Industrial application of APOLLO2 to Boiling Water Reactors

Vera Marotte [*1], François Clément[1], Sébastien Thareau[1], Stefan Misu[2] and Igor Zmijarevic[3]
[1] *AREVA NP SAS, Tour Areva, 92084 Paris La Défense, France*
[2] *AREVA NP GmbH, 43 Bunsentrasse, 91058 Erlangen, Germany*
[3] *CEA-Saclay DEN/DM2S/SERMA, 91191 Gif sur Yvette, France*

Abstract

AREVA NP – a joint's subsidiary of AREVA and Siemens– decided to develop a new calculation scheme based on the multigroup neutron transport code APOLLO2, developed at CEA, for industrial application to Boiling Water Reactors.

This scheme is based on the CEA93 library with the XMAS-172 energy mesh and the JEF2.2 evaluation. Microscopic cross-sections are improved by a self-shielding calculation that accounts for 2D geometrical effects and the overlapping of resonances. The flux is calculated with the Method of Characteristics. A best-estimate flux is found with the 172 energy group structure. In the industrial scheme, the computing time and the memory size are reduced by a simplified self-shielding and the calculation of the flux with 26 energy groups.

The results are presented for three BWR assemblies. Several BWR operating conditions were simulated. Results are accurate compared to the Monte-Carlo code MCNP. A very good agreement is obtained between the best-estimate and the industrial calculations, also during depletion.

These results show the high physical quality of the APOLLO2 code and its capability to calculate accurately BWR assemblies for industrial applications.

KEYWORDS: BWR, APOLLO2, assembly, lattice physics, industrial calculation

1. Introduction

The multigroup transport code APOLLO2 [1] is currently used by AREVA NP for industrial applications to Pressurized Water Reactors. Several calculation schemes – i.e. trains of physical and numerical models and assumptions- were developed [2] using the CEA93 nuclear data library based on the JEF2.2 evaluation. These calculation schemes were mainly based on the collision probability method and the Sn method. Their accuracy was checked against Monte-Carlo codes like MCNP and against critical experiments.

In the framework of the convergence on codes and methods between French, German and American practices inside AREVA NP, it was decided to develop a new calculation scheme based on the APOLLO2 code to extend the present industrial applications to Boiling Water Reactors. This new scheme was jointly developed with CEA with the latest APOLLO2 version that includes enhanced flux calculation with the Method Of Characteristics (MOC) on

* Corresponding author, Tel. (33) 1 47 96 26 40, E-mail: vera.marotte@areva.com

unstructured geometries [3] and the treatment of mutual shielding that accounts for resonance overlapping in a mixture of resonant isotopes [4]. These two methods permit to define a best-estimate scheme in which the finely shielded cross-sections are input directly in the MOC flux solver to obtain a fine energy solution for the exact geometry. As the computing resources required are too large for an industrial application, a less expensive scheme was derived with a simplified self-shielding and a MOC flux calculation with a reduced number of groups.

These schemes were settled on two simplified BWR assemblies and they were then applied to three actual BWR assemblies. These five assemblies and the two control blades are briefly described in the next section. The best-estimate and the industrial schemes are described in Section 3. The Section 4 is dedicated to the results. Conclusions are given in the Section 5.

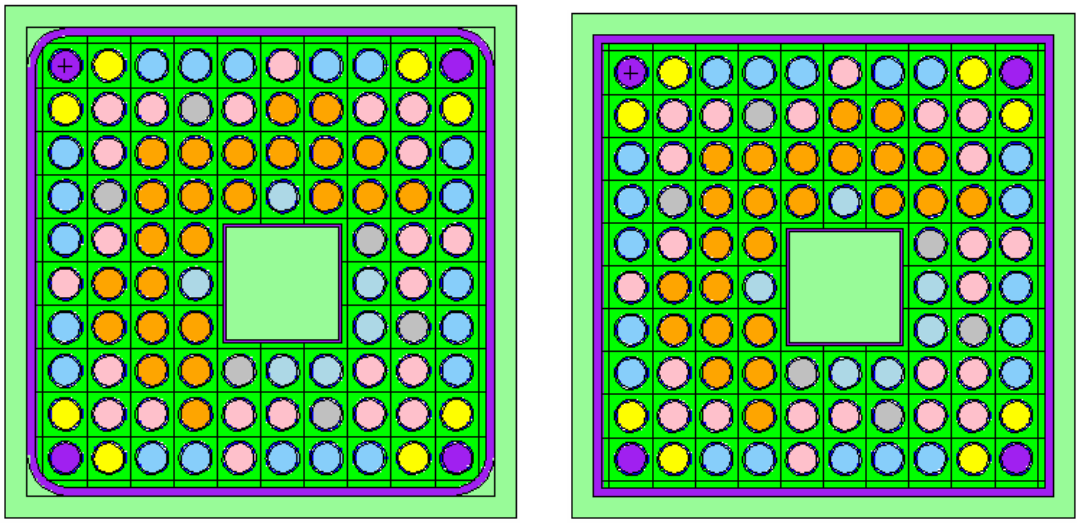
2. Assemblies and control blades description

The schemes were set up on two simplified BWR assemblies with squared channel box corners and two fuel loadings. They were then applied to three actual BWR assemblies. Two types of control blade were inserted between the assemblies. The five assemblies and the two control blades studied are briefly described hereafter.

