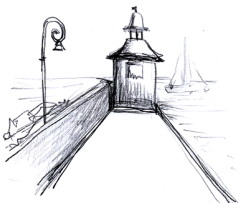

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FA2D Prediction Capability for NPP Krsko Fuel Assembly Calculation

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ABSTRACT

FA2D is 2D transport collision probability code developed at Faculty of Electrical Engineering and Computing, University Zagreb. It is used for calculation of cross section data at fuel assembly level. Usual application of obtained cross section data, including pin power reconstruction data, is within NRC's PARCS code for 3D core calculation.

This paper presents results of calculations performed for NPP Krsko 16x16 standard fuel and NGF fuel with different number of IFBA rods and for different Gd_2O_3 content. Two codes are used in calculation: FA2D and SCALE5 TRITON calculation sequence (based on discrete ordinates code NEWT). The results obtained with FA2D spectral code for reflected heterogeneous 2D fuel assembly geometry are presented for burn-up dependency of infinite multiplication factor, 2D pin power distributions, and for local peaking factor.

1 INTRODUCTION

The computer code FA2D is used for generation of cross sections and related data of pin cells and fuel assemblies at Faculty of Electrical Engineering and Computing (FEEC), University of Zagreb. The code is developed at FEEC and it solves transport equation using the collision probability method. The verification of the code has been performed by benchmark calculations at pin cell and fuel assembly level, as well as fuel management calculations for the NPP Krsko and IRIS reactors.

The 16x16 next generation fuel (NGF) and 16x16 standard fuel, containing IFBA rods and gadolinium rods as burnable absorbers, will be used for the subsequent cycles of the NPP Krsko. A generation of cross section for this type of the fuel we have performed by the calculational model of the FA2D code. We also modeled this type of the fuel by TRITON calculation sequence (based on the discrete ordinates code NEWT) of the SCALE5 computer code package. The calculational models are described in Section 2. The results obtained with two mentioned spectral codes are presented in Section 3. The conclusions of modeling NGF and standard fuel for NPP Krsko by FA2D and SCALE5 TRITON codes are given in Section 4. The Section 5 contains the references.

2 CALCULATIONAL TOOLS AND INPUT DATA

FA2D (Fuel Assembly 2Dimensional) is an advanced lattice physics code developed at the Faculty of Electrical Engineering and used for calculation of cross sections and related data for pin cells and fuel assemblies in two-dimensions (2D). The code solves transport equation using the collision probability method. The formal verification of the code predictive capability has not been performed in systematic way at the fuel assembly (FA) level, but the results of fuel management calculations obtained using the FA2D generated cross sections for the NPP Krsko [1] and IRIS reactor [2][3] are compared against the Westinghouse calculations (for NPP Krsko and IRIS) and ORNL calculations using HELIOS for IRIS benchmark core. The obtained cross section data were used within the NRC's PARCS v2.5 code and satisfactory preliminary results were obtained. Recently [4], the calculation was performed for LWR Next Generation Fuels Benchmark [5]. The benchmark was prepared by Nuclear Fuel Industries, (NFI) Ltd., Japan, with the aim to verify predictive capability in nuclear design for extended burnup regions. We compared FA2D results against published results obtained with HELIOS, CASMO and PHOENIX codes, and satisfactory results were obtained for UO2 fuel assembly with Gd burnable absorbers.

2.1 FA2D Description

The FA2D fuel pin and fuel assembly spectral code was developed mainly at the Faculty of Electrical Engineering and Computing, Zagreb, with some specific parts borrowed from other similar available codes (e.g., the ORNL arbitrary geometry package from MARSLIB [6] was used in implementation of arbitrary geometry capability of FA2D). FA2D performs flux and eigenvalue calculations in two-dimensions, and solves multi-group neutron transport equation using the method of collision probabilities. Its arbitrary geometry package similar to MARS [6] makes possible the calculations for square pin cells and fuel assemblies of current LWR design. Isotopic depletion is based on the predictor-corrector integration method. The code uses a nuclear data library with 97 energy groups based on Evaluated Nuclear Data File, ENDF/B-VI ver. 5. The current library includes around 290 nuclides with possibility to add additional ones when required. Eigenvalue calculations can be performed in the full micro-group energy structure or with a condensed group energy structure. 2D transport calculation can use homogenized cell cross sections or full heterogeneous treatment. The code performs the Dancoff calculation using a ray-tracing approach and resonance calculations for lumped fuel masses and burnable absorbers using a spatial variation function that accounts for self-shielding effects in multi-annulus and sectored material regions, similar to the one used in CPM-3 code [7]. The code performs a fundamental mode calculation to account for the effects of neutron leakage. The leakage spectrum data are then used to calculate nuclide reaction rates and to perform isotopic depletions.

