient operating margins; and finally
(d) to minimize the power generation cost.

In order to produce the required energy, the reactor must sustain the rated power level for the specified cycle length, and should be able to be properly started and shutdown at any time in the cycle. The required energy generation becomes possible only when the loaded fuel has sufficient reactivity that covers reactivity defects associated with startup, as well as reactivity loss due to fuel depletion [10,11]. The startup defects consist of temperature, power, xenon and neutron leakage.

3. VVER-1000 Nuclear Reactor

VVER or WWER [12-16] is an abbreviation for "water water energy reactor." It is a pressure vessel type nuclear reactor with water used both as moderator and coolant, resulting in a thermal neutron spectrum. WWER-1000 designates a unit of 1000 MWe (electrical power) as output.

A VVER-1000 nuclear reactor core contains 163 fuel assemblies, arranged in hexagonal geometry (Figure 1). Each fuel assembly consists of 311 fuel pins, 18 guide tubes for placing burnable absorber cluster (BAR) or for movement of absorber rods of control and protection system (CPSAR), one guide tube for keeping in core instrumentation detectors (ICID), and a slotted central tube for structural support [10,13]. All these fuel pins/tubes are held by a framework of 15 hexahedral spacer grids, and a supporting tail grid (Figure 2). In order to reduce parasitic capture of neutrons in the core, several components, like the fuel clad, guiding tubes, BAR clad and spacing grids, are made of zirconium alloy. There are 54 locations in the core where Rhodium type SPND detectors shall be installed in ICID tubes [13]. In the first cycle, 42 BAR clusters are used for the purpose of flux flattening and for ensuring negative moderator coefficient of reactivity. In further cycles, 18 BAR clusters are used. The number of CPSAR used in first cycle is 85, which is increased to 103 in subsequent cycles [13]. The main physical characteristics of the reactor core are given in Table 1 [12-16].

The core, as well as the peripheral components, including reflector and water channels, were carefully and comprehensively modeled in this work (Figure 1). This is necessary in order to obtain realistic neutron flux calculated not only in the core, but also in that leaking to the vessel.

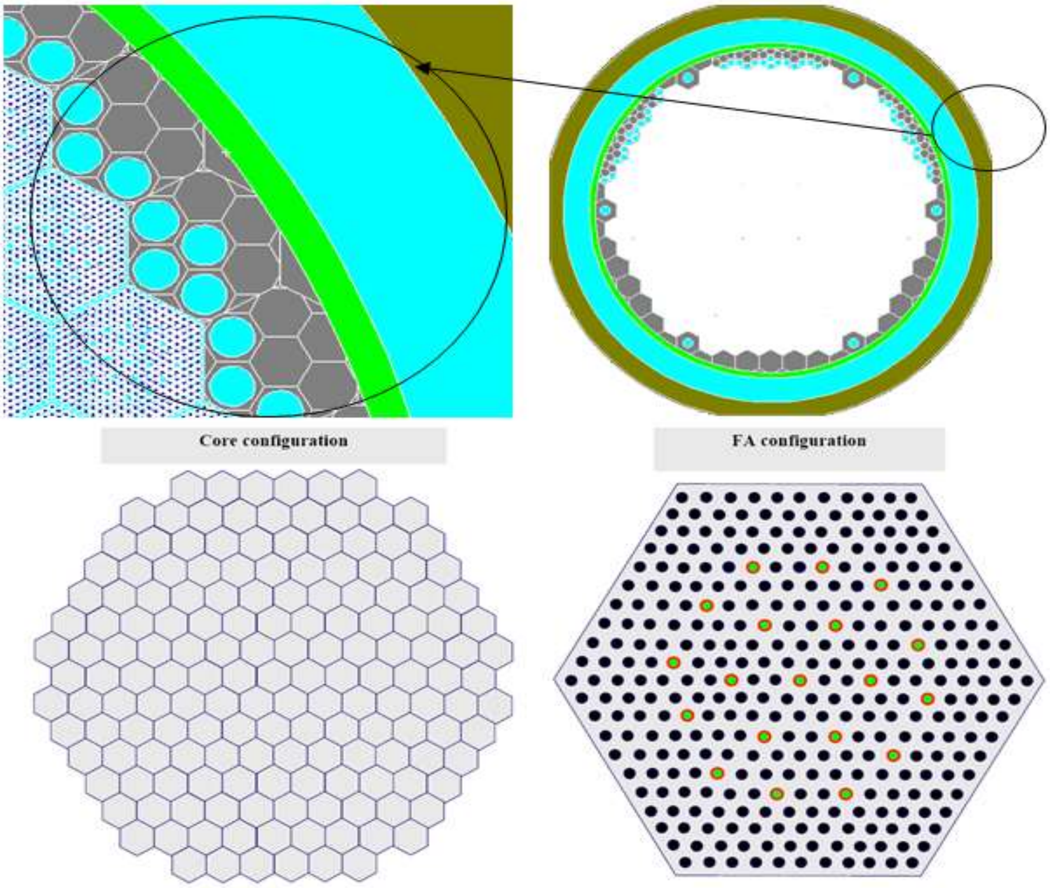

Figure 1. Upper panel: The reactor core and peripheral components were carefully modelled in this work (see text). Lower panel: Schematic view of the reactor core plan and fuel assembly [12]. Non-fuel element positions are identified.

