HPLWR EQUILIBRIUM CORE DESIGN WITH THE KARATE CODE SYSTEM

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Abstract

The High Performance Light Water Reactor (HPLWR) is the European version of the various supercritical water cooled reactor proposals. The paper presents the activity of KFKI-AEKI in the field of neutronic core design within the framework of the "HPLWR Phase 2" FP-6 and the Hungarian “NUKENERG” projects. As the coolant density along the axial direction shows remarkable change, coupled neutronic- thermohydraulic calculations are essential which take into account the heating of moderator in the special water rods of the assemblies. A parametrized diffusion cross section library was prepared for the HPLWR assembly with the MULTICELL neutronic transport code. The parametrized cross sections are used by the KARATE program system, which was verified for supercritical conditions by comparative Monte Carlo calculations. To design the HPLWR equilibrium core preliminary loadings were assessed, which contain insulated assemblies with Gd burnable absorbers. The fuel assemblies have radial and axial enrichment zoning to reduce hot spots.

1. Introduction

As the target average outlet temperature (500 C) and the maximum foreseen cladding temperature (~630 C) are very close to each other, the 3-pass core concept was proposed [1]. The hot spots in this proposal can be potentially eliminated by multiple flow of coolant through the active core with mixing after each passing. First the ascending coolant is heated in the central evaporator region of the core passing through the pseudocritical point, then after mixing in the upper mixing chamber flows downwards in the first superheater region. After leaving the lower mixing chamber, the coolant flows again upwards in the second superheater region located at the core periphery.

2. Testing the KARATE code system by the HPLWR Neutronic Benchmark

The KFKI Atomic Energy Research Institute uses the KARATE code system for core design. As the original KARATE [2,3] which is used for the calculation of VVER-440 reactors worked only in hexagonal geometry, we had to improve our code system to be able to treat the square assemblies planned in the HPLWR Phase 2 project. The improvement concerns both the MULTICELL neutronic transport code and the GLOBUS nodal code of the KARATE system. The more heterogeneous fuel assembly structure than usual, the smaller node size than usual, the steep axial and radial water density variation and last but not least the lack of experimental results require a thorough verification with the help of a Monte Carlo code. Realistic full core calculation of the HPLWR three pass core at prescribed thermo-hydraulic distribution with reflector was carried out with the help of the MCNP4C Monte Carlo code [4]. The arrangement of the HPLWR core is as follows.

In the HPLWR reactor core the basic unit is a closed fuel assembly with thermal insulation comprising of 40 fuel rods. Inside the assembly a moderator channel is applied which together with the assembly gap regions serves as moderator for the fuel pins in the tight lattice. One cluster (see Figure 1), the basic element of fuel shuffling consists of 9 assemblies with enrichment zoning. The control rods can be inserted into five of nine assemblies. The reactor core having 45 degree symmetry consists of 156 clusters surrounded by reflectors.

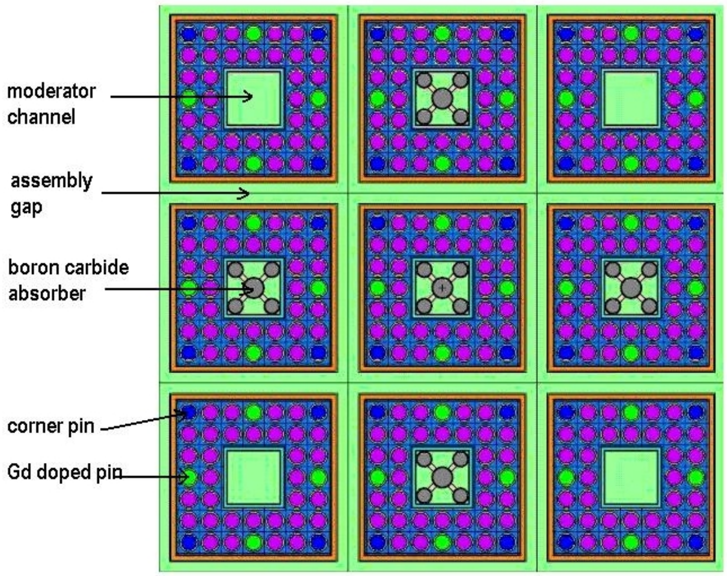

Figure 1 The MCNP model of the 9 assembly cluster with inserted control rods.