2.2 TRITON/NEWT Description

The TRITON/NEWT SCALE 5.1 calculational sequence was used in preparation of the data for NPP Krsko fuel assembly with Gd burnable absorber pins and for internal verification of FA2D calculation. SCALE is a modular system comprised of numerous sets of codes and data, with a broad range of functions and capabilities. Codes are classified as functional modules, control modules, or utilities. Functional modules include different basic physics codes. Control modules operate as sequence controllers, preparing input for functional modules, transferring data, and executing functional modules in the appropriate sequence for a particular analysis type. TRITON [8] is a SCALE control module that can be used for

problem dependent cross-sectional weighting, 2D transport calculations with NEWT [9], or 2D depletion calculations through a coupling of NEWT and ORIGEN-S. Given the mixtures and cell structures defined in the input, TRITON drives the cross-section processing operations using BONAMI to perform Bondarenko calculations for resonance self-shielding in the unresolved resonance range and CENTRM/PMC for the resolved resonance evaluation. The transport solution is followed by COUPLE and ORIGEN-S calculations. In depletion mode, NEWT creates a three group weighted library based on calculated and volume-averaged fluxes for each mixture. The NEWT transport solver provides a two-dimensional, unstructured-mesh discrete-ordinates solution for multigroup neutron transport calculations. NEWT employs a mesh defined by arbitrary polygons; extended step characteristic (ESC) discretization is applied to calculate transport between sides within each computational cell. While similar in concept to the Method of Characteristic (MOC) class of transport solution, ESC provides a more rigorous spatial representation of bodies and allows significantly more control of spatial mesh refinement. The arbitrary polygon mesh developed by NEWT can be used to closely approximate curved or irregular surfaces to provide the capability to model problems that were formerly difficult or impractical to model directly with discrete-ordinates methods. In this study we used the standard 44-group ENDF/B-VI.5 library with extended nuclides set used in depletion (232 nuclides). The CENTRM method was used for resonance treatment. The quadrature order used in integration was 8, with the overall convergence criterion of 1.0e-4. The maximum order of Legendre scattering was 3 for moderator and 1 for fuel and all other materials.

2.3 Model Input Data

The data were taken from NPP Krsko (NEK) Safety Analysis Report (SAR) [10] or obtained directly from plant personnel. A full fuel assembly is modeled in all cases due to asymmetry of 16x16 configuration. The fuel assembly layout for case with 24 Gd pins is shown in Figure 1. The corresponding calculational model as used in NEWT is shown in Figure 2.

Most of the following data directly applies to FA2D. The data and assumptions used in TRITON/NEWT were as similar as possible to the ones used in FA2D. 2D heterogeneous geometry was used. At fuel cell level two additional annular regions in UO_2 pellets and no angular sectors per cell were used. For fuel pins with Gd_2O_3 four annular regions in UO_2 pellets were used and four angular sectors per cell. The total number of flat flux regions in the model with 24 Gd pins was around 1790. Fuel rod gaps and inter-assembly gaps are explicitly modeled. The IFBA layer is not explicitly modeled and corresponding amount of B-10 was located in fuel rod gap. Cold dimensions were used in calculations. A spacer material is mixed with moderator to form a homogenous mixture. A moderator density is calculated from the input temperature and system pressure using IF97 water and steam properties (internal to FA2D, external for NEWT). Numerical integration parameters used in collision probability solver determine accuracy of the procedure. The distance between parallel lines used in discretization is 0.05 cm with the number of equally spaced angles being 16. The normalized calculated volume was used in calculation. A tracking distance used in ray tracing algorithm is 6 mean free paths. Reflective boundary conditions were assumed at outside boundaries.