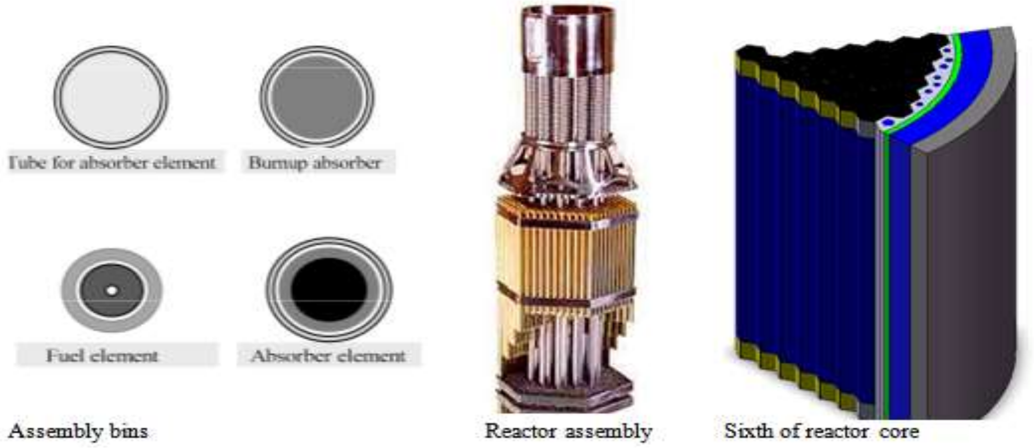

Figure 2. The design of FAs in the core (fuel and control rods).

Table 1. The main physical characteristics of the reactor core [12-16].

Characteristics	Value
Reactor nominal thermal power, MW	3000
Coolant pressure at the core outlet, MPa	15.7
Coolant temperature at the reactor outlet, C	321
Flow area of the core, m^2	4.14
Nominal duration of FA stay in the core, fuel cycle	3–4
Fuel height in the core in cold state, m	3.53
Equivalent diameter of the core, m	3.16
Pitch between FAs, m	0.236
Number of fuel assemblies inside the core	163
Mass of fuel in fuel assembly, kg	484.8
Nominal loading of reactor on UO_2, kg	79,840
Diameter of the vessel cylindrical part in the core, m	4.535

4. The Strategy for Development of VVER-1000 Fuel Cycle

The most important objective of this work is to choose a fuel loading scheme which provides:

1. A homogenized power production over the core which implies “flatness of power or low Peak Power Factor.”
2. A low reactor vessel neutron fluence (basic requirement) [2,5].
3. The best value of fuel burn-up, which reflects good fuel utilization.
4. A low NPP generation cost indicated by:
   a) The ability to increase reactor cycle time. This helps to raise the value of plant Availability Factor, which is achieved by rising the initial reactivity of the core by raising fuel enrichment up to 5.2%.
   b) Increasing the number of batches, implying small batch size. This helps to achieve high burn-up fuel value and low plant fuel fabrication cost. The latter is implied by the smaller number of assembles (small batch size).

Very high average discharge burn-ups require high average ^{235}U enrichment in the initial core, and this, in turn, requires more ability of adjustment for the initial reactivity of the core by using additional amount of absorbers [11,17,18].

The Moderator Temperature Coefficient MTC requirement could be the most limiting constraint. It must be maintained negative in any Hot Full Power HFP condition in such longer cycle fuel management schemes. This is a consequence of high boron concentration required in long fuel cycles, and thus MTC becomes less negative with increasing boron. Therefore, in order to assure inherent safety, other types of Burnable Absorber BA, such as gadolinia, must be used to control the initial reactivity and the differential in reactivity between the fresh fuel and the partly burnt fuel in the core, leading to radial power peaking factors.

In order to achieve these goals by comparing various loading schemes, we developed a simulator for WWER 1000 reactor which performs the required calculations.

5. Choice of Burnable Absorber

The choice of burnable absorber bin characteristics has an important consideration for very high burn-ups [10,17,18]. Gadolinia was chosen for controlling radial power peaking. It is very important to carefully optimize the fuel and burnable poison loading patterns to minimize the initial enrichment. Therefore, the first simulation results obtained in this work are described in what follows.