In order to avoid the possible buoyancy effects in the assembly a new flow path was proposed [5]. The new flow path of descending moderator flow, ascending gap flow and descending reflector water flow with mixing with the downcomer flow has been modeled in the KARATE code system at beginning of life, full power conditions. The calculated water density distributions served as a basis for defining the coolant, moderator channel and gap water densities in the benchmark (See Figures 2-4).

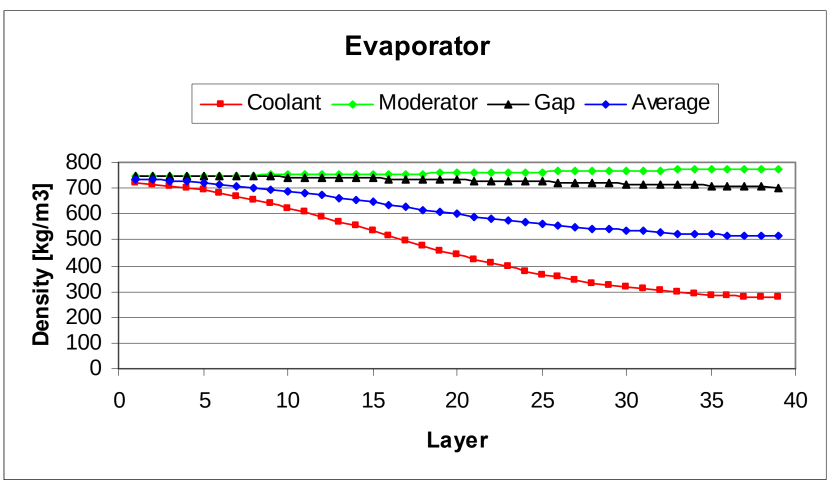

Figure 2 Prescribed water density distributions for the evaporator.

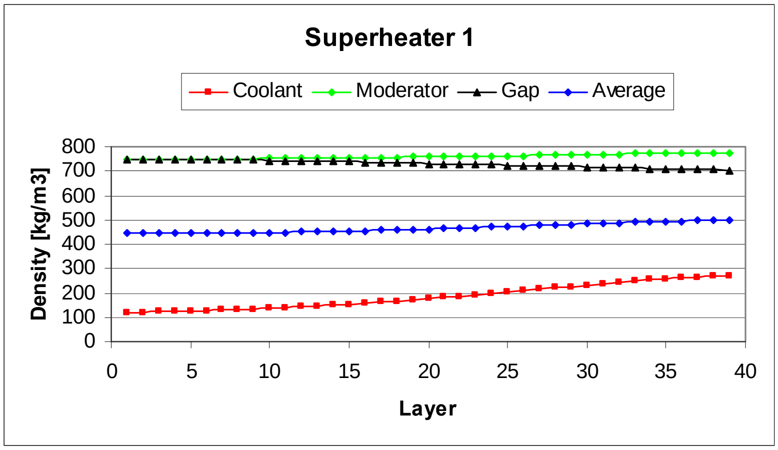

Figure 3 Prescribed water density distributions for superheater 1.

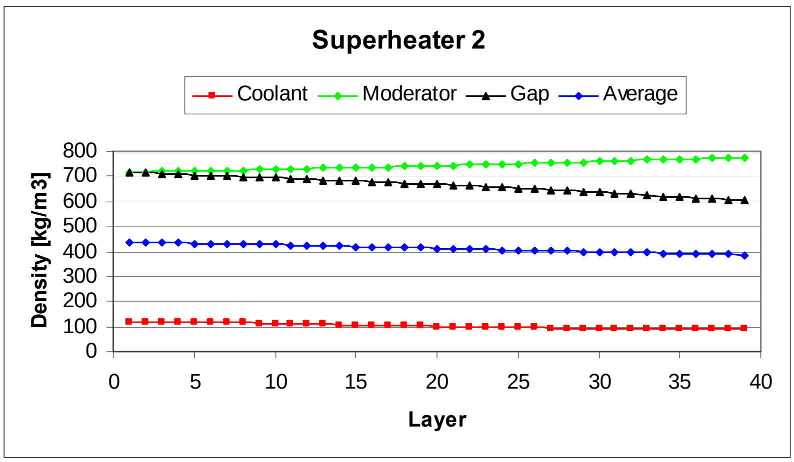

Figure 4 Prescribed water density distributions for superheater 2.

