Activities of the AZTLAN team on the OECD/NEA Benchmark on Fast Reactors

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Abstract

In the present paper, the activities of the AZTLAN Platform's Fast Reactor Group on the OECD/NEA Benchmark will be described. The main objective of these activities is to test the group's staff and capabilities as well as the domestic code reliability by putting them into test in this exercise with different institutions from around the world. Six different core configurations were treated; these are described in two different versions of the Benchmark document. The main tools used by the group were the Finnish stochastic Monte Carlo code Serpent for full core calculations and macroscopic Cross Sections (XS) generation, and the domestic deterministic code AZNHEX for full core calculations. Different calculations were performed, such as full core calculations under nominal conditions, with control rods fully and partially inserted and with the sodium voided in the active zone as well as different reactivity shift values due to various conditions of radial and axial expansion of the fuel elements and structural material. The results obtained in the full core calculations and most of the reactivity shift calculations obtained by our group were indeed comparable to the ones obtained by different institutions when using similar methodologies. Given these favorable results it can be said that the main objective was met and the group showed their capabilities, as well as its possibility to collaborate with other institutes, placing Mexico in a good position in fast reactor analysis. Future work will continue with the calculations not yet treated and with the new core specifications on the new versions of the Benchmark document.

1. INTRODUCTION

The *AZTLAN Platform* project [1] is a joint effort lead by the National Institute for Nuclear Research that gathers the main Mexican public universities which are the National Autonomous University of Mexico, National Polytechnic Institute and the Metropolitan Autonomous University, in an effort to place Mexico in a competitive position on reactor analysis matters. The project is funded by the Sectorial Fund for Energy Sustainability Conacyt-Sener and one of its main goals is to build up as well as strengthen the national development of specialized nuclear knowledge and human resources. This project aims to modernize, improve, and incorporate the neutronics, thermohydraulics and thermomechanical codes developed in the Mexican institutions of higher education, in an integrated platform, established and maintained for the benefit of the Mexican Nuclear knowledge.

The code AZNHEX [2] (AZtlan Neutronics HEXagonal) is included as part of the neutronics modules of the AZTLAN Platform; this code is a 3D diffusion module that solves numerically the time dependent neutron diffusion equations in Hexagonal-Z geometry for the calculation of the effective neutron multiplication factor (k_{eff}), neutron flux and power distribution. The geometry treated by AZNHEX makes it suitable for Fast Reactors (FR) analysis and thus, in order to verify and validate the code, a team for FR analysis was created in the AZTLAN Platform workgroup.

To test the team's and domestic code capabilities, as well as to collaborate with the international leaders on FR analyses, there was a special interest in participating in the "Benchmark for Neutronic Analysis of Sodium-cooled Fast Reactor Cores with Various Fuel Types and Core Sizes" published by the OECD/NEA. Participant institutions in this benchmark include:

- Argonne National Laboratory (ANL) from United States.
- Commissariat à l'Énergie Atomique et aux énergies alternatives (CEA Cadarache and Sacley) from France.
- Centre for Energy Research (CER-EK) from Hungary.
- Energy and Sustainable Economic Development (ENEA) from Italy.
- Helmholtz Zentrum Dresden Rossendorf (HZDR) from Germany.
- Institute of Nuclear Technology and Energy Systems (IKE) from Germany.
- Japan Atomic Energy Agency (JAEA) from Japan.
- Karlsruhe Institute of Technology (KIT) from Germany.
- Centre d'Étude de l'énergie Nucléaire (SCK•CEN) from Belgium.
- University of Illinois at Urbana Champaigne (UIUC) from United States.

In this paper the activities done by the FR team of the AZTLAN Platform workgroup are described. They will be sorted into four different sub-activities that had the objective of ending up as separate papers at the "International Conference on Fast Reactors and Related Fuel Cycles: Next Generation Nuclear Systems for Sustainable Development (FR17)".

