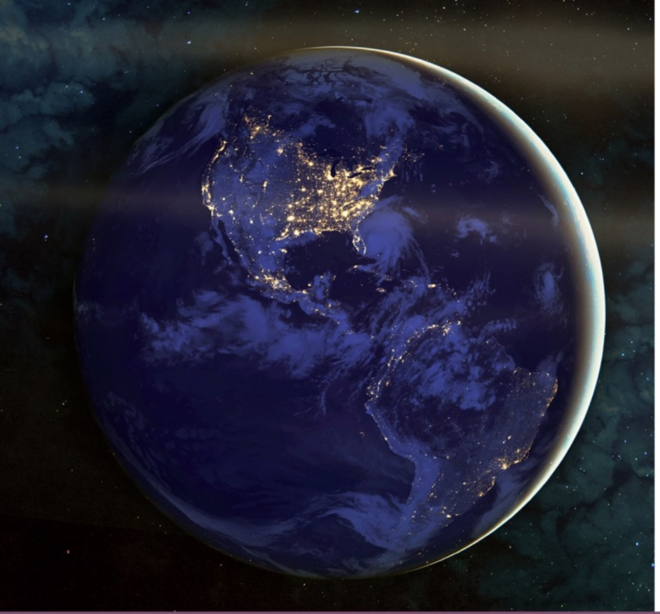

ANS Winter Meeting & Expo

2019 NUCLEAR TECHNOLOGY FOR THE U.S. AND THE WORLD

Determination of Bounding Axial Burnup Profiles for Criticality Analysis with Burnup Credit for Spent Fuels discharged from OPR-1000

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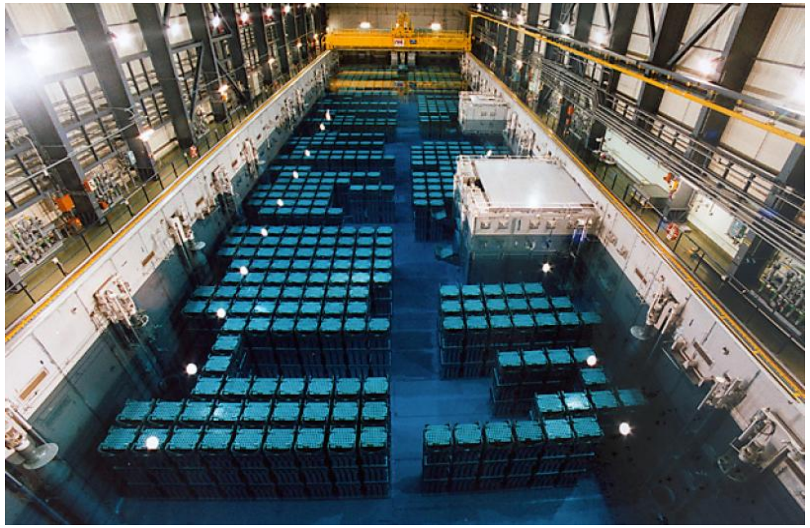
Introduction

- In South Korea, the dry storage of PWR spent fuels is one of the possible ways to solve the saturation of the spent fuel storage pools because the most spent fuel storage pools are expected to be saturated within 5 to 19 years.
- The criticality analysis of the spent fuel storage and transportation facilities is very important to show that they are kept under subcriticality under normal and accident situations.
- In particular, the burnup credit and end effect should be applied to spent fuel storage and transportation facility in order to increase spent fuel loading or to reduce the loading limits.
- When the criticality analysis with burnup credit is performed, the bounding axial burnup profiles are determined to give conservative end effects.
- The objective of this work is to determine the bounding axial burnup shapes for a cask loaded with spent fuels discharged from OPR-1000.



Calculation Method (Calculation Procedure)

Nuclear Design Report

Measured burnup

STREAM/RAST-K
Core follow calculation

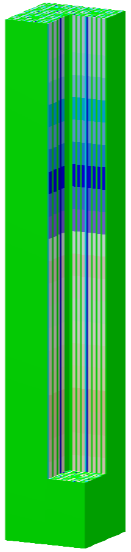
DeCART2D/MASTER
Core follow calculation

Validation

STREAM/RAST-K
OUTPUT

DeCART2D/MASTER
OUTPUT

NDR
(Nuclear Design Report)

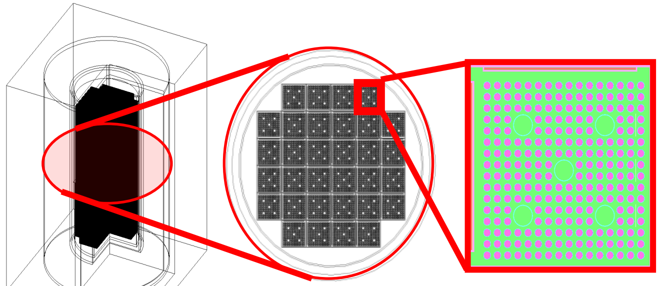


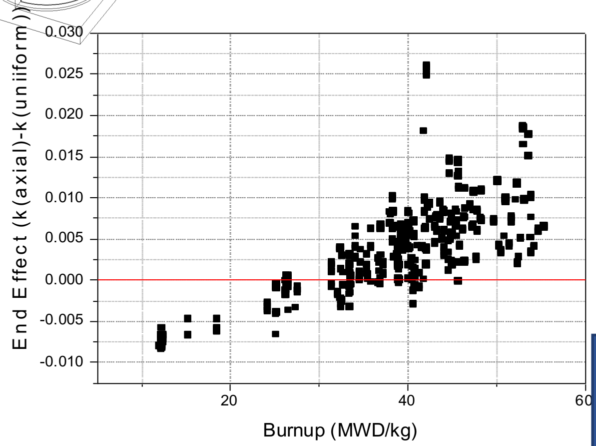
Node	BU (normalized)
1	0.39066
2	0.58831
3	0.77651
4	0.92437
5	1.03047
6	1.09079
7	1.11323
8	1.12067
9	1.12127
10	1.11958
11	1.11778
12	1.11643
13	1.11578
14	1.1157
15	1.11559
16	1.11413
17	1.10816
18	1.09277
19	1.05952
20	0.98602
21	0.87105
22	0.72476
23	0.55054
24	0.38754

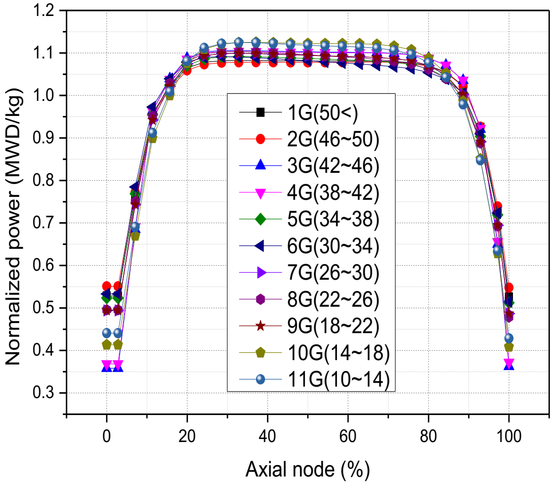
Assembly axial profile

Burnup Group (MWd/kg)	Number of FAs
50	92
46 – 50	76
42 – 46	192
38 – 42	171
34 – 38	59
30 – 34	60
26 – 30	33
22 – 26	19
18 – 22	8
14 – 18	3
10 - 14	45