The core main data can be found in [6]. The definition of cluster types in the test can be seen in Table 1.

Table 1 Definition of cluster types in the test.

Cluster type	Basic enrichment [w/o]	Corner rod enrichment [w/o]	Gd doped rod enrichment [w/o]	No. of Gd doped rods	Gd_2O_3 conten t [w/o]
1	4.0	3.0	-	-	-
2	5.0	4.0	-	-	-
3	6.0	5.0	5.5	4	2.0
4	7.0	6.0	6.5	4	2.0
5	3.0	2.0	-	-	-

The core map of cluster types is presented in Figure 5 for the quarter of the core. The central evaporator, the first superheater and the second superheater at the core periphery are separated by black lines.

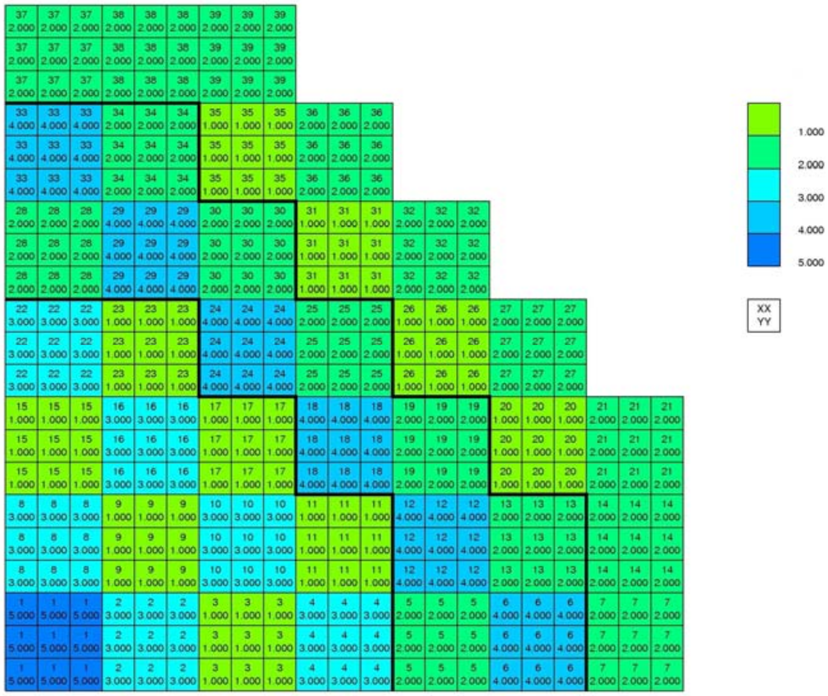

Figure 5 The core map of the test case. XX: Cluster sequence number. YY: Cluster type.

The rod positions in the three defined cases can be found in Table 2. The rod insertion is illustrated in Figure 6.

Table 2
Rod positions in the test.

Cluster	Case A Rod position [cm]	Case B Rod position [cm]	Case C Rod position [cm]
5,28	420.	0.	0.
11,23	420.	0.	0.
9	420.	0.	0.
18,24	420.	0.	210.
Rest	420.	420.	420.

