AREVA NP next generation fresh UO_2 fuel assembly shipping cask: SCALE - CRISTAL comparisons lead to safety criticality confidence

M. Doucet – R. Montgomery** – B. O' Donnell** – M. Landrieu**

*AREVA, AREVA NP
**AREVA, AREVA NP Inc.
michel.doucet@areva.com

AREVA NP as a worldwide PWR fuel provider has to have a fleet of fresh UO_2 shipping casks being agreed within a lot of countries including USA, France, Germany, Belgium, Sweden, China, and South Africa… and to accommodate foreseen EPR Nuclear Power Plants fuel buildings. To reach this target the AREVA NP Fuel Sector decided to develop an up to date shipping cask gathering experience feedback of the today fleet and an improved safety allowing the design to comply with international regulations (NRC and IAEA) and local Safety Authorities.

Based on pre design features a safety case was set up to highlight safety margins. Criticality hypothetical accidental assumptions were defined:

- Preferential flooding
- Fuel rod lattice pitch expansion for full length of fuel assemblies
- Neutron absorber penalty
- …

Well known computer codes, American SCALE package and French CRISTAL package, were used to check configurations reactivity and to ensure that both codes lead to coherent results. Basic spectral calculations are based on similar algorithms with specific microscopic cross sections ENDF/BV for SCALE and JEF2.2 for CRISTAL. The main differences between the two packages is on one hand SCALE's three dimensional fuel assembly geometry is described by a pin by pin model while an homogenized fuel assembly description is used by CRISTAL and on the other hand SCALE is working with either 44 or 238 neutron energy groups while CRISTAL is with a 172 neutron energy groups. Those two computer packages rely on a wide validation process helping defining uncertainties as required by regulations in force.

The shipping cask with two fuel assemblies is designed to maximize fuel isolation inside a cask and with neighboring ones even for large array configuration cases.

Proven industrial products are used:

- Boral™ as neutron absorber
- High density polyethylene (HDPE) or Nylon as neutron moderator
- Foam as thermal and mechanical protection

The cask is designed to handle the complete AREVA fuel assembly types from the 14x14 to the 18x18 design with a ^{235}U enrichment up to 5.0% enriched natural uranium (ENU) and enriched reprocessed uranium (ERU).

After a brief presentation of the computer codes and the description of the shipping cask, calculation results and comparisons between SCALE and CRISTAL will be discussed.

1. Introduction

Fresh fuel shipping casks are key components for safe nuclear transportation. To achieve a safe cask criticality safety margin is a major concern. As far as reactivity is driven by the fuel assembly characteristics the geometry i.e. moderator volume – fuel volume ratio has to be kept as close as possible to the nominal one. Fuel assembly mechanical and thermal protections are key points to maintain fuel assembly integrity.
As AREVA NP has to deliver fuel assemblies on several continents and to submit safety files to a rather large number of local Safety Authorities, it is important that the computer code package used for criticality safety analysis being validated on one hand benchmark comparison (1) and on the other hand on cross check with other computer codes (2). Item (1) will lead to uncertainty determination and item (2) will lead to trust the overall shipping cask results. Keeping in mind that SCALE and CRISTAL are using different basic cross section sets and different geometrical modeling, this paper will focus on gross reactivity results without any uncertainty.

2. Computer codes

2.1. CRISTAL code package

The CRISTAL code package [1] was developed during the late nineties by CEA (Commissariat à l'Energie Atomique) and IRSN (Institut de Radioprotection et de Sûreté Nucléaire), the advising body of the DGSNR (Direction Générale pour la Sûreté Nucléaire et la Radioprotection). This project was founded by the French nuclear industry together with the CEA and IRSN.

