APPLICATION OF CALCULATED MSM FACTORS USING TRIPOLI4® SEQUENCE ON BORON LINED PROPORTIONAL COUNTER ROD WORTH MEASUREMENT

J. COUYBES[1], S. NICOLAS[1]
L. CHABERT[1], L. MANIFACIER[1], T. BARBOSA[2], R. GERLIER[2]

[1]Safety and Power Plant Process
Neutronic – Shielding – Criticality Department
TechnicAtome, Aix en Provence, France

[2]TechnicAtome
Centre de Cadarache, Saint-Paul-lez-Durance, France

Corresponding email: julien.couybes@technicatome.com

ABSTRACT

The topic addressed deals with the determination of adjoint parameters for instrumentation relevance. This is a crucial subject for comprehension of subcritical levels in the frame of safety analysis. Indeed, such states require interpretation and raw data cannot be processed as such. To do so, the transcription of core reactivity through instrumentation located in the reactor periphery is considered with the use of MSM factors [1],[2]. We implement this method inside a TRIPOLI4® [3] sequence in order to establish predictive mapping of MSM factors and figure out optimal position for instrumentation location at the beginning of reactor operations.
Firstly, MSM factors are introduced, along with the designer point of view for geometry construction based on ROOT package [4]. At this point, the methodology of TRIPOLI4® calculation is explained in detail, including the sequencing associated to and how the Green Functions are performed within TRIPOLI4®.

In this second part and within the verification framework, the previous method is extended to a "fictitious core" developed in TechnicAtome for Monte Carlo [5] calculation and for different core pattern loadings. After the completion of these numerical validations gained on a High Performing Cluster, the method is then expanded to critical mock up [6] and challenged to recent experimental results for validation. The comparisons end up with a good agreement between predictive calculation and experimental values of reactivity worth.

Finally the document ends with a mid-term projection for outlooks and improvements, for ensuring an enhancement of the safety approach. Several items are discussed especially, fine tuning for the spatial meshing (regarding instrumentation size) and the impact on TRIPOLI4® Monte Carlo code with the development of new features. Then, the authors focus on sensitivity effect concerning delayed neutron spectrum and kinetics parameters. As a conclusion, this paper proposes to validate the method exposed in the near future, using experimental data coming from many years of critical mock up operations.

KEYWORDS: MSM factors, Rod worth measurement, TRIPOLI4®, spatial meshing, correction of raw measurements

1. INTRODUCTION

Nuclear reactors operation for any purpose requires the presence of neutronic poisons to instantaneously stop the chain reaction. These safety devices have several forms and the most commonly used is dropping rods made of neutron absorbing materials (Hf, AIC, B_4C …). The capability to stop the chain reaction is often checked using experimental Rod-Drops. The reactivity is measured before and after the rod drop by one or several detectors located around the core or at experimental positions, usually far from the core.

During scram, the flux shape changes brutally and heterogeneously. Indeed, global and local (near detectors) variations of neutron flux can evolve differently.

Monte-Carlo (MC) codes can reproduce with a high fidelity the different neutron flux distributions before and after scram, and can simulate neutron propagation all the way to detector sensitivity area (boron layer for example). This paper proposes a Monte-Carlo (MC) sequence to determine the so-called MSM (*Modified neutron Source Method*) factors [1] [2] and to use them in order to correct the raw rod control worth measurement from Boron-lined proportional counters.

First, theoretical framework is reminded. Then, the proposed calculation sequence is illustrated through an application using TRIPOLI4® MC [3] code on a "fictitious core" developed for dedicated purposes [5]. Third, results of MSM factors application on experimental measurements are described. Finally, perspectives and improvements of this calculation sequence are discussed.

2. THEORETICAL FRAMEWORK OF THE ROD-DROP METHOD AND MSM FACTORS

The Rod-Drop method [7] is commonly applied to measure the reactivity value of safety rods. It consists in studying the reactor transient response through the neutron population time behaviour after a fast-negative reactivity insertion. The Rod-Drop theoretical behaviour is shown on Figure 1 and can be described by traditional point-kinetics equation **(1)**:

1. Initially, reactor is stabilized, critical and with a stable situation ; global neutron flux is n_0 ;
2. Rod is dropped, and global neutron flux decreases immediately until ρ_{0+} reactivity (here, global neutron flux is n_{0+}) ;
3. Then, an asymptotic and subcritical state is reached after the Rod-Drop ; global neutron flux decreases exponentially.