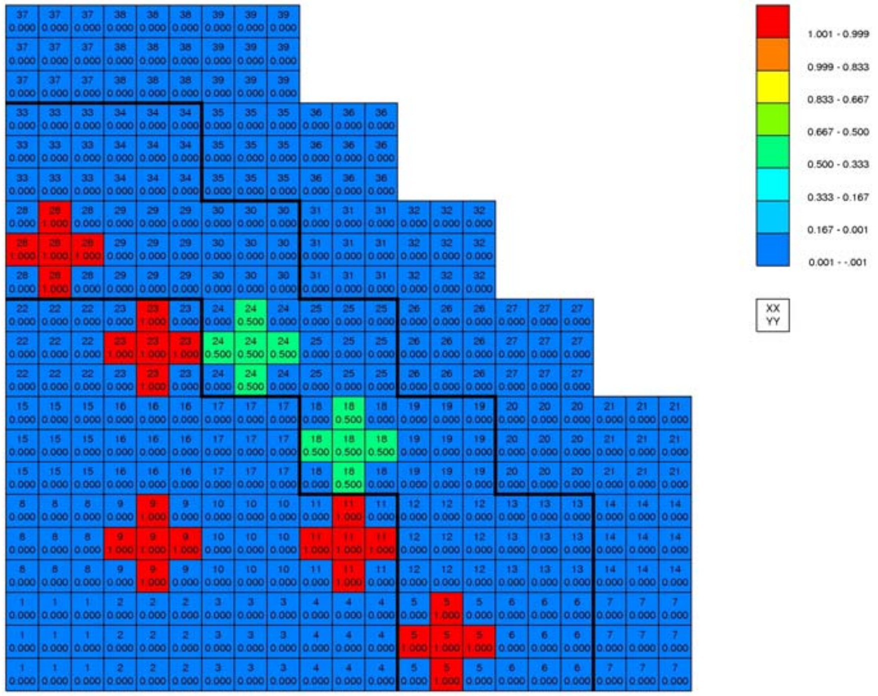

Figure 6 Control rod positions for case C. XX: Cluster sequence number. YY: Control rod insertion fraction (0: out, 1: in)

The MCNP model of the radial reflector can be seen in Figure 7.

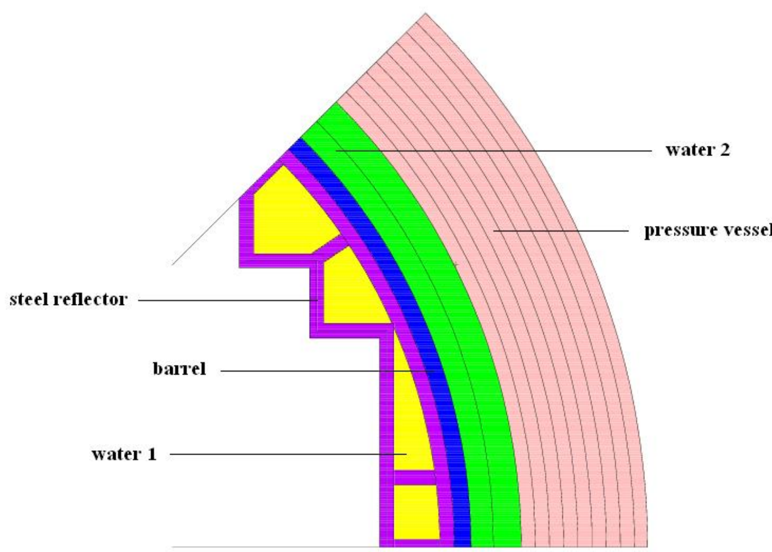

Figure 7 MCNP model of the HPLWR radial reflector.

The k_{eff} and 3D power distribution was calculated with the MCNP and GLOBUS codes. The reflector is modeled in detail in the MCNP calculations, while in the GLOBUS code albedo boundary conditions were used. The criticality results of the benchmark can be found in Table 3.

Table 3
The criticality results of the HPLWR full core benchmark

Test case	MCNP k_{eff}	MCNP St. dev.	GLOBUS k_{eff}	Reactivity difference
A	1.07003	0.00005	1.06486	-0.00454
B	1.02285	0.00005	1.01859	-0.00409
C	1.03199	0.00005	1.02629	-0.00538

The frequency of GLOBUS-MCNP nodal power distribution differences relative to the maximum can be seen in Figure 8 for all the test cases. On the basis of this figure an engineering safety factor of 1.09 was chosen which covers the hot spot power calculation uncertainty.