SCALE 6.1
STARBUCS sequence







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Calculation Method (Core Follow Calculation)

Code system of STREAM/RAST-K

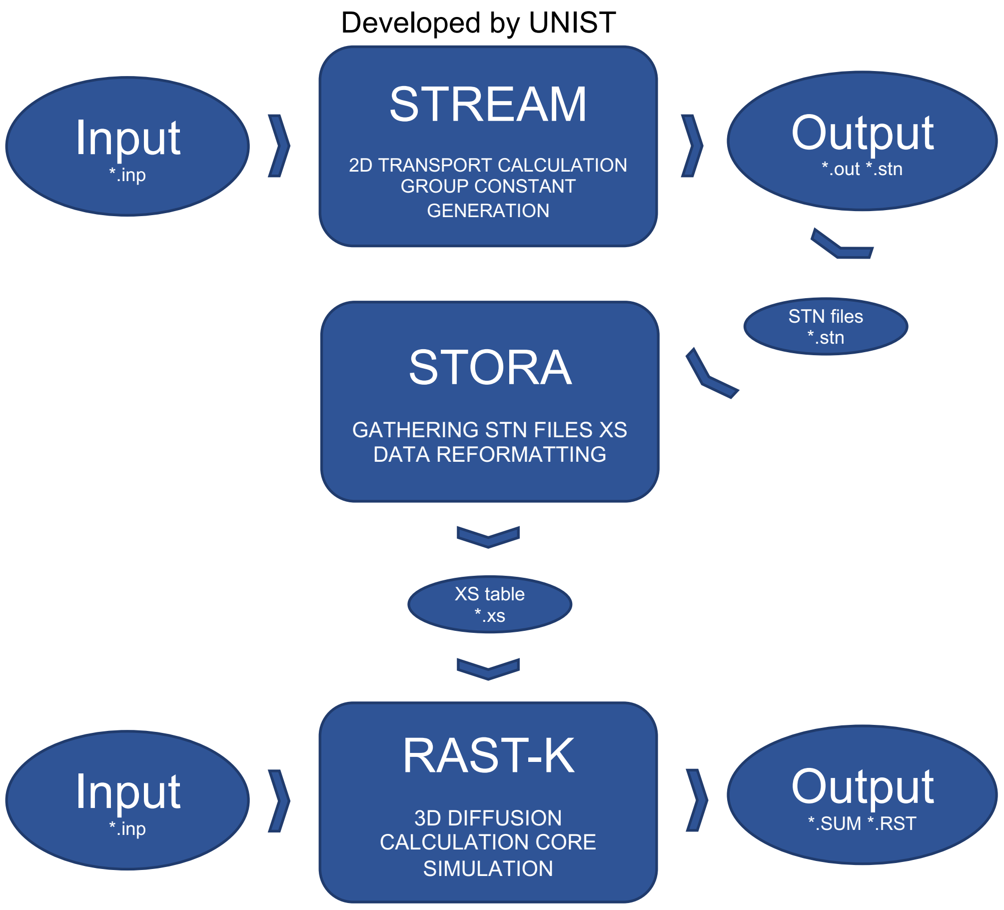

Fig. 1 Two-step core design system for STREAM/RAST-K

- The STREAM code developed by UNIST is an advanced lattice code, which solves multi-group transport equation with MOC(Method Of Characterized) for two-dimensional assembly and reflector models and generates homogenized fuel assembly cross sections and form function.
- The STREAM code is characterized by its PSM (Pin-based point-wise Slowing down Method) and equivalence theory for resonance self-shielding effect and by the CRAM(Chebyshev Rational Approximation Method) for depletion.
- The RAST-K code developed by UNIST is an advanced nodal diffusion code which uses the multi-group CMFD(Coarse Mesh Finite Difference) method coupled with 3D multi-group unified nodal method.

Calculation Method (Core Follow Calculation)

Code system of DeCART2D/MASTER

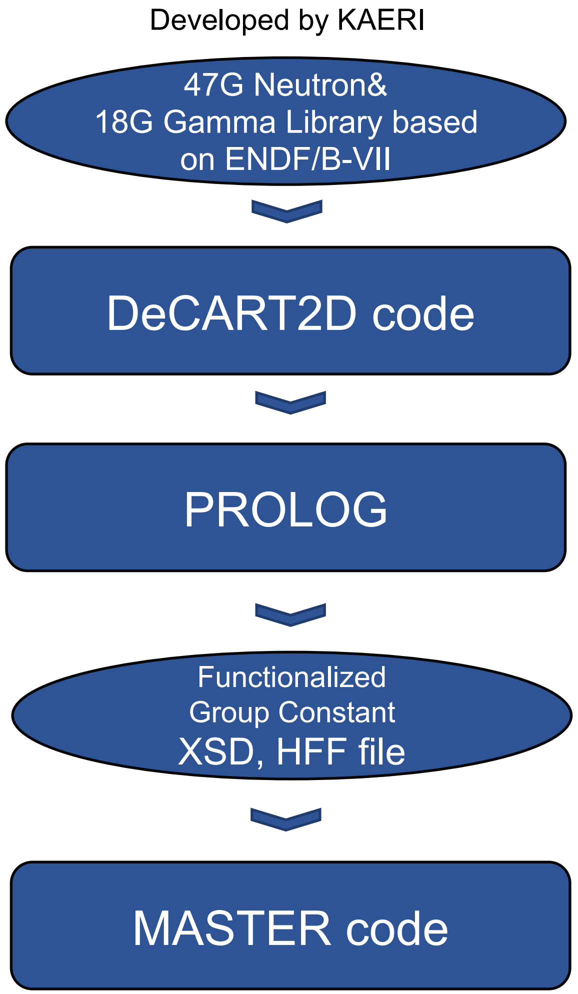

Fig. 2 Two-step core design system for DeCART2D/MASTER

- Additionally, we employed the DeCART2D/MASTER code system developed by KAERI.
- KARMA 47 group cross section libraries based on ENDF/B-VII is used.
- Assembly calculations were performed by using DeCART2D (Deterministic Core Analysis based on Ray Tracing for 2-Dimensional Core) code to generate few group homogenized neutron cross section data.
- PROLOG (PROcessor for Library Of Group constant) is a program to edit two group microscopic neutron cross section data.
- Core analysis was performed with MASTER(Multi-purpose Analyzer for Static and Transient Effects of Reactors) code developed at KAERI and the code is a nuclear analysis and design code which can simulate the PWR or BWR core in 1, 2, and 3-dimensional Cartesian or hexagonal geometry with the advanced nodal diffusion methods.