The neutron data library used in calculation (FA2D) is based on ENDFB 6.5 library with some isotopes taken from JENDL 3.3 library. The number of micro groups in library is 97. The fundamental mode B1 leakage spectrum was used for depletion and for calculation of 2-group cross section data. The integration is of the predictor corrector type. The burnup calculation was performed at fuel assembly level up to 60 GWd/tU (35 burnup steps). The depletion is at: the nominal power density (40.5 W/g of heavy metal), boron concentration of

500 ppm and average thermal-hydraulic conditions (fuel average temperature 810.93 K, clad temperature 616.5 K, moderator temperature 580.5 K, without CR insertion). The depletion conditions are the same for all calculated cases. The enrichment in fuel pins with Gd was decreased taking into account nominal enrichment and Gd_2O_3 weight fraction.

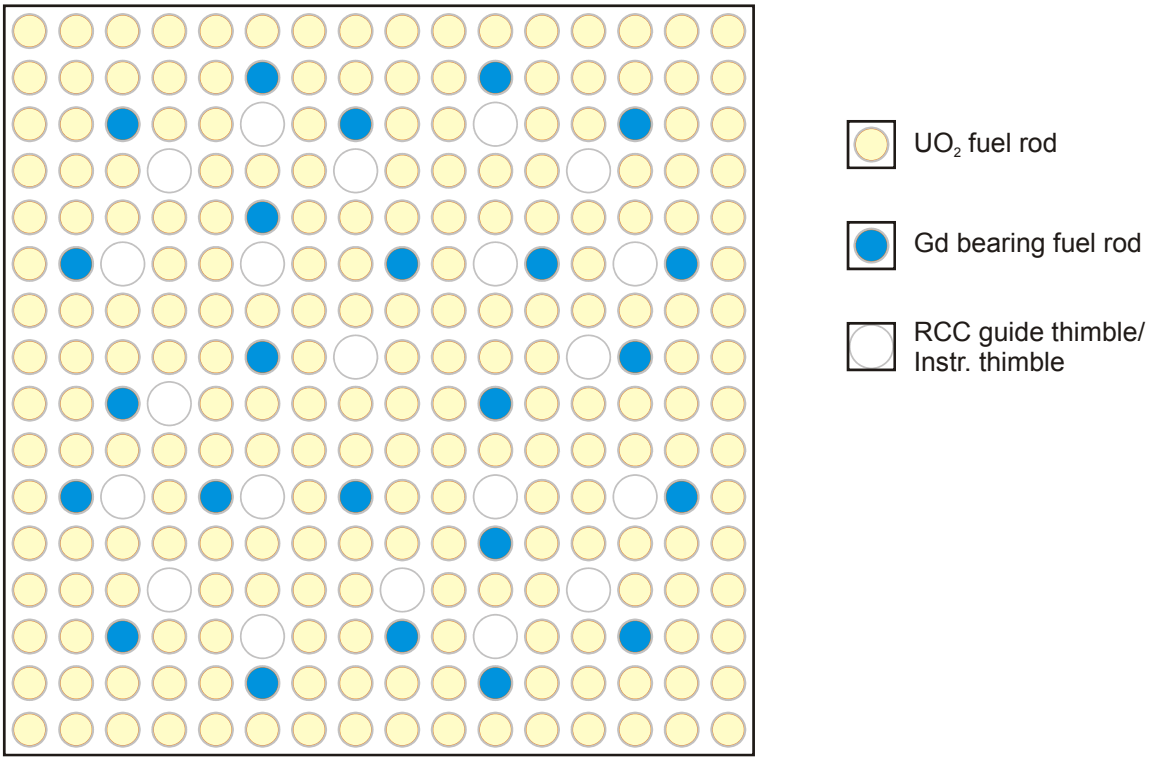

Figure 1**:** NEK 16×16 fuel assembly with 24 Gd_2O_3 fuel rods

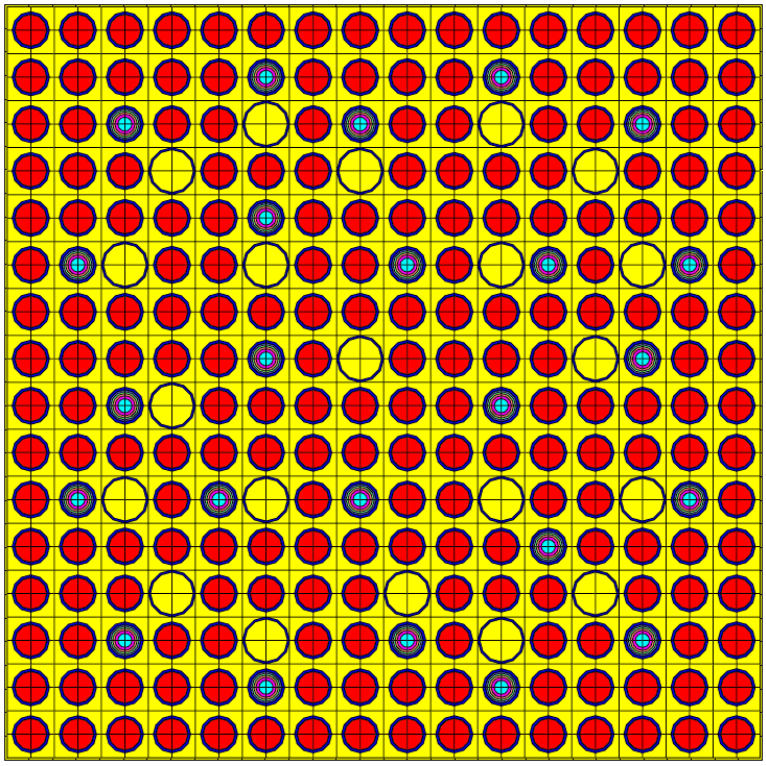

Figure 2**:** Material distribution and calculational mesh in NEWT model of FA

3 RESULTS

Two basic configurations were analyzed, present one (standard Westinghouse 16x16 (STD)), and proposed new one (Westinghouse Next Generation Fuel (NGF) configuration). The depletion characteristics of the fuel assemblies were calculated for FA without burnable absorbers and for FA with Integral Fuel Burnable Absorbers (IFBA) or Gd Burnable Absorbers (BA mixed with fuel). The calculations were performed for each fuel configuration and for two fuel enrichments (4.75% and 4.95%) FA layouts with 8, 12, 16, 20 and 24 Gd pins (Gd_2O_3 weight fractions 2% and 4%). For cases with IFBAs only 4.95% enriched fuel was used with 1.4x enriched B-10 coating on 104, 116, 132 and 148 rods. Taking into account runs for FAs without burnable absorbers, the total number of calculations was 52.

The infinite multiplication factor obtained for STD FA enriched at 4.95% with 24 Gd pins (weight fraction 4%) and without BA as calculated by NEWT and FA2D is shown in Figure 3. Rather big difference can be seen in both, FA without BA and FA with Gd pins. The NEWT predicts higher kinf values in whole burnup range. Up to 20 GWd/tU the difference