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Development of a computer code system for the analysis of prism and pebble type VHTR cores

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ABSTRACT

Korea Atomic Energy Research Institute (KAERI) is developing a new computer code system for an analysis of very high temperature gas-cooled reactor (VHTR) cores based on the existing HELIOS/MASTER code system. Several methodologies were developed in order for the original light water reactor (LWR) code system to treat the unique VHTR characteristics easily such as the so-called double-heterogeneity problem, the effects of a spectrum shift and a thermal up-scattering, a strong fuel/reflector interaction, etc. The method of a reactivity-equivalent physical transformation (RPT) and the equivalent cylindrical fuel (ECF) model are proposed to transform the double-heterogeneous fuel problem into a single-heterogeneous one in a cylindrical coordinate for both a prismatic fuel and a pebble-bed fuel. An eight energy group structure with appropriate group boundaries has been constructed in the MASTER diffusion nodal calculation, within which the issues of a spectrum shift and a thermal up-scattering are resolved. The concern about a strong fuel/reflector interaction can be handled easily by applying the equivalence theory to a simple one-dimensional spectral geometry consisting of the fuel and reflector regions. By combining all the methodologies described above, a well-known two-step core analysis procedure has been established, where HELIOS is used for the transport lattice calculation and MASTER for the 3-D diffusion nodal core calculation. The applicability of our code system was tested against several core benchmark problems. The results of these benchmark tests revealed that our code system is very accurate and practical for an analysis of both the prismatic and pebble-bed reactor cores.

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1. Introduction

In the Nuclear Hydrogen Development and Demonstration (NHDD) project (Chang et al., 2007), KAERI is developing a very high temperature gas-cooled reactor (VHTR) for a hydrogen production. For the successful completion of this project, a new computer code system for the analysis of VHTR cores is under development based on the existing HELIOS(STUDSVIK, 2000)/MASTER(Cho et al., 1999) code system. This code system was originally developed for a light water reactor (LWR) core analysis and it adopts a well-known two-step core analysis procedure. The two-step procedure which has been widely used and well-proven for the analysis of LWR cores consists of, for the first step, a transport lattice calculation for simple spectral geometries to generate a few group cross-section table-sets and, for the second step, a diffusion nodal core calculation by using these table-sets. HELIOS is responsible for the first step and MASTER is responsible for the second step in our code system.

In the VHTR reactor physics, there are several unique neutronic characteristics that cannot be handled easily by the conventional computer code system and the two-step analysis procedure for the analysis of LWR cores. Typical examples of such characteristics are the so-called double-heterogeneity problem due to the particulate fuels randomly dispersed in a graphite matrix, the effects of a spectrum shift and a thermal up-scattering due to a graphite moderator, a strong fuel/reflector interaction, etc. In order to facilitate in an easy treatment of such characteristics, we developed some specific methodologies for our code system.

The resonance self-shielding effect becomes significant if the fuel is lumped into small particles with multi-coating layers (so-called TRISO particles) and the particles are dispersed in a graphite matrix with a relatively low volume fraction. A simple volume-weighted homogenization (VWH) of a fuel zone with TRISO particles results in a significant reduction in the resonance self-shielding effect (Kim et al., 2004). This is a well-known double-heterogeneity problem which cannot be handled by most conventional lattice codes except for a few such as WIMS, APOLLO, and DRAGON (Halsall, 1996; Sanchez et al., 1987; Marleau et al., 2006). The method of a reactivity-equivalent physical transformation (RPT) has been proposed (Kim and Baek, 2005; Kim et al., 2006) to transform a double-heterogeneous fuel problem into a single-heterogeneous one, which renders the conventional lattice codes including HELIOS the capability to analyze the VHTR fuel elements.

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This method reduces the size of the homogenized fuel zone so that both the double-heterogeneous and the single-heterogeneous problems may provide an identical self-shielding effect.

Although the fuels are transformed into single-heterogeneous ones, the conventional two-dimensional lattice codes do not support a three-dimensional geometry consisting of spherical fuel pebbles randomly stacked in a pebble-bed type VHTR core. Therefore, we adopted the equivalent cylindrical fuel (ECF) model (Hogenbirk et al., 1995; Lee et al., 2005a) which transforms a spherical pebble stack into an equivalent cylindrical fuel bundle. This method conserves the material inventories and the mean chord length of the fuel zone during a transformation. Also, when this method is combined with the RPT method it enables our LWR based HELIOS code to treat double-heterogeneous fuels not only in a prismatic core but also in a pebble-bed core.

The spectrum shift due to the use of a graphite moderator and a higher operating temperature makes a resonance absorption and up-scattering more important in the VHTR cores. The adopted two-step procedure is based on an assumption that the cross sections are so independent of the environment that they can be calculated by a simple spectral geometry with a reflective boundary condition. The spectrum shift in the VHTR core may cause this premise to become unreliable if we use the two energy group structure widely adopted in a diffusion nodal core analysis for LWR cores. In order to appropriately handle the effects of a spectrum shift, we increased the number of energy groups to establish an eight-group structure with appropriate group boundaries (Kim et al., 2005), within which all the fuel and reflector cross sections become environment-free regardless of a state parameter change. In this energy structure, the cross sections may be calculated from a spectral calculation for a simple geometry consisting of a fuel block or a fuel pebble.

The long neutron diffusion length in a graphite moderated core and the long and thin design of the VHTR core result in a strong neutron interaction between the fuel and reflector regions. The sharp flux gradient caused by the control rods located in the reflector region may also increase the neutron leakage through the fuel/reflector interface. The diffusion theory may fail in modeling such a strong neutron interaction across the fuel/reflector interface. This concern was resolved easily by applying the equivalence theory (Smith, 1980) to a simple one-dimensional spectral geometry consisting of fuel and reflector regions. The transport effect can be captured in the diffusion nodal calculation by the equivalent parameters derived from the HELIOS transport lattice calculation for a simple spectral geometry.

Thanks largely to all the methodologies described above, a HELIOS/MASTER code system based on a two-step core analysis procedure has been established for an analysis of the two types of VHTR cores (Kim et al., 2005; Lee et al., 2005b; Kim et al., 2007). In this paper, we present an overview of the HELIOS/MASTER code system for these two types of VHTR cores while several issues specific to the analysis of prismatic VHTR cores were reported in a separate paper (Kim et al., 2007).

