Journal of Energy

VOLUME 65 - 1 | 2016

journal homepage: http://journalofenergy.com/

Nuclear and thermal hydraulic calculation of a representative I^{2}S-LWR first core

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ABSTRACT

The Integral Inherently Safe Light Water Reactor (I^{2}S-LWR) concept developed by Georgia Tech is a novel PWR reactor delivering electric power of 1000 MWe while implementing inherent safety features typical for Generation III+ small modular reactors. The main safety feature is based on integral primary circuit configuration, bringing together compact design of the reactor core (121 fuel assembly), control rod drive mechanism (CRDM), 8 primary heat exchangers (PHE), 4 passive decay heat removal systems (DHRS), 8 pumps, and other integral components. A high power density core based on silicide fuel and APMT (FeCrAl) stainless steel cladding is selected to achieve a high thermal power. Initial representative first core nuclear design is proposed by Westinghouse. Full core 3D depletion calculation was performed using PARCS code. The cross section library is prepared using FA2D code and verified using Polaris sequence from SCALE 6.2 beta5. The axial and radial reflectors are assumed to be homogeneous water-APMT mixtures. The axial reflectors are both assumed to be 12 inch (30.48 cm) sections composed of 30% APMT steel by volume. The radial reflector is assumed to be 90% APMT steel by volume. The reflector constants were calculated using SCALE TRITON sequence. The thermal hydraulic part of the model is based on COBRA subchannel code coupled to PARCS code. Initial depletion calculation is based on one thermal hydraulics channel per fuel assembly approach. The hot fuel assembly is determined using separate pin-by-pin COBRA subchannel model and pin power reconstruction data from PARCS. The objective of the paper is demonstration of LWR design methodology applicability to silicide fuel and identification of possible improvements in the first core design.

Keywords: *I^{2}S-LWR, core design, PARCS, COBRA, SCALE, FA2D*

1 INTRODUCTION

The Integral Inherently Safe Light Water Reactor (I^{2}S-LWR) concept [1][2] developed by team led by Georgia Tech is a novel PWR reactor delivering electric power of 1000 MWe while implementing inherent safety features typical for Generation III+ small modular reactors. The main safety feature is based on integral primary circuit configuration, bringing together compact design of the reactor core (121 fuel assembly), control rod drive mechanism (CRDM), 8 primary heat exchangers (PHE), 4 passive decay heat removal systems (DHRS), 8 pumps, and other integral components. The objective of the paper is to calculate 3D fuel depletion for one of the proposed 1st core concepts.

2 I²S-LWR FIRST CORE CONFIGURATION

The I²S-LWR core contains 121 fuel assembly (FA) with active height of 365.76 cm which is typical value for standard 2-loop PWRs [3]. The major difference compared to standard PWR FA is its higher power rating, giving I²S-LWR core thermal power of 2850 MWth. To accommodate such power increase, specific design features are introduced, such as 19x19 square pitch lattice, U_3Si_2 fuel and advanced stainless steel cladding and grids. The denser FA matrix with increased heat transfer surface area will provide slightly higher average heat flux compared to typical 4-loop PWR core which benefits DNB performance. Novel, high-conductivity, silicide fuel allows a reasonable margin against fuel melting during hypothetical accidents giving core dimensions compatible for integral configuration (Table 1).

Silicide fuel was selected as the primary option for several reasons, most important are higher heavy metal (H/M) ratio (about 17%) and higher thermal conductivity, compared to UO_2 fuel. These characteristics are enhancing fuel cycle and operational performance. Current design includes smaller pellet-gap compared to initial one (enabled by better understanding of fuel swelling under irradiation) and elimination of inner voids in pellets (Table 2). Safety considerations dictate the choice of advanced cladding steels (APMT), which must withstand high temperature (above 1200 °C) steam-water mixture without high oxidation rates and hydrogen generation customary for zirconium alloys. The corrosion resistance can be enhanced from properly tailoring the steel composition, together with mechanical properties (even compared to Zircaloy) and higher thermal conductivity under irradiation. On the other hand, some of the isotopes in the steel (especially Fe and Cr) have high neutron absorption cross-sections which lead to a significant reactivity penalty compared to Zr. Silicon carbide cladding has better neutronic properties than Zr and represents the secondary option for the I²S-LWR cladding material.

Table 1. Fuel property comparison: U_3Si_2 vs UO_2

Fuel	U_3Si_2	UO_2
Theoretical density (TD) (g/cm³)	12.2	10.98
HM TD (g/cm³)	11.3	9.68
Thermal conductivity (W/mK)	9 - 20 (300 - 1200 °C)	5 - 2 (300 - 2000 °C)
Specific heat (J/kgK)	230 - 320 (300 - 1200 °C)	280 - 440 (300 - 2000 °C)
Melting point (unirradiated) °C	1665	2840

Table 2. Fuel assembly main characteristics

Lattice type	19×19, square
Fuel/Cladding material	U_3Si_2/ APMT
Fuel rods per assembly	336
Fuel rod outer diameter (mm)	9.144
Cladding thickness (mm)	0.406
Pellet-clad gap width (mm)	0.1143
Pellet outer diameter (mm)	8.1026
Fuel rod pitch (mm)	12.116
Assembly pitch (mm)	231

Most of the core design effort within I[2]S-LWR project was spent on equilibrium core. We have decided to analyze first core design. Initial representative first core nuclear design is proposed by Westinghouse [3]. Full core 3D depletion calculation was performed using PARCS code [4]. The cross section library is prepared using FA2D code and verified using Polaris sequence from SCALE 6.2 beta5. Small benchmarking at Fuel Assembly (FA) level has been performed using SCALE TRITON and Polaris [5], SERPENT 2.1 [6], and FA2D. Both infinite multiplication factor and cross section data were reasonably similar (Figure 1).