2.1 The ATRIUM10 MOX assembly

It is a 10x10 rods assembly including a central 3x3 moderating water hole. It has a channel box with squared corners (Fig. 1a.) in the NEA benchmark description [5], and a channel box with rounded corners in the industrial application (Fig. 1b.). In both descriptions, the assembly is loaded with 6 types of MOX fuel with a total plutonium content of 2.69, 3.86, 5.2, 6.71, 7.55, 10.57 wt% and 6 UO_2-Gd_2O_3 rods with 1.5 wt% of gadolinium and 3.95 wt% of ^{235}U.

Figure 1: ATRIUM10 MOX assemblies



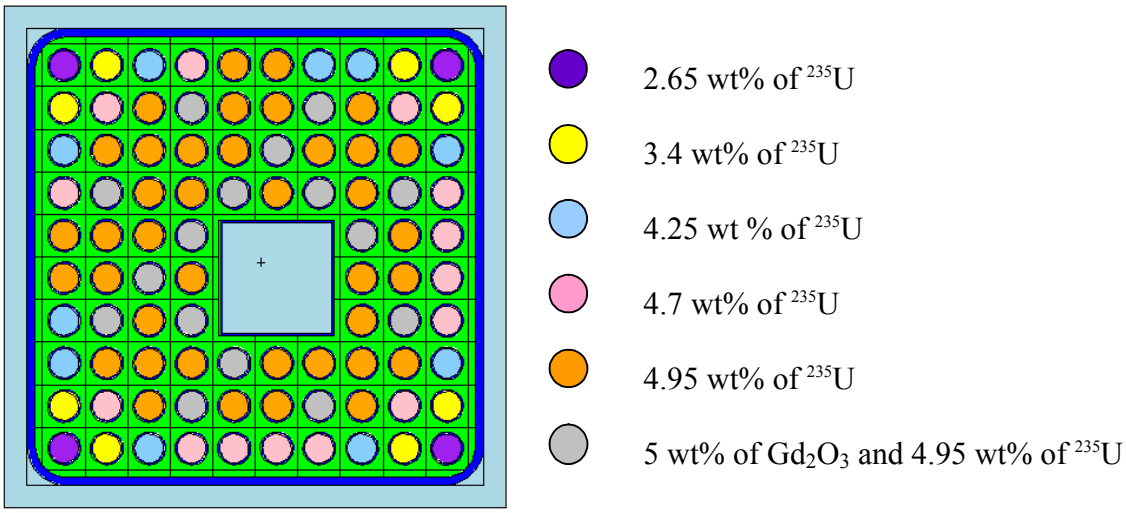
a. NEA benchmark b. Industrial

2.2 The ATRIUM10 UGD5 assembly

Geometries are equivalent to the ATRIUM10 MOX cases but it is loaded with UO_2 and UO_2-Gd_2O_3 fuel (Fig. 2): 5 types of UOX rods with ^{235}U enrichments of 2.65, 3.4, 4.25, 4.7, 4.95 wt%

and 16 UO_2-Gd_2O_3 rods with 5 wt% of gadolinium and 4.95 wt% of ^{235}U.