Calculation Method (Criticality Analysis)

SCALE6.1 STARBUCS sequence

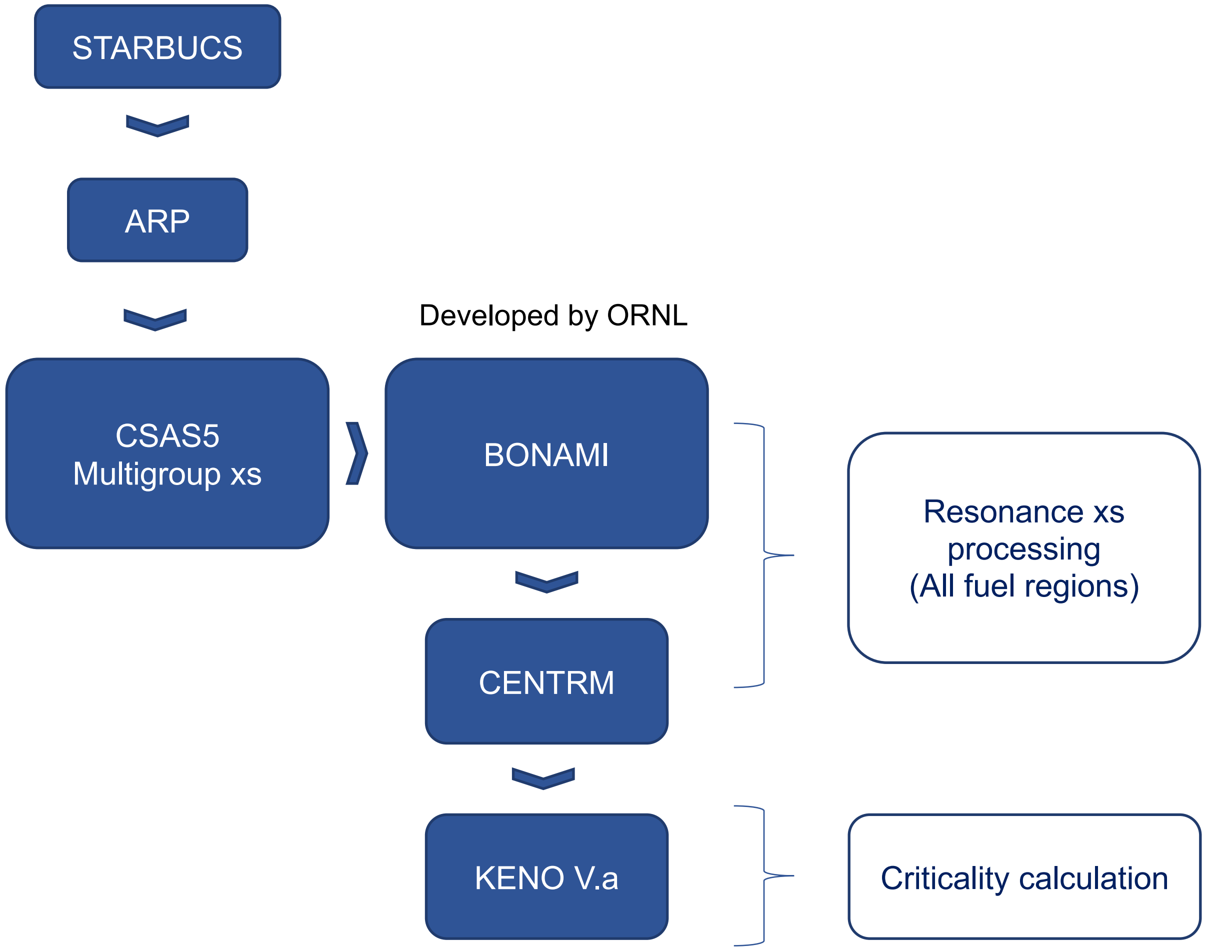

Fig. 3 STARBUCS sequence

- The STARBUCS sequence developed by ORNL automates the depletion calculations using the ORIGEN-ARP methodology to perform a series of cross-section preparation and depletion calculations to generate a comprehensive set of spent fuel isotopic inventories for each spatially-varying burnup region of an assembly.

- The spent fuel nuclide concentrations are subsequently input to CSAS5 and perform a criticality calculation of the system using the KENO V.a. For burnup credit application, we considered the following 12 actinides and 16 fission products: ^{234}U, ^{235}U, ^{238}U, ^{238}Pu, ^{239}Pu, ^{240}Pu, ^{241}Pu, ^{242}Pu, ^{241}Am, ^{95}Mo, ^{99}Tc, ^{101}Ru, ^{103}Rh, ^{109}Ag, ^{133}Cs, ^{147}Sm, ^{149}Sm, ^{150}Sm, ^{151}Sm, ^{152}Sm, ^{143}Nd, ^{145}Nd, ^{151}Eu, ^{153}Eu, ^{155}Gd, ^{236}U, ^{243}Am, ^{237}Np.

Calculation Results (Hanbit Unit 3 Cores)

Table 1 Specifications of Hanbit Unit 3(1)

Parameter	Value
Power (MWt)	2815
Power (MWe)	1000
No. of FAs in Core	177
Average linear heat rate (Watt/cm)	176.87
Average specific power (kW/kgU)	36.875
Active core height (cm)	381
Burnable poison type	UO_2/Gd_2O_3
Loading pattern type	3batch, Out-in

