Advanced Techniques for Monte Carlo simulations of ex-core responses supporting safety analysis: damage to Pressure Vessel

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Within the framework of simulations supporting safety analyses of nuclear power plants, Monte Carlo codes constitute a reference tool utilized in connection to the extension of the lifetime of currently operating nuclear power reactors, as well as the preparation of decommissioning procedures. In particular, the calculation of neutron and gamma damage to the reactor pressure vessel is a relevant step both in the design and in the licensing processes related to current Gen-II and innovative Gen-III pressurized water reactors (PWR). Thus, enhanced numerical methodologies are of great interest to achieve improved safety evaluations and are relevant tools for industry, utilities and regulators. In this work, an innovative methodology is applied to the evaluation of the radiation dose to the pressure vessel - taking into account the different contributions due to both neutron flux and gamma radiation. The calculations concern a Gen-III reactor model and are carried out with the Monte Carlo code MCNP (versions 5.1.4 and 6.1). The method implemented involves a modification to the MCNP code and allows to perform dose calculations at various positions in a single eigenvalue problem – without the preparation of a fixed “external” source based on the principal eigenfunction. The fundamental mode is recalculated within the same simulation in which the responses of interest are obtained. The presented approach is compared against a more classical method based on a decoupled methodology, with a fundamental mode calculation followed by a fixed source simulation in which ex-core results are obtained. A multiple response capability allows an approach in which the variance reduction parameters are optimized for a series of ex-core locations - achieving a calculation performance, which is compared to the standard approach.

1) Introduction

Within the framework of simulations supporting safety analyses of nuclear power plants, Monte Carlo codes constitute reference tools that have been utilized in the last decades for the determination of ex-core responses connected to safety issues concerning the ageing management of current nuclear power plants and the design phase of innovative systems.

A significant aspect is the estimation of the dose delivered to the reactor pressure vessel by both the neutron radiation and the gamma field inside the reactor pit. The effect of this dose causes atom displacement (dpa) in the material and impacts the vessel reliability during operation [1].

In particular, the physical quantities used to evaluate the radiation impact for ageing purposes are neutron dpa and gamma dpa, or neutron fluence (energy above 1 MeV) and gamma kerma.

These evaluations are relevant during the licensing process and are a subject of interest for technical support organizations (TSO) and nuclear safety agencies, concerning the lifetime extension of currently operating nuclear power reactors.

Monte Carlo transport codes are particularly utilized for the estimation of neutron and gamma contributions to dpa, since accurate and detailed responses are requested at particular points of the system, mainly in ex-core locations.

The Institute of Radiological Protection and Nuclear Safety (IRSN) acts as technical support to the French nuclear safety authority. In this context, one of its missions is to improve nuclear safety through the enhancement of knowledge regarding operating nuclear power reactors.

For some years, ENEA (The Italian National Agency for New Technology, Energy and Sustainable Economic Development) and IRSN have collaborated on Monte Carlo simulations, through the development and implementation of variance reduction techniques.
The innovative methodology here utilized is the Direct Statistical Approach (DSA) [2], which allows to determine variance reduction parameters for a set of responses, during the same simulation. In addition, the calculation is performed re-estimating the fundamental mode during the same run – without any decoupling and avoiding a subsequent fixed source calculation [3,4].
In this work, the new approach is compared to the standard procedure involving decoupling, for a number of neutron and gamma flux and dpa responses.
Neutron and gamma contributions to fast flux and dpa are evaluated at different points on the vessel surface of a Gen-III nuclear power reactor, by means of the MCNP transport code (versions 5-1.4 and 6.1) [5,6].
Results are presented and some characteristic features of the innovative method are shown, emphasizing the enhancements which are of interest for damage calculations as well as for other deeper penetration problems concerning radiation shielding and decommissioning.