can be mainly described as an offset, but for larger burnup the difference increases due to different (slower) decrease in NEWT case. Similar behavior, from point of view of burnup dependence, can be seen in Figure 4 where pin cell data are shown for pin with 4.95% enriched UO_2. In this case there is no offset found in FA calculation in the beginning and kinf results calculated by both codes are almost the same. We were not able to correct FA behavior in NEWT calculation in NEK case even though during the NFI benchmark calculation the initial difference between NEWT and FA2D calculation was small. Taking into account NEWT behavior for higher burnups and long calculation time (typical FA2D calculational time for FA depletion with 24 Gd pins is 5-6 CPU hours on Intel Core 2 Duo machine, and it is 2 to 3 times shorter than TRITON/NEWT CPU time for the same case) all other calculation were performed using FA2D only.

Some selected kinf results for IFBA and Gd assemblies in STD and NGF configuration for nominal enrichment of 4.95% are shown in Figure 5. As can be seen from spectral ratio (FA averaged fast neutron flux divided by thermal neutron flux) results for the same cases (Figure 6), NGF fuel is better moderated (the same pin cell with smaller diameter of the fuel rod), and has smaller spectral ratios and higher multiplication factors than corresponding STD fuel. For kinf results the situation is changed after 45-50 GWd/tU (exact value is dependent on amount and type of BA) when STD values are slightly above NGF values. The spectral ratio is typically larger for higher enrichment, but rest of the behavior is similar for both used enrichments. Almost all absorption influence of Gd BA disappears at about 15 GWd/tU (earlier for smaller number of pins or less GD in fuel pin) with some spectral influence present in both, kinf and spectral ratio. The configuration with 148 IFBAs produces little bit lower initial kinf than configuration with 24 Gd 4.0% pins. Due to faster depletion of Gd in this case, kinf increases with burnup much faster than for IFBA case and has peak at about 12 GWd/tU. According to our calculation after 25 GWd/tU kinf with 148 IFBAs is again higher than kinf for 24 Gd pins with 4.0% weight fraction.