Figure 3 depicts the result of studying the behavior of traditional fuel (e=4.4%) with different number of BA rods and different weight percent (w). As a result, we find that the following construction of the initial fuel is most suitable to start with:

1. The first (traditional) case (see Figure 3):
   a) The fuel bins: e =4.4%.
   b) The BA bins (Gd_2O_3):
      1. e =3.7%, w =4.0%.
      2. 18 fuel rod with BA.
   c) The Reactivity Equation which describes the average fuel burn up behavior in this case is:

$$\rho \approx \left| \begin{array}{ll} 0.165 & B \le 12.5 \\ 0.2451 - 0.0054B & B > 12.5 \end{array} \right.$$

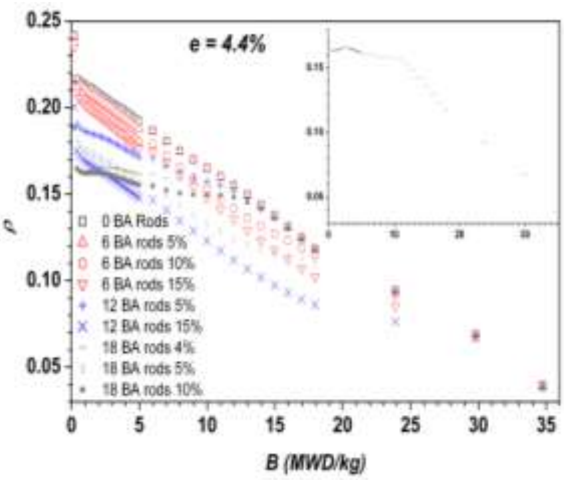

Figure 3. The behavior of variant traditional fuel constriction burn-up (e=4.4%). The inset illustrates the selected case for BA (see text).

While for fuel with enrichment e=5.2% suggested based on a study [17] presenting the relationship between the average discharge burn-up and the cycle length for different values of fresh fuel enrichment and different values of batch numbers, our study of fuel burn-up behavior with different number of BA rods and different weight percents implies the most suitable fuel construction as (see Figure 4):

2. The second (suggested) case:
   a) For the fuel rods: e =5.2%.
   b) For BA rods (Gd2O3)
      1. e =3.7%, w =4.0%.
      2. 30 fuel rods with BA.
   c) The Reactivity Equation which describes the average fuel burn up behavior in this case is:

$$\rho \approx \begin{cases} 0.178 & B \le 14 \\ 0.2555 - 0.0052B & B > 14 \end{cases}$$

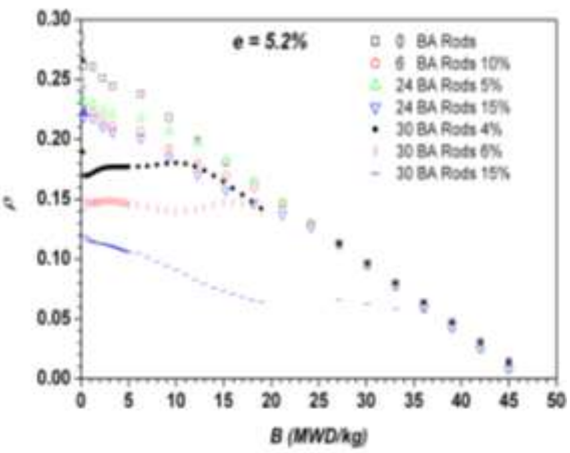

Figure 4: The behavior of variant fuel construction burn-up (e=5.2%).

It is worth noticing here that the load-up type symbol we adopt, like (1342), means the following: We divided the core into areas (Figure 5) of equal number of batches, and numbered areas from periphery toward center. The symbol is the numbers of areas from which fuel shaft on (fresh state until used state).

6. VVER -1000 Simulator and Simulation Results

In order to accomplish our objectives, the whole reactor core and peripheral component elements are modeled, in this work, using the MCNP4C2 and GETERA codes. Consequently, the neutronic parameters of the reactor core are calculated.

The simulator flowchart is depicted in Figure 6. The simulation procedures are based on:

1. The GETERA 90 code was used to perform the core burn-up calculations and the change in isotopic fuel components [19].
2. The Linear Reactivity model equations were programmed in this work in order to calculate burn-up characteristics for each FA according to its position in the reactor core [6,8,9].
3. The MCNP4C2 code was used to perform neutron flux and criticality calculation, in addition to peak power evaluation for each loading pattern [20,21].

As a conclusion from Section 5 above, the study considers two types of core fuel:

1. e = 4.4% + BA (Gd_2O_3, 18 rod, w = 4.0%);
2. e = 5.2% + BA (Gd_2O_3, 30 rod, w = 4.0%).

In order to validate our simulator, we compare the basic results for the traditional case (e=4.4) with referenced values before adopting results for the suggested case (e=5.2%). The comparative analysis presented in Table 2 provides confidence in our simulator and enables us to pursue the calculations.

For each loading scheme of the two cases, the simulator provides values of the peak power, burn-up, vessel neutron flux, K_{eff} and cycle time. The numerical values for these parameters are listed in Table 3 for the first case, and in Table 4 for the second.

The details as illustrated in the flowchart (Figure 6) are enormous, and in our assessment can't be contained in the text. Therefore, the detailed computational stages and analyses involved can be obtained directly from the authors.

Table 2. Comparative study for the traditional case (e = 4.4%).