The first steps taken were related only with the use of the finnish Monte Carlo code Serpent [3] in the modelling of the benchmark reactor cores. As soon as enough experience was achieved in the understanding of the subject, the capabilities of the domestic code AZNHEX together with the cross sections generated with Serpent were shown.

2. FULL CORES SIMULATION WITH SERPENT

Four cores were modeled and simulated on Serpent version 2.1.26 with both JEFF 3.1.1 and ENDFB/7.0 nuclear data libraries. Due to space limitation on the paper, here, only the main features and results of each core will be mentioned, for a more detailed description of the cores the reader is encouraged to visit reference [4]. The Table I shows the main characteristics of each core and the temperature of the materials used in these ones. Figures 1 and 2 show the radial and axial layout of the 3600-MOX and 1000-MOX cores respectively. It is important to note that at the moment of development of this part of the work, the Benchmark document included the four cores mentioned, while in future versions of the document only two were treated.

Table I. Features and temperatures used in the models in Serpent

	Power MWt	Fuel Type	Axial Reflector	Structure	Radial shielding	Radial Reflector	Fuel	Na/He Plenum	Control System	Sodium Channel
3600-CAR	3600	Carbide	600 K	600 K	600 K	600 K	1200 K	600 K	600 K	600 K
3600-MOX	3600	MOX	600 K	600 K	600 K	600 K	1500 K	600 K	600 K	600 K
1000-MET	1000	Metallic	600 K	600 K	600 K	600 K	900 K	600 K	600 K	600 K
1000-MOX	1000	MOX	600 K	600 K	600 K	600 K	1200 K	600 K	600 K	600 K