2. Methodology and test

2.1. Method of reactivity-equivalent physical transformation

In order to use the conventional lattice code systems for the VHTR fuels, the double-heterogeneous region needs to be converted to a homogeneous one. It is known that a simple volume-weighted homogenization (VWH) from a double-heterogeneous fuel zone results in a highly under-estimated reactivity. This is mainly because the resonance self-shielding effect of the fuel is substantially reduced in the homogeneous case.

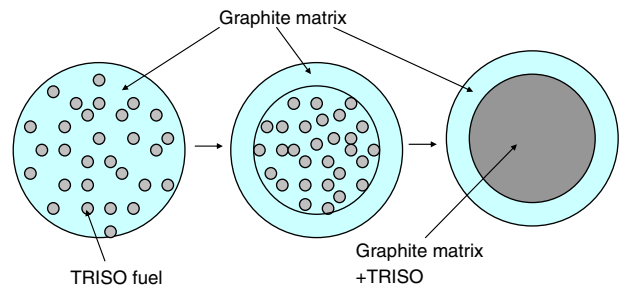

Fig. 1. RPT concept.

The concept of the RPT method is depicted in Fig. 1 for a cylindrical and spherical geometry: first, the fuel particles are moved into a smaller volume with a smaller surface area and then the inner double-heterogeneous fuel zone is simply homogenized in a volume-weighted sense. In the RPT method, the reduced radius (r_{rpt}) of the new homogeneous fuel zone is determined such that the resulting neutron multiplication factor should be equal to the reference one. It is worthwhile noting that reducing the fuel radius increases the reactivity since the self-shielding effect of the fuel is enhanced. The reference solution can be obtained by either a high-fidelity deterministic code or a Monte Carlo method.

The RPT method has been applied to a prismatic fuel assembly for the US NGNP design study (Macdonald et al., 2003) All the neutronic calculations were performed with the DRAGON code. Table 1 compares the RPT and VWH methods in terms of the four-factors (neutron emission per thermal neutron absorption in fuel η, thermal utilization factor f, resonance escape probability p, fast fission factor ε) for the two types of fuel assemblies, one without a burnable absorber (BA) and the other one with a B_4C BA. One should note that the large error for the VWH is mainly ascribed to a substantially smaller resonance escape probability. However, it is clearly observed that the RPT method provides almost identical four-factors. This is because the self-shielding effect is enhanced by reducing the fuel radius. The corrected self-shielding effects also lead to a more accurate neutron spectrum, thereby correcting other factors such as η, f, and ε. Further investigation (Kim et al., 2006) shows that not only the integral parameters such as k_{inf} or the four-factors but also the fine group microscopic cross sections are preserved well in the RPT process. It is also noteworthy that the RPT radius is very similar for the two significantly different fuel assemblies, and that it is much smaller than the original radius of the fuel compact.

Once the RPT radius is determined for an initial condition of a fuel assembly, all the other calculations can be done with the conventional methodologies. In Figs 2 and 3, the accuracy of the RPT is assessed during an assembly depletion for the two fuel assemblies. It is clearly observed that the RPT solution is matched very well up to a very high burn-up of 198 GWD/tU for the two cases, while the VWH solution had a large error. With respect to the assembly power distribution, the RPT method has a negligible error.

Next, the RPT method was applied to a pebble fuel with a 6 cm diameter. In this case, the calculations were done with the MC-CARD Monte Carlo code (Shim et al, 1999). A simple cubic arrangement of pebbles was assumed, and a UO_2 fuel kernel with a 10 wt.% enrichment was used. In an actual pebble fuel, about 15,000 TRISO particles are embedded in a pebble, which cannot be treated with the MC-CARD code due to the necessary memory requirements and computing time. Instead, 1000 TRISO particles with a kernel diameter of 1230 μm are explicitly modeled here. Due to the large kernel, the double-heterogeneity effect is huge, 11,418 pcm. Nevertheless, the RPT reveals a good agreement as shown in

Table 1
Effects of the RPT method on the four-factors

Method r_f = 0.6225 cm	Four-factors			
	η	f	p	ε
No burnable absorber				
Reference	2.07779	0.99218	0.70450	1.00075
VWH	2.08576 (0.38%)	0.99225 (0.01%)	0.66539 (−5.55%)	1.00066 (−0.01%)
RPT r_{rpt} = 0.3841 cm	2.07776 (0.00%)	0.99217 (0.00%)	0.70453 (0.00%)	1.00075 (0.00%)
With B_4C burnable absorber				
Reference	2.13095	0.73828	0.69633	1.00075
VWH	2.14117 (0.48%)	0.73982 (0.21%)	0.65766 (−5.55%)	1.00061 (−0.01%)
RPT r_{rpt} = 0.3830 cm	2.13085 (−0.01%)	0.73807 (−0.03%)	0.69659 (−0.04%)	1.00075 (0.00%)

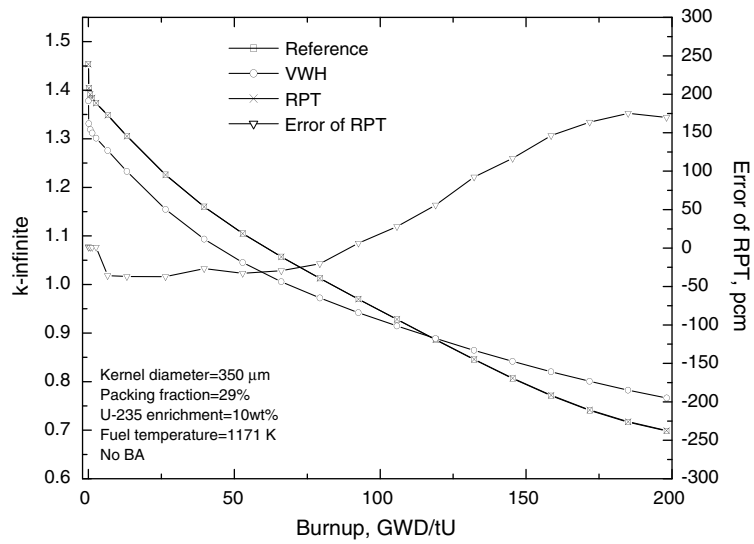

Fig. 2. Accuracy of the RPT during a depletion (no BA).