Parameter	Simulator value		Referenced value		Relative Error (%)[a]
K_{eff} BOC	1.0529		1.05352	[14]	0.06
K_{eff} EOC	0.9828		0.969	[15]	1.42
Cycle Time [Month]	9.98		9.72	[22]	2.67
Peak Power	BOC=1.46 EOC=1.38	Ave=1.42	1.41	[16]	0.71
Peri. FA power decrease	0.26		0.25	[18]	4.00
NRV neutron flux decrease	0.37		0.31	[10,23]	19.4

[a] Calculated as

$$\left| \frac{\text{Simulator value - Reference value}}{\text{Reference value}} \right| \times 100$$

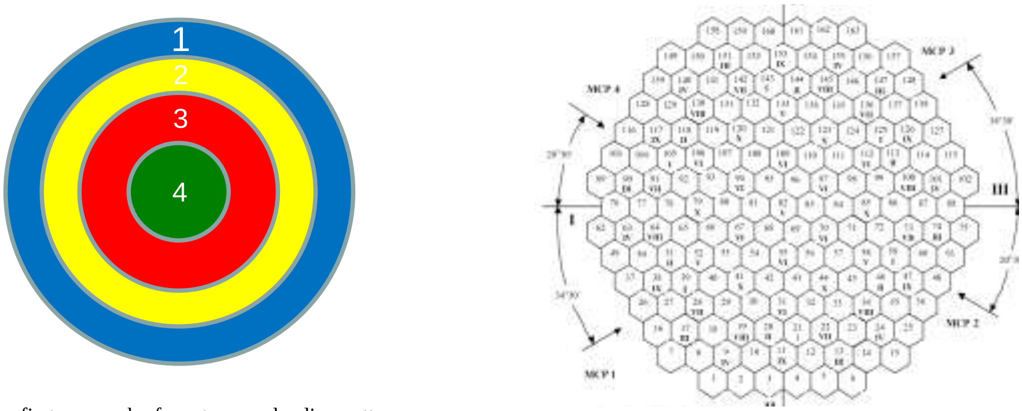

Figure 5. The first approach of reactor core loading pattern.

Table 3: Simulator results for the first case "e = 4.4% + 18 fuel rods with BA (Gd_2O_3)".

Pq fuel rod	Pq fuel AS	Vessel Neutron flux X 10^{11}	Burn-up Value				Cycle Time		K_{eff}		Load-up type →			NO. of batches
			1	2	3	4	day	month	K_{eff}	D K_{eff}				
1.82	1.18	1.9052	11.43	22.81	33.64	44.39	277.45	9.25	1.02	0.00			1234	
1.29	1.25	1.2530	11.49	22.93	34.83	44.67	279.194	9.31	1.01168	3E-04			1243	
1.67	1.28	1.5802	11.48	23.53	33.81	44.57	278.573	9.29	1.01902	2E-04			1324	
1.30	1.29	0.75523	11.57	23.71	35.59	45.03	281.424	9.38	1.02246	2E-04	BOC	OUT-IN	1342	
1.29		1.0926							0.96679	2E-04	EOC			
2.12	1.48	0.22402	11.60	25.02	35.29	45.22	282.648	9.42	1.02656	2E-04			1423	4
1.78	1.47	0.29544	11.63	25.07	35.90	45.36	283.499	9.45	1.04035	2E-04			1432	
1.48	1.46	0.61477	12.33	25.86	36.74	45.91	286.943	9.56	1.04497	2E-04			2431	
1.42	1.28	0.55916	12.27	24.49	36.43	45.59	284.94	9.60	1.04149	2E-04	BOC	LLLP	2341	
1.38		0.79066							0.9828	2E-04	EOC			
1.96	1.57	0.095717	14.69	27.08	37.64	47.02	293.88	9.80	1.08265	2E-04			4321	