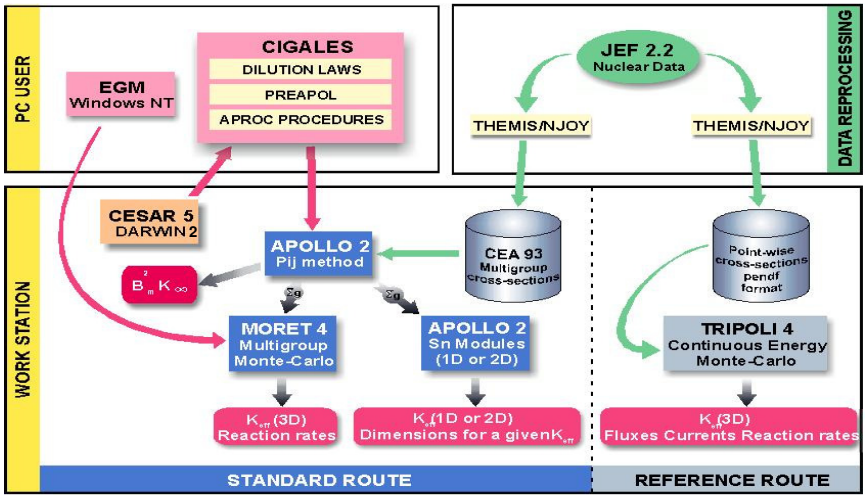

Figure 1

The functional architecture of the CRISTAL code package, presented on Figure 1, is organized around two calculation routes:

- A "Standard route" or "Industrial route" with a multigroup formulation of cross-sections of the CEA93 library based on APOLLO2 computer code [2], a spectral computer code, and MORET 4 [3] computer code, a three dimensional Monte Carlo computer code.

- A "Reference route" based on TRIPOLI-4 computer code [4] with point wise cross sections and continuous energy group.

These two calculation routes are using JEF 2.2, a basic microscopic cross library, via the CEA93 library with 172 energy group structure for the “standard route” and directly for the “reference route”.

This paper deals with the "Standard" route, the industry's preferred route for criticality safety evaluations.

The "Standard" route uses an input pre-processing Graphical User Interface, CIGALES [5]. CIGALES provides an efficient method to prepare APOLLO2 input decks and simplifies QA activities. Only basic "physical" data are used (fissile and structural media, shape, enrichment, dimension …). The APOLLO2 computer code performs cell or assembly spectrum calculations accounting for sophisticated self shielding process flux and macroscopic cross section determination. The resulting homogenized 172-group energy structures are directly linked with the three-dimensional Monte Carlo code, MORET 4, which provides the capability to model simple or complex geometries.

SCALE 4 code package

The SCALE 4.4a code is used with either a CSAS5 or CSAS6 option for KENO V.a or KENO VI, respectively. This code option further uses BONAMI and NITAWL. BONAMI and NITAWL perform resonance processing by default in the SCALE sequences. Both of these codes are able to process resolved and unresolved resonance data using different methods. A common characteristic of both modules is that neither BONAMI nor NITAWL treat resonance overlap or resonance interference. Both BONAMI and NITAWL perform temperature broadening at the user-specified problem temperature during resonance processing. NITAWL also performs a temperature interpolation of thermal-scattering data on the master library using a $\sqrt{T}$ law.

The BONAMI module self-shields cross sections with Bondarenko data using the shielding-factor methodology. Nuclides with Bondarenko data carry an infinite dilute cross section on the master library and tables of dilution-dependent shielding factors. BONAMI performs iterations for each nuclide and each energy group that has shielding factors. Convergence is achieved when the shielded total cross section changes by less than some minimum amount (ε) for all nuclides, groups, and zones. In this manner, the problem-dependent self-shielded cross sections for each nuclide and group are determined while approximately accounting for interactions. When CSAS6 calls BONAMI, heterogeneous geometry effects are accounted for in the escape cross section that is passed to BONAMI.