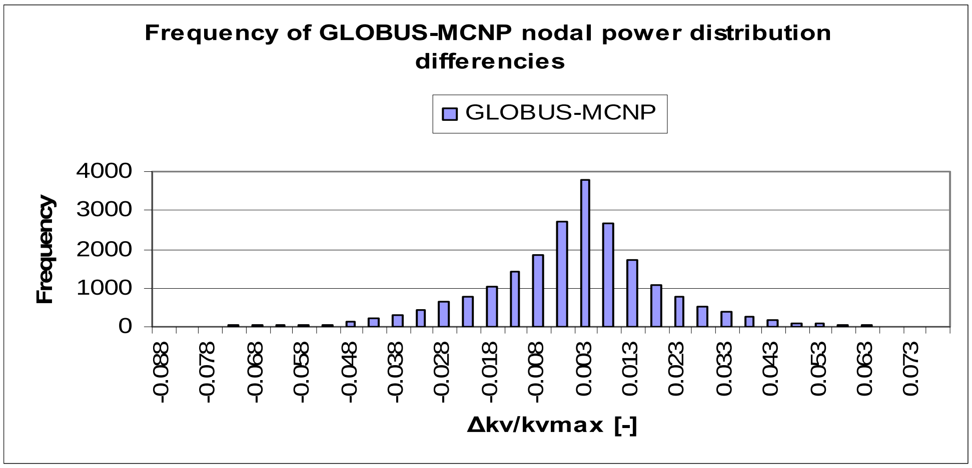

Figure 8 GLOBUS-MCNP nodal power distribution differencies relative to the maximum.

As there are no detailed pin wise calculations in the supercritical version of KARATE, the maximum intra assembly pin wise peaking factors were evaluated from this test for all the three flow regions (See Table 4).

Table 4
Maximum intra assembly pin wise peaking factors for the flow regions.

	Max (k_k) for cases A, B and C
Evaporator	1.22
Superheater 1	1.28
Superheater 2	1.30

3. Cross Section Generation for HPLWR

The KARATE code system applies the GLOBUS nodal code which uses the parametrized 2-group cross sections generated by the MULTICELL neutronic transport code. First burnup calculations were carried out at fixed, average technological parameters for assemblies with and without absorber cluster. At certain burnup points branch calculations were performed at wide range parameter combinations. After fitting the coefficients of cross section formulas the database of coefficients was prepared. The evaluation of the cross sections at the necessary parameters is performed during the nodal calculations. The interpolation of cross sections between the inserted rod and withdrawn rod cases is linear. The cross section dependence is as follows:

$$\Sigma\left(Bu, {}^{235}U, {}^{238}U, {}^{239}Pu, {}^{135}Xe, {}^{149}Sm, \rho_c, \rho_m, \rho_g, T_f, F\right)$$

, where
Bu: Burnup
^{235}U: ^{235}U concentration (optional)
^{238}U: ^{238}U concentration (optional)
^{239}Pu: ^{239}Pu concentration (optional)
^{135}Xe: ^{135}Xe concentration
^{149}Sm: ^{149}Sm concentration
ρ_c: Coolant density
ρ_m: Moderator rod water density
ρ_g: Gap water density
T_f: Fuel temperature
F: Absorber cluster insertion factor

The cross section methodology for HPLWR has been tested in the following way: Hypothetic fuel history calculations including water density, power density changes and control rod insertion for the bottom, middle and top slice of fuel assemblies were carried out in several cases. The reference is the MULTICELL transport code. The burnup history calculations with the help of the parametrized cross sections have been imitated using the MONOKLI code. The MONOKLI code calculates the same fuel history as the MULTICELL code but uses only the 2-group parametrized cross sections.

The MONOKLI code has two options:

- The concentrations of actinides ^{235}U, ^{238}U and ^{239}Pu are calculated directly by applying their microscopic cross sections.
- The concentrations of actinides ^{235}U, ^{238}U and ^{239}Pu are not calculated directly.