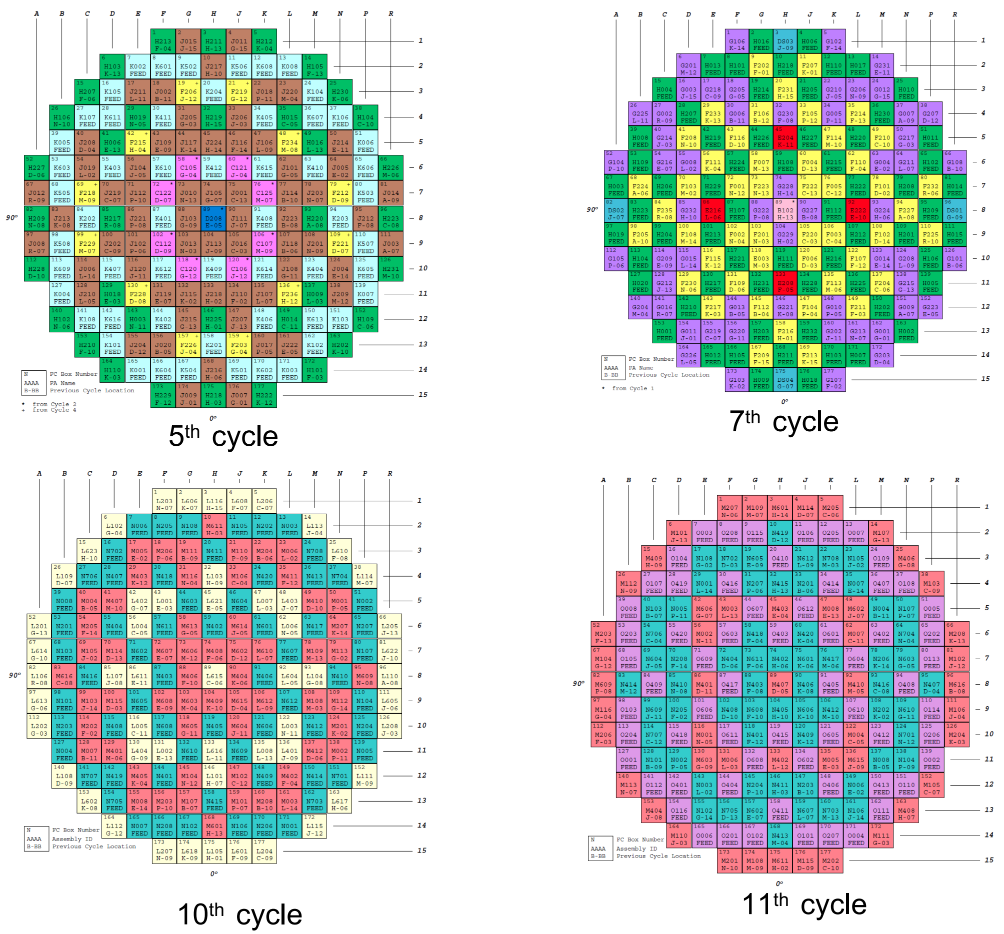

Fig. 4 Core Loading Patterns of Hanbit Unit 3

Calculation Results (Hanbit Unit 3 Cores)

Table 2 Specifications of Hanbit Unit 3(2)

Parameter	Value
Fuel material	UO_2
Fuel density(g/cm^3)	10.176
Number of fuel rods	236
Fuel pellet radius(cm)	0.41275
Cladding outer radius(cm)	0.48514
Cladding thickness(cm)	0.06350
Pin pitch(cm)	1.2852
Guide tube inner radius(cm)	1.1430
Guide tube outer radius(cm)	1.2446
Active fuel length(cm)	381
Assembly pitch(cm)	20.78
Cladding material	Zircaloy4

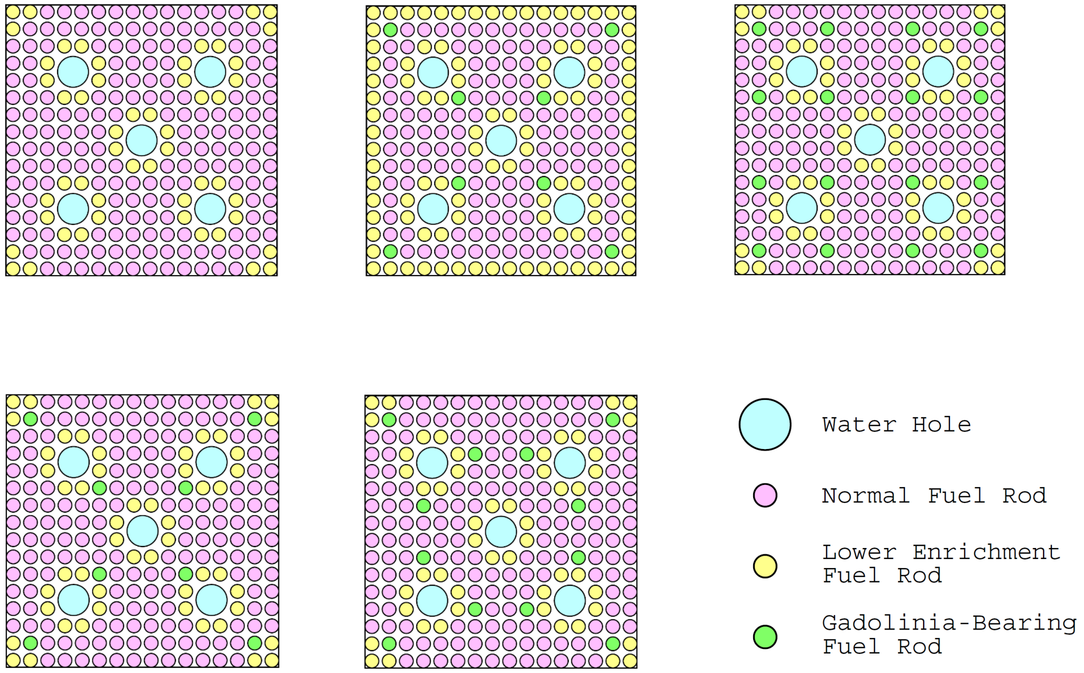

Fig. 5 Configuration of 16x16 fuel assembly

- Each fuel assembly has 16x16 fuel array lattice structure comprised of 236 fuel rods and 5 guide tubes.
- Actually, the three different type fuel assemblies are loaded into core depending on cycles.
  - 1st to 8th cycles : 16x16 KSFA
  - 9st to 10th cycles : 16x16 GUARDIAN
  - 11st to 12th cycles : 16x16 PLUS7

Calculation Results (CBC Evolutions over Cycles)

- The core following calculations from 1st to 12th cycles were performed using STREAM/RAST-K code system based on NDRs and measured burnup data for each cycle.
- For each cycle, the core depletion calculation was performed up to the measured cycle burnup
- The depletion calculations were performed up to the same burnups provided by KHNP for each cycle. But the depletion calculations in the NDR were performed down to 15ppm for 1st , 3rd , 4th cycles while to 10ppm for the other cycles .
- Maximum differences in CBC from NDR values :
  - STREAM/RAST-K : 59 ppm
  - DeCART2D/MASTER : 71 ppm

Calculation Results (CBC Evolutions over Cycles)