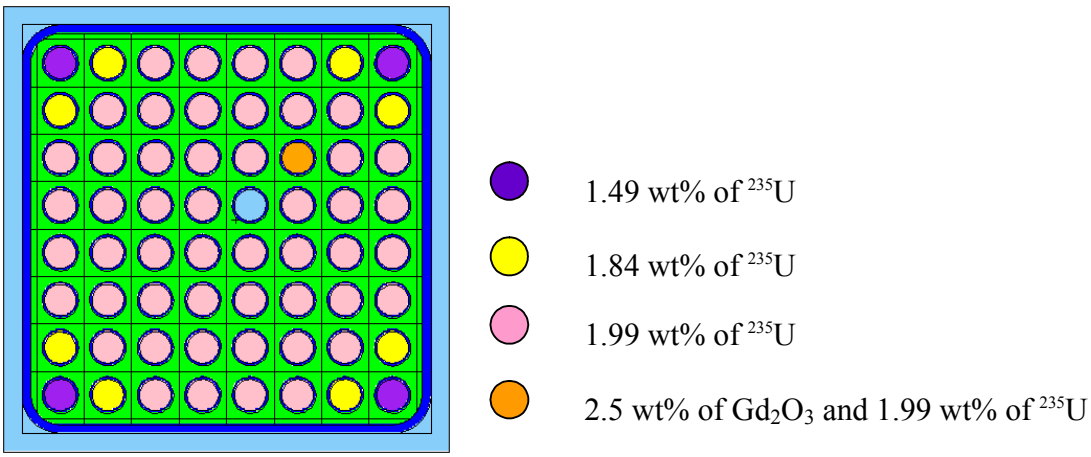
Figure 2: ATRIUM10 UGD5 assembly



2.3 The FA8 assembly

It is an assembly of 8x8 rods inserted in a channel box with rounded corners and including a single water tube (Fig. 3). It is loaded with 3 types of UOX fuel with uranium 235 enrichments of 1.99, 1.84, 1.49 wt% and 1 UO_2-Gd_2O_3 rod with 2.5 wt% of gadolinium and 1.99 wt% of ^{235}U.

Figure 3: FA8 assembly

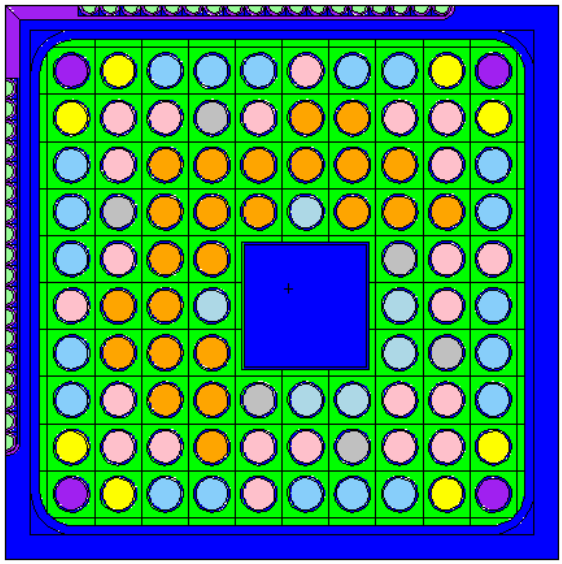


2.4 Control blades

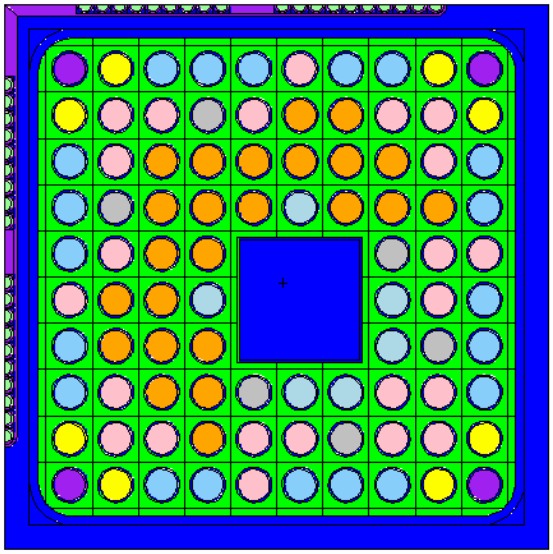
These assemblies were calculated with two types of control blades:

- CR1 containing 18 B_4C rods (Fig. 4a) and
- CR2 containing 19 B_4C rods and a stiffener (Fig. 4b).

Figure 4: Control blades CR1 and CR2 inserted in ATRIUM10 MOX assembly



a. CR1 inserted



b. CR2 inserted

3.2. Calculation schemes

These schemes use the CEA93 nuclear data library based on the JEF2 evaluation. They involve the new methods developed in the APOLLO2 code [2]: the flux calculation with the Method Of Characteristics [3] on unstructured geometries and the treatment of mutual shielding due to resonance overlapping [4]. These two methods permit to define a best-estimate scheme in which the best self-shielded cross-sections are input directly in the MOC flux solver to obtain a 172 energy group solution for the exact geometry. The computing time and memory size required are too large for an industrial application. Hence, a less expensive multi-level scheme was derived with a simplified self-shielding calculation and a MOC flux calculation on a reduced number of groups. Critical calculation is performed and efficiently accelerated for the industrial scheme. Depletion is performed with the predictor-corrector solver of APOLLO2. 24 heavy isotopes and 85 fission products are explicitly modeled; the remaining fission products are lumped into 8 pseudo-isotopes.