The MONOKLI code evaluates the k_{eff} with the parametrized cross sections using the buckling from the MULTICELL criticality calculation. The deviation of the k_{eff} of MONOKLI from one characterizes the accuracy of cross section parametrization. k_{eff} is presented in Figure 9 at irradiation history parameters of the middle of core height. Figure 9 shows the superiority of the first option, so the explicit calculation of the above actinides will be applied.

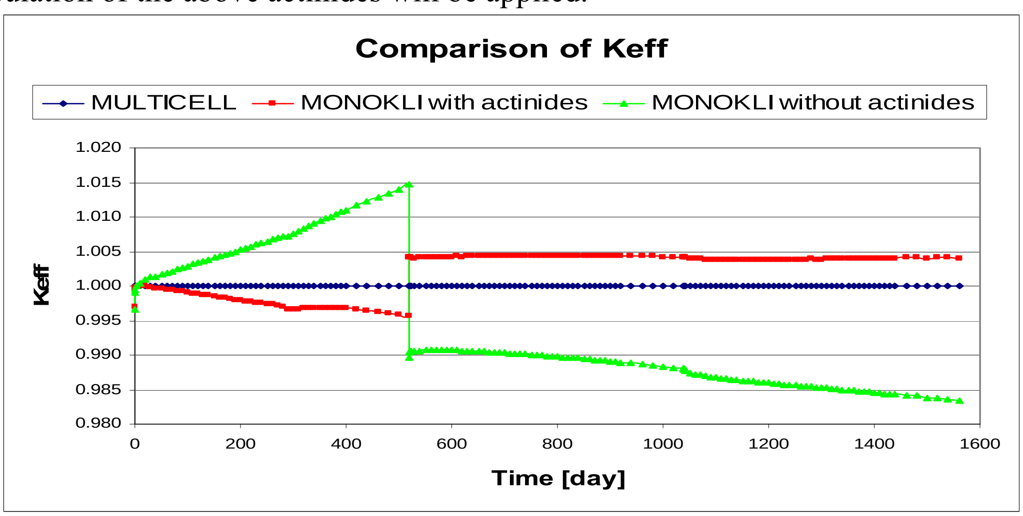

Figure 9 Comparison of few-group parametrization accuracy at the middle height for three cycles.

4. Equilibrium core design with the KARATE code system

In the HPLWR reactor the thermal properties show steep variation near the pseudocritical point. The coupled neutronic-thermohydraulic calculation is essential for determining the proper criticality and power distribution data. For thermohydraulic calculations the modified SPROD code [7] of the Tokyo University using the Watt and Jackson-Hall heat transfer correlations [8,9] was applied. In the parallel channel SPROD code the IAPWS-IF97 water property functions are used. To control the maximum cladding temperatures cluster orifices were applied to tune the mass flow distributions in the three flow regions of the core. As a consequence of the application of burnable absorbers considerable power distribution change can be observed in the HPLWR core which results in the mass flow redistribution among the assemblies affecting the cladding temperatures. To take into account this redistribution the pressure drop calculation of the coolant in the core was implemented which includes the local pressure losses and the distributed pressure loss along the core. The distributed pressure loss of Rehme [10] was applied for the assembly with wire spacers. During the equilibrium core calculations at full power conditions equilibrium Xe was reached.

The main parameters of the calculation:
- Nominal pressure=25. MPa
- Total power=2.3 GW
- Flow fraction to downcomer=0.50
- Flow fraction to moderator=0.50
- Total mass flow=1179. kg/s
- Pressure vessel inlet temperature=280. C
- Core outlet temperature=502. C

Starting from the first loading of the HPLWR, equilibrium cycle was reached, which contains assemblies with Gd integrated poison. Applying 6.7 % average enrichment and reloading one third of the fuel clusters, 355 day cycle length was achieved. The following main considerations were taken into account in the equilibrium cycle design at normal operation conditions:
- Keeping the linear power limit of 390 W/cm
- Keeping well below the fuel centerline temperature under the melting point
- Keeping the maximum cladding temperature below 630 C
- Achieving as high discharge burnup as possible
- Compensation of excess reactivity mainly by Gd burnable poison
- Burnup of Gd burnable poison of 1^{st} cycle fuel clusters for achieving longer cycle length