The escape cross section, which appears in the Wigner rational approximation for the fuel escape probability, is determined from the system geometry specified in the cross-section processing portion of the SCALE input. The geometry type, materials, characteristic dimensions, and the Dancoff factor are all used to determine an escape cross section that with “units” equivalent to a macroscopic cross-section. The escape cross section is added to other cross sections to account for geometry effects. The equivalence theorem of lattice physics allows all nuclides to be processed by BONAMI as infinite homogeneous media in the CSAS sequences.

The NITAWL-III module shields cross sections with resonance data utilizing the Nordheim integral transport method. In the SCALE implementation, the infinite dilute multigroup cross

sections are adjusted by a correction value determined in NITAWL. This correction is calculated by first determining the infinite dilute contribution of each resonance to the group cross section and then by calculating what the contribution would be if the resonance were shielded for the specific problem. The geometry type, materials, characteristic dimensions, and Dancoff factor are all passed to NITAWL for determining the details of the approximations used to self-shield the cross sections. NITAWL uses two internal moderators when reconstructing the shielded flux. The slowing-down mass and scatter cross section for the principal material (first moderator) mixed with the fuel are used explicitly. The remaining materials (second moderator) are treated using an averaged slowing-down mass and scatter cross section. A fundamental assumption of the Nordheim method is that resonances are widely spaced, both within a particular nuclide and between nuclides.

KENO (V or VI) is a three dimensional Monte Carlo eigenvalue, keff code. Group cross-section data are used to randomly transport particles throughout a system containing fissile material. Particles are grouped in generations, with the fission particles for one generation providing the starting particles for the next generation. The primary purpose of KENO is to calculate a system's keff; however, many other physics parameters are calculated during the random walk. Other calculated quantities include neutron lifetime, generation time, energy-dependent leakages, energy- and region-dependent absorptions, fissions, fluxes, fission densities, energy of the average lethargy of fission (EALF), and system mean-free path.

KENO-VI is an extension of the KENO V.a Monte Carlo criticality program. KENO-VI contains features currently in KENO V.a while allowing more complex geometry modeling. The geometry package in KENO-VI is capable of modeling any volume that can be constructed using quadratic equations. In addition, such features as more predefined geometry volumes, geometry intersections, body rotation, hexagonal and dodecahedral arrays, and array boundaries have been included to make the code more flexible. These features allow the user to readily solve large geometrically complex problems whose computer storage requirements and geometric complexity preclude solution by the previous versions of KENO.

Over 20 predefined geometry shapes have been incorporated into KENO-VI. Additional volumes can be constructed using the QUADRATIC geometry record, which allows the user to specify any volume that can be modeled using quadratic equations. The ability to intersect the volumes makes it possible to model exactly such things as pipe intersections, which is impossible in KENO V.a. The ability to rotate bodies means volumes no longer must be positioned parallel to a major axis. Hexagonal arrays were added to simplify the construction of triangular pitch arrays. Dodecahedral arrays were added to facilitate the modeling of pebble-bed–type fuel arrangements. The use of array boundaries makes it possible to fill non-cuboid volumes with an array, specifying the boundary where a particle leaves and enters the array. KENO-VI retains KENO V.a features such as flexible data input, a Pn scattering model in the cross sections, a procedure for matching lethargy boundaries between albedos and cross sections to extend the usefulness of the albedo feature, and restart capabilities.

The 238-group ENDF/B-V library is a general-purpose criticality analysis library and the most complete library available in SCALE. It has been extensively tested and validated over a wide range of materials, systems, and energies. The library contains data for more than 300 nuclides as processed by the AMPX-77 system. The library has 148 fast groups and 90 thermal groups (below 3 eV). Most resonance nuclides in the 238-group ENDF/B-V library have resonance data (to be processed by NITAWL-III) in the resolved resonance range and Bondarenko factors (to be processed by BONAMI) for the unresolved range. This library

contains resolved resonance data for s-wave, p-wave, and d-wave resonances (R = 0, R = 1, and R = 2, respectively). The 235U ENDF/B-V data result is slightly too reactive in the epithermal range, while the 238U data result has slightly too much resonance capture.