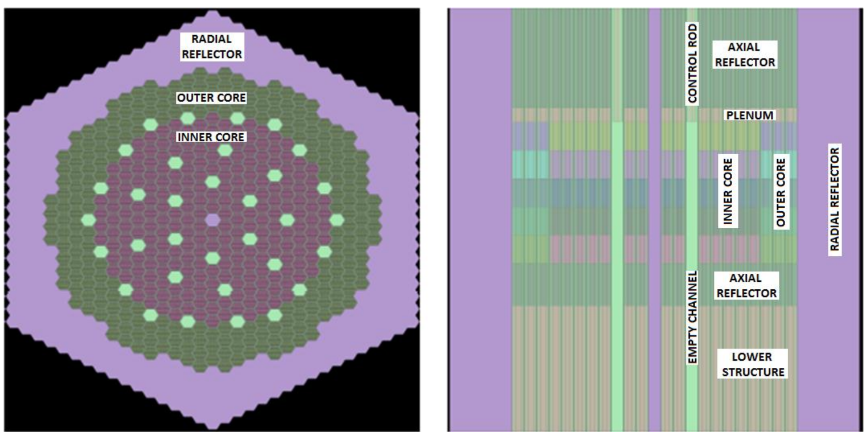

Figure 1. 3600-MOX core model in Serpent

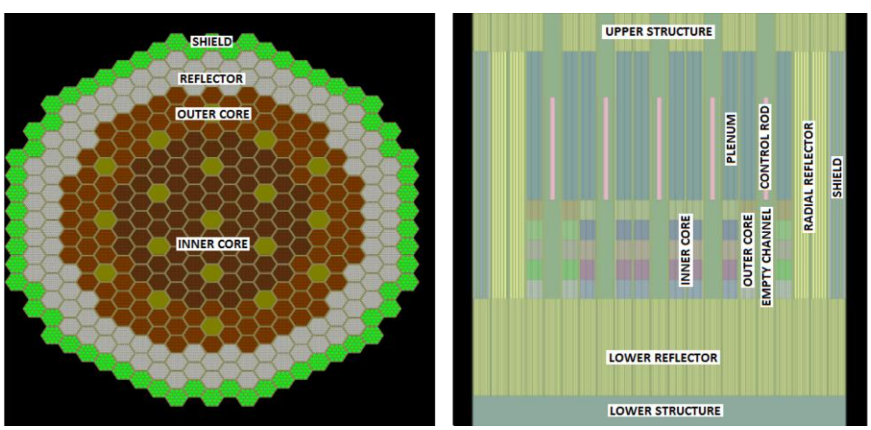

Figure 2. 1000-MOX core model in Serpent

Tables II, III, and IV show the results of k_{eff}, sodium void worth (replacing the sodium by void in the active zone) and delayed neutron fraction (β) obtained by the AZTLAN group: ININ-1 (using JEFF 3.1.1) and ININ-2 (using ENDF 7.0) and the results are compared with the ones obtained by other institutes with similar methodologies. The results show good agreement between ININ's results and others shown in green and that used similar methodologies.

Table II. Results of k_{eff} for all cores

ID	XS Library	Code	3600-MOX	3600-CAR	1000-MET	1000-MOX
ANL-2	ENDFB 7.0	MCNP5	1.00750	0.99970	1.02420	1.02230
ANL-3	JEFF 3.1	MCNP5	1.01370	**1.00850**	1.03730	1.03030
CEA-10	JEFF 3.1.1	TRIPOLI-4	1.01970	1.01220	**1.04290**	**1.03530**
ENEA	ENDFB 7.0	MCNPX	1.01080			
HZRD	ENDFB 7.0	Serpent	1.01040			
JAEA-3	JENDL 4.0	MVP	1.01390		1.03400	
JAEA-5	JENDL 4.0	MVP/Diff			1.03400	
CEN-1	ENDFB 7.1	MCNPX/ALEPH2.5				1.02560
CEN-2	JEFF 3.1.2	MCNPX/ALEPH2.5				1.03480
UIUC-1	JEFF 3.1.1	Serpent	1.02340	1.02100	1.03590	1.02580
UIUC-2	ENFB 6.8	Serpent	**1.02940**	1.02780	**1.03800**	**1.02370**
UIUC-3	ENDFB 7.0	Serpent	1.01930	1.01560	1.02690	1.02000
IKE-1	JEFF 3.1	MCNP 5			1.04260	
IKE-2	JEFF 3.1	MCNP 5			1.03740	
ININ-1	JEFF 3.1.1	Serpent 2.1.20	**1.03428**	**1.00913**	**1.04140**	**1.03303**
ININ-2	ENDFB 7.0	Serpent 2.1.20	**1.02965**	**1.00326**	**1.03226**	**1.02719**
		AVERAGE	1.01380	1.00900	1.03550	1.02870
		±SD	0.00405	0.00620	0.00780	0.00620

Table III. Results of void worth (in pcm) for all cores

ID	XS Library	Code	3600-MOX	3600-CAR	1000-MET	1000-MOX
ANL-2	ENDFB 7.0	MCNP5	2033	2289	2238	2002
ANL-3	JEFF 3.1	MCNP5	2078	2312	2273	2050
CEA-10	JEFF 3.1.1	TRIPOLI-4	1963	2122	**1858**	**1621**
ENEA	ENDFB 7.0	MCNPX	1940			
HZRD	ENDFB 7.0	Serpent	1860			
JAEA-3	JENDL 4.0	MVP	2009		2164	
JAEA-5	JENDL 4.0	MVP/Diff			2164	
CEN-1	ENDFB 7.1	MCNPX/ALEPH2.5				1760
CEN-2	JEFF 3.1.2	MCNPX/ALEPH2.5				1789
UIUC-1	JEFF 3.1.1	Serpent	**1559**	**1465**	1032	**1508**
UIUC-2	ENFB 6.8	Serpent	1696	1911	1251	1642
UIUC-3	ENDFB 7.0	Serpent	**1569**	**1750**	1128	**1526**
IKE-1	JEFF 3.1	MCNP 5			2257	
IKE-2	JEFF 3.1	MCNP 5			2520	
ININ-1	JEFF 3.1.1	Serpent 2.1.20	**1554**	**1698**	**1831**	**1562**
ININ-2	ENDFB 7.0	Serpent 2.1.20	**1544**	**1688**	**1844**	**1566**
		AVERAGE	1937	2120	2024	1831
		±SD	158	225	407	228

Numerically, the results are in the order of the ones obtained by other institutes following similar methodologies, which gives confidence in the results obtained by the methodology here followed.