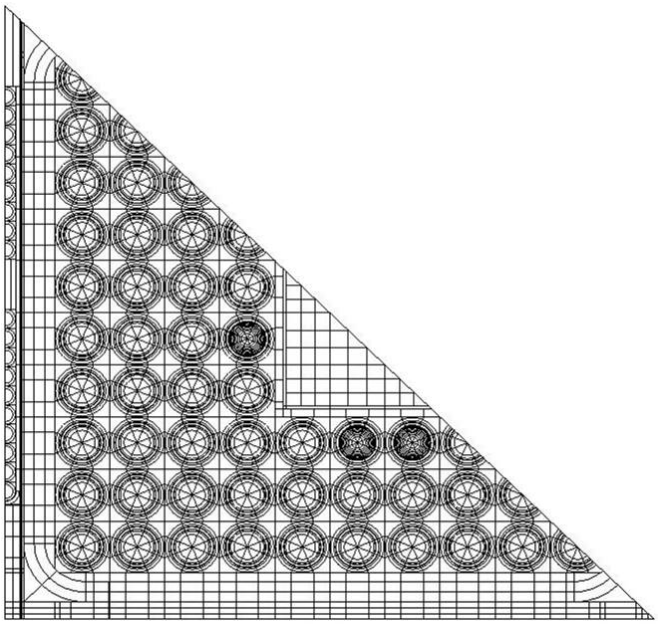
3.1 The best-estimate scheme

The fuel cross-sections of the CEA93 library are self-shielded on the assembly geometry. Rings are defined in the fuel rods: 4 in UOX and MOX rods in order to take into account RIM effect and 11 in UO2-Gd2O3 rods to correctly represent the depletion if this black absorber. Rods facing the water hole or the channel box are self-shielded separately in such a way that the Dancoff effect is explicitly treated. The self-shielded fuel isotopes are: uranium 235, uranium 238, plutonium 238, plutonium 239, plutonium 240, plutonium 241, plutonium 242, gadolinium 155, gadolinium 156, gadolinium 157 and gadolinium 158. In the energy range between 19.45 to 204 eV where main resonances overlap, all the self-shielded uranium and plutonium isotopes are treated with the new mixture formalism [4]. Zirconium of the water hole box and the channel box is self-shielded on slab geometry.

The flux is calculated for the 172 energy groups with the Method of Characteristics that permits to calculate unstructured geometries, modeling without approximation real BWR

assemblies. The spatial mesh used is shown on Fig. 5. About three thousands regions are calculated. Rings of the fuel rods are kept and the coolant and the moderator are very finely described with sub zones. The tracking is also very fine: distance of 0.05 cm and 36 azimuth angles.

Figure 5: MOC calculation mesh for the ATRIUM10MOX assembly with CR2 inserted



Criticality is obtained by adjustment of the DB^2 leakage term and iteration on the flux calculation until K_{eff} is equal to one.

Self-shielded cross-sections are up-dated during depletion taking into account the fuel inventory changes. Every ring of every rod is depleted separately with its own flux calculated with the method of Characteristics.

3.2 The industrial scheme

The best-estimate scheme utilizes the most accurate and direct methods available in APOLLO2 for BWR assembly calculation. Its results are very satisfactory as compared to a Monte-Carlo code like MCNP as shown in Section 4. It is thus a rather interesting tool for highly accurate analysis. Nevertheless, the computing resources required by such an approach are much too high for an industrial application in which many calculations have to be performed in order to build the library of cross-sections for the core calculation. Thus, a simplified and less expensive calculation scheme was developed for industrial applications.

The self-shielding was simplified by choosing a simpler geometry on one hand and by reducing the energy domain where mixtures of resonances are used. So, the self-shielding is performed on the cell geometry. In order to take into account the Dancoff effect for the main isotopes, only uranium 238 is self-shielded on the assembly geometry. Also, the mixture of resonant isotopes is used in the two main relevant domains:

- From 19.45 to 22.6 eV for ^{235}U, ^{238}U and ^{240}Pu and
- From 33.72 to 204 eV for ^{235}U, ^{238}U, ^{239}Pu and ^{240}Pu.

This simpler self-shielding method permits to reduce CPU time by a factor of 3.

The flux calculation, which takes about 50% of the CPU time of the best-estimate calculation, is accelerated by reducing the number of energy groups from 172 to 26 [6]. The cross-sections for the 26 group mesh are obtained by collapsing the self-shielded cross-sections with a reconstructed flux [7]. This flux is reconstructed from two calculated fluxes:

- A first one which is the fine energy spectrum for detailed parts of the assembly and
- A second one which is the 2D spectrum for the whole assembly described with homogeneous cells. Indeed, an accurate calculation of the pin-by-pin reaction rates with a reduced number of groups requires a 2D description of cross-sections: if cross-sections are input per type of fuel, about 0.3% is lost in terms of root mean square of the pin fission rate distribution and about 1% on the maximum discrepancy.

The first flux calculation involves the collision probability method with the interface-current approximation. The second flux calculation is performed with the discrete ordinates using the method of characteristics as spatial differencing scheme and 44 energy groups.