The axial and radial reflectors are assumed to be homogeneous water-APMT mixtures. The axial reflectors are both assumed to be 12 inch (30.48 cm) sections composed of 30% APMT steel by volume. The radial reflector is assumed to be 90% APMT steel by volume. The reflector constants were calculated using FA2D and SCALE TRITON sequence.

The thermal hydraulic part of the model is based on COBRA subchannel code coupled to PARCS code. Initial depletion calculation is based on one thermal hydraulics channel per fuel assembly approach.

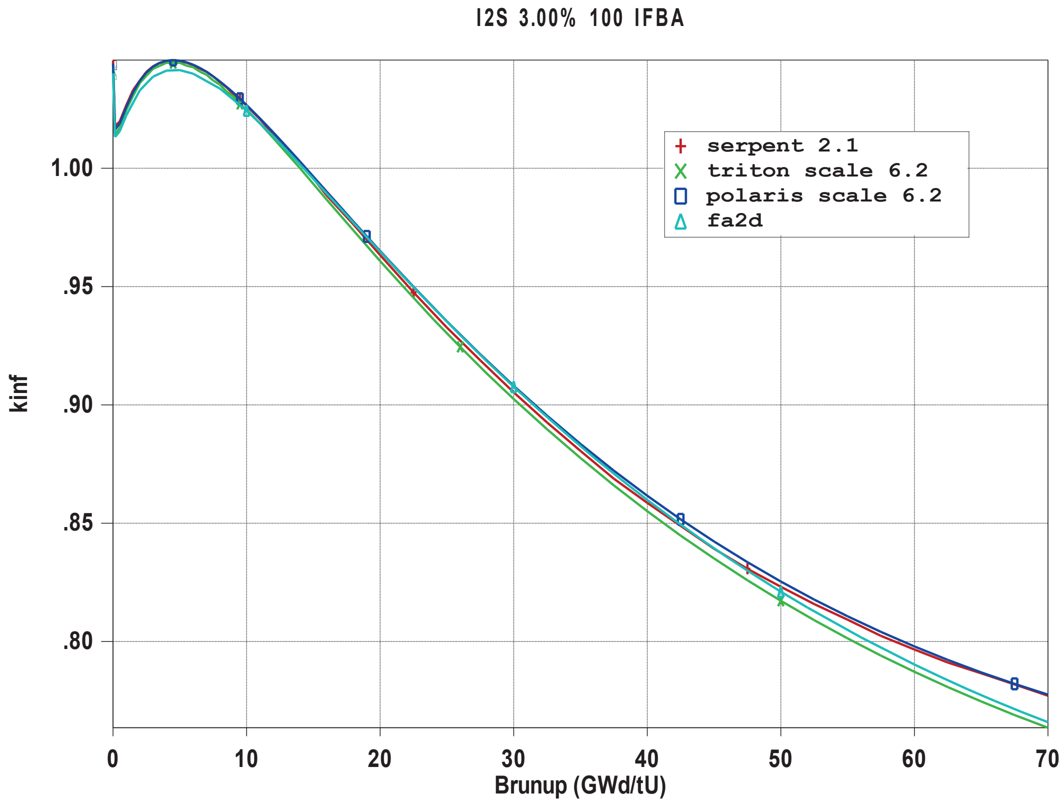

Figure 1: Fuel assembly calculation and cross section preparation benchmarking

Initial core loading scheme is shown in Figure 2. FAs labeled with A have enrichment of 2.5%, labeled with B have enrichment 3.0% and assemblies labeled with C have enrichment 4.0%. All fuel assembles have axial blankets enriched at 2.5%. IFBA burnable absorbers were used in the core design (batch A has 84 IFBAs, batch B has 100 and 156 IFBAs, and batch C has 100 IFBAs). Classical Westinghouse scheme with IFBA layer in the middle of the assembly was initially used (6-6-120-6-6 inches), Figure 3. Due to higher than wanted Axial Offset (AO) values the IFBA layer is shifted toward bottom for 6 inches, and then optimization is performed for additional 6 inches IFBA segment at the top. If we take geometry shown in Figure 3 as a reference it is possible to add additional 6 inches to the top of shifted 120 inches layer and then perform sensitivity calculation by decreasing that length in steps of 1 inch from 6 inches to 0 inch. That was done in our calculation uniformly across all IFBA FAs. In original WEC design additional 2 inches were present for 2.5%

84-IFBA and 3.0% 100-IFBA assemblies, and additional 4 inches for 3.0% 156-IFBA and 4.0% 100-IFBA assemblies (that was obviously result of additionally performed nuclear peaking factor optimization done by WEC).