Table IV. Results of β (in pcm) for all cores

ID	XS Library	Code	3600-MOX	3600-CAR	1000-MET	1000-MOX
ANL-2	ENDFB 7.0	MCNP5	360	365	330	326
ANL-3	JEFF 3.1	MCNP5	354	378	332	335
CEA-10	JEFF 3.1.1	TRIPOLI-4	**370**	**377**	**343**	**334**
ENEA	ENDFB 7.0	MCNPX	352			
HZRD	ENDFB 7.0	Serpent	361			
JAEA-3	JENDL 4.0	MVP	363		339	
JAEA-5	JENDL 4.0	MVP/Diff			339	
CEN-1	ENDFB 7.1	MCNPX/ALEPH2.5				315
CEN-2	JEFF 3.1.2	MCNPX/ALEPH2.5				344
UIUC-1	JEFF 3.1.1	Serpent	**371**	**382**	**350**	**337**
UIUC-2	ENFB 6.8	Serpent	360	367	335	324
UIUC-3	ENDFB 7.0	Serpent	**358**	**368**	**335**	**326**
IKE-1	JEFF 3.1	MCNP 5			352	
IKE-2	JEFF 3.1	MCNP 5			341	
ININ-1	JEFF 3.1.1	Serpent 2.1.20	**371**	**379**	**343**	**333**
ININ-2	ENDFB 7.0	Serpent 2.1.20	**359**	**368**	**334**	**323**
		AVERAGE	367	382	345	333
		±SD	13	16	10	15

3. XS GENERATION WITH SERPENT AND IMPLEMENTATION IN AZNHEX

The code AZNHEX has already been briefly described in the introduction section, due to space limitations there will not be given a deeper description but the reader is encouraged to visit references [2] [5] [6] for a more complete description on the code and its methodology.

In this work, the studied cores were the two included in the newest (at that time) version of the Benchmark [7], they are updated versions of the 3600-MOX and the 1000-MET cores with geometric and material differences from the previous version. The objective is to calculate the effective neutron multiplication factor (k_{eff}) under nominal conditions, with the control rods (CR) fully inserted and the sodium voided in the active zone.

The macroscopic XS were generated using the Serpent code and following the methodology previously used in the literature [8]. The main characteristics of the methodology will be described for material type being: non-fuel elements, fuel elements and fuel elements in the most external ring.

For non-fuel elements, such as radial and axial reflector, Na and He plenums, shielding or control systems, the main characteristics on the modeling were:

- 2D model.
- Radial reflection.
- Supercell consists on non-fuel element surrounded by half of fuel assemblies (Figure 3).
- 1,000,000 neutron histories per cycle, 330 active cycle, 30 inactive cycles.

Considerations made for fuel elements in both inner and outer not belonging to the most external ring of the outer zone, i.e. the one next to the radial reflector, are:

- 3D model.

- Radial reflection and no axial reflection.
- Whole active zone simulated at a time (supercell consists of five different axial layers, see Figure 4).
- XS generated for each fuel zone included in the whole active zone.
- 1,000,000 neutron histories per cycle, 330 active cycle, 30 inactive cycles.

A special treatment is needed for the most peripheral fuel elements to take into consideration the contribution of the reflector on the softening of the neutron spectrum in that region. The considerations were the following:

- 3D model.
- Radial reflection and no axial reflection.
- Three types of materials included: radial reflector, peripheral (which is in contact with the reflector) fuel and regular fuel.
- Regular fuel and peripheral fuel are identical but defined as two different materials in order to treat them separately.
- Whole active zone simulated at a time (supercell consists in five different axial layers in the two fuel regions, see Figure 5).
- XS generated only in the fuel region belonging only to the peripheral fuel assemblies.
- 1,000,000 neutron histories per cycle, 330 active cycle, 30 inactive cycles.