The maximum relative pin power for selected FA cases is shown in Figure 7. The relative pin powers are normalized on the basis of 256 pins. The maximum relative pin power values decrease for FA without BA. For FA with large number of IFBAs they are rather low in the beginning, but they increase after about 5 GWd/tU and for higher burnup they are similar to values obtained in other configurations. For configurations with more Gd pins and more Gd in each pin initial maximum relative powers are rather large, but the values decrease rapidly with depletion of Gd content. In all cases STD configuration has larger maximum relative powers than NGF. The maximum values are slightly higher for higher nominal enrichment of the fuel.

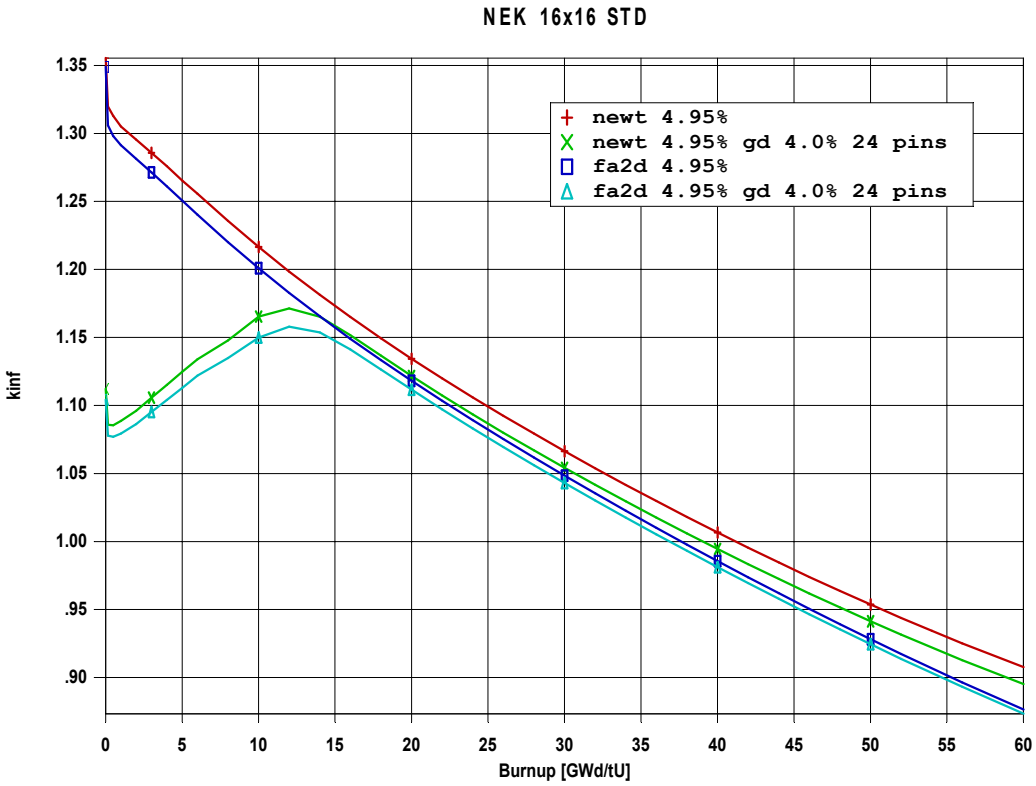

Figure 3**:** Infinite multiplication factor for NEWT and FA2D fuel assembly calculations