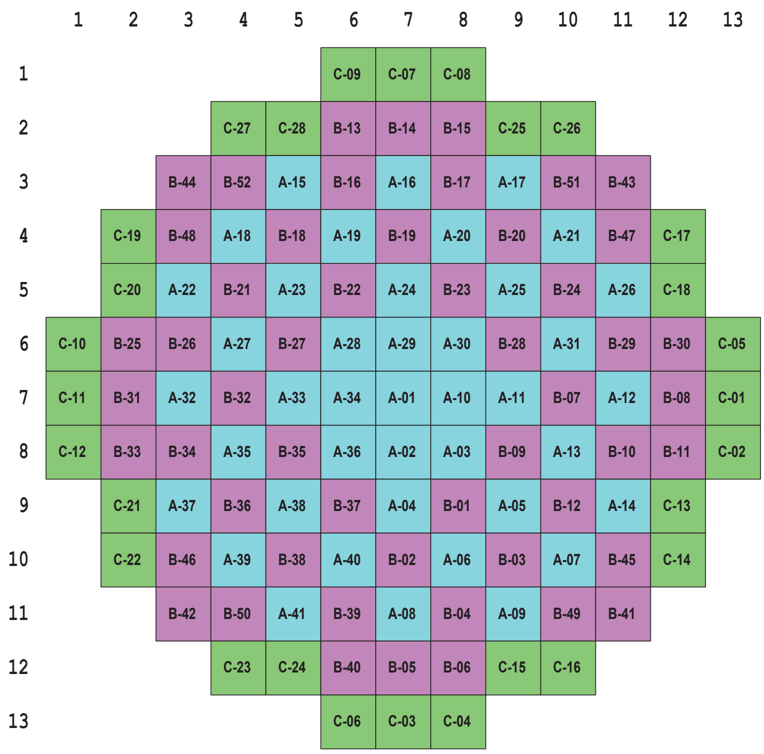

Figure 2: I^{2}S first core loading pattern

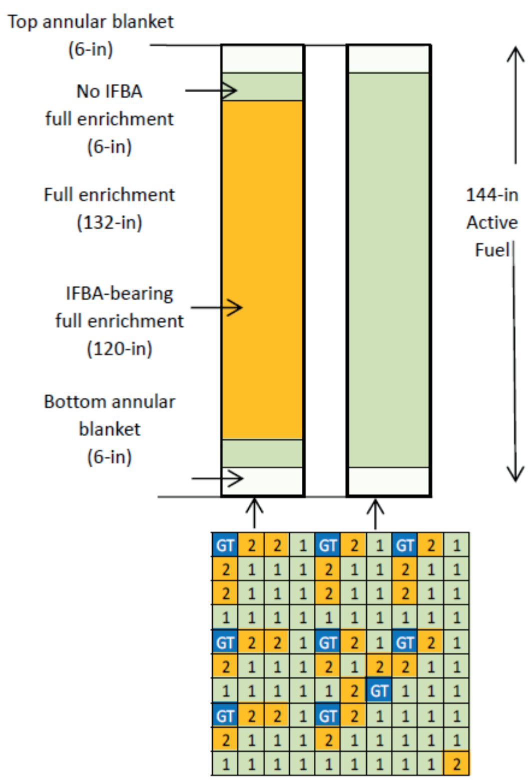

Figure 3: Axial representation of Westinghouse IFBA and non-IFBA fuel assembly

3 ANALYSIS AND RESULTS

Initial PARCS calculation was performed for loading scheme proposed by Westignhouse, but with classical symmetric IFBA layer. In following figures that case was labeled with ***ori***. For all additional cases two numbers were used to describe analyzed configuration. First number is length of node without IFBA layer at original FA enrichment and second number is length of node with IFBA layer (cases ***6-6, 5-7, 4-8, 3-9, 2-10, 0-12***). The sum of two numbers is always 12 inches. The case ***6-6*** is with additional 6 inches IFBA layer (total length 126 inches) and ***0-12*** is case with shift of original 120 inches IFBA layer for 6 inches down. Label ***wec*** means Westinghouse results are used when available.

Boron concentration versus burnup is shown in Figure 4. It is clear that small variation in IFBA layer length and position has limited influence on overall reactivity of the core. In addition results compare reasonably with WEC results. In Figure 5, showing AO versus burnup, it is clear that AO is very sensitive to variation in IFBA layer length and position. AO can be very negative (-20%) for classic central IFBA layer or very positive (+25%) if IFBA layer is just shifted toward bottom. Our case ***4-8*** is close to WEC case with variation of IFBA layer length depending on position, enrichment and number of IFBAs.

The results for total and axial peaking factors are shown in Figure 6 and Figure 7. The variation in IFBA layer length again has significant influence on power peaking values both before and after IFBA depletion. The differences between our ***4-8*** and ***wec*** case are small for axial peaking factor and rather large, early in cycle, for total peaking factor. That is due to radial peaking factor that is additionally optimized in WEC case by variation of IFBA length depending on enrichment and number of IFBAs. It should be mentioned that radial core peaking factor (not shown) is rather insensitive to the described variation in IFBA layer.

The corresponding relative axial power distribution for selected cases for 0.15 and 13 GWd/tU are shown in Figure 8. The influence of IFBA axial variation is significant before IFBA depletion and small later. Using the proposed scheme it is possible to keep it within allowable range.

Rather symmetric radial power and burnup distributions at EOL conditions are presented in Figure 9 and Figure 10. The temperature of fuel rod centre line (one average rod per FA) is shown at EOL in Figure 11. It is as expected lower than for UO_2 fuel due to better thermal conductivity of used fuel. The results of pin power reconstruction calculation for beginning of life (0.15 GWd/tU, first core quadrant) and for end of life (13 GWd/tU, forth core quadrant) are shown in Figure 12 and Figure 13, respectively. Rather large variation in power distribution is initially present, but within allowable values. For end of life conditions due to depletion of both fuel and IFBA usual, more flat, distribution is obtained. The obtained pin powers can be used for detailed hot fuel assembly calculation in COBRA subchannel code.