$$\frac{dn}{dt}(t) = \frac{\rho(t) - \beta_{eff}}{\Lambda_{eff}} n(t) + S_d + S_{ext} \qquad (1)$$
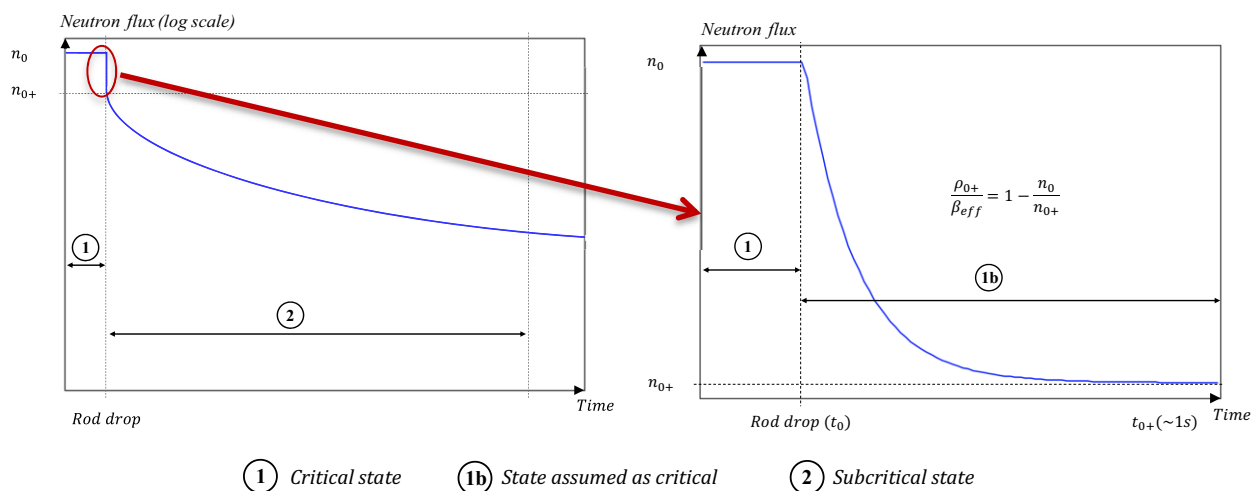

Figure 1 – **Schematic representation of neutron flux behavior during rod drop (right = zoom)**

In equation **(1)**, S_d stands for delayed sources, S_{ext} represents external sources, β_{eff} effective delayed neutron fraction and Λ_{eff} is effective mean generation time.

After rod drop (0^+ point), neutron population is essentially composed of delayed neutrons, and so driven by delayed neutrons production. Spatial distribution of precursors (*i.e.* delayed neutron source) is proportional to fission rate distribution. Nevertheless, delayed neutrons energetic spectrum is lower than that of fission neutrons [10]. This has to be considered in calculations.

Theoretical resolution of **(1)** is similar to *Maillot et al.* methodology [2]: both global reactivity and local-detector neutron flux variations are linked by **(2)**. This is due to, all-together:

- a quasi-static approach ($dn/dt(t)$ term can be neglected),
- a constant delayed neutron source before and after the scram,
- starting point is critical and stable ($S_{ext} = 0$),
- no reactivity feedback (due to a very low initial power),
- shape neutron flux profile variations is taken into account in MSM factors (see later).

Moreover, considering dynamics reactivity equals to measured reactivity, equation **(3)** can be written.

$$\frac{\rho_d}{\beta_{eff}} = 1 - \frac{n_0}{n_{0+}} = 1 - \frac{m_0}{m_{0+}} \quad \textbf{(2)} \qquad \Longrightarrow \qquad \frac{\rho_m}{\beta_{eff}} = 1 - \frac{n_0}{n_{0+}} = 1 - \frac{m_0}{m_{0+}} \quad \textbf{(3)}$$

$$\text{Considering } \rho_d = \rho_m$$

With:

- ρ_d and ρ_m = dynamic reactivity and reactivity measured by detector ;
- n_0 and n_{0+} = neutronic population respectively before and immediately after scram ;
- m_0 and m_{0+} = counter signal respectively before and immediately after scram.