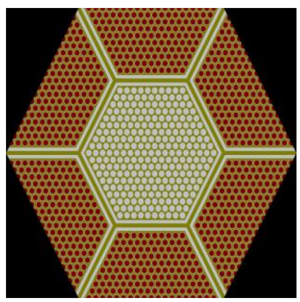

Figure 3. Layout of the axial reflector supercell of the 1000 MW core

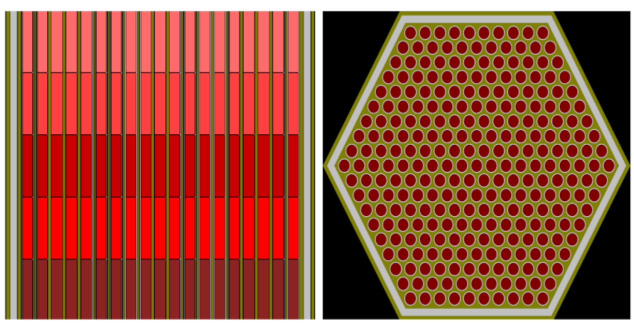

Figure 4. Side cut (left) and cross section (right) of a given fuel assembly of the 1000 MW core

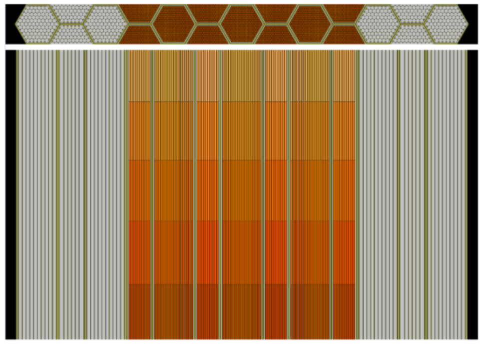

Figure 5. Side cut (down) and cross section (up) of the supercell used for XS generation on the peripheral fuel assembly of the 1000 MW core

The energy spectrum is segmented into 24 groups following the methodology previously described; the upper energy limits of each group are shown in Table V.

Table V. Neutron energy groups limits

Group	Upper Limit [MeV]	Group	Upper Limit [MeV]	Group	Upper Limit [MeV]
1	2.0000E+01	9	3.0197E-01	17	5.5309E-03
2	1.0000E+01	10	1.8316E-01	18	3.3546E-03
3	6.0653E+00	11	1.1109E-01	19	2.0347E-03
4	3.6788E+00	12	6.7379E-02	20	1.2341E-03
5	2.2313E+00	13	4.0868E-02	21	7.4852E-04
6	1.3534E+00	14	2.4788E-02	22	4.5400E-04
7	8.2085E-01	15	1.5034E-02	23	3.1203E-04
8	4.9787E-01	16	9.1188E-03	24	1.4894E-04

As mentioned before, three cases for simulations were considered: a) core operating under nominal conditions, b) the core has all the CR completely inserted, and c) the fuel assemblies have no sodium inside. In the Table VI the results of the simulations are presented. The mentioned simulations were done with Serpent using full-core modeling and AZNHEX using the XS generated by Serpent.

Table VI. Results of k_{eff} on simulated cores

	1000 MW Metallic Core		Error* [pcm]	3600 MW MOX Core		Error* [pcm]
	Serpent	AZNHEX		Serpent	AZNHEX	
Nominal Conditions	1.01989	1.02192	-199	1.01326	1.01157	167
CR inserted 100%	0.92797	0.92358	473	0.95366	0.94998	386
Na voided	1.04114	1.05008	-859	1.02734	1.03549	-794

*Relative error calculated as: $\frac{Serpent - AZNHEX}{Serpent} * 1x10^5 \ pcm$

As it can be seen in Table VI, the results obtained with AZNHEX show good agreement if are compared with the ones obtained with Serpent in the case of nominal conditions and with larger differences in the other cases.