4 CONCLUSION

The objective of the paper is to demonstrate applicability of our LWR design methodology to silicide fuel and to identify possible improvements in the first core design. We were able to apply classical PWR core design calculation methodology, with some adjustments of fuel and cladding thermal properties, to I^2S-LWR calculation. Obtained length of the cycle and most of the calculated core parameters are acceptable for an initial first core development. Axial power distribution was proven to be sensitive to change in initial core axial properties. The variation in IFBA layer position and size, proposed by WEC, can be an effective way to control axial power shape, but some additional optimization will be needed. Another observation, as expected, is a significant influence of steel reflector on overall core reactivity, and need for careful preparation of reflector cross sections.

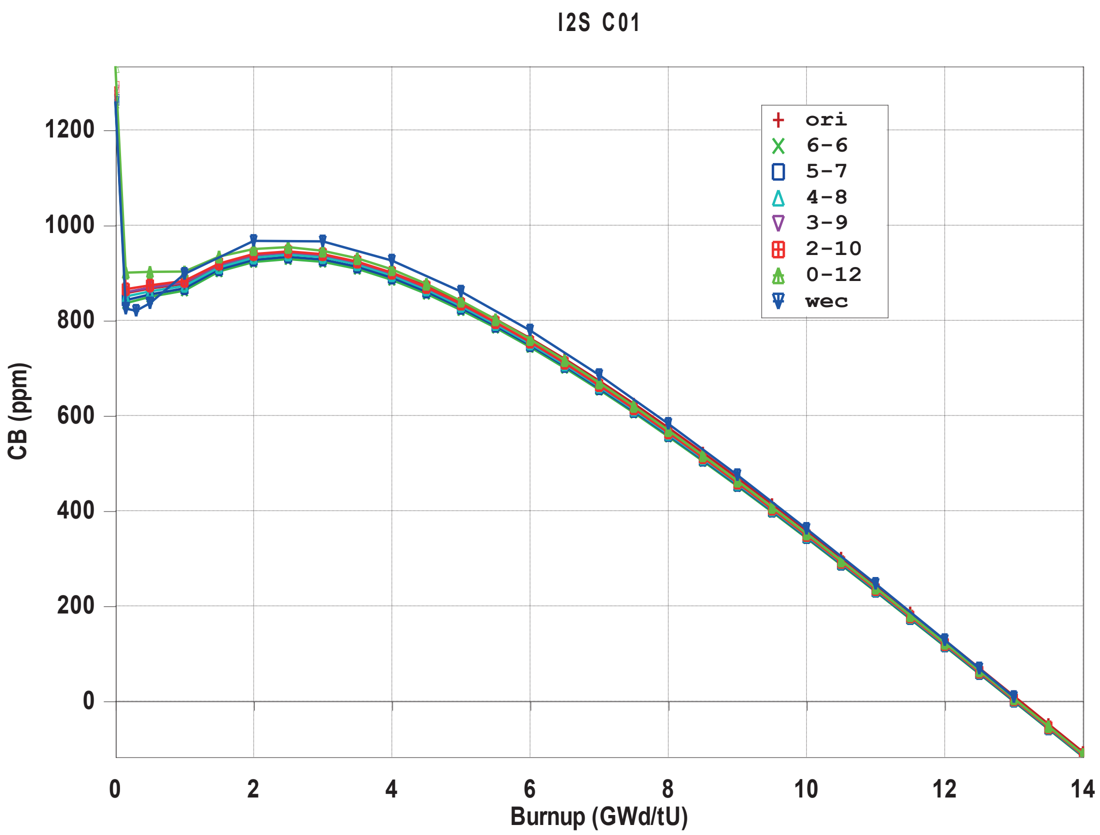

Figure 4: Boron concentration vs.burnup

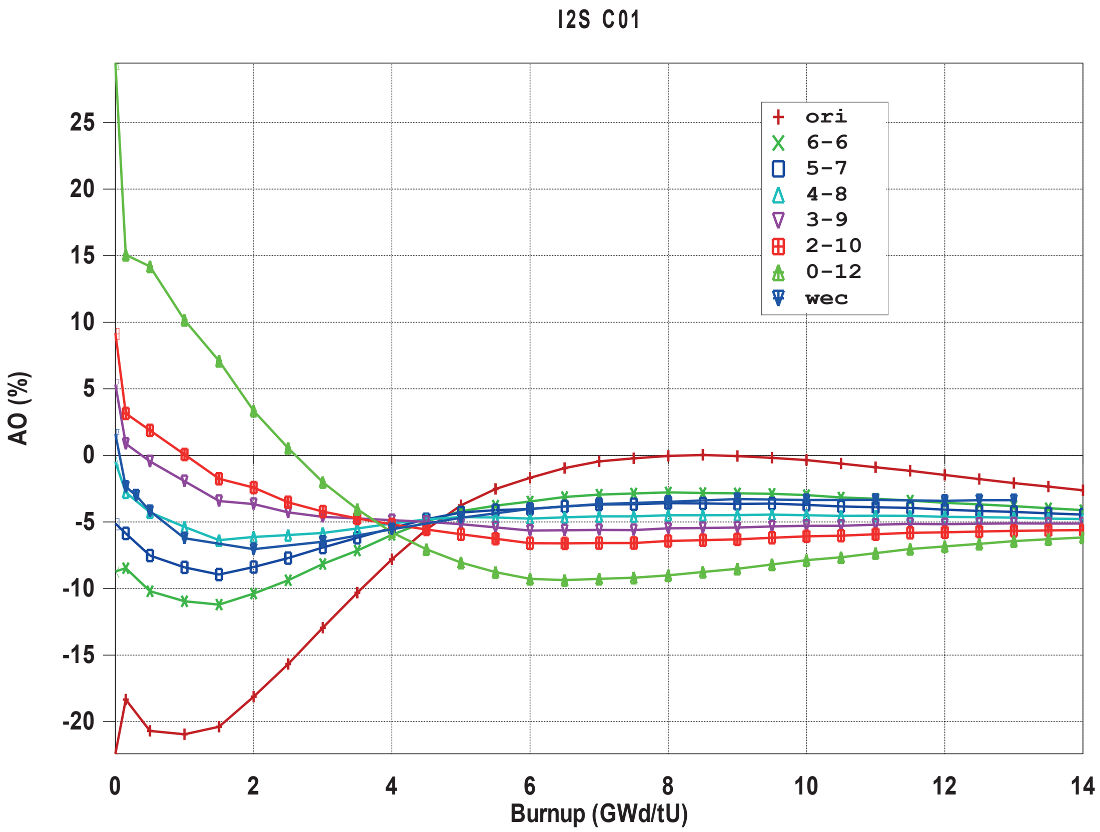

Figure 5: Axial Offset vs. burnup

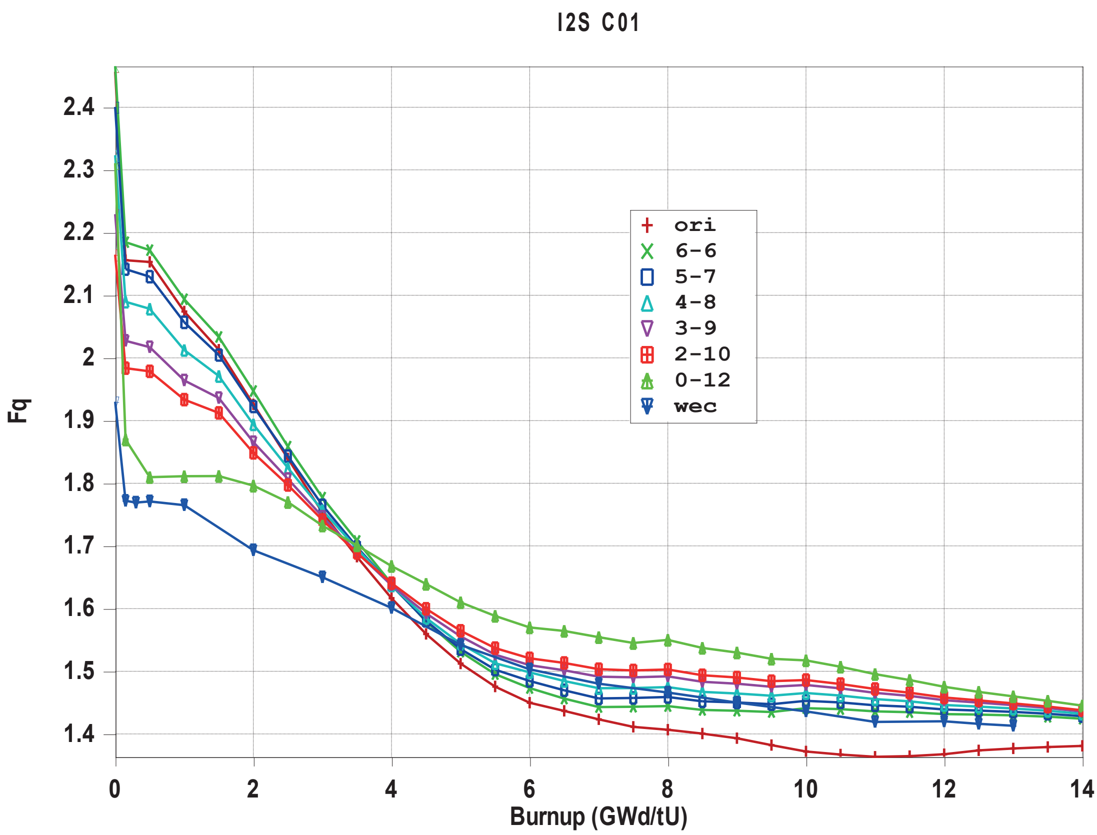