Then, MSM factor (f_{MSM}) definition [1] [2] given in **(4)**, is the ratio of detectors count rates. It characterises the modifications of neutron flux distribution before and after the Rod-Drop: especially the difference between both global and local-counter variations of neutron flux magnitude. Finally, using f_{MSM} definition [1], [2], **(2)** and dynamic reactivity definition **(3)**, corrected value of measured reactivity ($\rho_{m,corrected}$) with detector is given by the equation **(4)**.

$$\rho_{m,corrected} = \beta_{eff} \cdot (1 - f_{MSM}) + \rho_m \cdot f_{MSM} \qquad \text{with} \qquad f_{MSM} = \frac{m_0/n_0}{m_{0+}/n_{0+}} \qquad \textbf{(4)}$$

3. APPLICATION OF CALCULATED MSM FACTORS USING TRIPOLI4® SEQUENCE

In this section, the transposition of the theoretical methodology described above to an industrial calculation sequence is presented. Verification, and validation are applied on a "fictitious core" [5] with fuel assemblies similar to standard experimental TechnicAtome cores. Calculation sequence is fully based on TRIPOLI4® MC code [3] associated with JEFF3.1.1 nuclear data.

3.1. "Fictitious core" model

The "fictitious core" model composed of 36 fuel assemblies, including 32 control assemblies (*cf.* Figure 2) is used for verification, and numerical validation of the methodology. All geometrical and composition data are detailed in [2]. Geometrical construction is based on the ROOT software [4]. C++ objects and ROOT features are applied within TechnicAtome studies, thereby gaining representativeness, especially by significantly reducing modelling biases and by describing components closer to reality.

Absorbers are divided in 2 families:
- 4 safety absorber groups (SA) located in the 4 core corners,
- 4 control absorber groups (CA) adjusted (bank strategy) to reach criticality (critical position is 13 cm below the fuel mid core plane).

Main neutronics parameters are summarized in Table I. Thermal neutron flux (< 0.625 eV) distribution at 30 cm below mid core (position of interest for detectors) for critical situation is shown on Figure 3.

Table I. Main neutronics parameters of "fictitious core" (uncertainties are given for 2σ)

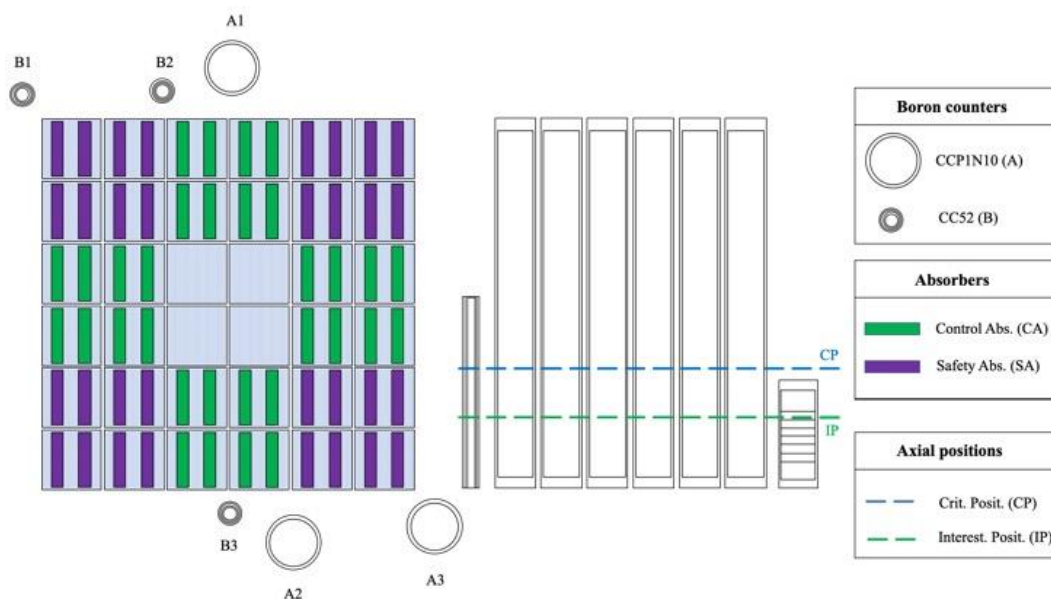| k_{eff} – All rods in (ARI)	k_{eff} – All rods out (ARO)	β_{eff} (pcm)	Λ_{eff} (μs)
0,77429 ± 3 pcm	1,12954 ± 2 pcm	751,2 ± 0,4	24.7 ± 0,002