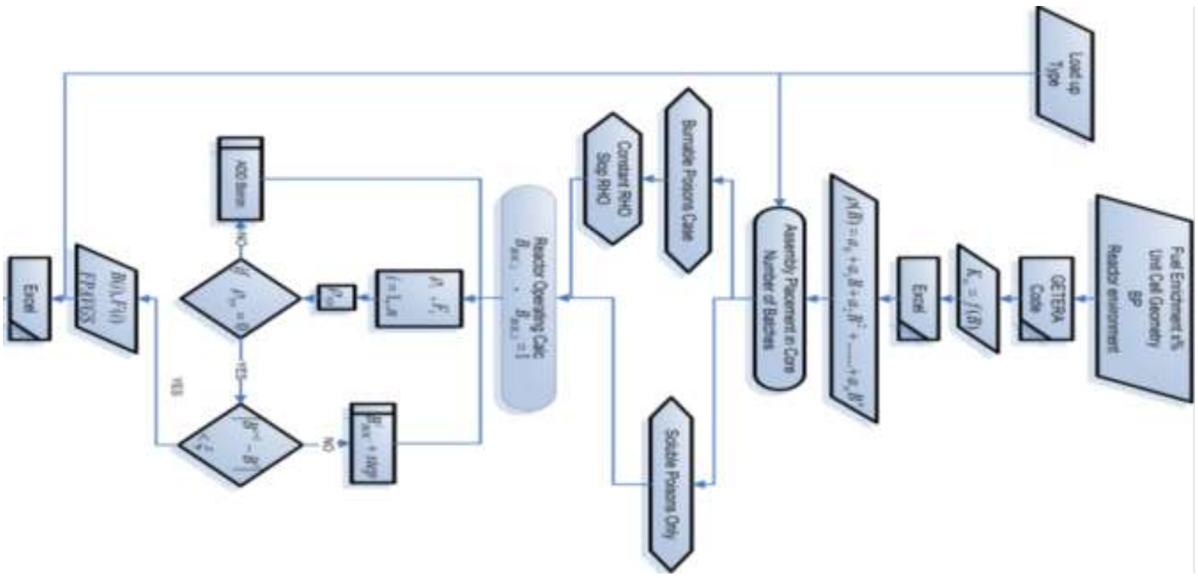

Figure 6. VVER-1000 simulator flow chart.

Table 4: The simulator results for the second case “e = 5.2% + 30 fuel rods with BA (Gd_2O_3)”.

Pq fuel rod	Pq fuel AS	Vessel Neutron flux X 10^{11}	Burn-up Value					Cycle Time		K_{eff}		Load-up type →		NO. of batches
			1	2	3	4	5	day	month	K_{eff}	D K_{eff}			
2.19	1.31	1.3441	13.88	27.38	39.75	51.58	63.06	373.71	12.46	1.02194	3E-04		12345	
1.95	1.31	1.7799	13.90	27.41	40.37	51.69	63.18	374.42	12.48	1.01479	2E-04		12435	
1.43	1.31	1.4336	13.93	27.49	40.49	52.97	63.41	375.75	12.52	1.0057	2E-04		12453	
1.88	1.31	1.0755	13.95	27.51	41.31	52.62	63.52	376.43	12.55	1.01767	2E-04		12534	
1.94	1.31	0.39094	13.97	27.54	41.36	53.14	63.61	376.92	12.56	1.02089	3E-04		12543	
2.62	1.33	1.2598	13.98	27.39	39.68	52.29	63.24	374.73	12.49	1.07186	3E-04		13254	
1.63	1.33	1.1751	13.99	27.85	40.87	53.36	63.69	377.40	12.58	1.01582	2E-04		13452	
1.45	1.32	0.68538	13.98	27.84	41.65	52.78	63.72	377.60	12.59	1.04211	2E-04	LLLP	13524	
1.36	1.39	1.2522	13.99	28.51	40.68	53.18	63.66	377.24	12.57	1.00683	3E-04	OUT-IN	14253	5
1.54	1.40	0.90246	14.00	28.55	40.95	53.46	63.81	378.14	12.60	1.01814	2E-04		14352	
1.72	1.39	0.47518	14.03	28.61	42.39	53.52	64.03	379.45	12.65	1.03427	3E-04		14523	
2.12	1.50	0.42833	14.05	29.61	41.76	53.56	64.06	379.63	12.65	1.04402	2E-04		15243	
2.13	1.51	0.28722	14.05	29.59	41.94	53.12	64.09	379.77	12.66	1.04019	2E-04		15324	
1.97	1.50	0.18540	14.08	29.66	42.57	53.72	64.24	380.67	12.69	1.05111	2E-04		15423	
2.75	1.39	3.4074	14.54	29.17	42.99	54.32	64.53	382.38	12.75	1.08228	3E-04		24531	
3.43	1.50	3.5376	14.57	30.18	42.58	54.42	64.63	382.99	12.77	1.10407	3E-04		25341	
3.21	1.50	3.5743	14.59	30.22	43.18	54.52	64.74	383.63	12.79	1.09816	3E-04		25431	
2.75	1.61	2.8616	16.91	31.61	44.19	55.55	65.93	390.67	13.02	1.18591	3E-04		54321	

7. The Load-Up Schemes

First case e=4.4 %

By comparing the different fuel load-up schemes in terms of the radial power peaking factor Pq and reactor vessel RV neutron flux, we distinguish the “1342” OUT-IN fuel loading type, and the “2341” Low Leakage Loading Pattern LLLP scheme, as illustrated in Figure 7. This choice satisfies the lowest reactor vessel neutron fluence for the LLLP while keeping acceptable increase of Pq value.

The numerical characteristics of the “1342” OUT-IN and the “2341” LLLP schemes are listed in Table 5. We presumed a decrease of the NRV fluence by about 30% using LLLP instead of OUT-IN scheme, which could comprise a positive consequence on NRV life time. The simulator results provided a good uniformity with referenced values as shown in Table 5. This positive outcome furnishes the ground for exploring new cases using the simulator.

The thermal, the up thermal and the fast neutron flux distributions for the LLLP scheme extracted via the simulator over the reactor components are illustrated in Figure 8.