Figure 6: Fq vs. burnup

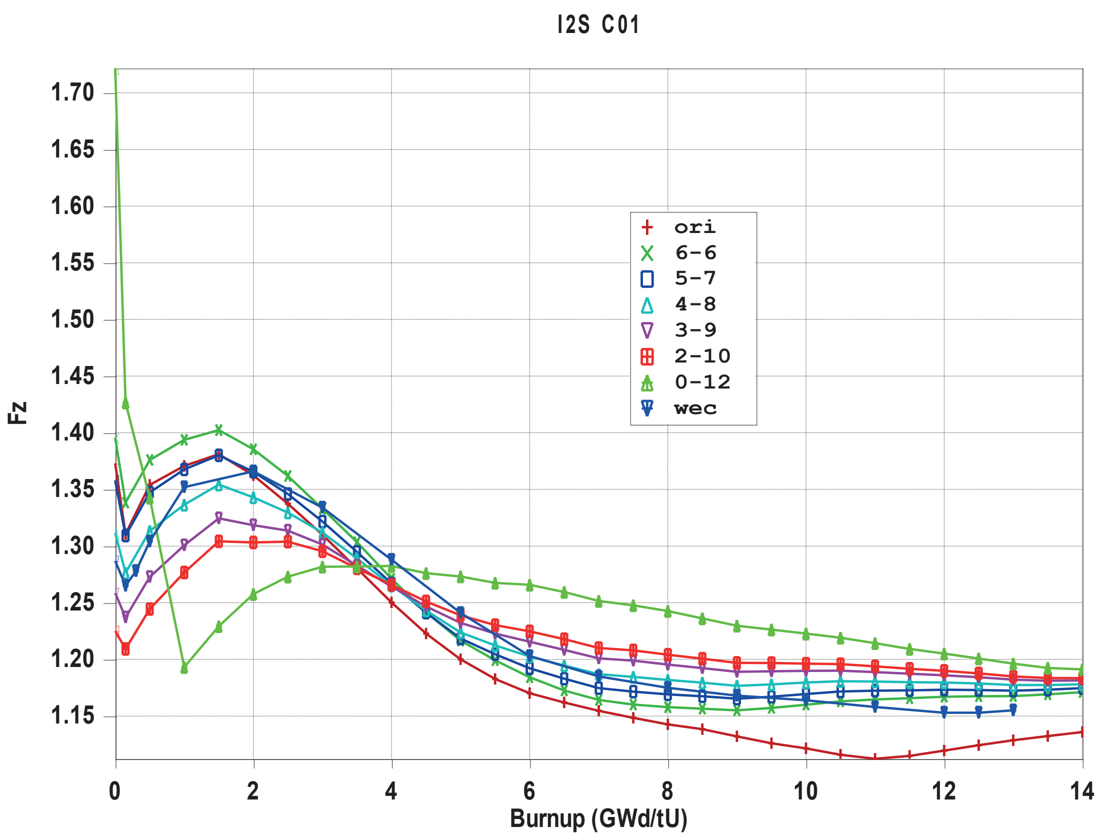

Figure 7: Fz vs. burnup

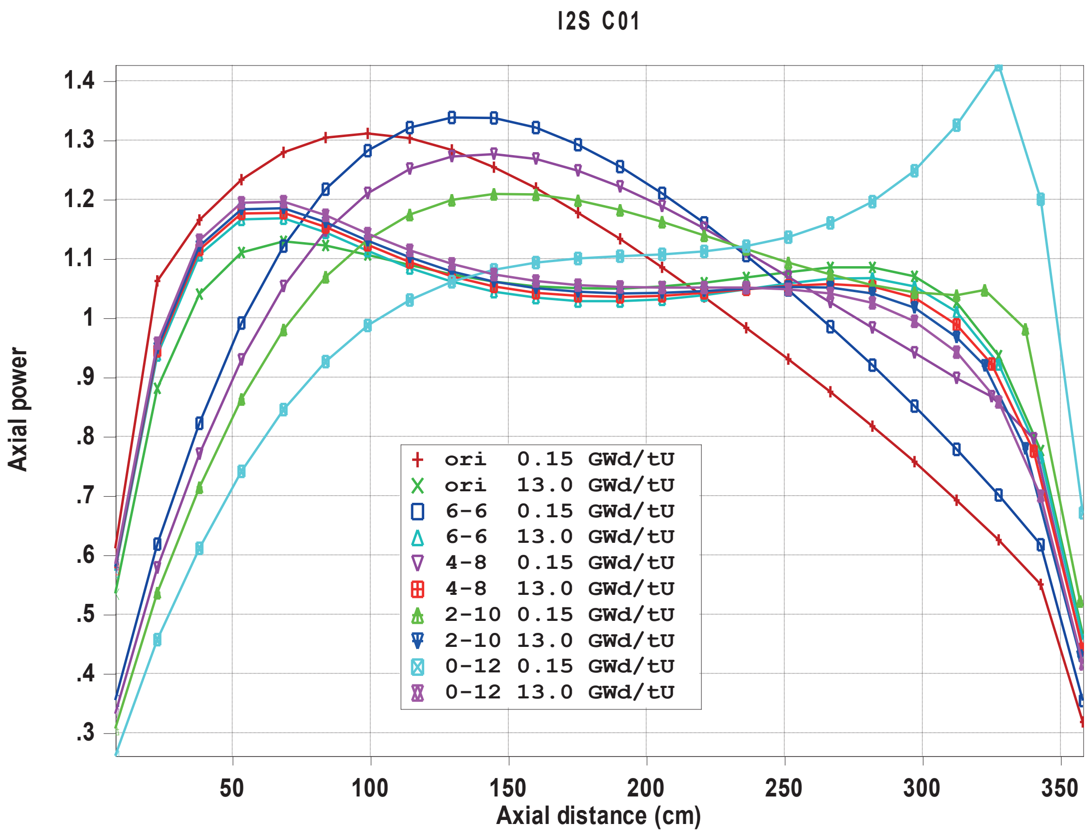

Figure 8: Axial power distribution

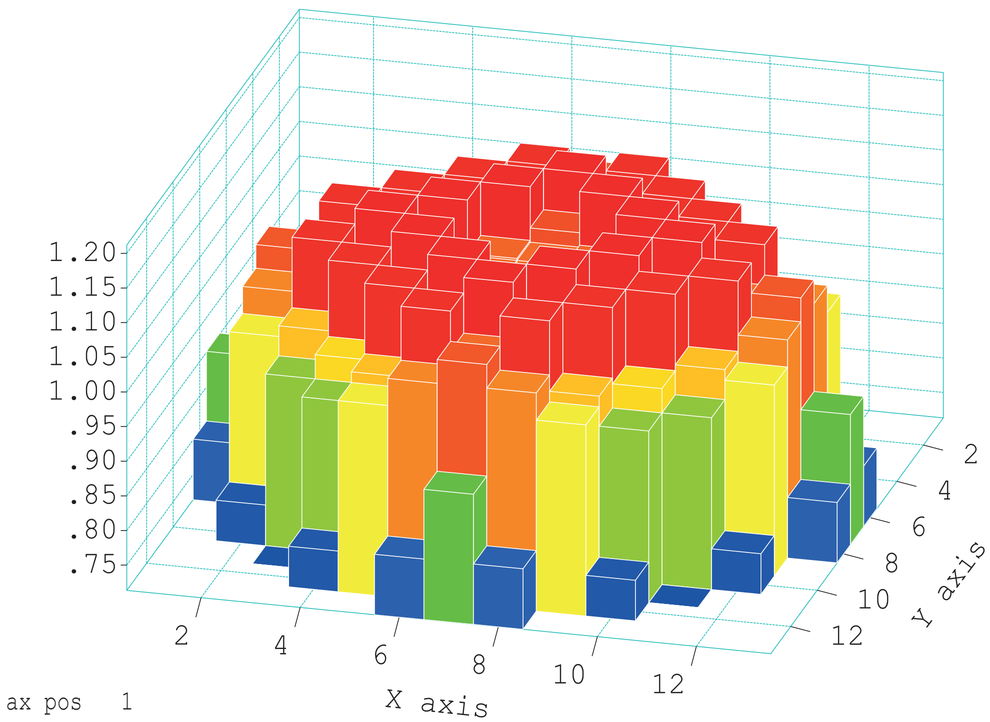

Figure 9: Relative power distribution at Bu = 13 GWd/tU

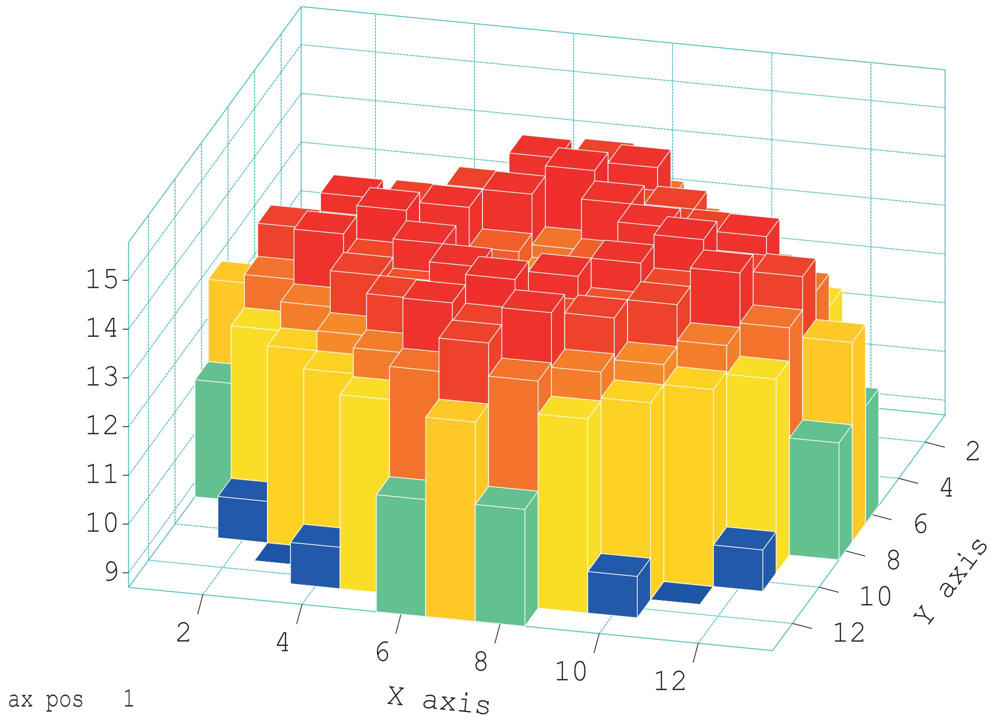

Figure 10: Burnup distribution at Bu = 13 GWd/tU