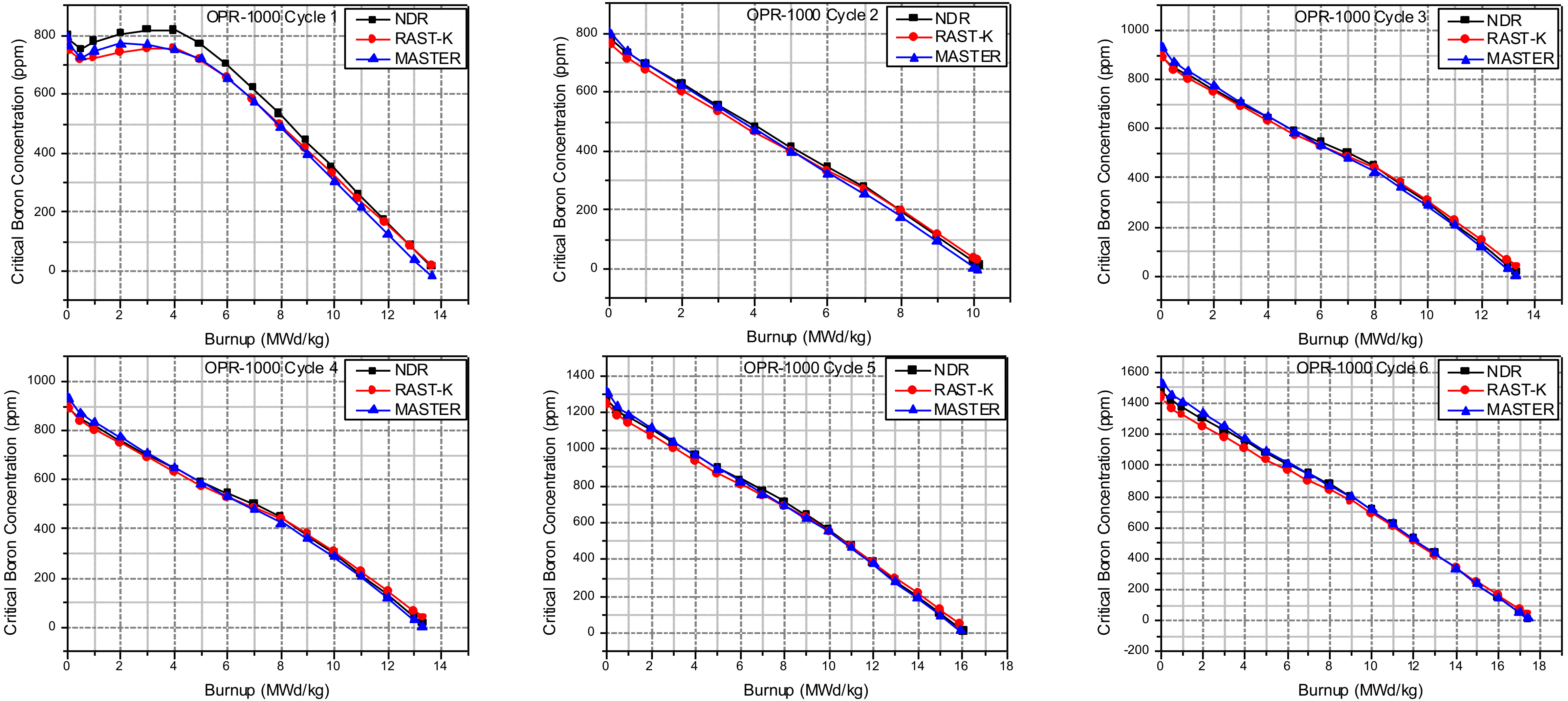

Fig. 6 Comparison of changes of CBC for 1st - 6th cycles

Calculation Results (CBC Evolutions over Cycles)

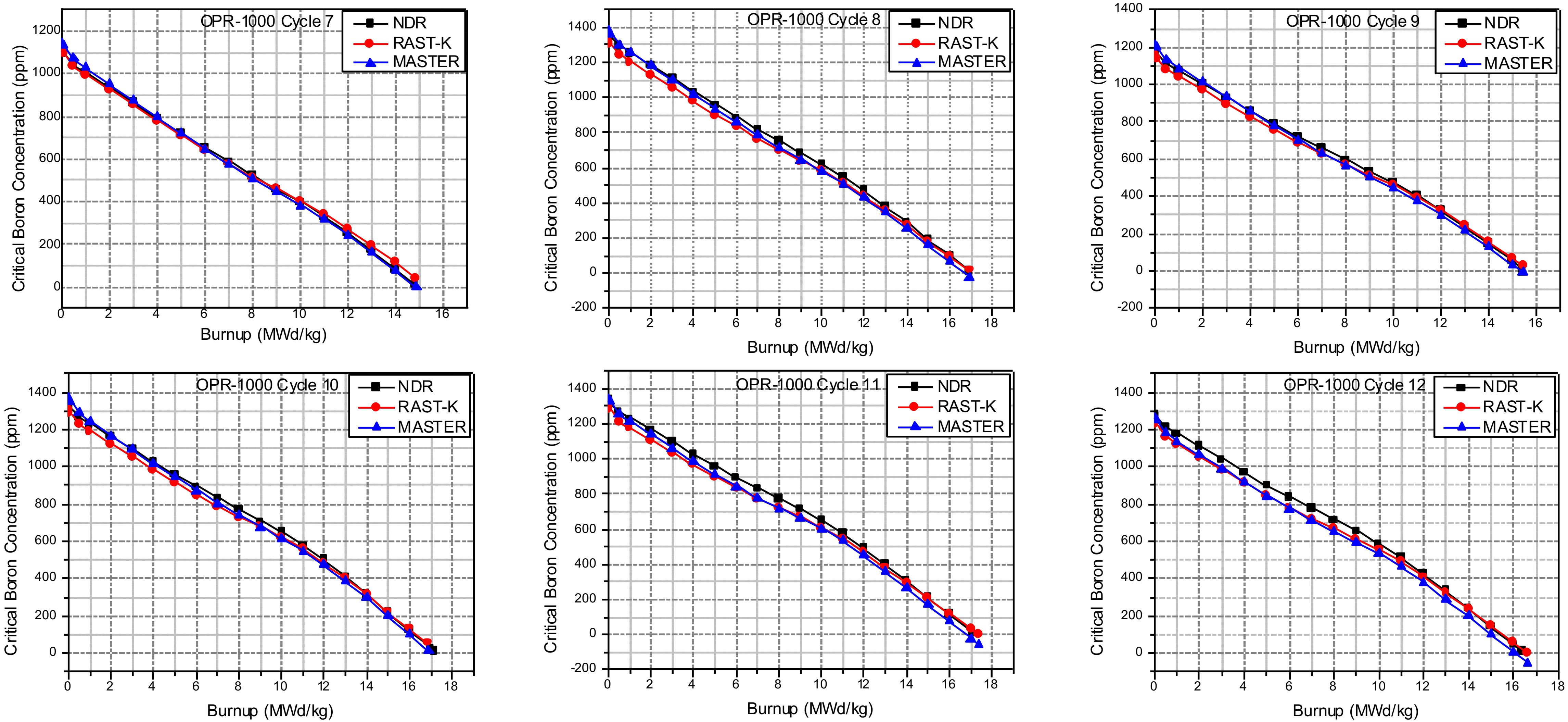

Fig. 7 Comparison of changes of CBC for 7th - 12th cycles

Calculation Results (Assembly-wise Power)