2) Reactor Model and Responses

In this work, the neutron and gamma fluxes on the inner surface of the vessel are evaluated for damage determination. The model of the Gen-III reactor is prepared by means of the Monte Carlo code MCNP, which allows a detailed 3D geometrical description of the reactor core, the internals within the vessel and the reactor pit. The model is based on data available to the authors.
The fuel compositions correspond to those of the end of an equilibrium cycle and are unique for each assembly. The poisoned rods are included. No axial burn-up effect is taken into account. The geometry of the assemblies is detailed pin-by-pin as reported in figure 1.
Pressure vessel and reactor core are described in figure 2. Dose responses are requested at the inner vessel surface and at core mid-plane, at four angular positions: 0 degrees, 22.41 degrees, 36.26 degrees, 45 degrees.
These locations are chosen in order to determine where the neutron and gamma fluxes reach a maximal value.
Nuclear data set utilized is ENDF/B-VII.1 [7] evaluated at both 293 K and 600 K for cold regions and hot portions inside core, respectively. Data tables for thermal neutrons are ENDF/B-VII.0 [8] for water at both 293 K and 600 K – for reasons of format compatibility between MCNP5 and MCNP6.

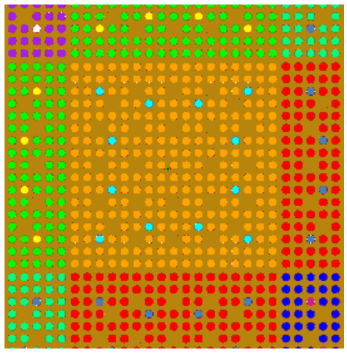

Figure 1.
MCNP model of the pin-by-pin geometry of the assemblies

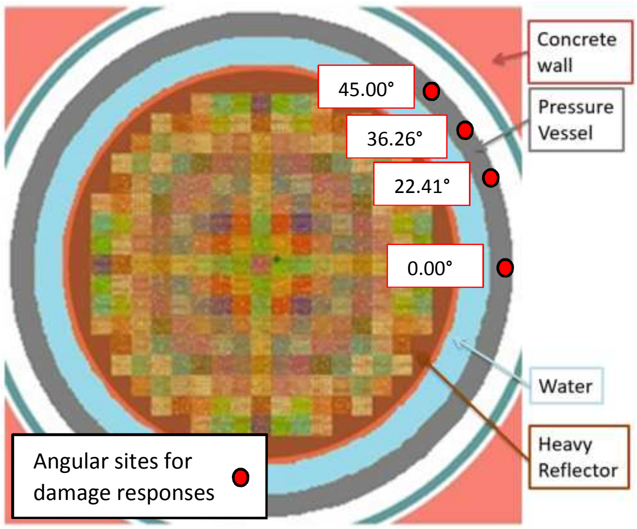

Figure 2.
MCNP reactor model and outside core (at core mid-plane)

Three different neutron responses and one gamma response are requested. The neutron damage responses consist of a dpa tally with a response function for each of the components of the steel pressure vessel from Konobeyev dosimetry data [9,10], the total flux above 1 MeV and the total flux above 100 keV. The gamma dpa response consists of 14 energy groups between the threshold of 700 keV and 20 MeV from [11] and is only for iron.

All responses are requested at the four angular positions; thus 16 tallies are considered in total.

3) Methodology

The method utilized here is the Direct Statistical Approach (DSA). It was originally developed for fixed source problems and has been recently extended to eigenvalue problems.

The DSA optimizes splitting and Russian roulette parameters at surfaces in space and in energy, either independent of, or dependent on, the weight of the progenitor arriving at the surface [2]. An innovative feature of the DSA consists of estimating and enhancing the second-moment and the calculation time, in contrast to classical approaches that rely on the first-moment.

The aim is to find the optimum trade-off between over-splitting and under-splitting: the former wastes time and instead under-splitting induces too high variance per source history.

Moreover, the DSA allows to produce variance reduction parameters for a number of responses simultaneously – so called multi-response capability - both in the fixed source version and in fundamental mode one.