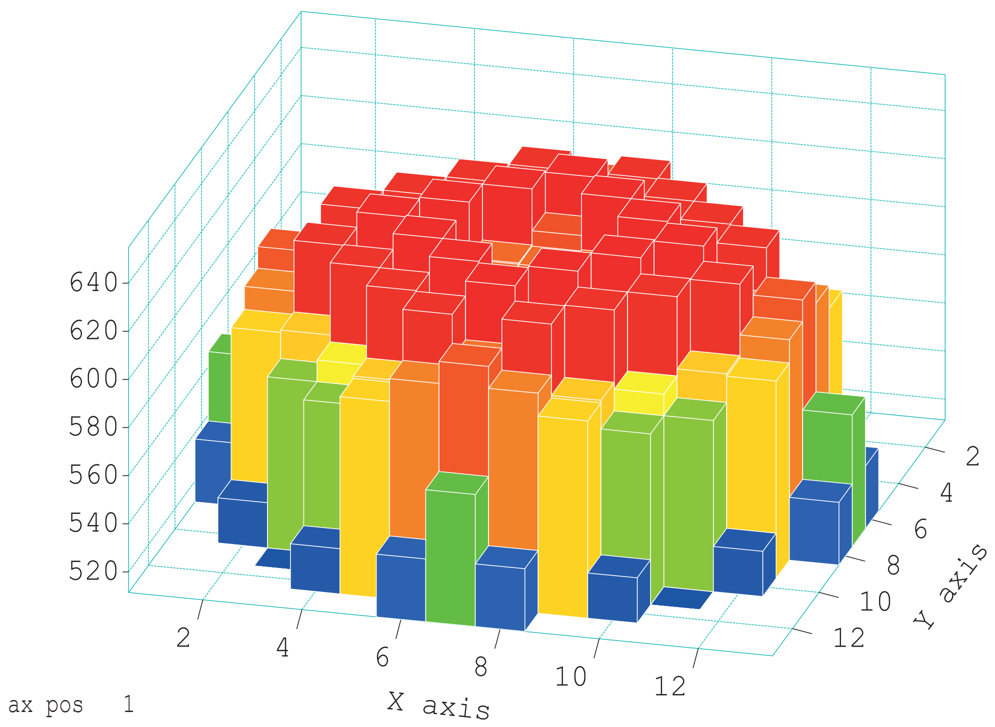

Figure 11: Fuel center line temperature distribution at Bu = 13 GWd/tU

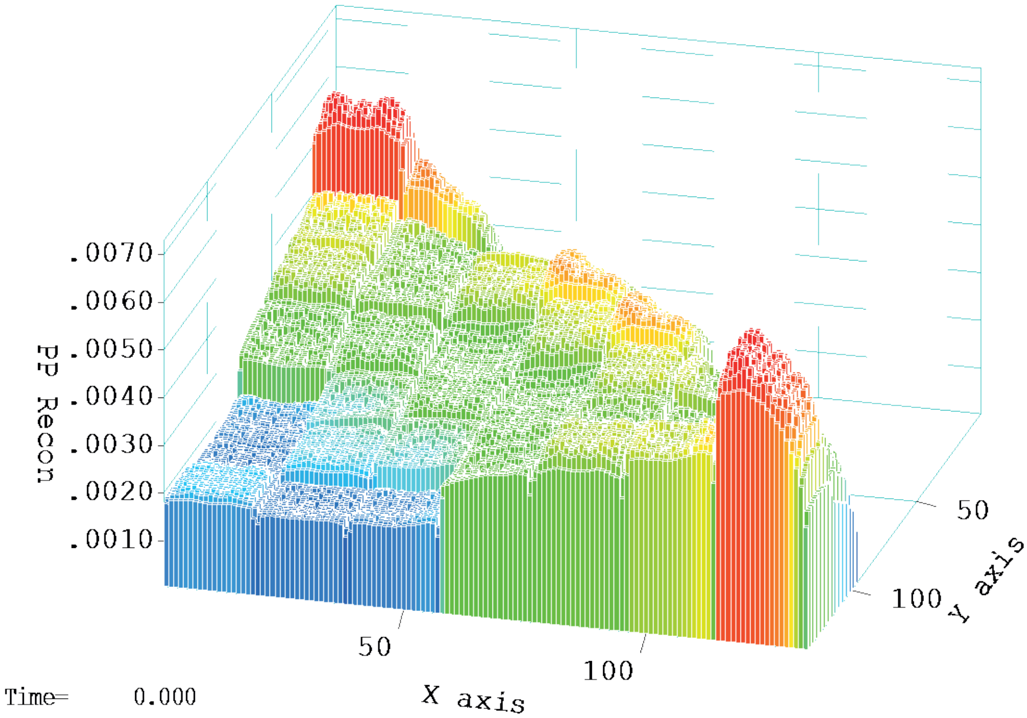

Figure 12: Relative pin powers at Bu = 0.15 GWd/tU, 1st quadrant

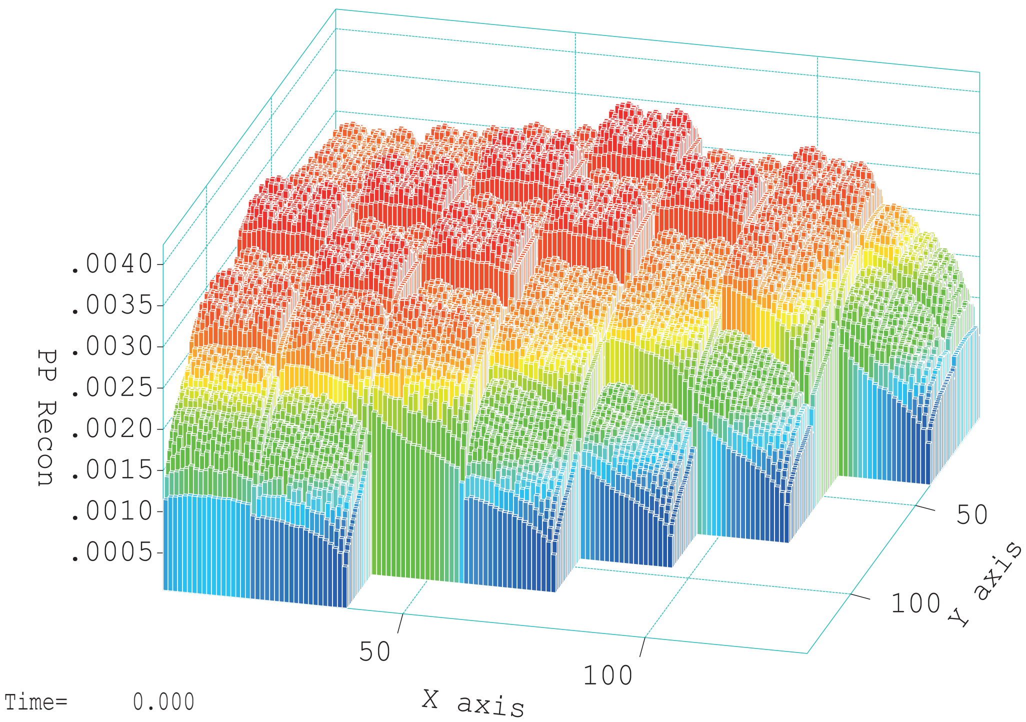

Figure 13: Relative pin powers at Bu = 13 GWd/tU, 1st quadrant

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