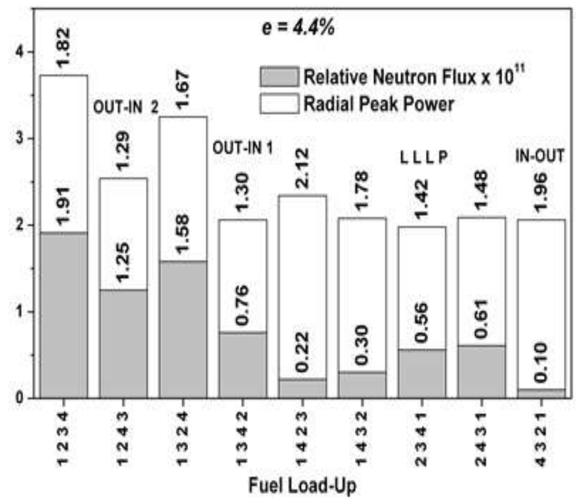

Figure 7. Comparison of different fuel load-up schemes according to Pq and RV neutron flux (e=4.4 %).

Figure 8 depicts the thermal, up thermal and fast neutron flux distributions over the core for LLLP “2341”.

Table 5. The "1342" OUT-IN scheme & the "2341" LLLP scheme characteristics for the first case (e=4.4 %).

NRV neutron flux decrease	FA power decrease	Fluence decrease	Fluence	Peak Power	Cycle Time MONTH	K_{eff}	Load-up Type →		
0.37	0.26	0.31	3.6E+18	1.24	9.80	1.04948	BOC	OUT-IN	1342
				1.29		0.96679	EOC		
			2.4E+18	1.46	9.98	1.0529	BOC	LLLP	2341
				1.38		0.9828	EOC		
0.31	0.25	0.3	7.27E+19	1.41	9.72	1.05352 [14]	Referenced value		
[10] [23]	[1]	[10] [23]	[24]	[16]	[22]	0.969 [15]			

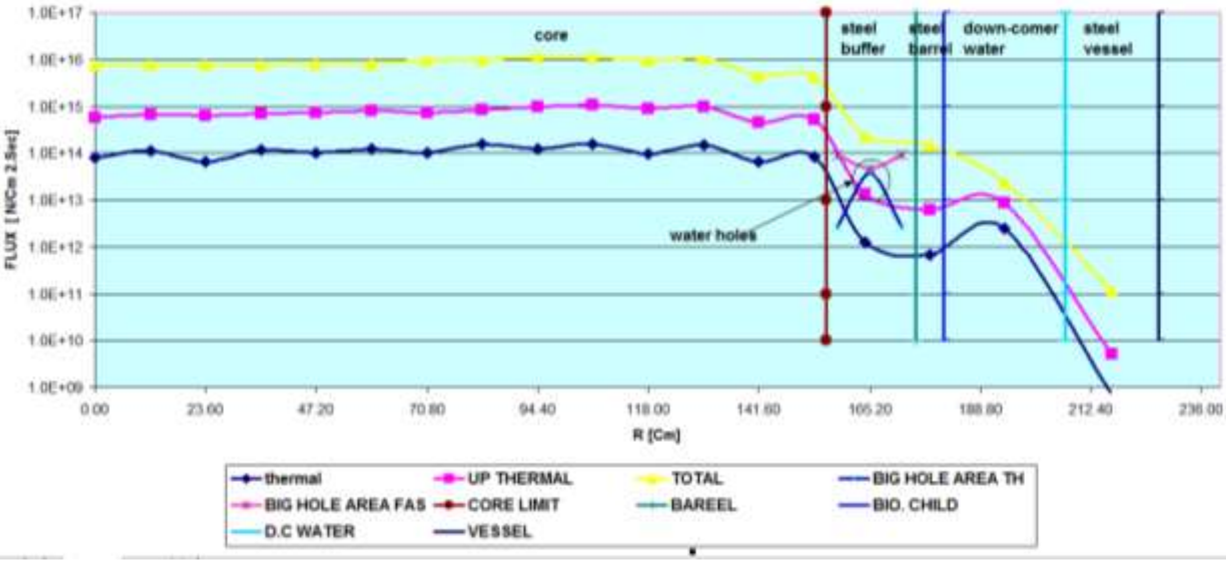

Figure 8. The thermal, up thermal and fast neutron flux distributions over the core for e=4.4% and LLLP "2341".

Second case e=5.2%

Following the same argument for the proposed case (e=5.2%), we distinguish the "13524" LLLP scheme, with the best ($0.69x10^{11}$ n/cm^2.s) flux and a corresponding Pq value of 1.45 (Figure 9).

Figure 10 depicts the thermal, up thermal and fast neutron flux distributions over the reactor for the selected scheme; namely LLLP "13524".

Finally, it is worth mentioning that Tallies F4, *F4, F7 and *F7 were used to calculate neutron flux densities in different core cells and power in fuel cells. The obtained power distributions were used to obtain radial peak power (Pq). The RV neutron flux was obtained using F2 tally for internal reactor vessel surface for fast neutrons (> 0.5 MeV). The results were used to get a sectional distribution of neutron flux through the core and the surrounding components as shown in Figures 8 and 10.