The relative error for results under nominal conditions is only around two hundred pcm, this is a notorious result given the difference of methodologies followed by the solvers (Serpent is a stochastic/continuous-energy code and AZNHEX is a deterministic/multi-group code). The neutron spectrum of a fast reactor is also a factor in these results. Fast reactors have, in general, a longer mean free path, the results are not much affected by the fact that Serpent considers the heterogeneity of the geometry and AZNHEX treats each region with a homogenized XS; in the case of thermal reactors, special treatment must be done to take into consideration these heterogeneities.

As mentioned before, in the case of the core with the CR inserted and especially in the case with no sodium in the fuel zones, the discrepancy between codes is considerably larger than in nominal conditions. One explanation for this can be that this is an effect of the methodology for XS generation itself, most of the XS were calculated isolated (except for the peripheral fuels where the impact of the neighbor reflector was considered) and no special treatment was used for materials that are next to others. This can become an issue when the regions have widely different absorption XS next to each other (such as fuel/absorbent vicinity), as in the case of cores with the CR inserted; and it can have a much smaller effect in the nominal conditions where the CR are above the active zone where most of the neutronic activity is taking place.

4. VERIFICATION OF AZNHEX BY COMPARING IT WITH DYN3D AND PARCS

In this section, the main goal is to compare the results obtained by AZNHEX against the ones obtained by other institutes with different codes such as PARCS [9] and DYN3D [10]. The three codes need a previously generated macroscopic XS set. The XS were generated by a different institute, as well as the simulations with PARCS and DYN3D and delivered as they are to the FR group, so the work of the group was to implement the XS in AZNHEX and do the full-core calculations with Serpent.

A brief description of PARCS and DYN3D will be presented here for the reader to know about the characteristics of the code.

PARCS is a deterministic 3D code developed at Purdue University and endorsed by the United States Nuclear Regulatory Commission. Its capabilities [9] include:

- Neutron diffusion and transport solutions.
- Time-dependent solutions for transients and burnup.
- Treatment of Cartesian and hexagonal geometries.
- Transient simulation capabilities.
- Corrections for control rod treatment.
- Decay heat and Xe/Sm treatment

Regardless the capabilities of the PARCS code, there is one thing that it is not capable of doing and is the generation of Cross Sections (XS) sets for its calculations, these must be given by the user for the specific case of the core simulated.

The code DYN3D is another deterministic 3D code originally developed for Light Water Reactors (LWR) but extended for Sodium Fast Reactors (SFR) [10]. Its capabilities are very like those of PARCS. Thermalhydraulic modules have been implemented for one-phase and two-phase coolant flow treatment. As well as PARCS, XS for the specific problem need to be generated prior to the use of DYN3D.

As the XS were not generated by the group, no details on the generation is available except that the methodology is somewhat similar to the followed previously [8] and that in Serpent 1500 active cycles were used and 200 skipped with 640,000 neutron histories per cycle, giving a total of 960 million of active neutron histories.

In Tables VII and VIII the results between codes are compared. Four factors were calculated being these:

- k_{eff}: Effective neutron multiplication factor.
- K_D: Doppler constant.
- $\Delta\rho_{Na}$: Sodium void worth.
- $\Delta\rho_{CR}$: Control rod worth.

To calculate the Doppler constant two reactivities need to be calculated, one at nominal conditions (1500 K) and one at perturbed conditions (3000 K), and it is calculated as:

$$K_D = \frac{\rho_{per} - \rho_{nom}}{\ln \frac{T_{per}}{T_{nom}}} \tag{1}$$

The sodium void worth is calculated as the difference in reactivity due to the extraction of all the sodium (void) in the active zone and the reactivity on nominal conditions:

$$\Delta\rho_{Na} = \rho_{void} - \rho_{nom} \tag{2}$$

The control rod worth is calculated by the difference between the reactivity in nominal conditions and its value when all CR are inserted:

$$\Delta\rho_{CR} = \rho_{CR} - \rho_{nom} \tag{3}$$

Table VII. Results of core simulations

	Serpent	DYN3D	PARCS	AZNHEX
k_{eff}	1.01070	1.00940	1.00984	1.00873
K_D (pcm)	-852	-867	-868	-878
$\Delta\rho_{Na}$ (pcm)	1864	1951	1945	2019
$\Delta\rho_{CR}$ (pcm)	-6046	-6173	-6227	-6046