Used in fundamental mode simulations [12], the DSA allows to score ex-core (or in-core) local responses while keeping control of the fundamental mode through “global” responses (typically neutron production or heating in the appropriately segmented fissile region).

Superhistories are employed to extend each cycle by increasing the number of fission generations between normalization points in the source-iteration scheme. (It has been found [13] that employment of superhistories dampens to acceptably low levels fluctuations in the fundamental mode that are stimulated by the variance reduction.)

The damage responses are determined through the two following calculation approaches:

1) Decoupled approach: an analog fundamental mode calculation produces the source that is then employed in a second step in which variance reduction parameters are produced with the DSA (in fixed source mode) and damage responses are generated.

2) Non-decoupled approach: a simulation generates the damage responses in a single calculation (with the DSA in eigenvalue mode). This utilizes variance reduction parameters within a fundamental mode simulation.

4) Standard Decoupled Approach

The standard approach for the evaluation of ex-core responses during a Monte Carlo calculation consists of a two-step procedure. First, a fundamental mode simulation is run, tallying fission neutron production according to some radial binning (which may be at a pin-by-pin level) and a specific axial binning. Afterwards, a fixed source problem is solved using the previous tally output as a source term. Variance reduction parameters are produced during this second step and optimized for the required responses.

This two-step approach contains some approximations based on certain hypotheses:

- the binned source has a pin-by-pin pattern with a given axial discretization
- the source in each bin is homogenized both radially and azimuthally
- the fixed source has a constant energy distribution and further fission reactions are disabled

The impact and the limits of this approach constitute the topic of this work and one of the reasons for the comparison with the single calculation.

4.1) Fundamental Mode Calculation

The fission source is produced during a fundamental mode calculation, through 1200 cycles consisting of 10^6 histories each – with no inactive cycles since the transient stage of generating the fundamental mode source was already made. Only neutrons are transported and neutron fission production is tallied on a pin-by-pin pattern in all fuel assemblies in 52 axial bins. The length of the bins is about 3-4 cm at the core bottom and about 10-11 cm from some 150 cm below core mid-plane to 150 cm above core mid-plane. Then, in the upper part of the active zone, the length of the bins is 5-8 cm.

The relative standard deviation provided by MCNP for fission neutron production - without taking into account the correlation between the cycles - is below 5% for about 55% of the bins, 5%-10% for 18% of the bins, and the remaining 18% of the active bins show a relative standard deviation of 10%-20%.

4.2) Fixed Source Calculation

Following the fundamental mode step, a fixed source simulation is performed in which the only source term considered is constituted by the fission neutrons, previously tallied. (For gamma responses, the activation of structures outside the core is not taken into account in this study, although in principle it is not negligible).

The results of the previous calculation are managed through an external Fortran routine, that prepares the customized fixed source, according to the core pin-by-pin pattern.

Previous studies showed that a flat radial distribution within an assembly leads to an overestimation of the responses outside the vessel [14]. Thus, in this analysis a pin-by-pin pattern is preferred together with an axial binning to provide each pin with its own axial variation. No radial subdivision is made in each pin.

The source position is sampled employing a patch that reads the output from the first step while the source energy distribution is taken as the standard MCNP Watt spectrum for thermal fission.

The damage responses previously described are requested in the second calculation step. Then, appropriate variance reduction parameters are prepared with the DSA (fixed source version) and optimized for all tallies simultaneously with the multi-response capability [2]. These parameters can then be converted to a Weight Window to be implemented in a standard MCNP input.

There are 7 Weight Window energy groups for neutrons and 4 for photons, with upper limits for neutrons: 1 eV, 1 keV, 20 keV, 0.2 MeV, 2 MeV, 5 MeV, 20 MeV and upper limits for photons: 1.5 MeV, 3 MeV, 5 MeV, 20 MeV (with a 700 keV threshold).

The fixed source simulation is run up to a total of 10^{10} histories.