All nuclides in the 238-group LAW Library use the same weighting spectrum consisting of Maxwellian spectrum (peak at 300 K) from 10-5 to 0.125 eV, a 1/E spectrum from 0.125 eV to 67.4 keV, a fission spectrum (effective temperature at 1.273 MeV) from 67.4 keV to 10 MeV, and a 1/E spectrum from 10 to 20 MeV.

All nuclides use a P5 Legendre expansion to fit the elastic and discrete-level inelastic scattering processes in the fast range. A P3 fit is used for thermal scattering. Thermal scattering kernels are provided at various temperatures (K) as presented in the SCALE manual. All other scattering processes use P0 fits.

3. Shipping cask description

Illustration of the shipping cask is shown Figure 2:

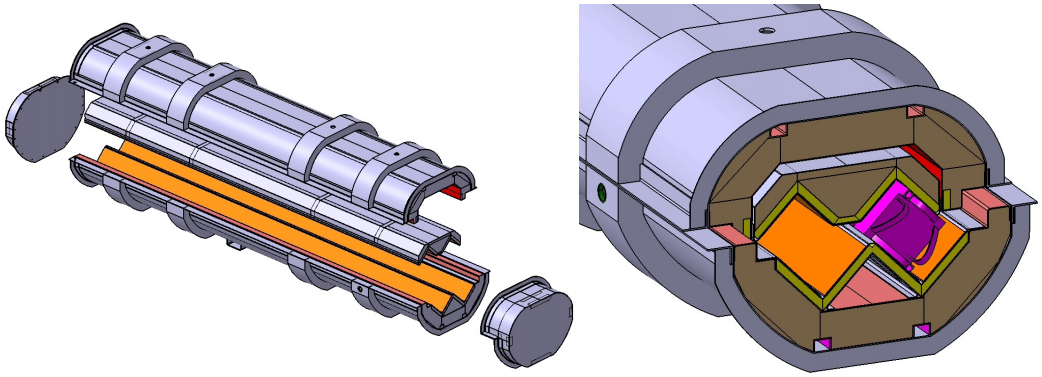

Figure 2

The package consists of a rectangular stainless steel shell with round corners and a horizontal axis, housing internal equipment comprising the following main components:

- a W-shaped frame fixed to the lower shell for receiving the 2 fuel assemblies,
- an inverted W-shaped frame fixed to the upper shell for clamping the 2 fuel assemblies,
- 2 end plates including covers dampers serve to close the cavities.

When assembled, the above components form two symmetric neutron cavities, in which the assemblies are placed.
The external shell is modeled with 3 mm thickness (the real thickness is 5 mm). The external dimensions used in calculation are 950 mm x 700 mm.
Spacings between the W-shaped frames and the shell are filled with foam for thermal protection goal. The bundles are rotated 45 degrees and are spaced 350 mm center-to-center.
The W-shaped frame is made from the assembly to outside, of layers of:

- 5 mm void gap,
- stainless steel with 2 mm thickness,
- Boral with 3 mm thickness,
- HDPE (high density polyethylene) with 25 mm thickness.

For the preliminary study shown in this paper we account for the 12 feet fuel assembly design.

4. Calculation Case definition

4.1. Methodology and characteristics

Preliminary studies show that among the AREVA's fuel assembly design the Mk-B11 produced the highest Keff values. The Mk-B11 is the most reactive fuel assembly under flooded conditions. Therefore, all subsequent calculations are performed with the Mk-B11 assembly type.
For this study, the container array size is conservatively modelled as a 15x15x3 (675 packages). Assuming that this analysis would cover arrays of both damaged and undamaged packages, this array size would justify a Criticality Safety Index (CSI) of 0.4.
The configuration is an array (15x15x3) of damaged packages (rectangular modelling) and a reflector consisting of 30 cm water surrounding the array.
Some more calculations have been performed on an infinite array of damaged packages as well as for a single package.