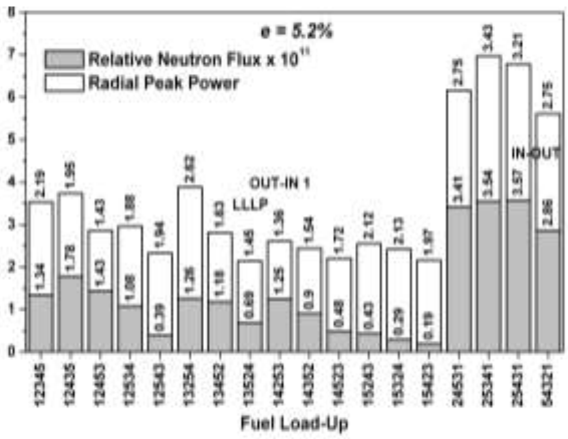

Figure 9: Comparison of different fuel load-up schemes according to Pq and RV neutron flux (e=5.2 %).

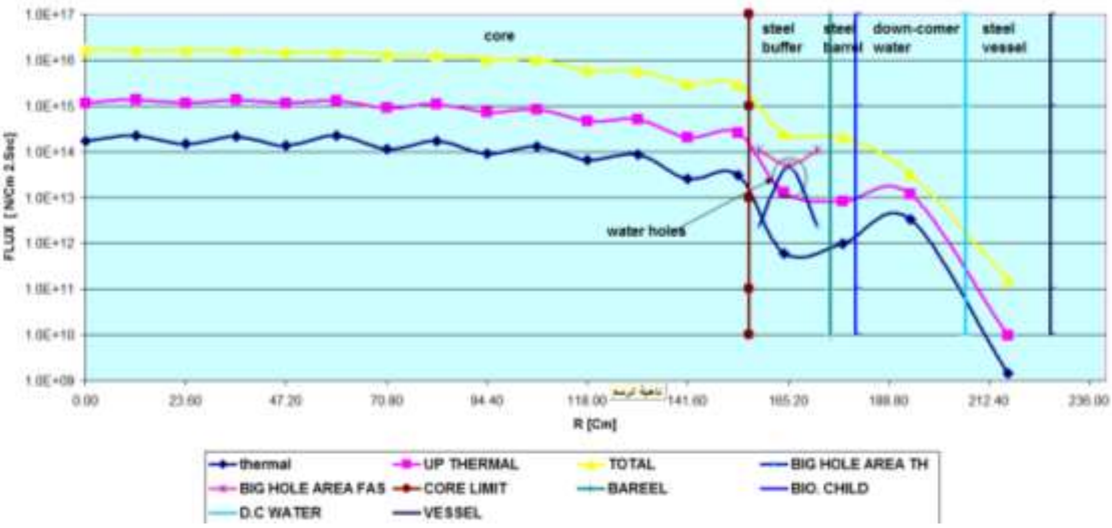

Figure10: The thermal, up thermal and fast neutron flux distributions over the core for e=5.2% and LLLP ”13524”.

8. Conclusion

In this study, a method for the first approach of optimizing the VVER-1000 nuclear reactor loading pattern was introduced. Using this method, we evaluated the pressure vessel integrity by comparing the vessel neutron fluence for each fuel loading scheme. A simulator was developed to calculate the basic characteristics (peak power, vessel fluence and average burn-up value). The most significant results were:

- For the first standard case e =4.4%:

The use of LLLP scheme instead of OUT-IN load pattern permitted to reduce the reactor vessel neutron fluence by 31%, and the reactor vessel neutron flux by 26%. On the other hand, the LLLP scheme increased the batch time by 2.3% comparing to the OUT-IN load pattern. One inconvenience of the LLLP scheme involved the increase of Pq by about 9.23% (from 1.3 to 1.42) compared to literature values of Pq=1.41 and Pq critical=1.55 [4,8,14].

- For the second proposed case e =5.2%:

The LLLP (5 batches + e=5.2%) allowed to lower the reactor vessel neutron flux by 9.3%, to increase the batch time by 19.3%, and to raise the average burn-up value by 41.5% (up to 63.73). In addition, the reactor spent fuel production rate is reduced from 41 assemblies by 9.4 months, to 33 assemblies by 11.2 months.

Furthermore, the Pq value for the selected LLLP scheme increased by about 11.5% (from 1.3 to 1.45) which is still beneath the critical value of 1.55.

References

[1] B. Boehmer, J. Konheiser, K. Noack, A. Rogov, G. Borodkin, E. Polke, P. Vladimirov, “Neutron and gamma fluence and radiation damage parameters of ex-core components of Russian and German light water reactors”. Proceedings of the 11th International Symposium on Reactor Dosimetry, 18-23 August 2002 in Brussels, Belgium. World Scientific Publishing Co. ISBN #9789812705563; 2003, 286-294.

[2] O.Ö. Gülol, Ü. Çolak, “Comparison of pressure vessel integrity analyses and approaches for VVER 1000 and PWR vessels for PTS conditions”. Nuclear Engineering and Design, Vol. 226 (2003) 231–241.

[3] A. Kryukov, D. Erak, L. Debarberis, F. Sevini, B. Acosta, “Extended analysis of VVER-1000 surveillance data". International Journal of Pressure Vessels and Piping, Vol. 79 (2002) 661–664.