5) Single Eigenvalue Calculation

So as to avoid decoupling and the approximations inherent in such an approach, a single calculation is performed with damage responses at the vessel surface. A fundamental mode simulation is run in which damage responses are requested and variance reduction parameters are generated by the DSA.

Employing variance reduction parameters normally distorts the fundamental flux shape. To mitigate this, additional global responses – distinguished from the local vessel damage responses – are inserted inside the core: in our case the core is subdivided into 4 non-equal axial segments and 4 radial segments and thus 16 in-core fission-heating tallies are requested.

There is then a trade-off between a necessary distortion of the particle population aimed at the local responses (damage tallies) and the fundamental mode restoration according to the global in-core responses (fission heating) [3,4,12].

It was found furthermore that the fundamental mode could only be properly balanced through the employment of superhistories [13]. Here a superhistory of 10 fission generations was employed, a number shown to be sufficient to achieve this goal [13].

6) Results and Comparison

Monte Carlo results obtained through both the single eigenvalue approach and the decoupled approach are compared. The decoupled approach is considered as reference, since it is common practice in ex-core problems. Notwithstanding, the decoupled approach implies a certain degree of approximation.

Firstly, there is the source discretization: radial homogenization of the source throughout each pin does not take into account that neutron production is dominant in the outer part of the pin. (Instead we have seen that axial binning tends to be less important for most ex-core responses.)

Secondly the adoption of a single fission spectrum (^{235}U Watt, thermal fission) certainly represents an approximation as there is a non-negligible quantity of Pu in the core.

The DSA method in a single fundamental mode calculation attempts to quantify such approximations.

Note all the reported results have a relative standard deviation less than 1%. Furthermore, the two approaches are statistically independent.

Foremost, the standard decoupled approach shows a systematic underestimation of the results, compared to the single fundamental mode calculation: this occurs for all tallies at core mid-plane and for both neutron-based and gamma-based responses.

Neutron flux (above 1 MeV) differs between 10% to 14%, going from 45° to the 0° position, as shown in figure 3.

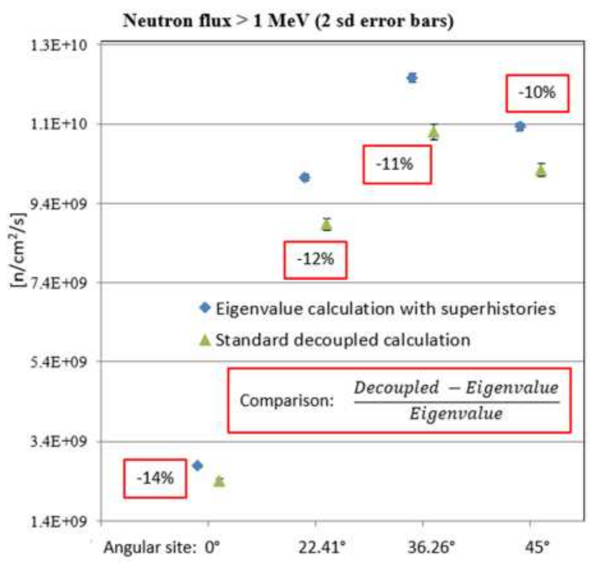

Figure 3.
Single eigenvalue and decoupled approaches:
comparison of neutron flux (above 1 MeV)

Considering neutron dpa evaluations the trend is similar (see figure 4). Comparison yields an underestimation of the standard decoupled approach of 11%-12% with respect to the single eigenvalue approach.

