State of the art of Monte Carlo technics for reliable activated waste evaluations

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ABSTRACT

This paper presents the calculation scheme used for many studies to assess the activities inventory of French shutdown reactors (including Pressurized Water Reactor, Heavy Water Reactor, Sodium-Cooled Fast Reactor and Natural Uranium Gas Cooled or UNGG). This calculation scheme is based on Monte Carlo calculations (MCNP, [2]) and involves advanced technics for source modeling, geometry modeling (with Computer-Aided Design integration), acceleration methods and depletion calculations coupling on 3D meshes. All these technics offer efficient and reliable evaluations on large scale model with a high level of details reducing the risks of underestimation or conservatisms.

I. Context

The prediction of activities inventory of structures surrounding the core of a reactor drives the cost of its dismantling. This requires several approaches: measurements and calculations ([1]). Indeed the activities inventory is the input for waste classification assessment, doses evaluation leading to define the decommissioning scenario and schedule and related safety analyses.

This paper presents the calculation scheme based on Monte Carlo calculations (MCNP transport code, [2]) to evaluate accurately the activities and the subsequent waste category. This scheme implies:

- To model source distribution of the core according the fuel management history,
- Adequate level of details in the 3D geometry model,
- The use of advanced variance reduction methods,
- The coupling between MCNP flux results and depletion calculation codes.

The calculation scheme has been used for many studies supporting EDF DP2D efforts to predict the activities of French shutdown reactors. These include sodium fast breeder reactor, heavy water reactor, Natural Uranium Gas Cooled or UNGG reactors (Uranium Naturel Graphite Gaz, natural uranium fuel, graphite moderated and CO_2 cooled) and light water pressurized reactors (see Figure 1). The versatility of MCNP permits to carry calculations in all of that type of reactor with an adequate calculation of neutron spectrum effect.