The critical calculation is accelerated by computing the leakage at the second flux calculation and input it to the MOC calculation. The leakage is then corrected for the reconstructed flux. This avoids the 3 or 4 iterations with the MOC calculation.

Self-shielded cross-sections are up-dated during depletion taking into account the fuel inventory changes. Every ring of every rod is depleted separately with its own reconstructed flux.

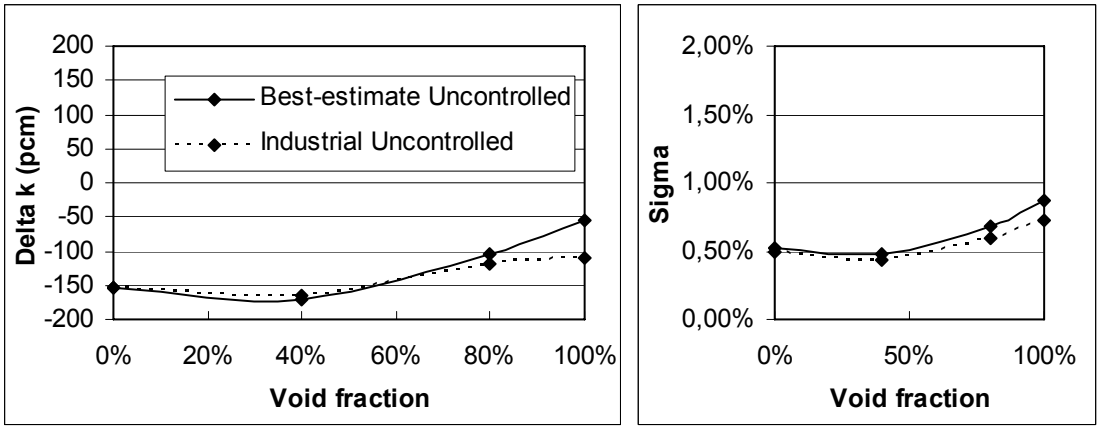
4. Results

The results are presented in this chapter for the actual assemblies: ATRIUM10MOX, ATRIUM10UGD5 and FA8. At beginning of life, the results of the best-estimate and the industrial scheme are compared to the Monte-Carlo code MCNP [8]. MCNP calculations were carried out with 10 millions histories so that the error on the multiplication factor is about 20 pcm at 1 sigma and about 0.03 on the pin fission rate.

4.1 ATRIUM10MOX assembly

The results compared to MCNP for the uncontrolled assembly as a function of void fraction are presented on Fig.6. Reactivity discrepancies are lower than 200 pcm and discrepancies on pin-by- pin fission rates lead to a sigma lower than 1%.

Figure 6: Discrepancy to MCNP with void fraction for the ATRIUM10MOX assembly



The results for the controlled assembly as a function of void fraction are presented on Fig. 7. The results for the Doppler case –i.e. fuel temperature decreased by 300K- are shown on Fig. 8. Finally, the results for cold conditions are shown on Fig. 9; a case with 1000 ppm of boron is included.

These results show the accuracy as compared to MCNP of the APOLLO2 code for the calculation of BWR assemblies at beginning of life. It shows also the very good accuracy of the industrial scheme with a CPU time decrease of 80% as compared to the best-estimate scheme.