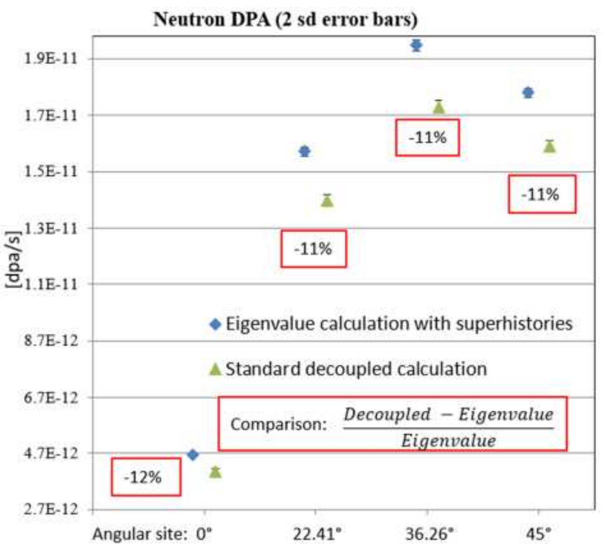

Figure 4.
Single eigenvalue and decoupled approaches:
comparison of neutron dpa

Conversely, the gamma dpa results are less sensitive to the simulation approach. In fact, the standard decoupled approach underestimates the gamma dpa by only some 2% to 5%, as reported in figure 5.

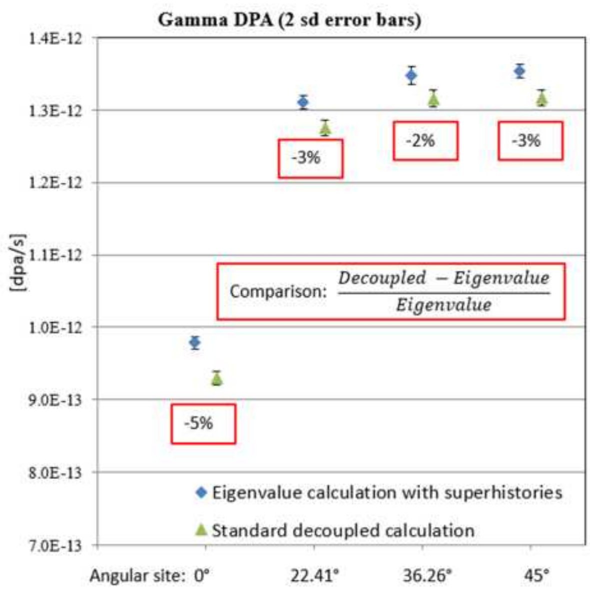

Figure 5.
Single eigenvalue and decoupled approaches:
comparison of gamma dpa

The systematic underestimation of the standard method – with respect to the DSA – constitutes the effective result of the work, which summarizes all the evaluations – for both neutron and gamma tallies.

7) Conclusions

Within the framework of nuclear safety analyses supporting licensing and design of existing and innovative nuclear reactors, Monte Carlo codes are utilized to obtain detailed responses: in particular, fast neutron flux and dpa dose delivered by both neutrons and gammas to the pressure vessel.

Therefore, innovative methods to improve calculation efficiency as well as precision of the results are key tools to enhance safety analyses.

Currently, a decoupled approach is used to obtain ex-core responses in critical systems, through a first fundamental mode simulation in which a source is prepared for a subsequent decoupled fixed source calculation - provided with ad-hoc variance reduction parameters.

The innovative approach here proposed is based on the Direct Statistical Approach (DSA) [3,4,12,13]. It is implemented as a patch applied to MCNP and allows variance reduction for multiple responses within the same fundamental mode simulation without the need for decoupling. This approach is applied here to a Gen-III PWR model, in which neutron and gamma damage

responses are requested at the vessel surface at different angular positions at the core mid-plane.
Results show an interesting trend: the decoupled approach underestimates the tallies compared to the single eigenvalue approach by about 10%-14% for neutron-based responses (flux or dpa). Moreover, gamma dpa results are underestimated by less than 5%.
In addition, it is worth notice how calculation times between methods are similar.
Further simulations and tests are necessary in order to establish which approximation is mainly responsible for the differences.
This method can then provide an enhanced and more efficient methodology to support safety analyses of nuclear power plants, with interesting possible applications both in the radiation shielding and decommissioning domains.

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