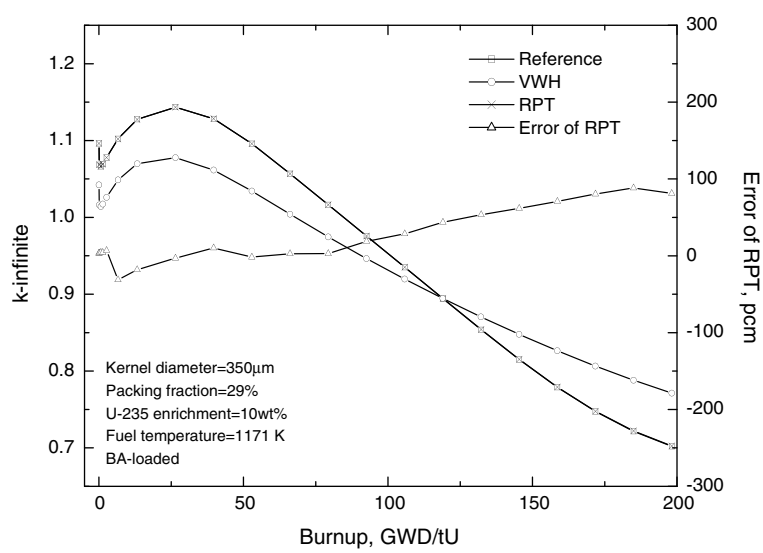

Fig. 3. Accuracy of the RPT during a depletion (with BA).

Fig. 4. The RPT performance would be much better for an actual pebble with a substantially smaller double-heterogeneity effect.

2.2. Equivalent cylindrical fuel model

Even after the fuel pebbles stacked in a pebble-bed core are transformed into single-heterogeneous ones by the RPT method, they should be transformed again into equivalent cylindrical fuels to be handled by a conventional two-dimensional lattice code such as HELIOS. The equivalent cylindrical fuel (ECF) model plays this role by preserving the material inventories and the mean chord length of a fuel during this transformation.

The mean chord length of a convex body is given by

$$\bar{l} = 4V/S, \tag{1}$$

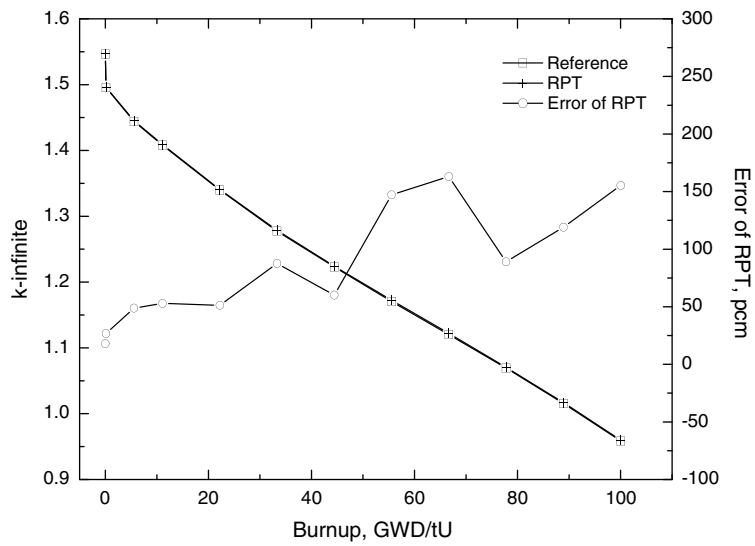

Fig. 4. Depletion with the RPT for a pebble fuel.

where V and S are the volume and the surface area of the body respectively. The average chord length is $4R/3$ for a sphere with radius R and $2r$ for an infinite cylinder with radius r.

The radius of the equivalent cylinder can be determined by preserving the chord length of the fuel region. The graphite radius and the boundary of the cylinder model can be determined by preserving the volume fractions of each region. It should be noted that it is mathematically impossible to preserve the average chord lengths of both the fuel region and the outer graphite region simultaneously. Therefore, we did not preserve the average chord of the outer graphite region but that of the fuel region.

The ECF model for the 3-D spherical pebbles was verified by using the MC-CARD code. A spherical pebble with a BCC lattice structure was taken as a reference case. The fuel zone radius of the pebble was 2.5 cm, the thickness of the graphite shell was 0.5 cm, the packing fraction of the pebbles was 0.61, the enrichment of the fuel was 9.6%, and the temperature of the pebble and the helium coolant were assumed to be 800 °C and 750 °C respectively.

The spherical pebble described above was transformed into three equivalent cylinder models. They are a 1-D cylinder model, a 2-D cylinder model with a hexagonal lattice structure, and a 2-D cylinder model with a square lattice structure. In the 1-D cylinder model, the boundary surface of a unit cell was approximated by a cylindrical surface. Two boundary conditions, a reflective one and a white one, were applied at the boundary of each model.

The ECF model should also work in different conditions from the reference condition. To investigate this, we compared the 3-D spherical pebble model with a BCC lattice structure and the ECF model with a square lattice structure in several different conditions from the reference condition. In Case 1, we lowered the temperature of the pebble and helium gas to a temperature of 27 °C. In Case 2, we raised the enrichment of the uranium fuel to 19.5%. In Case 3, half of the pebbles were replaced with moderator pebbles to simulate the initial pebble-bed core.

To verify the accuracy of the ECF model during a depletion calculation, we also performed a MC-CARD depletion calculation for the two models. The validity of the equivalent cylinder fuel loaded into a core should also be verified. Fig. 5 shows the spherical pebble fuels and the equivalent cylinder fuels loaded into an infinite slab reactor with a packing fraction of 0.5. Note that the pitches of the cylinders in the x-and y-directions were slightly adjusted to space them evenly in the x-direction. The power distributions and the effective multiplication factors of the two cores from the MC-CARD calculation were compared.