Figure 7: Discrepancy to MCNP for the controlled ATRIUM10MOX assembly

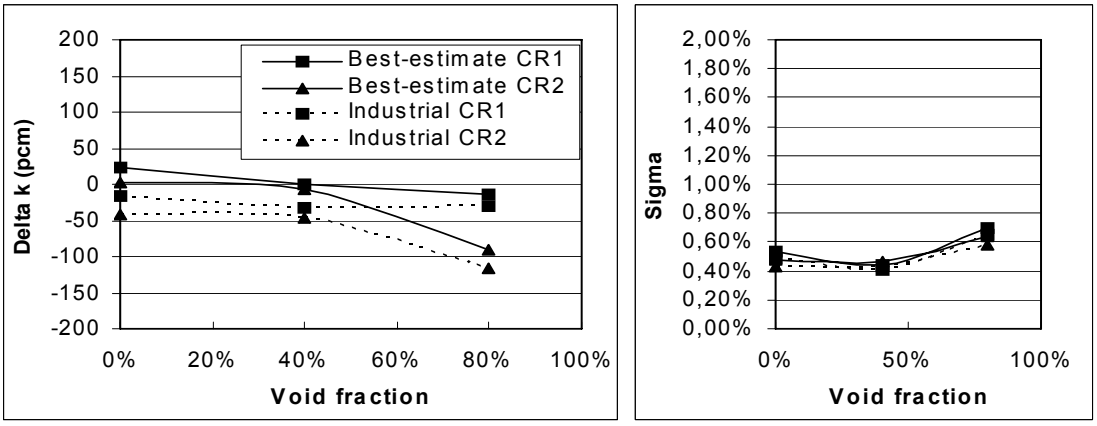


Figure 8: Discrepancy to MCNP with Doppler Effect for the ATRIUM10MOX assembly

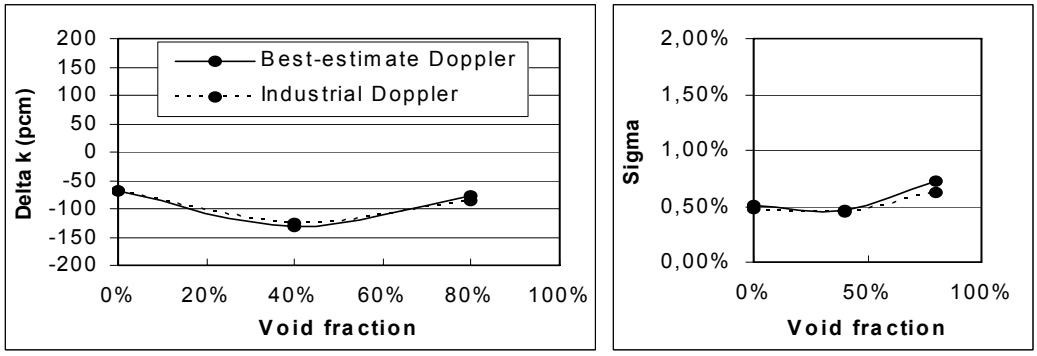
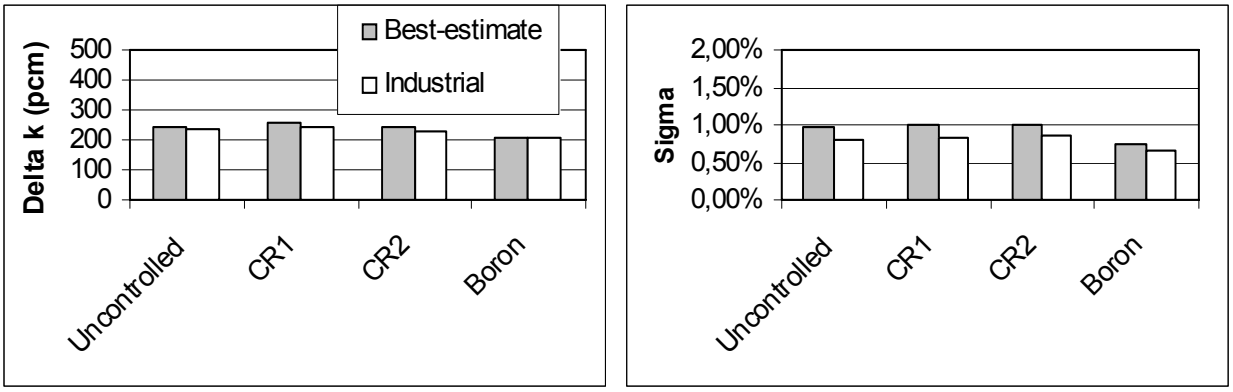


Figure 9: Discrepancy to MCNP in cold conditions for the ATRIUM10MOX assembly



The results between the best-estimate and the industrial scheme during the depletion are presented on Fig. 10 for reactivity and pin-by-pin fission rates at 0, 40 and 80% void and on Fig.11 for isotopic content during depletion at 40% void. The discrepancy remains smaller than:

- 100 pcm in absolute value for reactivity,
- 0.5% on the sigma of the pin-by-pin distribution and
- 2% in absolute value for the main isotopes.

Figure 10: Discrepancy between the best-estimate and the industrial scheme for the ATRIUM10MOX assembly