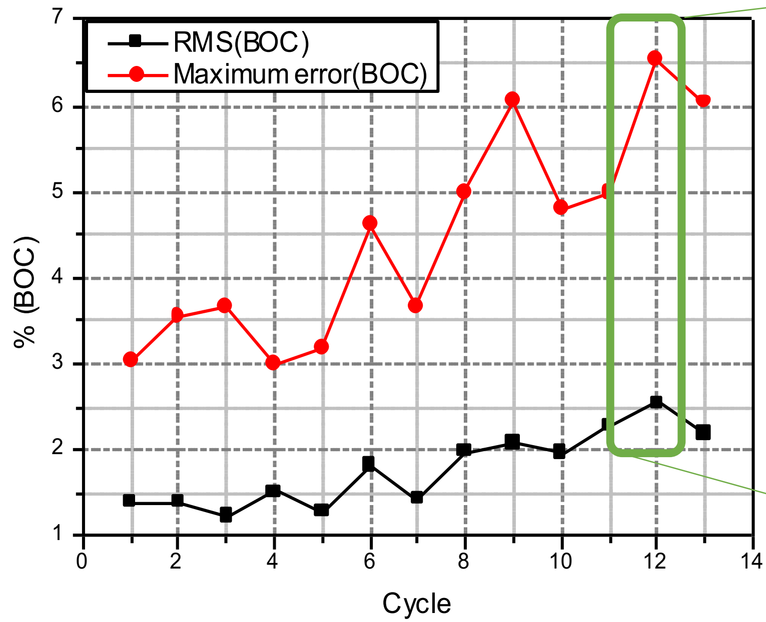

Fig. 8 RMS and Maximum error comparison with NDR and RAST-K

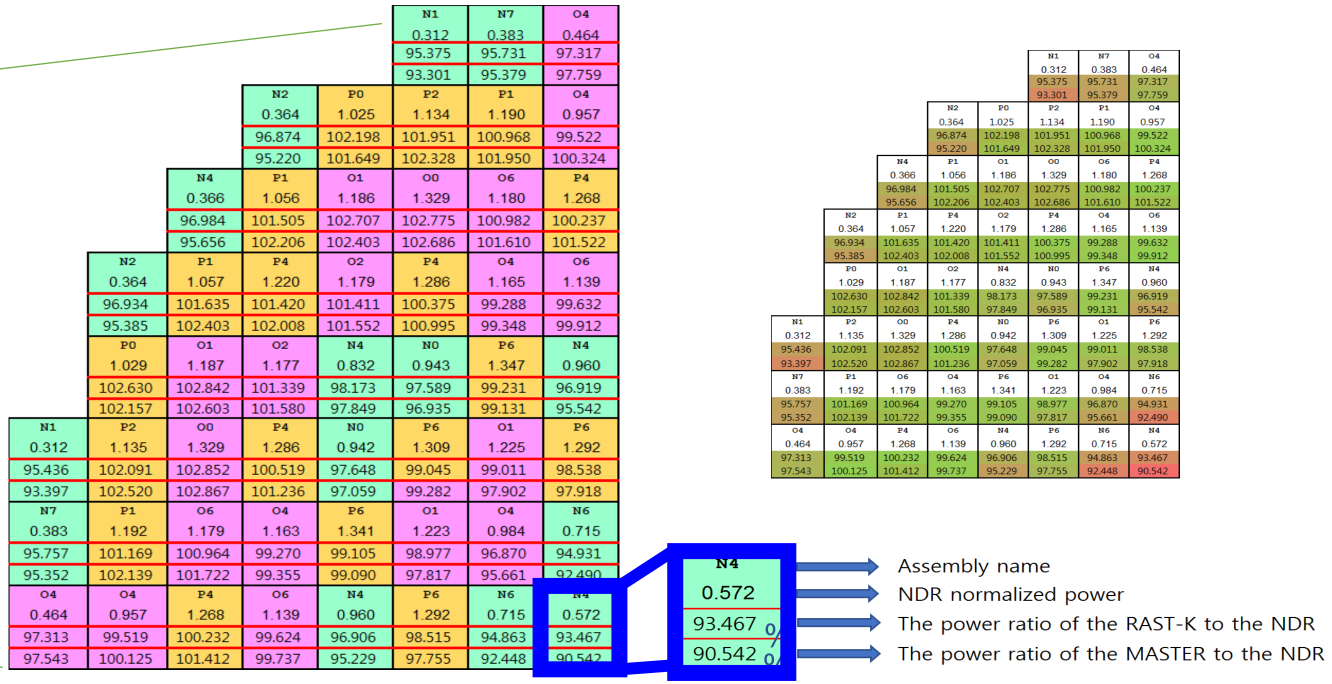

Fig. 9 Comparison the assembly-wise power distributions obtained with the NDR, RAST-K and MASTER at 12th cycle.

- The assembly-wise power maximum error with respect to NDR values is about 6.5 % at 12th cycle.
- The maximum RMS(Root Mean Square) for assembly power is about 2.5 % at 12th cycle.
- Fig. 9 shows the assembly-wise power distributions obtained with the NDR, RAST-K and MASTER at 12th cycle.

Calculation Results (Axial Burnup Profiles)

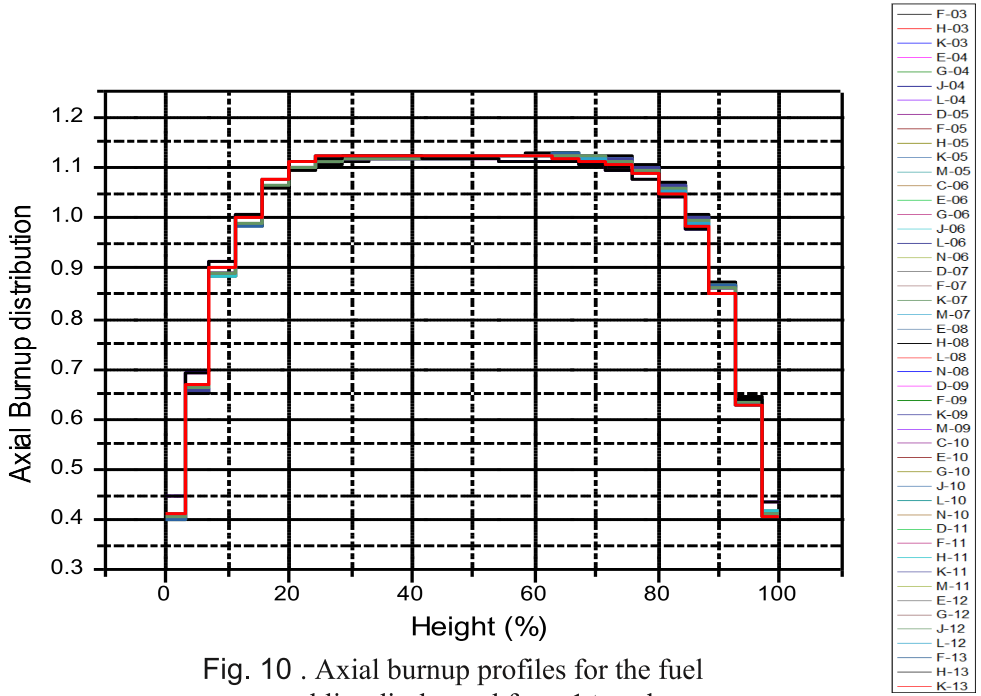

Fig. 10 . Axial burnup profiles for the fuel assemblies discharged from 1st cycle

- The axial burnup profiles for the 24 axial nodes are generated through the core follow calculation for all the assemblies discharged from 1st to 12th cycles.
- The axial burnup profiles for the 24 axial nodes are renormalized for the other 24 axial node divisions in which two end nodes occupy 2.8 % of the total length and each of the other 22 nodes occupies 4.29 %.
- As shown in this figure, the axial burnups of the low regions are higher than those of the top regions due to the higher moderator densities of the lower regions and the axial burnups of the end regions are lower than the central regions due to the large neutron leakages in the end regions.