Table 2 shows the k_{inf}'s and their standard deviations for the models obtained from the MC-CARD calculation. Except for the 1-D cylinder model with a reflective BC, all the equivalent cylinder models seem to be acceptable, even though the average chord length of the outer graphite region is not preserved. However, the 1-D cylinder model with a white BC has a relatively large difference. Very small k_{inf} differences are observed for both Case 1 and Case 2 while a relatively large k_{inf} difference is observed in Case 3, in which half of the pebbles were moderator pebbles. The relatively large difference in k_{inf} is ascribed to the fact that the average chord length in the graphite region plays a more important role than in the other cases, because the amount of graphite is larger in this case than the other cases.

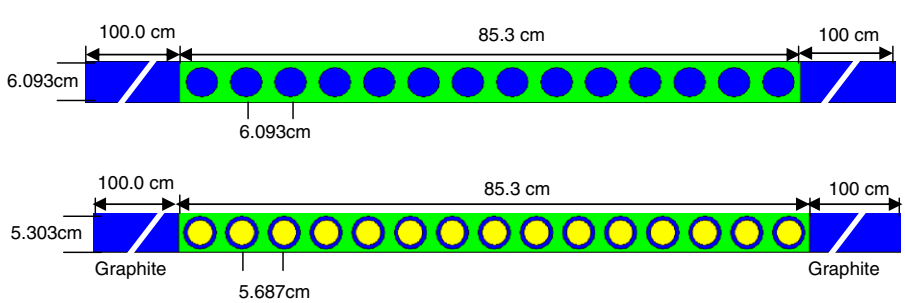

Fig. 5. Infinite slab reactor model.

Table 2
Comparison of various ECF models

	3-D Sphere	Equivalent cylinder model		Error (pcm)
Reference Condition	1.32476 ± 12 pcm	1-D RBC	1.33760 ± 12 pcm	+1284
		1-D WBC	1.32335 ± 12 pcm	−141
		2-D Sq. RBC	1.32504 ± 12 pcm	+28
		2-D Sq. WBC	1.32394 ± 12 pcm	−82
		2-D Hex. RBC	1.32539 ± 12 pcm	+63
		2-D Hex. WBC	1.32395 ± 12 pcm	−81
Cold state (27 °C)	1.42404 ± 12 pcm	2-D Sq. RBC	1.42461 ± 12 pcm	+57
High Enr. (19.5%)	1.39517 ± 12 pcm	2-D Sq. RBC	1.39539 ± 12 pcm	+22
Fuel:Mod. (1:1)	1.54304 ± 10 pcm	2-D, Sq. RBC	1.54140 ± 11pcm	−164

Sq: Square; Hex: Hexagonal; RBC: Reflectice; WBC: White.

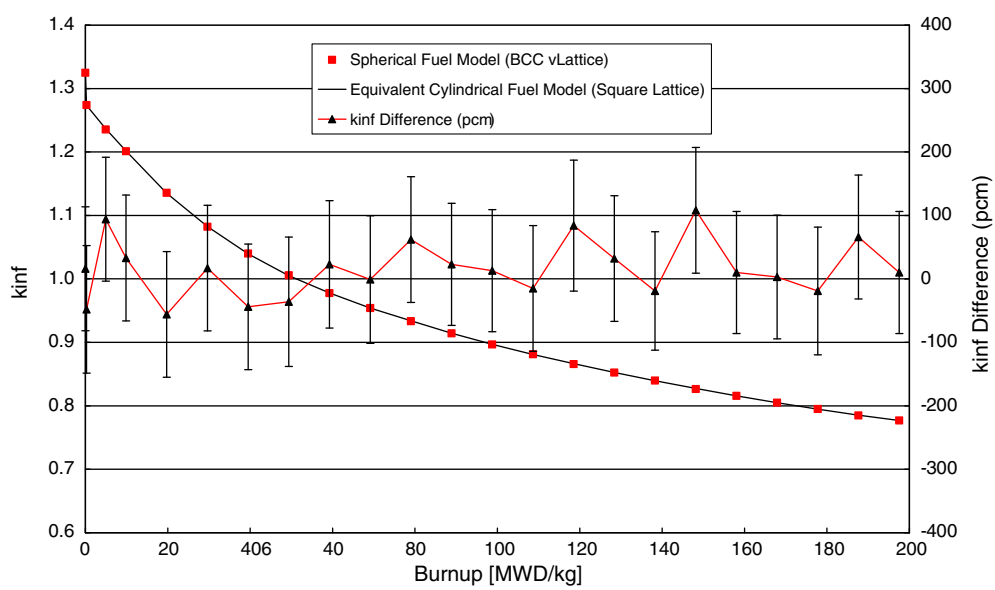

Fig. 6. Accuracy of the ECF model during a depletion.

Fig. 6 shows the k_{inf} of the two models during the depletion calculation. At all the burn-up points except for only one point, the k_{inf} values of the two models agree with each other to within 2σ. Though the difference in the k_{inf} values is larger than 2σ at a burn-up point, the value is around 100 pcm, which is very small and still acceptable for a neutronics calculation.

Fig. 7 shows the power distribution of the infinite slab reactor cores loaded with spherical pebble fuels and equivalent cylinder

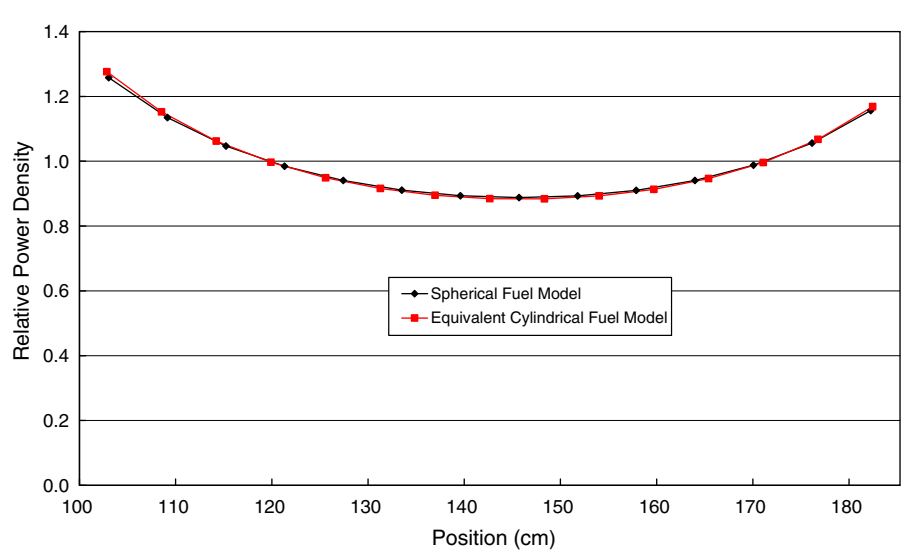

Fig. 7. Power distribution for an infinite slab reactor.