Figure 2 – Core loading with detectors and absorbers distribution - xy plane (left) and xz (right)
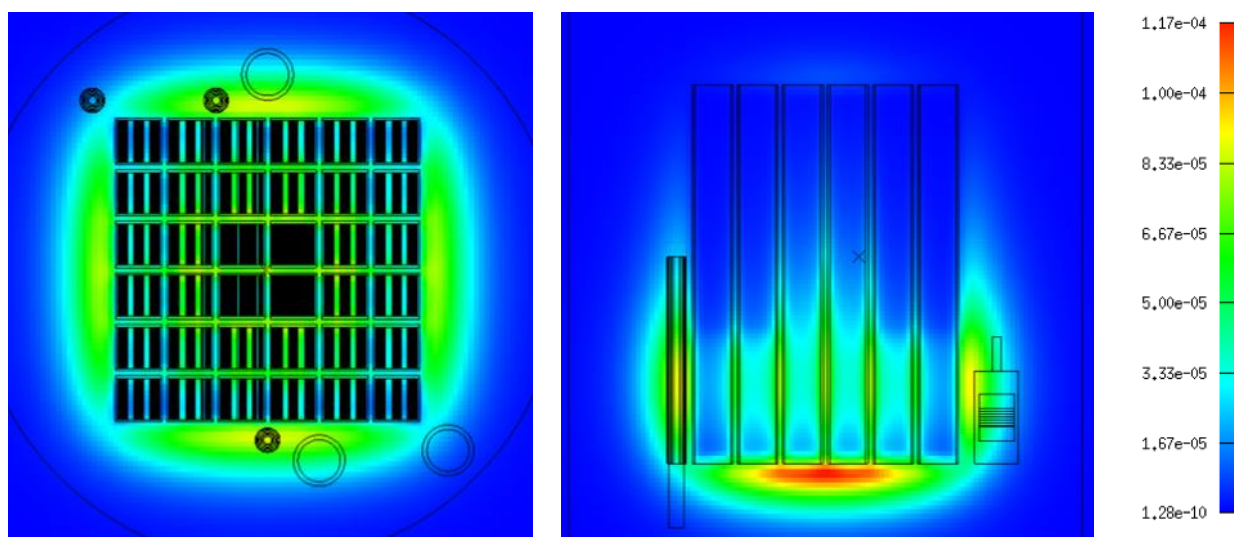

Figure 3 – Thermal flux distribution given with source neutron normalized to 1 n/s at critical state xy plane (left) and xz (right)

3.2. Safety absorbers worths calculation

Safety absorbers worths and associated thermal flux maps are calculated for 0^+ state (critical simulation) using TRIPOLI4®: *cf.* Table II. The control rod worths calculated for subcritical levels are also added (fixed sources situation). Initial state is considered to a null reactivity. Through experimentation, safety absorbers worths are taken immediately after rod drop: it is assumed that 0^+ state is reached and so critical worths are considered for control rod worths.

To describe this method, the boron-lined detectors positions considered are shown in both Figure 2 and Figure 3. These positions are chosen to scan the largest range of corrections to apply. Two different boron-layer detectors (both size and geometry) are considered to get f_{MSM} sensitivity for various detectors designs: CCP1N10 is a wide and short detector with six boron trays, and CC52 is a narrow and long one with four concentric tubes and two concentric boron layers.

Table II. Safety absorbers worths (pcm with 2σ) & thermal flux calculated with TRIPOLI4®

Rod	1 SA	2 SA (diagonal)	2 SA (adjacent)	3 SA	All 4 SA
Crit. (0^+)	1705 ± 2	4071 ± 2	3530 ± 2	6316 ± 2	10357 ± 2
Subcrit.	1748 ± 2	4235 ± 2	3767 ± 2	7068 ± 2	11880 ± 2
Thermal flux for IP at 0+ state (with source neutron normalized to 1 n/s)					

3.3. General description of calculation sequence route

The proposed calculation sequence for f_{MSM} determination deals with the following sequence route:

1. Critical calculation before scram to determine at the beginning of rod drop:
   - (n, α) reaction rates on detectors with results given for 1 source neutron, inducing $n_0 = 1$ in **(4)** ;
   - Fission rate mesh, with EXTENDED_MESH score (FMESH equivalent in MCNP).
2. Subcritical calculation after scram to assess (n, α) reaction rates on detectors at the end of rod drop.

The subcritical calculation uses fission rate mesh calculated within previous critical calculation: fission rates (1st step output) are converted to neutron sources map (2nd step input). Indeed, both fission rate and delayed neutron distribution are considered similar, within first approximation. Moreover, the same distribution is considered before and after scram (Constant Delayed Neutron Source Approximation): the decay constants are mostly longer than the rod-drop duration (around 1 second).