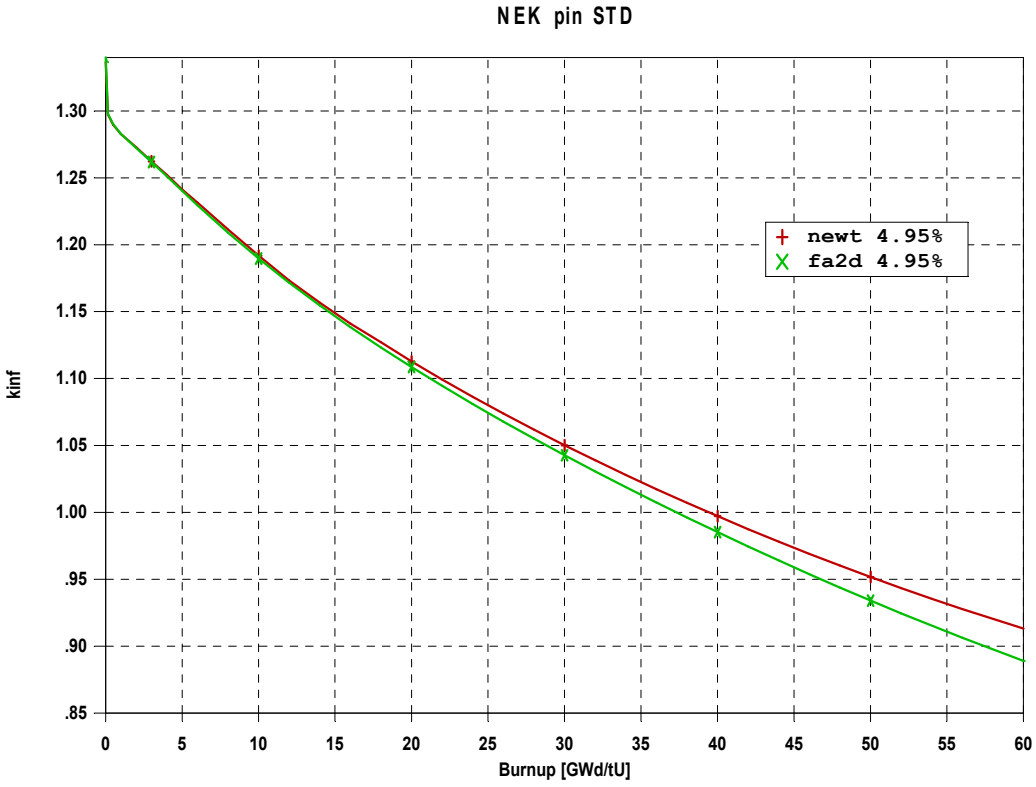

Figure 4**:** Infinite multiplication factor for NEWT and FA2D fuel pin calculations

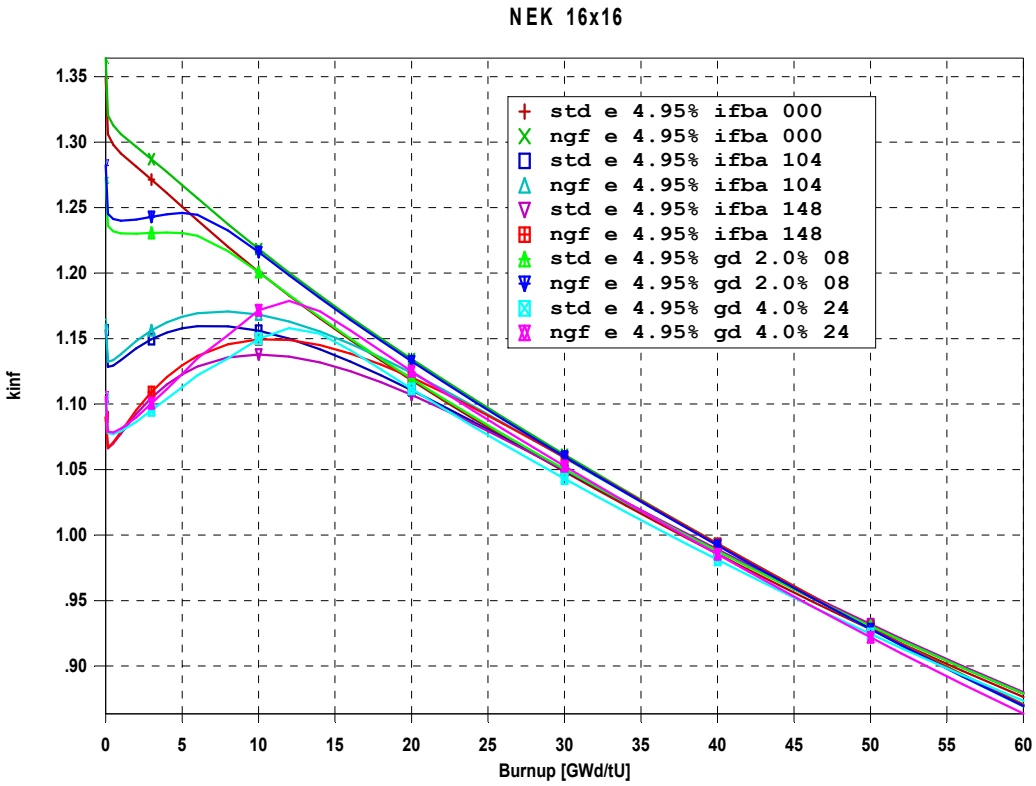

Figure 5: Infinite multiplication factor for selected FA2D fuel assembly calculations