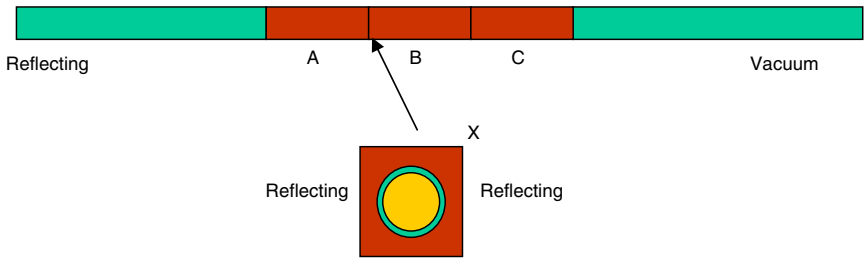

Fig. 8. One-dimensional VHTR core model.

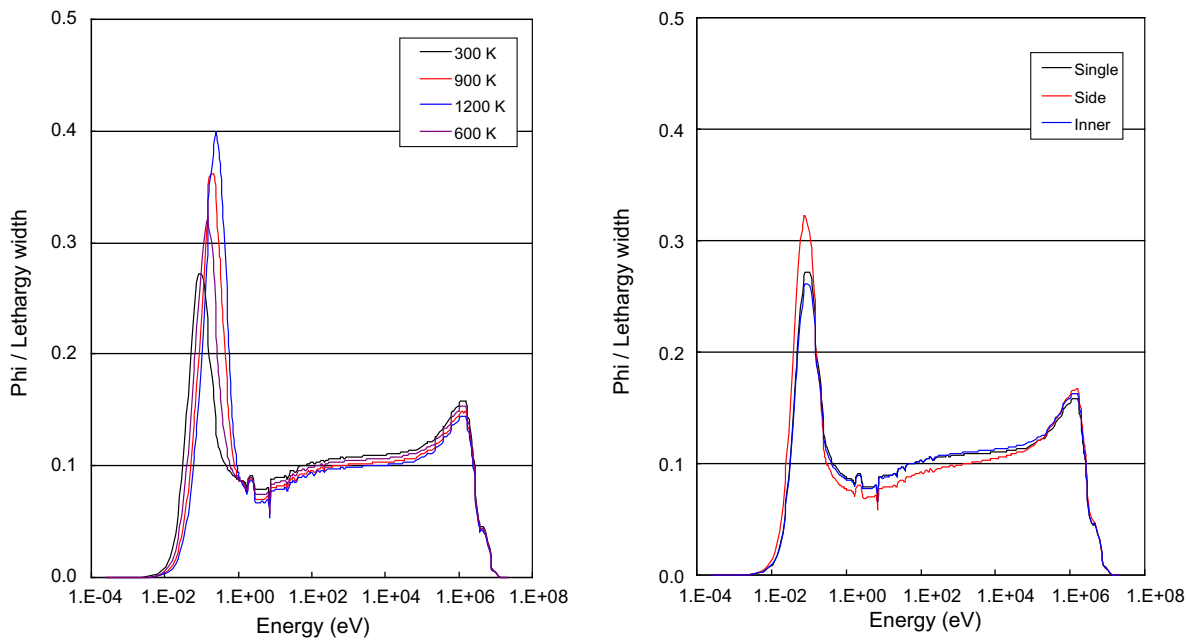

Fig. 9. Neutron spectra for the VHTR blocks.

fuels. The power distributions of the two cores are almost identical. The effective multiplication factor of the core loaded with spherical pebble fuels and that of the core loaded with equivalent cylinder fuels are 1.28144 and 1.27948 respectively.

2.3. Eight energy group structure

For the energy group study, a 1-D core model as shown in Fig. 8 was constructed from a typical prismatic VHTR core. This model consists of three fuel blocks, each of which includes 14 fuel pins. The double-heterogeneity of a pin was assumed to have already been eliminated by the RPT method. By using this 1-D model, we determined the number of energy groups and their boundaries, with which all the cross sections become environment-free regardless of a fuel temperature change, so that the cross sections may be calculated from a spectral calculation for a simple geometry. For this purpose, the neutron spectrum changes according to the location of the fuel blocks in the 1-D core and a temperature variation were investigated. Fig. 9 shows that the neutron spectrum becomes hardened as the temperature increases and that the thermal neutron spectrum of a fuel block accompanied by a reflector is higher than that of an inner block. Few group macroscopic cross

Table 3
Comparison of the block-wise k_{inf}

Temperature (K)	# of groups	Block				Core X	$A + B + C$ (pcm)
		X	A (pcm)	B (pcm)	C (pcm)		
300	4	1.52497	1629	30	1570	1.58425	−1034
	6	1.52497	323	−88	315	1.58424	−154
	7	1.52497	218	−213	205	1.58424	−128
	8	1.52343	40	−44	39	1.58425	−17
	9	1.52497	45	−41	44	1.58425	−18
	12	1.52497	−22	−42	−21	1.58424	23
300	8	1.5251	106	−5	103	1.58498	−63
600	8	1.47864	75	−4	70	1.54500	−43
900	8	1.44421	62	−8	56	1.51453	−34
1200	8	1.41771	61	−9	55	1.49065	−32