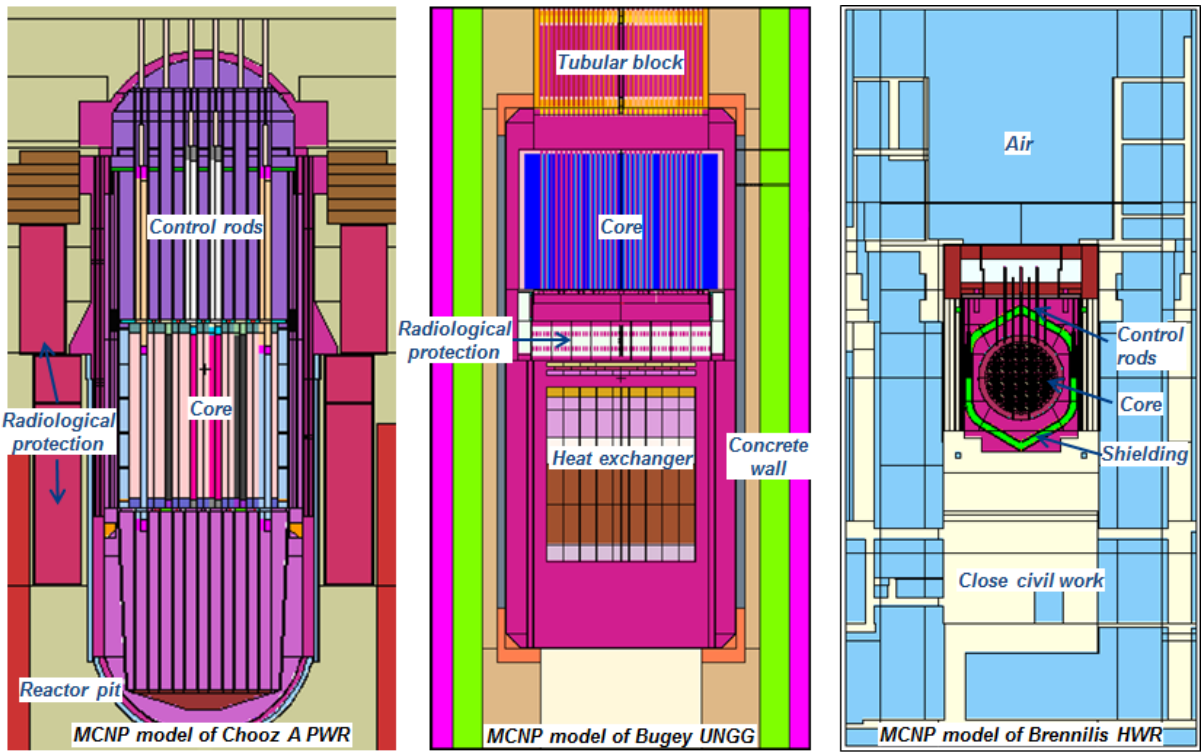

Figure 1: Vertical cross-sections of several MCNP models

II. Source and geometry modeling

The AREVA methodology implies a high fidelity for the source modeling (axial and radial power distributions along the fuel assembly pins) as shown figure 2.

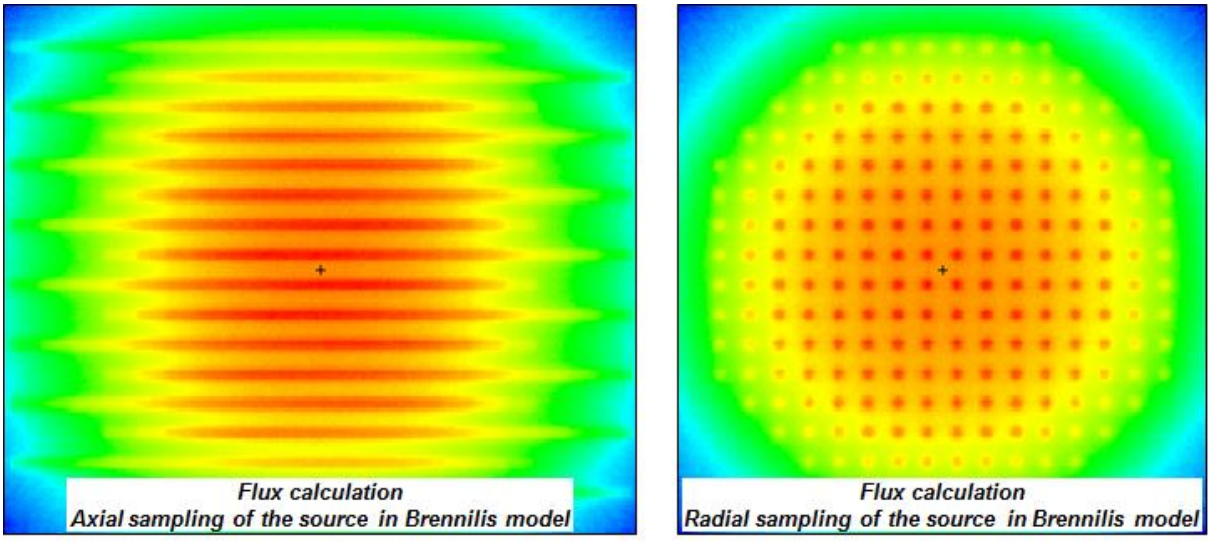

Figure 2: Illustration of the sampling of a source by calculating the flux in a void model

The AREVA methodology is also based on a 3D geometry modeled with high details thanks to an internal software taking benefits of CAD (Computer-Aided Design) files (see Figure 4).