Calculation Results (Criticality Analysis)

Criticality analysis by using STARBUCS

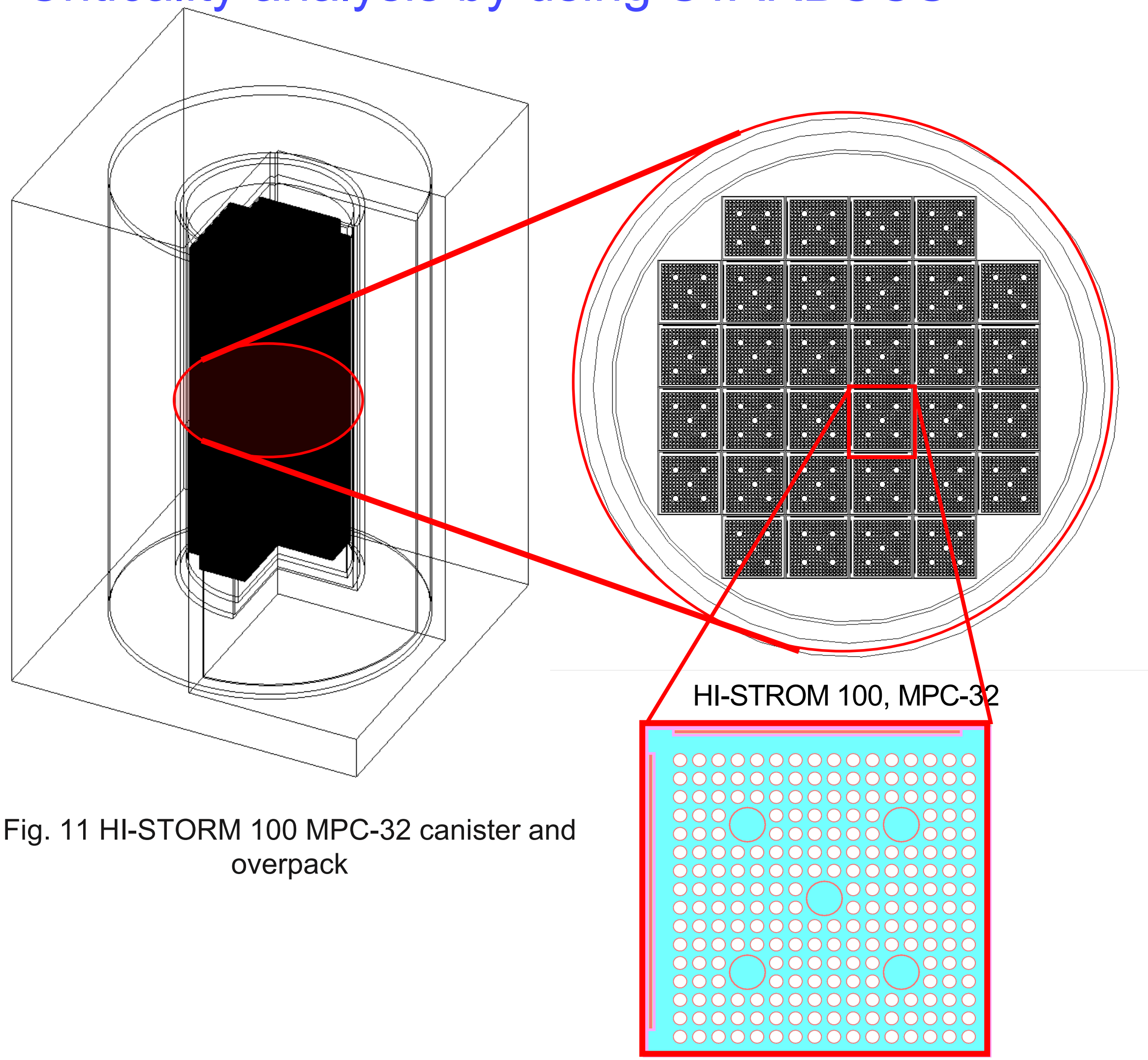

Fig. 11 HI-STORM 100 MPC-32 canister and overpack

- HI-STORM 100 system consists of a helium-pressurized stainless canister that is loaded into a vertical steel-lines concrete overpack.
- This canister applied to burnup credit and additional boron credit in the water.
- Fig. 11 represent the configuration of the HI-STORM 100 cask geometry. The detailed specification of the HI-STORM 100 cask system are taken from the final safety analysis report of HI-STORM
- The criticality analysis using STARBUCS sequence was performed for 758 axial burnup profiles.
  - Specific power : 37 MWd/kg
  - Enrichment : 4 wt%
  - Cooling time : 8.8 y
- In the criticality calculation, we used 2000 active cycles, 100 inactive cycles and 1000 particles for each cycle.
- Maximum standard deviation is 14 pcm.

Calculation Results (Criticality Analysis)