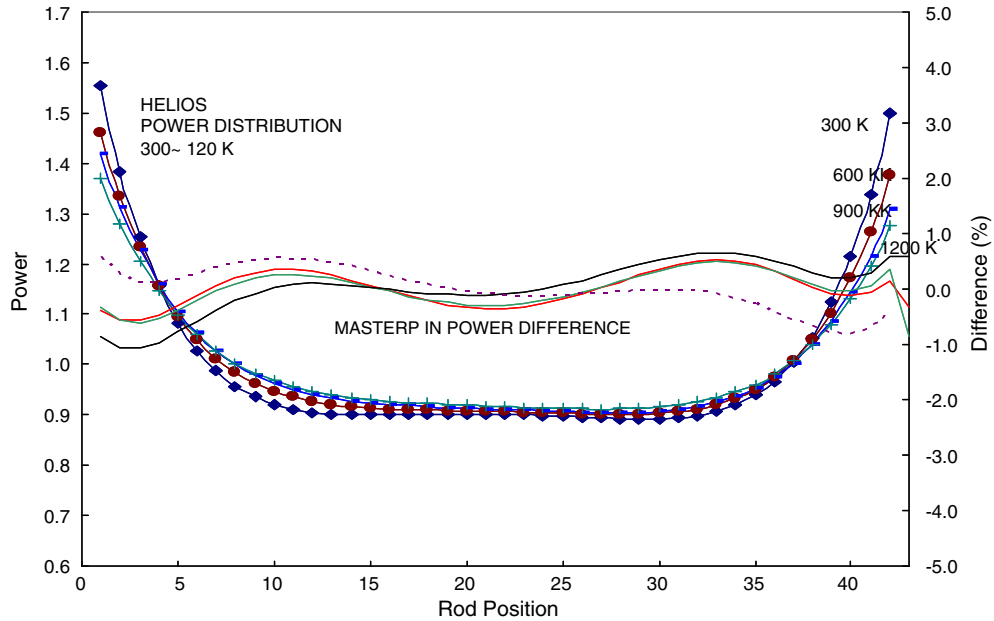

Fig. 10. Comparison of the pin power distributions.

sections and spectrum were extracted from the 190 group HELIOS calculations for blocks A, B, C, and X in Fig. 8.

Group boundaries were adjusted to minimize the differences in the infinite multiplication factors ($k_{\text{inf.}}$) of the blocks and the core as shown in Table 3. Here the k_{inf}'s for a block and a core were calculated using the spectra of block X and the core (blocks A + B + C), respectively. The result of the optimization calculations performed at 300 K by changing the number of energy groups from 4 to 12 reveals that when the number of groups is larger than 8, there is almost no improvement, and the maximum reactivity difference is about 45 pcm. Therefore, the eight-group case, which is regarded as an optimum choice at this temperature, was verified further at various temperatures. The results reveal that the reactivity differences are less than 105 pcm.

Using this eight-group structure may avoid a pin power reconstruction after the nodal calculation during an practical application, because the homogeneous power shapes calculated by MASTER agreed well with those of the heterogeneous HELIOS calculations, as shown in Fig. 10.

Table 4
Comparison of k_{eff} for the 3-D benchmark problem

MASTER (C)	MC-CARD (R)	C-R (pcm)
1.21342	1.20928 ± 3 pcm	+414

2.4. Equivalence theory application

To capture the effect of a strong spectral interaction between a core and a reflector, the equivalence theory was applied to a 1-D spectral geometry consisting of a fuel region and a reflector region, as shown in Fig. 8. The discontinuity factors and the reflector cross sections derived from the HELIOS lattice calculation for this

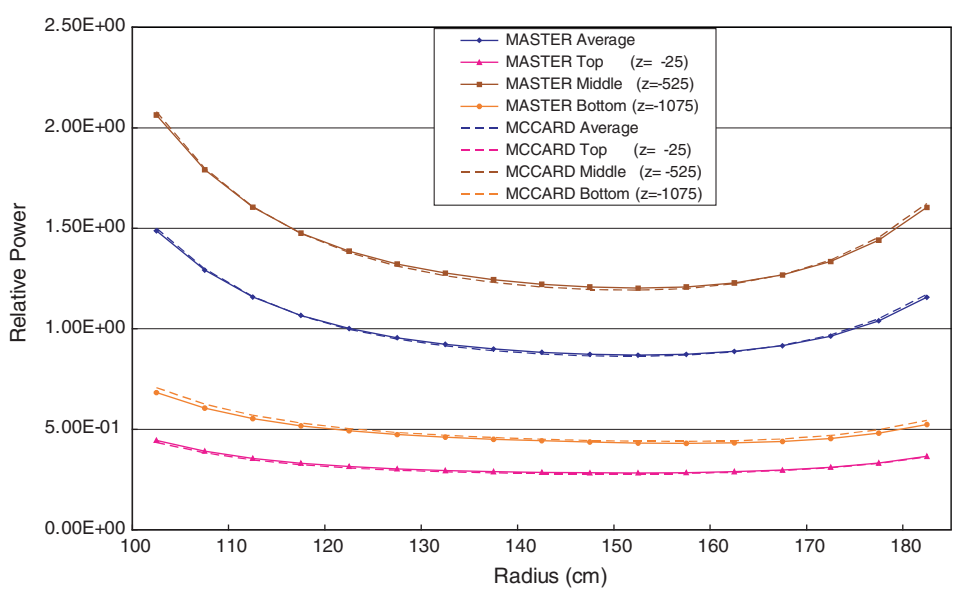

Fig. 11. Comparison of the radial power distributions.