The definition of cluster types and the age of clusters used in the equilibrium cycle calculations is presented in [11]. On the basis of previous calculations with flat axial enrichment distribution resulting in bottom peaked power, axial enrichment profile has been introduced. Concerning the assembly-wise radial burnup distribution at the core mid-plane in EOC state, it is worth recognizing the orientation of the clusters: the fuel assemblies in superheater 1 and 2 near to the flow region boundaries with higher water densities have the highest burnup in their clusters [11]. The assembly-wise radial power distributions normalized to the entire core show considerable power redistribution between BOC and EOC states. In spite of the advantageous burnup orientation of the clusters mentioned earlier, the assembly-wise peaking factors normalized to flow regions in BOC and EOC states are quite high for superheater 1 and 2, exceeding the design expectations [1].

To reduce the maximum cladding temperatures in the hot spots of superheaters 1 and 2 cluster orifices were applied at the inlet of some clusters. With the application of this by far not optimized orifice distribution a 20 C degree of node average cladding temperature reduction was achieved, but according to an estimation based on additional subchannel analyses, taking into account fuel assembly bending, not perfect mixing in mixing chambers, assigning uncertainties to subchannel codes and heat transfer, the cladding temperature limit can not be kept yet [11]. The average linear power generation axial redistribution from BOC to EOC state in the flow regions can be seen in Figure 10.

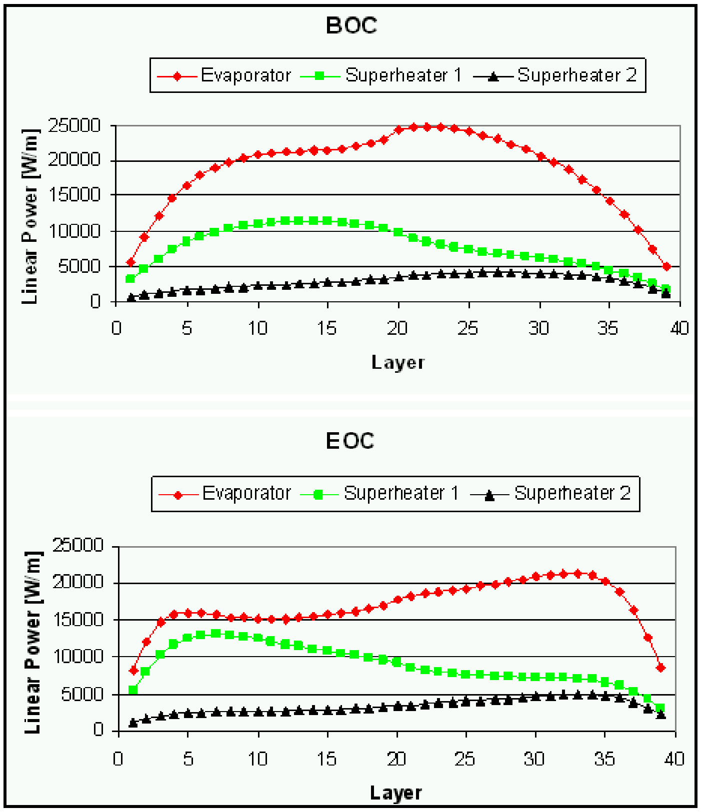

Figure 10 Average linear power generation in the flow regions. Axial redistribution from BOC to EOC.

The average discharge burnup of the core is rather low, 32.5 [MWd/kgU] , which can be accounted for the use of more absorbing steel materials instead of the zircalloy and for the application of more structural materials than in conventional water cooled reactors. Table 5 summarizes the linear power hot spot data at BOC and EOC conditions.

Table 5
Equilibrium core calculational results.

	BOC	EOC
k_v	2.8186	2.3546
k_k	1.22	1.22
P_{lin}^{av} [W/cm]	97.5	97.5
f_{eng}	1.09	1.09
$P_{lin}^{hot} = P_{lin}^{av} \cdot k_v \cdot k_k \cdot f_{eng}$ [W/cm]	365.4	305.3
$T_{fuel,centerline}^{hot}$ [K]	2390.	2156.