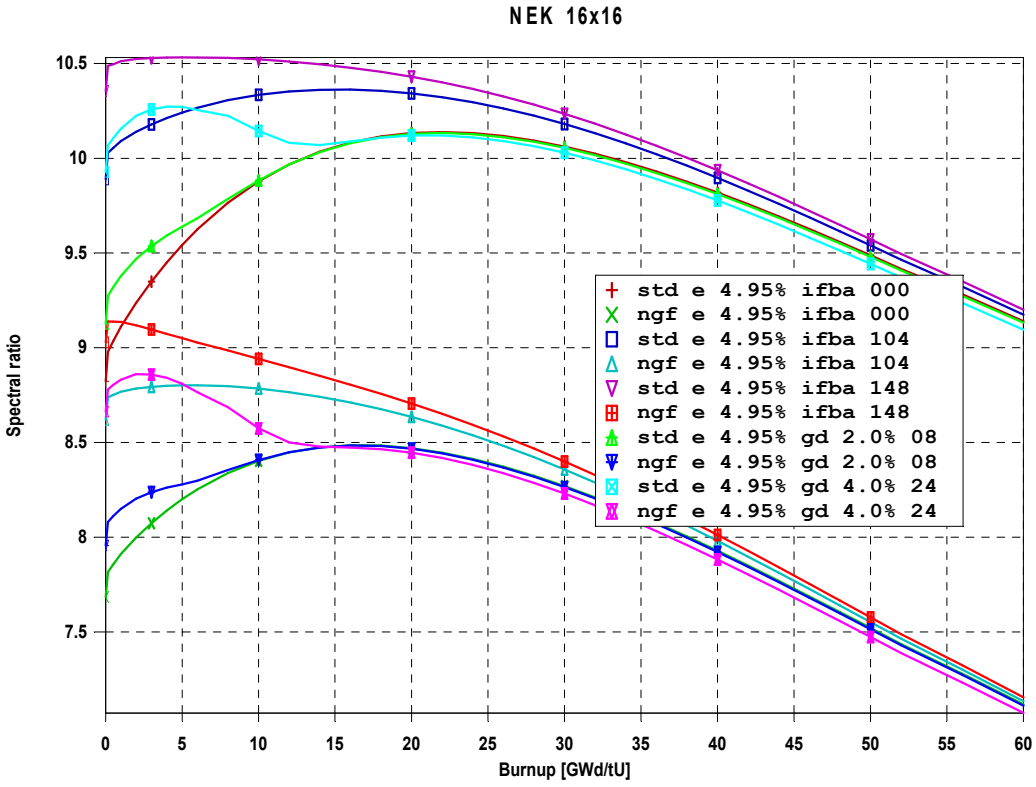

Figure 6: Spectral ratio for selected FA2D fuel assembly calculations

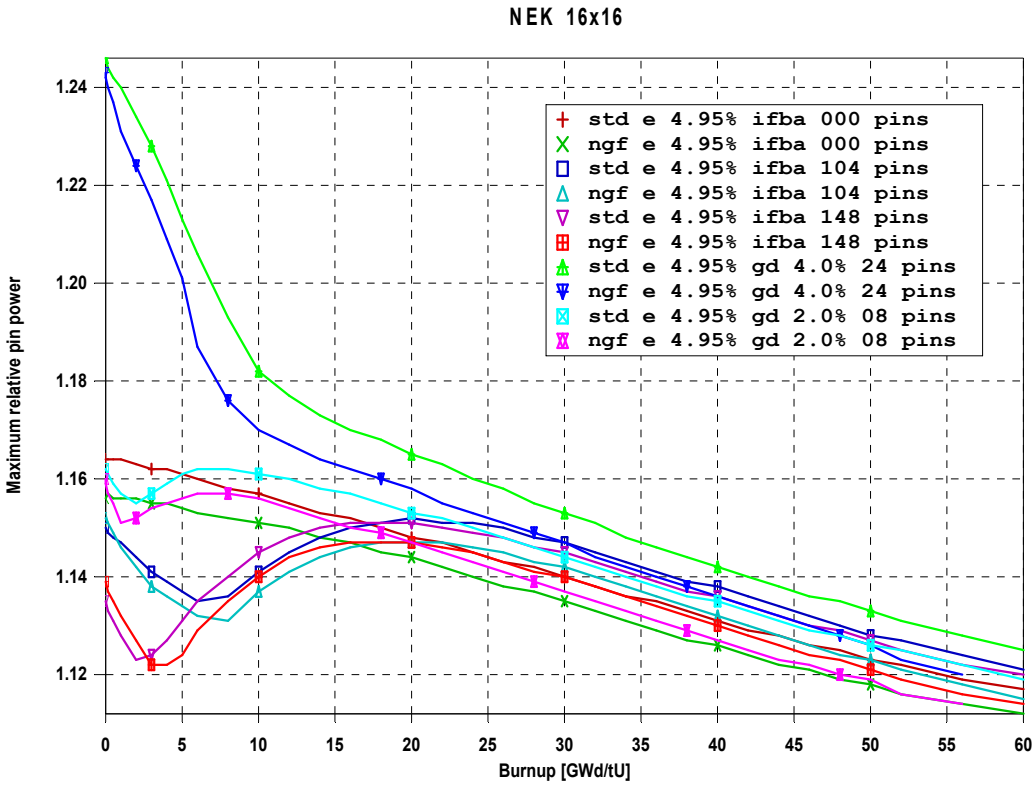

Figure 7**:** Maximum relative pin power for selected FA2D fuel assembly calculations

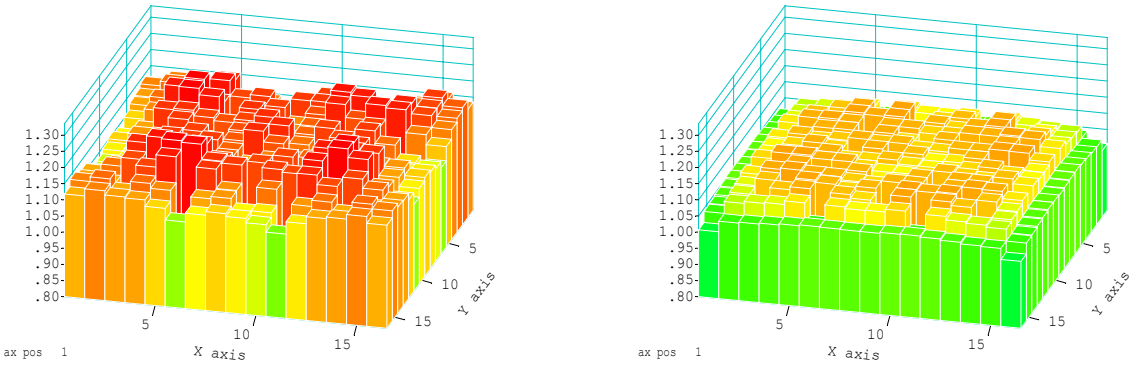

Figure 8**:** 2D relative pin power distribution for STD FA (0 and 60 GWd/tU)

The largest changes in local power distribution are in the vicinity of Gd pins. Initially depressed, the power in Gd pins increases with burnup due to lower Gd content at higher burnups. 2D relative pin power distributions for 4.95% enriched STD fuel with 24 Gd pins (4.0% weight fraction of Gd) for burnup of 0 and 60 GWd/tU are shown in Figure 8. Most of the local differences in pin powers disappear after burnup of 20 GWd/tU.

4 CONCLUSION

A number of calculations was performed using FA2D transport code to see influence of different BA (IFBA and Gd) usage in two 16x16 fuel assembly configurations (STD and NGF). Preliminary results were shown in terms of multiplication factors, spectral ratios, and 2D pin power distributions. Just basic trends can be seen from the FA results and for real evaluation whole core design calculation will be necessary. As an extension of this effort we are planning to perform additional FRAPCON calculation for selected pins in different FA configurations in future.

We were not able to verify FA2D prediction capability for fuel assemblies with Gd because NEWT code, that was mentioned for comparison, was not performed as expected. We will try to find out what is the cause for such a behavior.

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