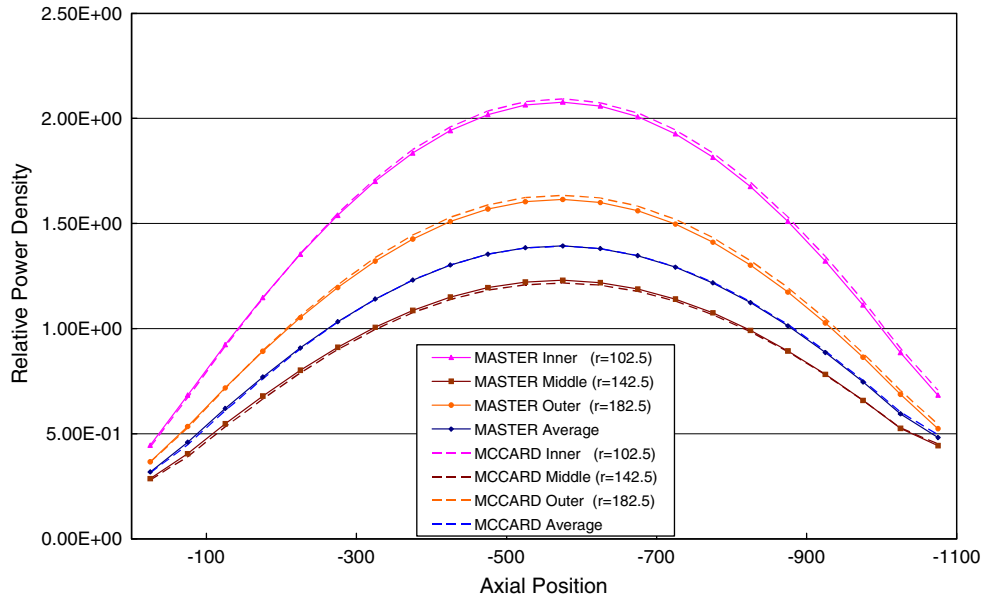

Fig. 12. Comparison of the axial power distributions.

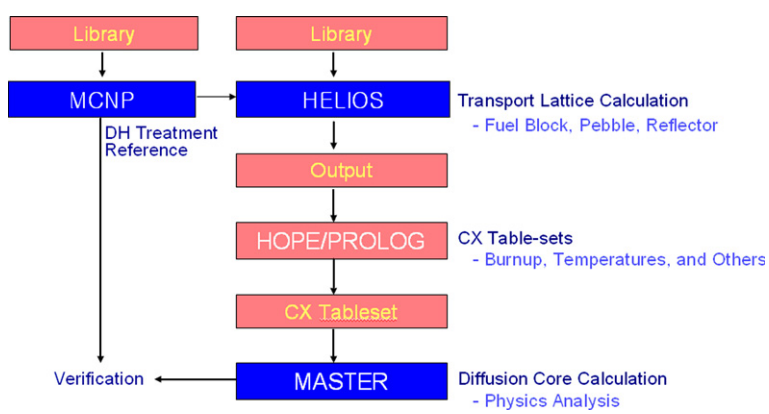

Fig. 13. HELIOS/MASTER code system.

spectral geometry were used in the three-dimensional diffusion nodal core calculation performed by MASTER.

This equivalent procedure was verified against a 3-D benchmark problem derived from a typical pebble-bed VHTR core. In Table 4 and Figs. 11 and 12, the diffusion solution and the Monte Carlo solution were compared for this benchmark problem. The equivalent cross sections generated from the 1-D slab spectral geometry were used in the diffusion calculation. The diffusion solution predicts the Monte Carlo solution well and the errors are acceptable.

3. HELIOS/MASTER code system

All the methodologies described in the previous chapter were incorporated into the HELIOS/MASTER code system shown in Fig. 13.

Table 5
Comparison of k_{eff} for the 3-D prismatic core problem

Temperature (K) (pcm)	MCNP	HELIOS/MASTER Error (pcm)
300	1.42476	179
600	1.39789	189
900	1.37452	83

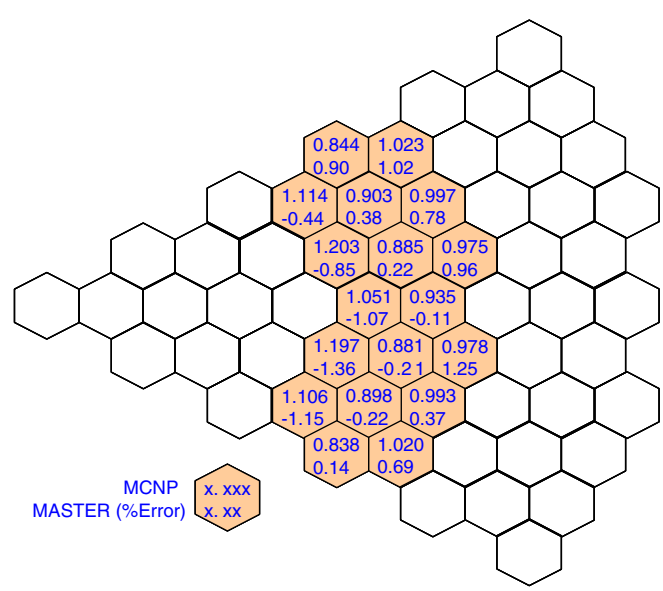

Fig. 14. Comparison of the radial power distribution.

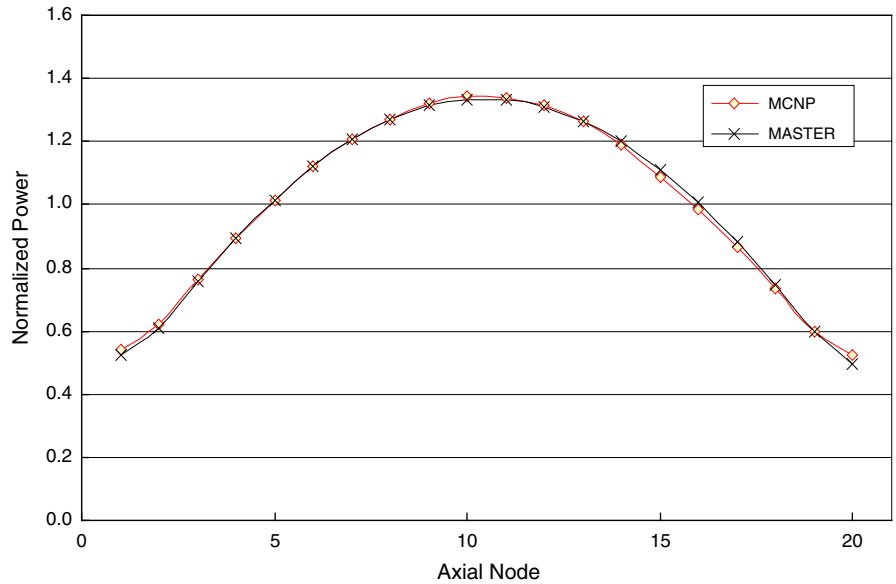

Fig. 15. Comparison of the radial power distribution.