3.4. 1st step: critical calculation representing situation before scram

The initial situation considered is the critical situation, with safety absorbers in their initial position. Critical calculation is performed: (n, α) reaction rates on boron detectors are summarized in Table III. Fission rate output map is represented in Figure 4. Here, meshes are 1 cm³ (1 cm x 1 cm x 1 cm).
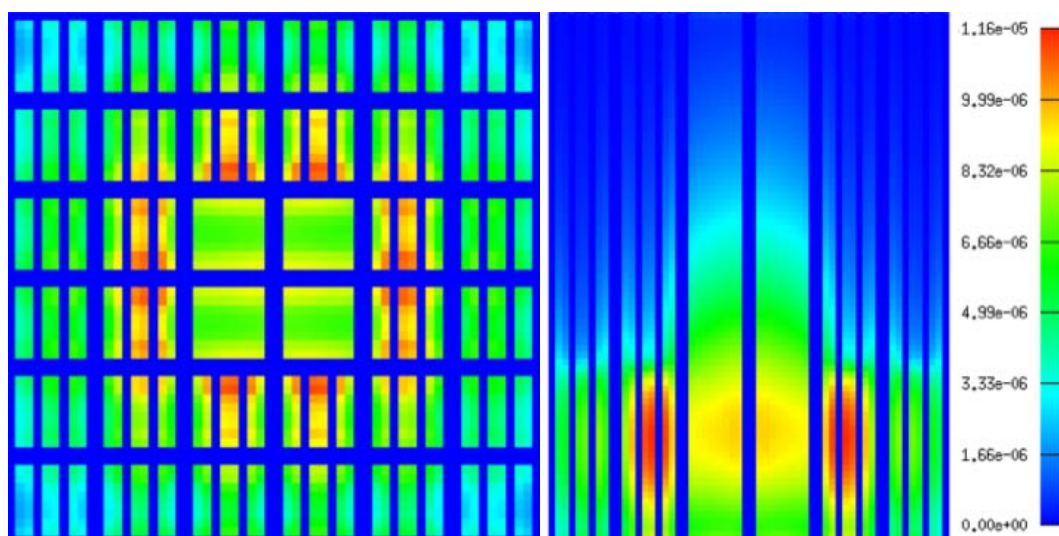
Figure 4 – Fission rate map before scram - xy plane (IP) and xz plane (given for with source neutron normalized to 1 n/s)

3.5. 2nd step: subcritical calculation representing situation after scram

Before performing calculation, characterization of subcritical sources (delayed neutron) is required. Delayed neutron sources distribution comes from post-treatment of fission rates, using a homemade tool for transcription in TRIPOLI4 format. Results from a fixed sources criticality calculation are given for $n_{0+} = I_{source} \times M$, with I_{source} = global source intensity and M = subcritical amplification. For simplification sake, geometrical intensities are calculated and a global source intensity of 1 source neutron is given. A last step consists in finally dividing by M, enabling to have $n_{0+} = 1$ in **(4)**.

The delayed neutron energetic spectrum is assumed as a Watt spectrum with parameters a = 10 and b = 100. This simple model fits correctly measured delayed neutron energetic spectrum for [235]U cores [10].
Then, subcritical calculation determines (n, α) reaction rates in boron-lined detectors after scram: *cf.* Table III. FIXED_SOURCES_CRITICALITY (*i.e.* subcritical) calculation mode simulates both prompt and delayed events. In this case, the "prompt" source is delayed neutron source, so delayed contributions must be cancelled (already taken into account), possible due to PROMPT_FISSION_ONLY simulation option in TRIPOLI4®.

Three strategies have been identified for f_{MSM} calculations and answer several aims. (n, α) reaction rates are calculated either:
- directly on detectors: sum of volumic scores on all boron layers. This is considered as reference method (no geometrical approximation), but requires perfect knowledge of detectors localisation. This is mainly applied for experiments interpretation, if detector positions are well-known ;
- on EXTENDED_MESH using microscopic reaction rates, *i.e.* reaction rate considering infinitesimal boron concentration: $\int \sigma_{10B(n,\alpha)}(E)\phi(E)dE$. This methodology requires the same operation for previous critical calculations. However, accurate detector position knowledge is not required: it is useful for preparation of experiments or if position uncertainties are significant ;
- using Green functions associated with fission rate distribution [5]. Actually, this method is currently in validation phase and is still a perspective: associated results won't be given here but interest for this method will be discussed later.

As an example, reaction rates on boron-lined proportional counters (reference method) for one SA drop situation are summarised in Table III. MC statistic uncertainties are lower than 1% (2σ).