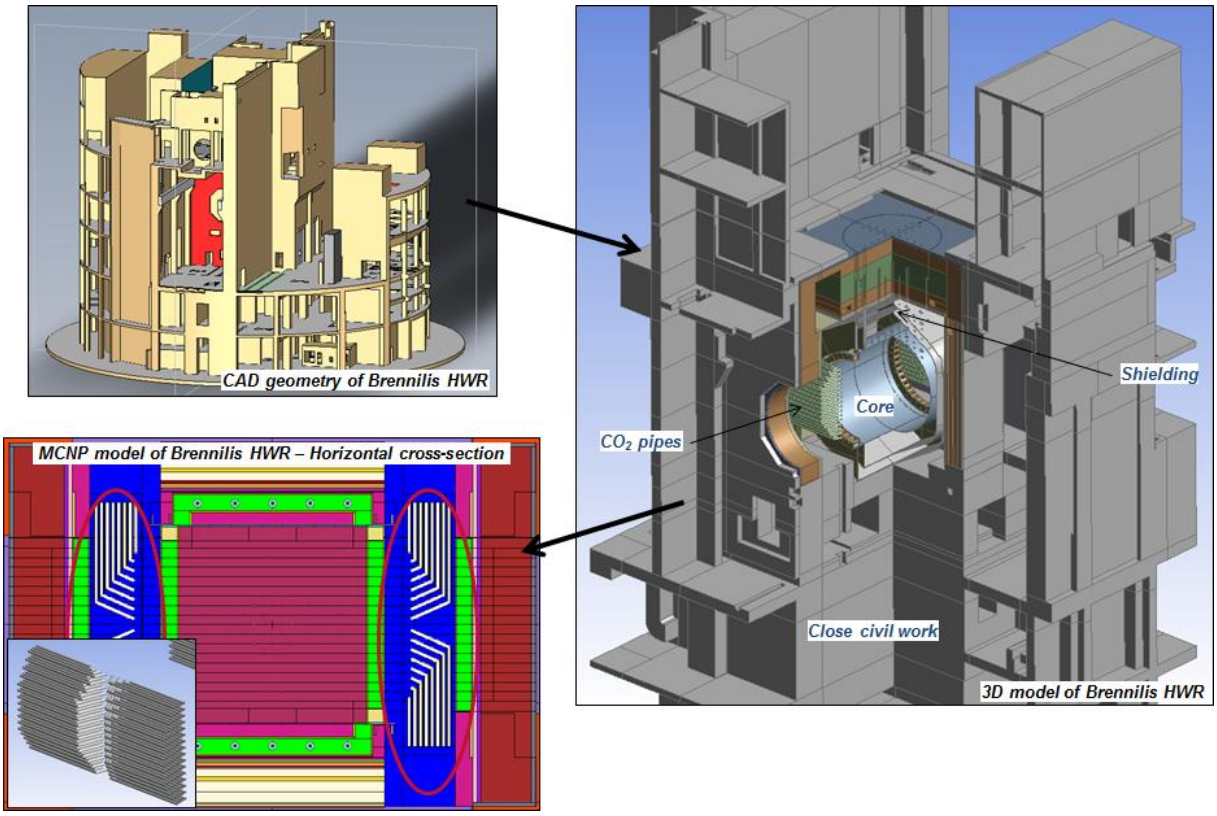

Figure 4: Illustration of the use of CAD file in the creation of Brennilis model

The 3D complex geometries give accurate results which are confirmed by measurements comparison ([3]). Indeed geometry simplification could lead to overvaluation, wrong evaluation of local spectrum effects or wrong evaluation due to edge effects. Moreover, the simplification can lead to the cancellation of neutron pathways. The Figure 5 presents the neutron distribution along the concrete structure surrounding the core of a UNGG. A tiny hole in the lower structure of the core leads to an additional flux peak. The use of approximate or simplified geometrical description could not simulate properly this kind of 3D effects.