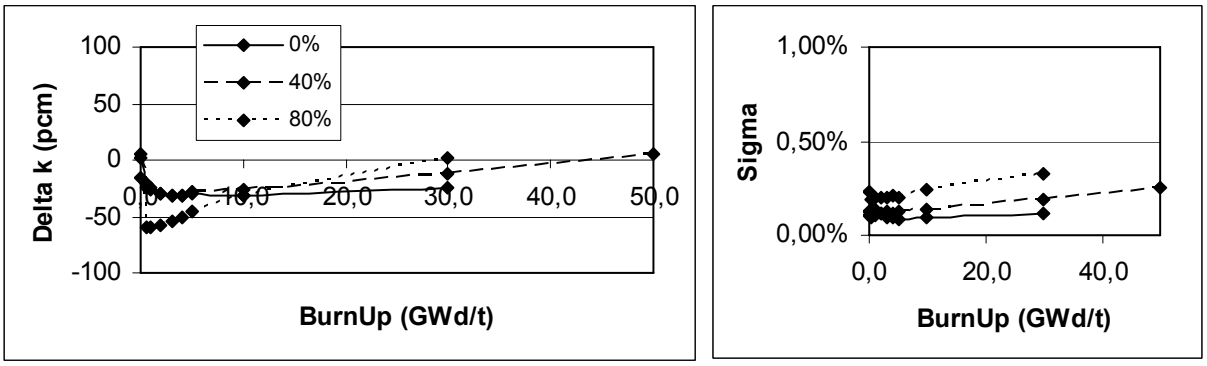
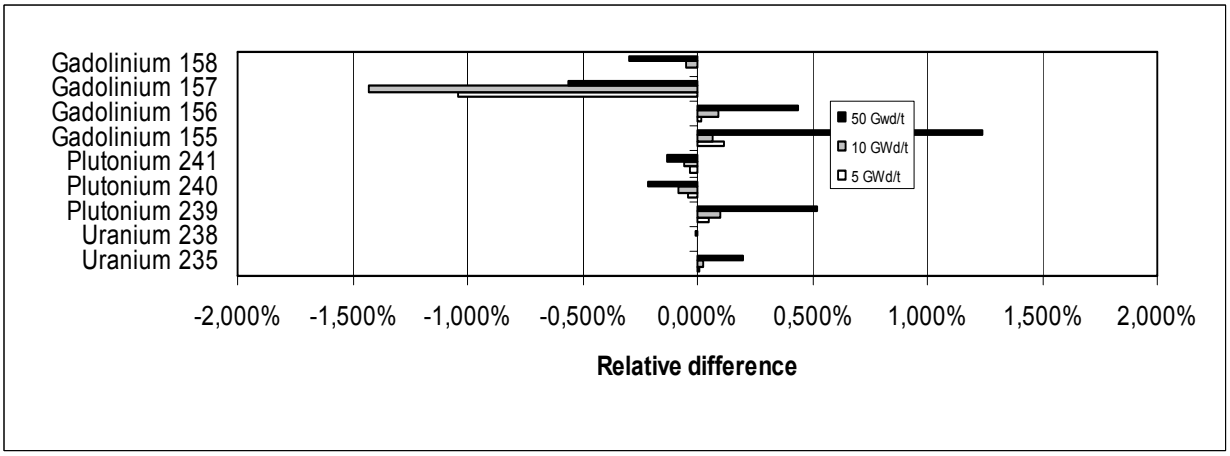


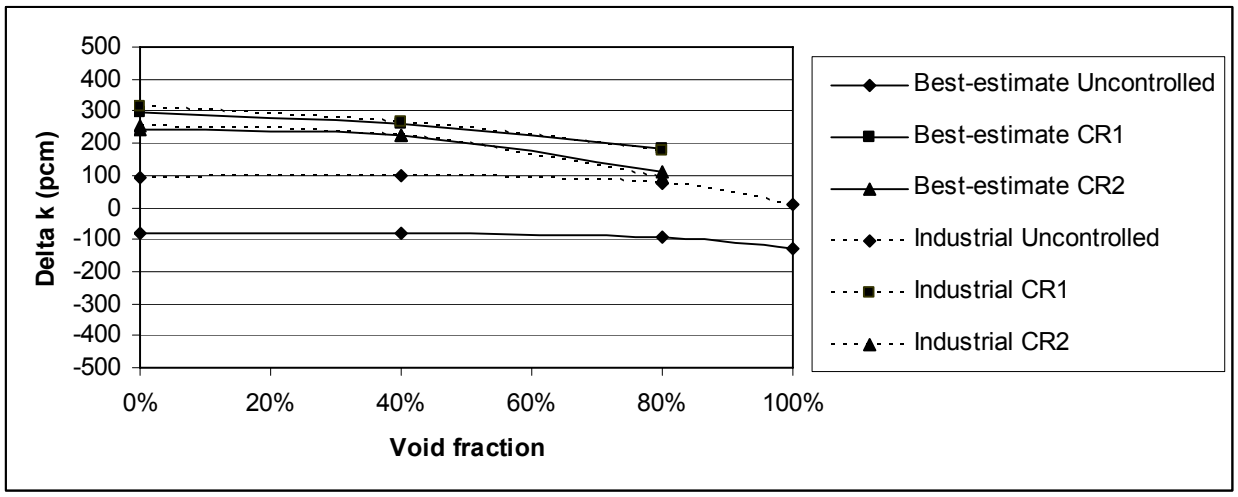
Figure 11: Discrepancy between the best-estimate and the industrial scheme for the ATRIUM10MOX assembly



4.2 ATRIUM10UGD5 assembly

The calculations performed for the ATRIUM10 MOX were also run for the ATRIUM10UGD5. As an example, Fig. 12 shows the discrepancy on reactivity to MCNP versus the void fraction for the uncontrolled and controlled assembly.