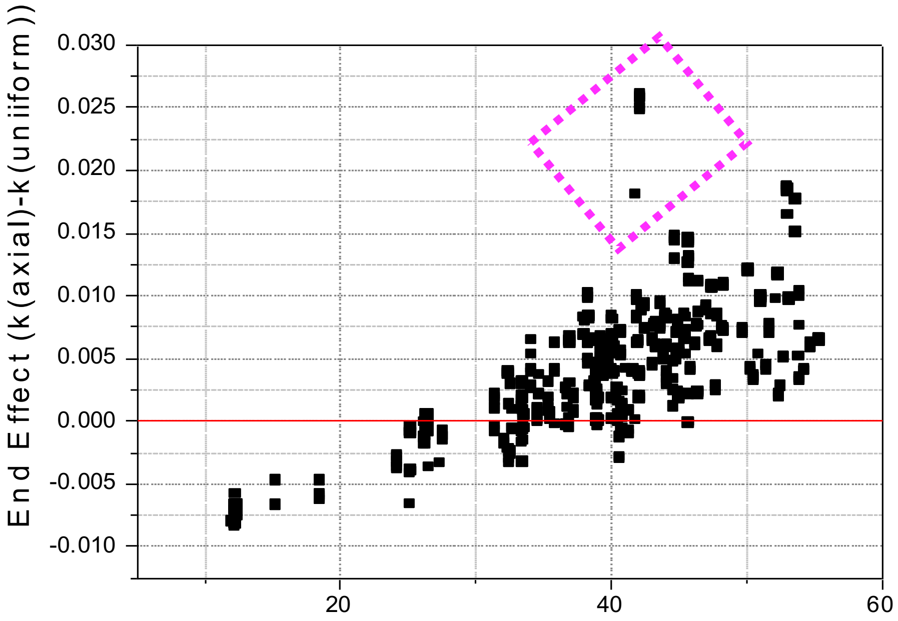

Fig. 12 Distribution of the end effect versus discharge burnup

Table 3 Average and maximum end effects in each group

Group of Burnup (MWd/kg)	Number of FAs	Average Burnup (MWd/kg)	Average End Effect	Maximum End Effect
1 (>50)	92	52.5202	0.00854	0.01858
2 (46 - 50)	76	47.6596	0.00719	0.01106
3 (42 - 46)	192	44.0184	0.00766	0.02588
4 (38 - 42)	171	39.7383	0.00439	0.01817
5 (34 - 38)	59	35.8534	0.00209	0.00661
6 (30 - 34)	60	32.9968	0.00029	0.00379
7 (26 - 30)	33	26.6903	-0.00060	0.00047
8 (22 - 26)	19	25.0614	-0.00240	-0.00050
9 (18 - 22)	8	18.6339	-0.00626	-0.00594
10 (14 - 18)	3	15.3707	-0.00488	-0.00488
11 (10 - 14)	45	12.3854	-0.00748	-0.00671

- Table 3 summarizes the number of the assemblies for each group, the average burnup, the estimated average end effects and maximum end effects for each group.
- From Table 3, it is shown that the average end effect becomes positive from 6th burnup group at 30-34 MWd/kg and it increases as burnup.
- The largest number of discharged fuel assemblies are in the 3rd burnup group having 42 to 46 MWd/kg and this burnup group has average and maximum end effects of 0.766 and 2.59 % Δk, respectively.
- It is noted that this 3rd burnup group has the largest maximum end effect. The large end effects in the 3 and 4 groups occur at the eleven assemblies having axial blankets.

Calculation Results (Criticality Analysis)

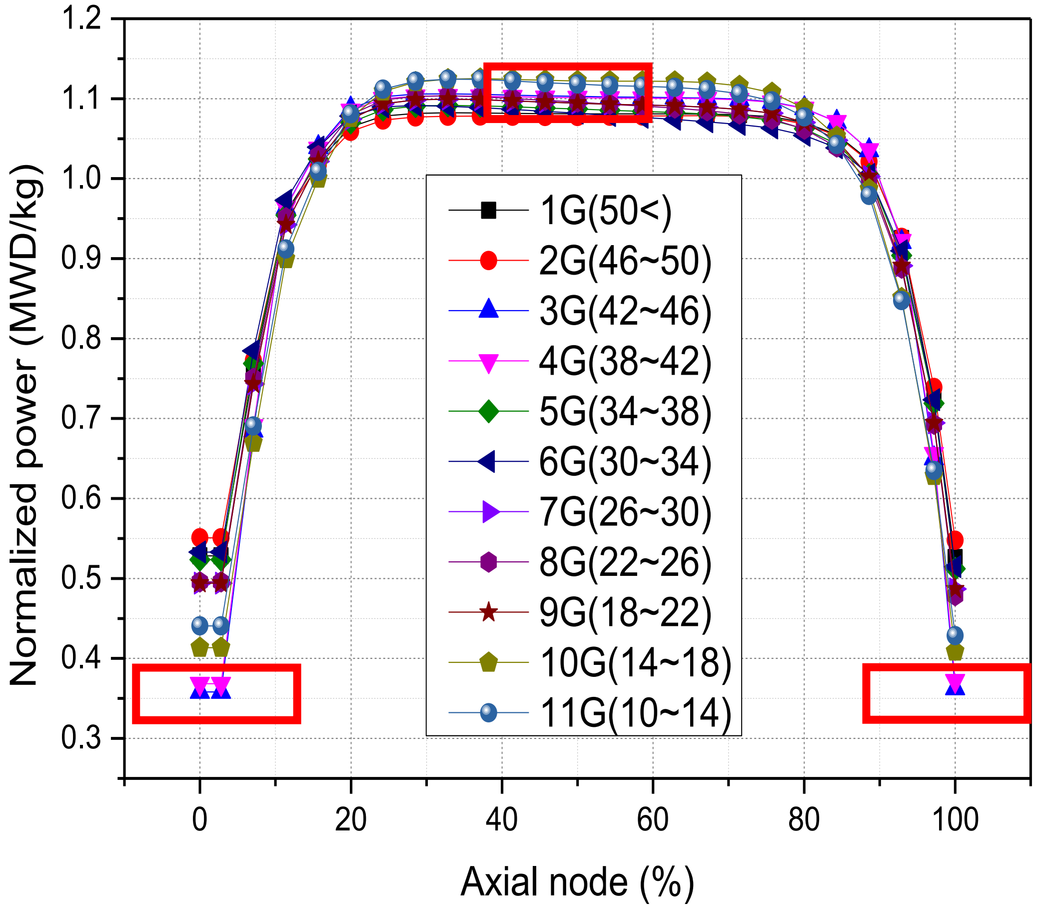

Fig. 13 Comparison of the bounding axial distributions for burnup group

- Fig. 13 shows the bounding axial burnup profiles giving a largest end effect for each burnup group.
- It is noted from Fig. 13 that for the burnup groups containing largest number of spent fuel assemblies (i.e., 3rd and 4th burnup groups), the bounding axial burnup profiles have very low normalized burnups of ~0.35 in the top and bottom nodes.
- These bounding axial burnup profiles correspond to the assemblies having axial blanket (but we did not consider the lower uranium enrichment for axial blankets in criticality calculations).

Conclusion

- In this work, the bounding axial burnup profiles for the spent fuels discharged from HANBIT Unit 3 were evaluated by estimating the end effects with STARBUCS sequence for MPC-32 canister.
  - ➢ This evaluation was performed using 758 axial burnup profiles which were produced through core follow calculations with STREAM/RAST-K core analysis code system for 1^{st} to 12^{th} cycle cores of HANBIT Unit 3.
  - ➢ The criticality calculations characterized the average and maximum end effects for the eleven burnup groups.

- From the analysis, it was found that
  - ➢ The positive average end effects start to occur from the burnup group of 30 to 34 MWd/kg
  - ➢ The maximum end effect of 2.58 % Δk occurs in the burnup group (42 - 46 MWd/kg) containing largest number of spent fuels. → This large end effects occur in the spent fuel assemblies having axial blanket.

- In the future, we will consider the effects of lower uranium enrichment in axial blanket on the end effects.

Thank you for your attention.