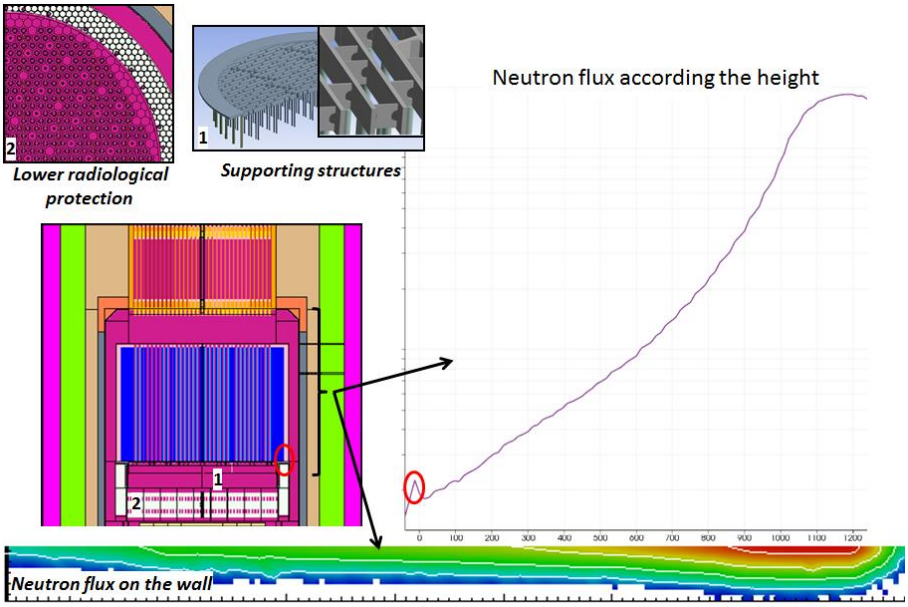

Figure 5: Illustration of a neutron "leak" on Bugey UNGG

Moreover, in most of the cases, inventory assessments required to deal with whole 3D models. Considerations of partial geometrical description could conduct to wrong evaluations of the neutron streaming. Indeed the following figures show how the neutron pathways have to be estimated to have a reliable assessment of the neutron flux and activation induced. Complex and complete geometries assure to identify bypassing ways (see Figure 6a and Figure 6b), defaults in shielding or also streamings leading to neutron beams.

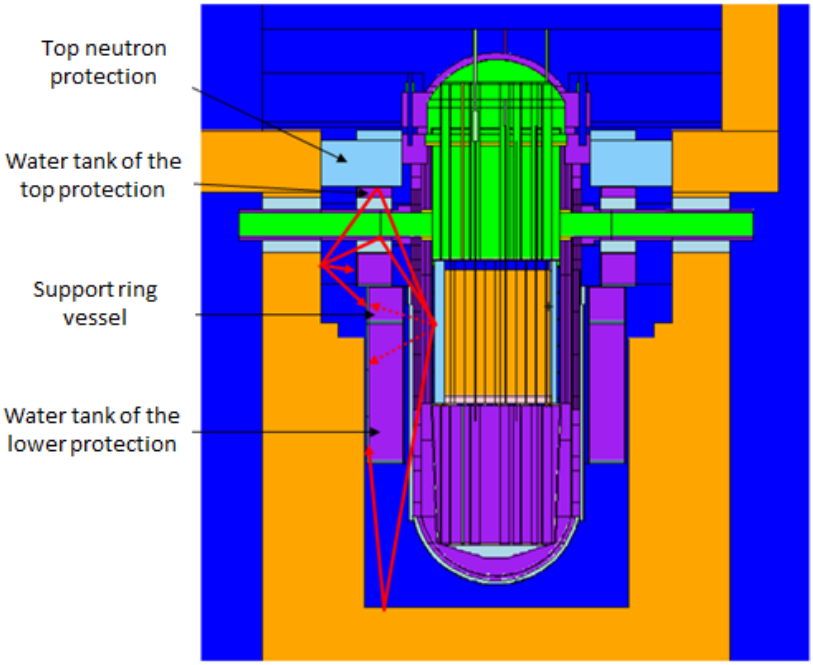

Figure 6a: Direct contributions (dashed lines) vs. bypassed contributions (plain lines) on a neutron flux calculation on Chooz A PWR

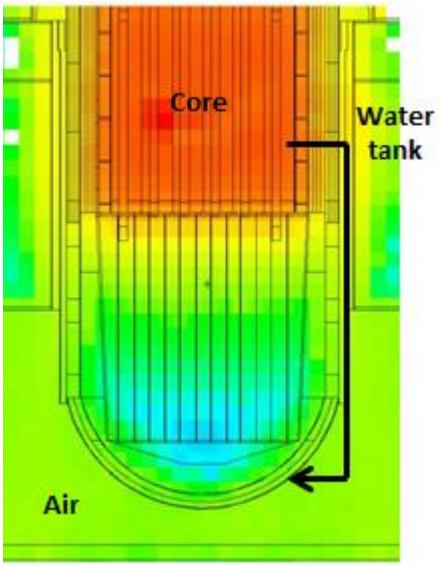

Figure 6b: Illustration of the bypass of the neutron protection by the neutron flux toward the bottom vessel on Chooz A PWR