Table VIII. Difference in pcm (absolute value) of AZNHEX vs other cores

	AZNHEX vs Serpent	AZNHEX vs DYN3D	AZNHEX vs PARCS
k_{eff} (pcm)	194.9	66.37	109.9
K_D (pcm)	26	11	10
$\Delta\rho_{Na}$ (pcm)	155	68	74
$\Delta\rho_{CR}$ (pcm)	0	127	181

In general, the results obtained with AZNHEX showed very good agreement. The differences in k_{eff} of AZNHEX vs. DYN3D and AZNHEX vs. PARCS are 66 pcm and 109 pcm respectively and 194 when comparing directly with Serpent. These results are considered acceptable and give confidence that the methodology of the solver inside AZNHEX is well implemented.

5. COLLABORATION OF FR GROUP ON CALCULATIONS OF BENCHMARK PARAMETERS

The intention of this work was to collaborate with different institutions such as CEA Cadarache, CEA Saclay, Argonne National Laboratory, and others, in the elaboration of an article where results obtained by these international research centers and the ones obtained by the FR group of the AZTLAN Platform.

The parameters to calculate were:

- k_{eff} on nominal conditions
- β (fraction of delayed neutrons)
- Control Rod worth (CR fully inserted)
- Control Rod worth (CR inserted 5cm from top)
- Doppler constant
- Na void worth
- Δρ 1% Sodium density variation (Na density multiplied by 0.99 in the whole assembly)
- Δρ 1% Wrapper density variation (Wrapper density multiplied by 0.99 in the Active Zone "AZ")
- Δρ 1% Cladding density variation (Cladding density multiplied by 0.99 in the AZ)
- Δρ 1% Fuel density variation (Fuel density multiplied by 0.99 in the AZ)
- Δρ 1% Fuel density variation + 1% axial expansion (Fuel density multiplied by 0.99 in the active zone, the active zone is expanded 1% of length and CR are moved to remain above AZ)
- Δρ 1% Pitch variation (Assembly pitch is increased by 1.01 but masses are conserved, except for Na which volume is increased and density is fixed)

The used code by the FR Group was Serpent 2.1.27 with JEFF 3.1.1 XS library. The simulated cores were the same as the ones in Section 3 consisting in one 3600 MWt MOX fueled core and one 1000 MWt metallic-fueled core; the results are summarized in Tables IX and X.

The acronyms of each participant institution are shown here:

- ANL, Argonne, USA
- CEA Cadarache, France
- CEA Saclay, France
- CER, Budapest, Hungary
- GRS, Garching, Germany
- HZDR, Dresden, Germany
- IKE, Stutgart, Germany
- ININ, Edo. de México, Mexico
- IPPE, Obninsk, Russia