The fuel is modelled by combining the following parameters:

- diametric gap flooded or voided,
- 3 pellet diameters: minimal, nominal or maximal.

The following hypothetical accidental conditions (HAC) were considered:

- Fuel assembly fully flooded,
- Preferential flooding, loosed part of the cask are voided,
- Accounting for tolerances for the pellet dimensions (+ and -),
- Accounting for flooded and voided fuel rod gap,
- Accounting for outer shell SS thickness (or not).

The main difference between SCALE 4 and CRISTAL is the way to model the fuel assemblies. On one hand SCALE is using a "fine" fuel assembly pin by pin geometry and on the other hand CRISTAL is using an "homogenised" fuel assembly geometry. The models used for computations are shown Figure 3 and Figure 4 hereafter.
Another difference is the microscopic cross section data set, ENDF/BV with 238 neutron energy groups for SCALE and JEF2.2 with 172 neutron energy groups for CRISTAL.
Both calculations methodologies are described in section 2.1 for CRISTAL and 2.2 for SCALE.
Calculations have been performed by two independent and qualified individuals after a common input data review.

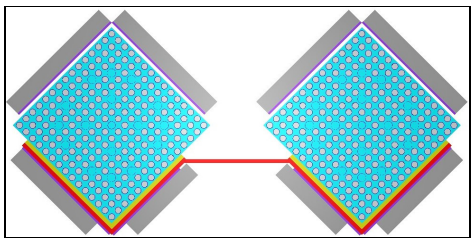

Figure 3: SCALE modeling

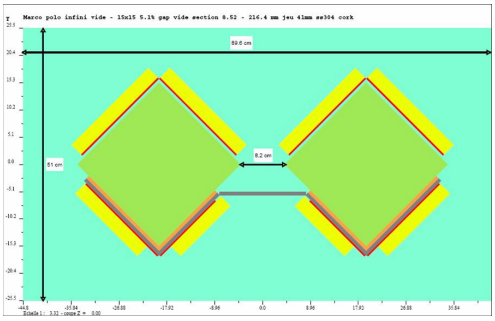

Figure 4: CRISTAL modeling

4.2. Calculation results

Table 1 lists the preliminary results. They do not account for any bias and uncertainties; they are only given as a gross comparison between the 2 code packages.

Array	Inside shell	SS304 plate modelling	Cork modelling	Fuel gap	Average SCALE Keff	Average CRISTAL Keff
15x15x3	void	void	void	water	0.950	0.941
15x15x3	void	void	void	void	0.941	0.934
15x15x3	void	ss304	water	water		0.961
15x15x3	void	ss304	water	void		0.951
15x15x3	void	ss304	cork	water		0.949
15x15x3	void	ss304	cork	void		0.945
15x15x3	void	ss304	void	water		0.948
15x15x3	void	ss304	void	void		0.937
infini	void	void	void	void	0.948	0.952
infini	void	ss304	cork	water		0.960
infini	void	ss304	cork	void	0.953	0.954
infini	void	ss304	void	water	0.953	0.960
single	water	void	void	water		0.852
single	water	void	void	void		0.842
single	water	ss304	cork	water		0.841
single	water	ss304	cork	void		0.830
single	void	ss304	cork	water		0.828
single	void	ss304	cork	void		0.820

Table 1

As it can be seen in the above table the discrepancy between the two calculation routes is small. The results overlap within three Monte Carlo statistical uncertainties (3σ).
Some sensitivity studies were performed with CRISTAL to get feelings on supplemental hypothetical accidental conditions.

5. Conclusion

The approach presented in this paper is rather theoretical and concerns mainly a comparison between two well known criticality computer codes used for shipping cask packages evaluations. Nevertheless their application to the pre design features of a new shipping cask which will be used in a near future to supply AREVA fuel assemblies all over the world lead to get confidence in calculation methodology as far as criticality safety is concerned. Furthermore getting certificate or agreement following NRC and IAEA regulations is an important item regarding worldwide local Safety Authorities.