The comparison between calculations and measurements shows a slight overvaluation of the radioactive inventory ([3]). This overvaluation can be linked to the type of structures tallied but can also be linked to the uncertainties attached to the measurement, to the nuclear data, to the fuel management, to the irradiation condition history and to the statistical error of the calculation (increasing with the distance between the active part of the geometry and the location of the tally)... All of those uncertainties are not appreciated easily. That is why, it is better to consider complex geometries to reduce the uncertainty calculation due to the precision of the model. Moreover, with such a model, those overvaluations are reasonable as the waste classification is the same.

Finally, the tools and internal software used by AREVA allow the users get precise geometries readable by the code MCNP quite fast. Indeed, according the size of the CAD geometry and its initial quality (in term of "precision"), a user can spend only a few hours to several days to adapt the geometry. The last step, the translation of this 3D geometry into an MCNP input format, being automatically done by an internal software, allow the user to get a MCNP model that fits the limits of the last versions of the code.

III. Variance reduction

The inherent drawbacks of the Monte Carlo methods are got around thanks to parallel calculations and the use of hybrid variance reduction methods.

The CADIS (Consistent Adjoint Driven Importance Sampling) method, widely presented by Oak Ridge National Laboratory publications [4], is included in an internal routine enabling the reaching of the convergence criteria on the hardest locations (see Figure 7). This method is based on the calculation of adjoint flux distributions with SN 3D codes in order to compute flux contributions to a target location. In the right part of the Figure 7, the color scale, from blue to red, quantifies the probability of a neutron particle to contribute more (blue) or less (red) to the target tally. The

quantification of these contributions is energy dependent and is directly equal to the biaising factor applied within MCNP of the neutron transport problem (WW cards: Weight Windows cards).