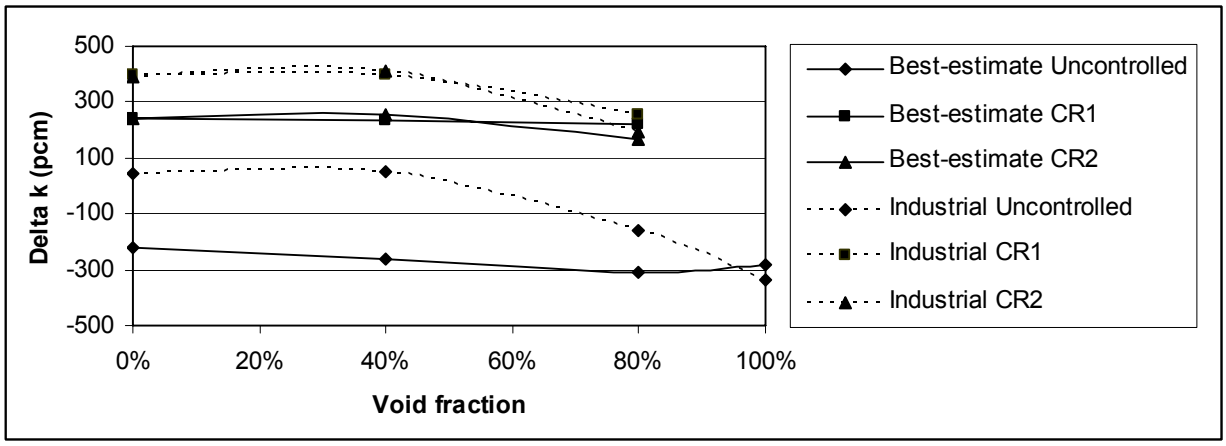
Figure 12: Discrepancy to MCNP with void fraction for the ATRIUM10UGD5 assembly



4.3 FA8 assembly

The last example of industrial application of APOLLO2 to BWRs is the FA8 assembly. Fig. 14 shows the discrepancy on reactivity to MCNP versus the void fraction for the uncontrolled and controlled assembly.

Figure 13: Discrepancy to MCNP with void fraction for the FA8 assembly



5. Conclusion

In the framework of the convergence on codes and methods between French, German and American practices inside AREVA NP, it was decided to develop in close cooperation with CEA a new calculation scheme based on the multigroup neutron transport code APOLLO2 to extend the present industrial applications to Boiling Water Reactors.

This scheme, based on the CEA93 library with the XMAS-172 energy mesh and the JEF2.2 evaluation, involves the latest developments available in APOLLO2: self-shielding treatment of the overlapping resonances and flux calculation with the Method of Characteristics. The computing time and memory size are reduced for industrial needs by a simplified self-shielding and the calculation of the 2D-MOC flux on 26 energy groups thanks to a new flux reconstruction method and an adequate multi-level strategy.

The results presented for three BWR assemblies and several BWR operating conditions show a very good agreement with the Monte-Carlo code MCNP and a very high accuracy and efficiency of the industrial scheme.

These results show the high physical quality of the APOLLO2 code and its capability to calculate accurately BWR assemblies for industrial applications.

References

1) R.Sanchez, J.Mondot, Z.Stankovski, A.Cossic, I.Zmijarevic, "APOLLO II: A user-oriented, portable, modular code for multigroup transport assembly calculations", 1988, Nuclear Science and Engineering 100, 352-362
2) A.Santamarina, C.Collignon, C.Garat, "French Calculation Schemes for Light Water Reactor Analysis", Proc. of Int. Conf. PHYSOR2004, Chicago (USA), April 25-29, 2004
3) S.Santandrea, "A new multi-domain DPN technique to accelerate the method of characteristics in unstructured meshes", MC2005, September -, 2005, Avignon
4) M.Coste-Delclaux, S.Mengelle, "New resonant Mixture Self-Shielding Treatment in the Code APOLLO2", PHYSOR-2004, April 25-29, 2004, Chicago
5) "Physics of Plutonium Recycling – Volume VII: BWR MOX benchmark – Specification and Results", Nuclear Science OCDE/NEA, 2003
6) A.Santamarina, N.Hfaiedh, R.Letellier, V.Marotte, S.Misu, A.Sargeni, C.Vaglio, I.Zmijarevic, "Advanced neutronics tools for BWR design calculations", Proc. of ICONE 14, July 17-20, 2006, Miami (Florida), USA
7) I.Zmijarevic, E.Masiello, R.Sanchez, "Flux reconstruction methods for assembly calculations in the code APOLLO2", submitted to Physor-2006 conference
8) J.F.Briesmeister, "MCNP-A General Monte Carlo N-Particle Transport Code, Version 4C", LANL-13709-M (2000)