This code system is based on a two-step core analysis procedure where a two-dimensional lattice calculation is accomplished by using HELIOS to generate a few group cross-section table-sets and a three-dimensional diffusion core calculation is accomplished by using MASTER. MCNP (Breismeister et al., 1997) or MC-CARD is used to determine the RPT radius prior to the HELIOS calculation. Once the HELIOS calculation is performed for a simple spectral geometry consisting of a single prismatic block or a single pebble with a reflecting boundary condition, the cross sections are tabulated as functions of the burn-up, fuel temperature, and moderator temperature. Using these cross sections in the table-set form, MASTER performs a whole core depletion calculation. MCNP may also be used to provide reference solutions for a verification of our code system.

To verify this code system, we solved a typical three-dimensional prismatic VHTR core in a perfect one-sixth symmetry. The MCNP model with an explicit treatment of all the TRISO particles served as a reference for our code system. The only common thing shared by both the MCNP and HELIOS/MASTER calculations is the starting cross-section data library ENDF/B-VI. The results are summarized in Table 5 and Figs. 14 and 15. These results reveal that HELIOS/MASTER predicts the MCNP solutions very well and the maximum errors for the effective multiplication factor, block-wise powers, and axial powers are only 190 pcm, 1.4%, and 2.4%, respectively.

4. Conclusions

Several methodologies were developed in order for the original LWR code system to treat the unique VHTR characteristics easily such as the so-called double-heterogeneity problem, the effects of a spectrum shift and a thermal up-scattering, a strong fuel/reflector interaction, etc.

The RPT method was proposed to transform the double-heterogeneous fuel problem into a single-heterogeneous one in such a way that the two problems provide an identical reactivity. We proposed the addition of the ECF model which can handle a spherical pebble fuel as an equivalent cylindrical fuel. These two methods when combined together enabled our LWR based code system to treat double-heterogeneous fuels not only in a prismatic core but also in a pebble-bed core.

In order to handle the effects of a spectrum shift, an eight energy group structure was constructed, within which all the fuel and reflector cross sections become environment-free regardless of a fuel temperature change, so that these cross sections may be calculated from a spectral calculation for a simple geometry with a reflective boundary condition.

Any concern about a strong fuel/reflector interaction can be resolved easily by applying the equivalence theory to a simple one-dimensional spectral geometry consisting of fuel and reflector regions.

With all the methods developed in this study, a HELIOS/MASTER code system based on a two-step core analysis procedure was established for an analysis of two VHTR core types. In this code system, HELIOS is used for the transport lattice calculation to generate a few group cross sections, and MASTER for the 3-D core calculation to perform a reactor physics analysis.

The applicability of this code system was verified against some VHTR benchmark problems. The results of these benchmark tests revealed that our code system is very accurate and practical for an analysis of both the prismatic and pebble-bed reactor cores.

Further tests of this code system for depleted core configurations are in progress. And its capability to simulate a pebble flow movement will be added in the future to simulate the depletion process of a pebble-bed reactor core.

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References

Breismeister, J.F. et al., 1997. MCNP – A General Monte Carlo N-Particle Transport Code. Version 4B. LA-12625-M.

Chang, J. et al., 2007. A study of nuclear hydrogen production demonstration plant. Nuclear Engineering and Technology 39, 111–122.

Cho, B.O. et al., 1999. MASTER-2.0: Multi-purpose Analyzer for Static and Transient Effects of Reactors. Korea Atomic Energy Research Institute Technical Report (KAERI/TR-1211/99). Daejeon, Korea.

Halsall, M.J., 1996. WIMS7, an overview. In: Proceedings of International Conference on the Physics of Reactors (PHYSOR 96). Mito, Japan.

Hogenbirk, A. et al., 1995. HTR-Proteus Benchmark Calculations, Netherlands Energy Research Foundation (ECN) Technical Report (ECN-C–95-087). Petten, The Netherlands.
Kim, T.K. et al., 2004. Preliminary assessment of lattice physics capabilities for VHTR analysis. Trans. Am. Nucl. Soc., 91.
Kim,Y., Baek, M., 2005. Elimination of Double-Heterogeneity through a Reactivity-Equivalent Physical Transformation. GLOBAL2005, Tskuba, Japan, October 9–13 (CD ROM).
Kim, K.S. et al., 2005. Two-step procedure development for the VHTR physics analysis using a 1-D core model. Trans. Kor. Nucl. Soc. Autumn Mtg. (CD ROM).
Kim, Y., Kim, K.S., Noh, J.M., 2006. Preservation of fuel characteristics in the RPT method. Trans. Kor. Nucl. Soc. Spring Mtg (CD ROM).
Kim, K.S. et al., 2007. Development of a physics analysis procedure for the prismatic very high temperature gas-cooled reactors. Annals Nucl. Energy 34, 849.
Lee, H.C. et al., 2005a. The equivalent cylinder models for the homogenization of pebble-bed reactor cores. Trans. Am. Nucl. Soc. 93, 961.
Lee, H.C. et al., 2005b. Two step procedure using a 1-D slab spectral geometry in a pebble-bed reactor core analysis. Trans. Kor. Nucl. Soc. Autumn Mtg (CD ROM).
Macdonald, P.E. et al, 2003. NGNP Preliminary Point Design – Results of Initial Neutronics and Thermal-Hydraulic Assessment. INEEL/EXT-03-00870 Rev. 1.
Marleau, G. et al., 2006. A User Guide for DRAGON IGE-174 Rev. 6. Ecole Polytechnique de Montreal.
Sanchez, R. et al., 1987. Apollo-II: A user-oriented, portable, modular code for multigroup transport assembly calculations. In: Proceedings of the International Topical Meeting Advances in Reactor Physics, Mathematics and Computation. Paris, France.
Shim, H.J. et al., 1999. Monte Carlo Depletion Analysis of a PWR with the MCNAP. M&C'99. Madrid, Spain.
Smith, K.S., 1980. Spatial Homogenization Methods for Light Water Reactor Analysis. PhD Thesis, Massachusetts Institute of Technology.
STUDSVIK SCANDPOWER, 2000. HELIOS 1.7 Methods.