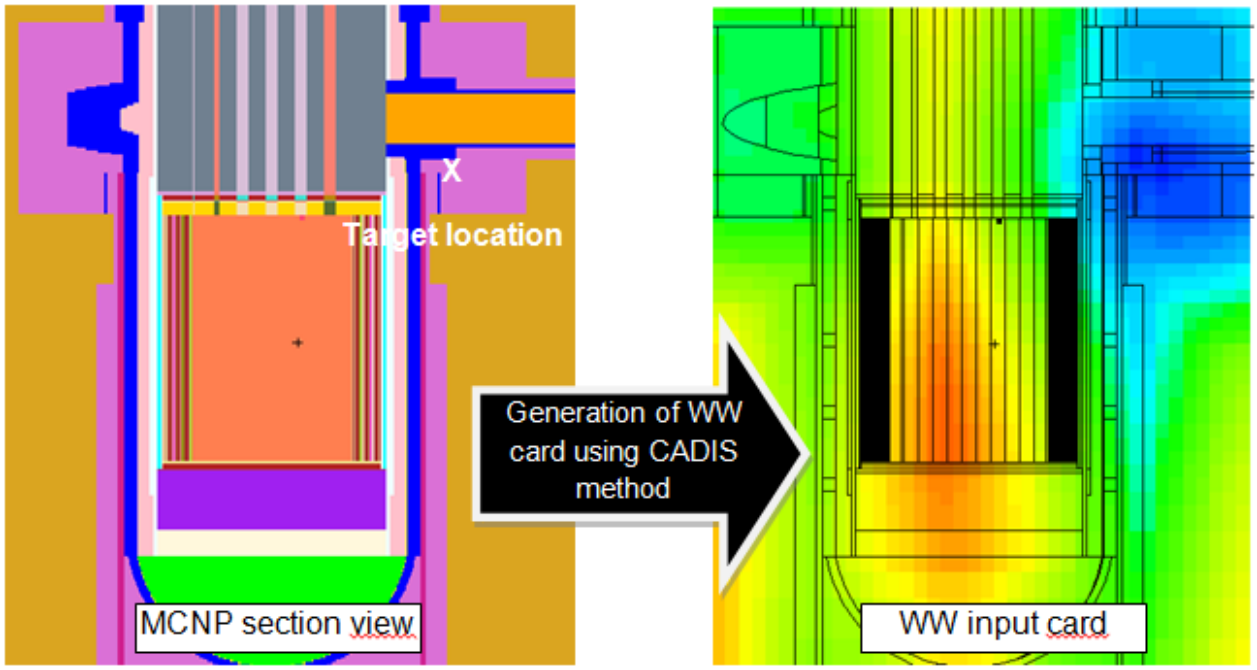

Figure 7: Illustration of the use of the CADIS method on a MCNP model

This internal routine also enables the generation of Weight Windows cards (WW cards) allowing the calculation not only on one location but on large scale areas (full scale optimization problem). This routine is helpful to deliver neutron flux mapping along deep components (concrete walls). The use of those techniques is extremely efficient on wide models where full characterization is needed. As an example Figure 8 gives the results of neutron calculation on the quarter of a reactor building. The size of the model is 24.90m x 24.90m x 60m. This figure presents 400 000 neutron tallies results inside concrete walls after decades of hours of calculation.

The results are stored in a 3D mesh format allowing efficient data transfer to depleting codes as describe in the following section.

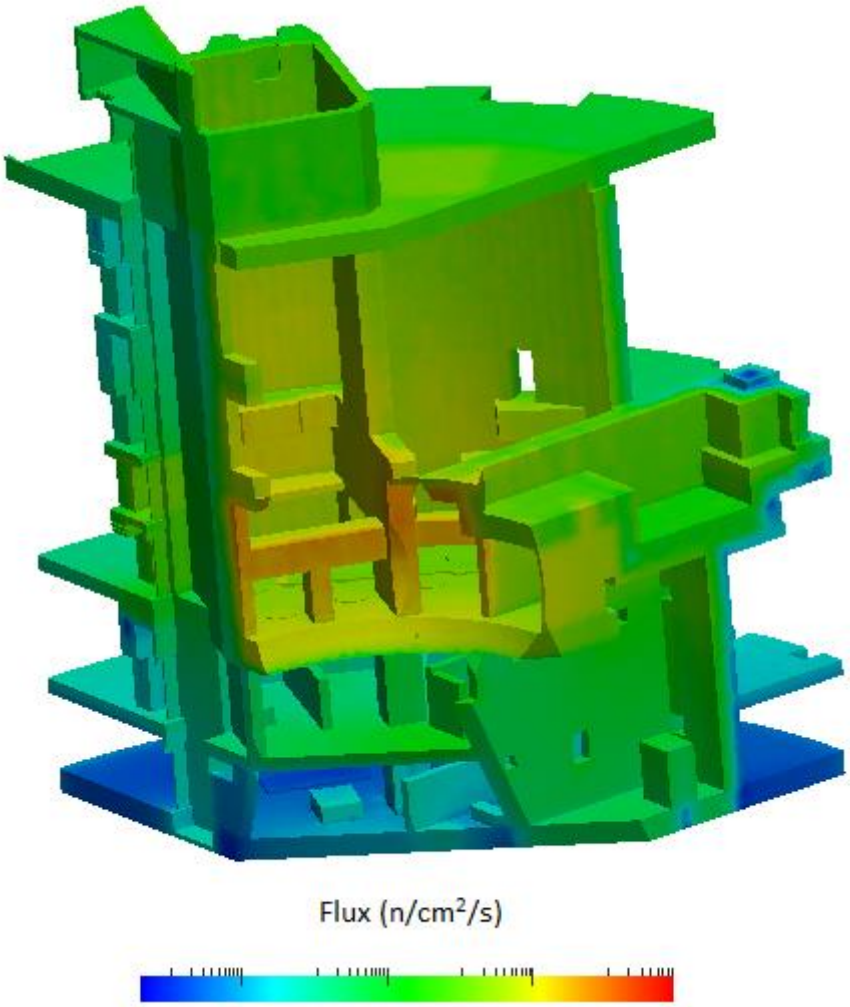

Figure 8: Illustration of neutron flux distribution on the quarter of the civil work of a reactor building using full scale optimization method (systems and reactor pit hidden)

Statistical criteria, for most difficult tallies located outside of the active part, are reached within only a couple of decades of hours of calculation by the use of those last variance reduction techniques and the parallel computations of the last versions of MCNP.