[4] J.J. Duderstadt, L.J. Hamilton. Nuclear Reactor Analysis. 1st edition. Wiley; 1976.

[5] G.L. Guthrie, “Preliminary study of the use of fuel management techniques for slowing pressure vessel embrittlement”. 4th International ASTM-EURATOM Symposium on Reactor Dosimetry, Gaithersburg, MD, USA, 22 Mar 1982, 111–120.

[6] C.W. Bagnal, G.P. Cavanaugh, L.B. Tarko. Survey of vessel fluence reduction techniques. Palo Alto, California, Electric Power Research Institute EPRI; 1984.

[7] A.H. Fadaei, N.M. Moghaddam, E. Zahedinejad, M.M. Fadaei, S. Kia, "Fuel management optimization based on power profile by Cellular Automata". Annals of Nuclear Energy, Vol. 37 (2010) 1712–1722.

[8] Sh. Jiang. Nuclear fuel management optimization using estimation of distribution algorithms. Ph.D. Thesis, Imperial College, University of London, UK; 2009.

[9] E. de Klerk, C. Roos, T. Terlaky, T. Illés, A.J. de Jong, J. Valkó, J.E. Hoogenboom, “Optimization of nuclear reactor reloading patterns". Annals of Operations Research, Vol. 69 (1997) 65–84.

[10] B.D. Murphy, J. Kravchenko, A. Lazarenko, A. Pavlovitchev, V. Sidorenko, A. Chetverikov. Simulation of Low-Enriched Uranium (LEU) Burnup in Russian VVER reactors with the HELIOS code package. Oak Ridge National Laboratory, Oak Ridge, Tennessee. Report ORNL/TM-1999/168; 2000.

[11] I. Panayotov, N. Mihaylov, K. Ilieva, D. Kirilova, M. Manolova, "Burnup influence on the VVER 1000 reactor vessel neutron fluence evaluation". 19th Symposium on AER for VVER Reactor Physics and Reactor Safety, St. Constantine, International House of Scientists, Bulgaria, 2009.

[12] IAEA. WWER-1000 Reactor Simulator. Training Course Series No. 21, Second Edition, IAEA Vienna, ISSN 1018–5518; 2003.

[13] S. Kaichao. MCNP modeling of hexagon VVER fuel. M.Sc. Thesis, Department of Reactor Physics, Royal Institute of Technology, Stockholm, Sweden; 2008.

[14] G. Todorova, N. Petrov, N.P. Kolev, P. Bellier, "2D Core calculations for VVER 1000 with APOLLO2". International Conference on the Physics of Reactors, Nuclear Power: A Sustainable Resource, Interlaken, Switzerland, September 14-19, 2008.

[15] G. Todorova, N. Petrov, N.P. Kolev, F-X. Hugot, "2D Core Solutions for VVER 1000 with APOLLO2 and TRIPOLI4". 19th Symposium on AER for VVER Reactor Physics and Reactor Safety, St. Constantine, International House of Scientists, Bulgaria, 2009.

[16] J. Lee, N.Z. Cho, "AFEN method and its solutions of hexagonal three-dimensional VVER-1000 benchmark problem". Progress in Nuclear Energy, Vol. 48 (2006) 880-890.

[17] Nuclear Energy Agency. Very high burn-ups in light water reactors. NEA No. 6224, ISBN 92-64-02303-8, Paris, France; 2006. <http://www.oecd-nea.org/science/pubs/2006/nea6224-burn-up.pdf>

[18] P.N. Alekseev, A.V. D'yakov, A.S. Kolokol, A.A. Proshkin, A.L. Shimkevich, "Improvement of the operating characteristics of VVER oxide fuel". Atomic Energy, Vol. 102 (2007) No. 2, 129-134.

[19] N. Belousov, S. Bychkov, Yu. Marchuk *et al.*, "The Code GETERA for cell and polycell calculations: Models and Capabilities". Proceedings of the 1992 Topical Meeting on Advances in Reactor Physics, Charleston, SC, USA, 8–11 March, 1992, Vol. 2 (1992) 516–523.

[20] J.F. Briesmeister (Ed.). MCNP – A general Monte Carlo N-Particle transport code (Version 4C2). RSICC# CCC-700, Oak Ridge National Laboratory; 2000.

[21] J.S. Hendricks. MCNP-4C manual. University of California, Los Alamos National Laboratory, U.S. Department of Energy; 2000.

[22] P.V. Petkov, D.V. Hristov, "VVER-1000/V320 decay heat analysis involving TVS-M and TVSA fuel assemblies". Nuclear Engineering and Design, Vol. 238 (2008) 3227-3239.

[23] D. Fraklin, T. Marston. Investigation the flux reduction option in reactor pressure vessel integrity. Palo Alto, California, Electric Power Research Institute EPRI; 1983.

[24] J. Böhmert , H-W. Viehrig, H. Richter. Irradiation response of VVER pressure vessel steels: First results of the Rheinsberg irradiation programme. Wissenschaftlich-Technische Berichte, Forschungszentrum Rossendorf, FZR-284 (2000) 71-76.