The denominations in the table are as follows:

BOC	Beginning of cycle
EOC	End of cycle
k_v	Volumetric peaking factor
k_k	Pin-wise peaking factor
P_{lin}^{av}	Average linear power density
f_{eng}	Engineering factor of linear power density
$T_{fuel,centerline}^{hot}$	Fuel centerline temperature in the hot spot

The BOC maximum pin power $P_{lin}^{av} \bullet k_v \bullet k_k \cdot f_{eng}$ value is 365.4 [W/cm], which is rather close to the 390 W/cm limit. Only 6% margin for normal transients is reserved.

5. Conclusions

Fine mesh calculations are necessary to follow up the fuel rod linear power evolution during the whole cycle. At the studied cases the cladding temperatures are determined only for the most loaded fuel nodes. Taking into account the intra assembly power peaking, the maximum cladding temperature would be higher, but the applied spiral wrap enhances mixing, which alleviates the consequences, so subchannel analysis is also needed for further assessments. By introducing newer assembly types and applying self-consistent assemblywise orificing more optimal core loading can be potentially achieved for reducing the calculated clad surface temperatures and linear powers.

6. Acknowledgements

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7. References

[1] T. Schulenberg, J. Starflinger and J. Heinecke, "Three pass core design proposal for a high performance light water reactor", *Progress in Nuclear Energy* 50**,** 2008, pp. 526-531.

[2] J. Gadó, Cs. Hegedűs, Gy. Hegyi, A. Keresztúri, M. Makai, Cs. Maráczy and M. Telbisz, "KARATE - A code for VVER-440 core calculation", Transactions of ANS Winter Meeting, Washington DC, 13-17 November, 1994

[3] Cs. Hegedűs, Gy. Hegyi, G. Hordósy, A. Keresztúri, M. Makai, Cs. Maráczy, F. Telbisz, E. Temesvári and P.Vértes, "The KARATE program system", PHYSOR 2002, Seoul, Korea, 7-10 October, 2002

[4] J. F. Briesmeister, "MCNP™–A general Monte Carlo n–particle transport code, Version 4C", LA-13709-M, Los Alamos National Laboratory, December, 2000

[5] C. Kunik, B. Vogt and T. Schulenberg, "Flow phenomena in the gap volume between assembly boxes", 4th International Symposium on Supercritical Water-Cooled Reactors, Paper No. 04, Heidelberg, Germany, 8-11 March, 2009

[6] Cs. Maráczy, A. Keresztúri, I. Trosztel and Gy. Hegyi, "Safety analysis of reactivity initiated accidents in a HPLWR reactor by the coupled ATHLET-KIKO3D code", *Progress in Nuclear Energy* 52, 2009, pp. 190–196.

[7] A. Yamaji, Y. Oka and S. Koshizuka, "Three-dimensional Core Design of SCLWR-H with Neutronic and Thermal-hydraulic Coupling", GLOBAL 2003, New Orleans, USA, November, 2003

[8] A. Yamaji, T. Tanabe, Y. Oka, J. Yang, J. Liu, Y. Ishiwatari and S. Koshizuka, "Evaluation of the nominal peak cladding surface temperature of the Super LWR with subchannel analyses", Proceedings of GLOBAL 2005, Tsukuba, Japan, 9-13 October, 2005

[9] J. D. Jackson and W. B. Hall, "Forced convection heat transfer to fluids at supercritical pressure", Turbulent Forced Convection in Channels and Bundles edited by S. K a k a c and D. B. Spalding, Vo1.2, Hemisphere Pub., 1979, p. 563.

[10] K. Rehme, "Pressure drop correlations for fuel element spacers", *Nucl. Techn.*, 17, 1973, pp. 15-23.

[11] T. Schulenberg, Cs. Maráczy, W. Bernnat and J. Starflinger, "Assessment of the HPLWR thermal core design", The 5th International Symposium on SCWR, Paper 057, Vancouver, British Columbia, Canada, 13-16 March, 2011