IV. MCNP and depletion calculation codes coupling

The neutron waste characterization of full reactor induces challenges in data storage and management. AREVA internal software development allows coupling between MCNP flux calculations and activation and depletion calculation codes (DARWIN [5] and/or SCALE [6]). 2D/3D visualization of results is also available for easy analysis and post-treatment (see Figure 9).

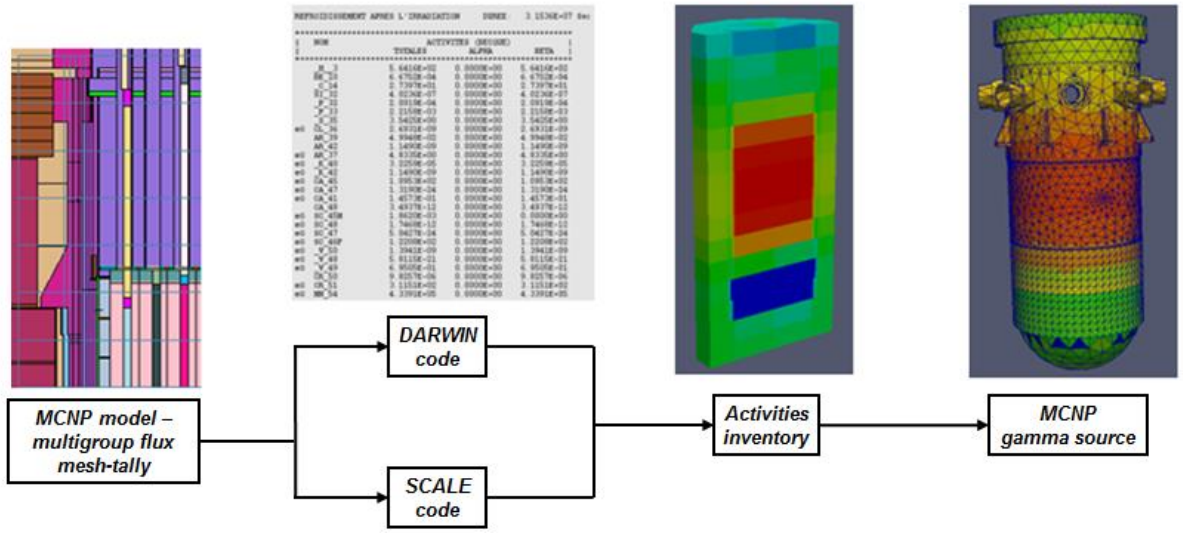

Figure 9: Illustration of the coupling between MCNP flux calculations and activation and depletion calculation codes

The coupling of depletion calculation codes can be done with several MCNP calculation points or with MCNP neutron flux 3D meshes of tallies. The scalars fields on each cells are nuclide inventories, waste classification indicator, gamma source terms on user specified decay time steps. At that step the knowledge of material impurities contents is essential.

The use of the CADIS method with full scale optimization enables the study of vast areas of a reactor and the determination of the activation in the same areas. This coupling avoids human error in the management of these huge quantities of data (storage of neutron spectrum and history, creation of depletion calculation code inputs, post-treatment of extraction of inventories quantities). The activation gradient can be so calculated on an adequate 3D mesh avoiding considering penalizing values for all the equipment/area. This is another example of how 3D considerations with appropriate techniques could lead to reduce conservatisms.

V. Conclusions

Both 3D modeling with efficient tools and variance reduction optimization warrant the release of reliable evaluations in a short time frame. The outputs of the studies are also directly used to design and optimized the segmentation of activated waste and packaging and run ALARA (As Low As Reasonably Achievable) approach of personal exposure.

This calculation scheme is an efficient tool to demonstrate the robustness of the waste assessment before and during the decommissioning (supporting visualization of results).

For all the steps of the calculation scheme AREVA has acquired and developed efficient, affordable and reliable advanced techniques.

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