Table IX. Results of the 3600 MWt. MOX fuel core

Institute	ANL	CEA Cadarache	CEA Saclay	CER	GRS	HZDR	IKE	ININ	IPPE
Library	ENDF / B-VII.1	ENDF / B-VII.1	JEFF 3.1.1	ENDF / B-VII.1	ENDF / B-VII.1	ENDF / B-VII.1	ENDF / B-VII.1	JEFF 3.1.1	ABBN-RF (ROSFOND)
Code	MC^2 / VARIANT	ERANOS	TRIPOLI 4	Serpent	KENO-IV	Serpent	MCNP	Serpent	TRIUM (MMKK)
K-effective	1.0162	1.0102	1.0185	1.0289	1.0164	1.0134	1.0075	1.0164	1.0087
β [pcm]	351	372	361	348	344	361	353	360	361
CR worth (fully inserted) [pcm]	-6360	-6511	-6135	-5556	-6218	-6315	-6439	-6111	-6206
CR worth (5cm inserted) [pcm]	-140	-139	-146	-126	-134	-133	-138	-127	-136
Doppler Const. [pcm]	-857	-929	-875	-758	-848	-778	-800	-791	-787
Na Void Worth [pcm]	1863	2005	1768	1726	1677	1821	1690	1851	1889
1% Sodium [pcm/K]	0.42	0.448	0.466	0.446	0.523	0.5	0.366	0.828	0.48
1% Wrapper [pcm/K]	0.023	0.022	0.025	0.019	0.021	0.017	0.019	0.027	0.027
1% Cladding [pcm/K]	0.036	0.041	0.038	0.041	0.043	0.047	0.034	0.051	0.039
1% Fuel [pcm/K]	-0.3	-0.31	-0.304	-0.292	-0.295	-0.306	-0.312	-0.31	-0.318
1% Fuel + Axial [pcm/K]	-0.127	-0.133	-0.12	-0.144	-0.125	-0.139	-0.128	-0.127	-0.152
1% Grid [pcm/K]	-0.745	-0.755	-0.758	-0.726	-0.757	-0.761	-0.822	-0.614	-0.811

Table X. Results of the 1000 MWt metallic fuel core

Institute	ANL	CEA Cadarache	CEA Saclay	GRS	ININ	IPPE
Library	ENDF/B-VII.1	ENDF/B-VII.1	JEFF 3.1.1	ENDF/B-VII.1	JEFF 3.1.1	ABBN-RF (ROSFOND)
Code	MC2-3/VARIANT	ERANOS	TRIPOLI 4.9	KENO-IV	Serpent	TRIUM (MMKK)
K-effective	1.0171	1.0128	1.0299	1.0197	1.0284	1.0215
β [pcm]	332	352	342	324	342	343
CR worth (fully inserted) [pcm]	-9905	-10029	-9540	-9796	-9640	-9542
CR worth (5cm inserted) [pcm]	-239	-230	-241	-232	-233	-241
Doppler Constant [pcm]	-383	-407	-394	-378	-384	-351
Na Void Worth [pcm]	1327	1219	1579	1370	1247	1423
1% Sodium [pcm/K]	0.383	0.34	0.405	0.261	0.565	0.393
1% Wrapper [pcm/K]	0.021	0.022	0.022	0.023	0.032	0.023
1% Cladding [pcm/K]	0.043	0.05	0.05	0.049	0.07	0.04
1% Fuel [pcm/K]	-0.553	-0.568	-0.538	-0.567	-0.594	-0.57
1% Fuel + Axial [pcm/K]	-0.257	-0.265	-0.26	-0.277	-0.307	-0.267
1% Grid [pcm/K]	-1.137	-1.115	-1.074	-1.093	-1.097	-1.162

As can be seen in Tables IX and X, the results obtained by the FR Group (ININ) are comparable with the rest of the results obtained by other institutions making us sure that the group is in the right path.

6. CONCLUSIONS

One of the objectives after forming a FR group on the AZTLAN Platform was to test its capabilities and experience through participating in various international exercises to compare results with other institutions. On that matter, the objective has been accomplished by the participation of the group in the Benchmark organized by OECD/NEA.

The results obtained by the group using the Serpent code are very good and comparable with the obtained by institutions that used similar methodologies, which gives the confidence that the group can carry out international collaborations and quality work.

Based on the numerical results, it can be concluded that the AZNHEX code is a promising tool to the study of nuclear reactor cores with hexagonal-z geometry. Regarding the numerical results, it is important to point out that the differences are bigger than those taken as references when the core exhibit a localized larger absorption which can be diminished once that discontinuity factors may be included. Nonetheless the fact that differences are less than 200 pcm for smooth scenarios and 800 for non-smooth ones motivates the AZTLAN neutronic team to improve AZNHEX code to obtain better results than the ones here above given and to study its behavior for time-dependent problems.

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