Table III. (n, α) reaction rates (in s-1 with source neutron normalized to 1 n/s) on boron detectors for one SA Rod-Drop

	A1	A2	A3	B1	B2	B3
Before scram	$1,15.10^{-7}$	$1,17.10^{-7}$	$3,62.10^{-8}$	$8,11.10^{-8}$	$2,63.10^{-7}$	$2,70.10^{-7}$
After scram	$5,60.10^{-6}$	$7,81.10^{-6}$	$2,34.10^{-6}$	$5,25.10^{-6}$	$1,54.10^{-5}$	$1,83.10^{-5}$

3.6. MSM factor determination

Depending on strategies previously chosen, the f_{MSM} determination is done accordingly to **(4)**:
- Using a ratio between both before and after scram (n, α) reaction rates on each detector ;
- Using a ratio between both before and after scram (n, α) reaction rates for each mesh. The mean value is then calculated using a homemade tool which determines what meshes intercept each detector and determine mean value.

The f_{MSM} results for all rod drops studied with "fictitious core" are summarised in Table IV. Values far from 1,0 characterize significant flux distribution variations during scram. Concerning validation of f_{MSM} map method, mean deviation between both reference and meshing use is 1,1%, showing satisfactory results.

Table IV. f_{MSM} results for all studied situations (ref. = on counter, map = on meshes)

	1 SA		2 SA (diag.)		2 SA (adj.)		3 SA		4 SA	
detector	ref.	map	ref.	map	ref.	map	ref.	map	ref.	map
A1	0,86	0,85	1,07	1,06	1,06	1,05	1,34	1,33	1,20	1,20
A2	1,18	1,17	1,25	1,25	0,74	0,72	0,75	0,73	1,01	0,99
A3	1,14	1,13	1,36	1,34	0,32	0,31	0,39	0,38	0,55	0,54
B1	1,14	1,14	1,36	1,36	1,44	1,43	1,75	1,75	0,52	0,50
B2	1,03	1,02	1,26	1,25	1,28	1,27	1,60	1,59	0,96	0,96
B3	1,20	1,19	1,08	1,08	1,07	1,05	0,90	0,90	1,20	1,21
M	56,8		23,6		26,6		14,2		8,4	

4. EXPERIMENTAL ROD WORTH CORRECTION ON A CRITICAL MOCK-UP

Within the experimental validation framework, this methodology has been applied on an internal critical mock-up [6] for several experimental configurations. These layouts are similar to "fictitious core" applications presented above.

f_{MSM} application ends up with a significant enhancement based on Calculation/Experiment ratios (C/E). Currently, the validation basis is composed of almost one hundred C/E points. Despite small persistent discrepancies, the use of f_{MSM} allows a C/E convergence close to a null value. Analytically, the mean point cloud is about 4,3 % accompanied by a 6,7 % standard deviation.

Method exposed in this papier shows an improvement in sub critical state understanding. As a result, the MSM application has been deployed giving a consistent C/E comparison database and will be completed by forthcoming both calculations and measurements.

In order to improve the methodology, investigations will be further continued along with sensitivity analysis (delayed neutron energetic spectrum simplification, fission rate mesh fine tuning…) and uncertainties studies (nuclear data and measurements, counter uncertainties, detector positions…).

5. CONCLUSIONS AND PERSPECTIVES

This paper presents a calculation route for f_{MSM} determination, correct rod worth measurements from boron lined proportional counter, to consider the different motion of both global and local neutron fluxes during rod drop. The methodology deals with 2 linked steps:

- first, critical situation before scram is simulated for determination of both fission rate and (n, α) reaction rates (directly on counter, or on mesh) distributions ;
- fission rate distribution is transformed into delayed neutron source distribution for second step, considering that the delayed neutron distribution doesn't change during scram.

Finally, subcritical simulation is done for after scram situation, and (n, α) reaction rates (directly on counter, or on mesh depending on the choice made in the first step) are calculated. The methodology has been industrially tested and is now often deployed in studies, which could be illustrated by a sampling of calculation/measurement deviations determined on experimental-critical mock-up. The calculation/measurement discrepancies decrease due to use of f_{MSM}: with reducing both standard and maximum deviations.

Main perspectives are focused around faster determination of these factors, and uncertainties mastering:

- First, TRIPOLI4® Green function module allows to determine the contributions of the source areas for each detector response. Independently, fission rate distribution can be calculated with MC or deterministic code. Indeed, Green function utilisation for a given fission distribution is instantaneous, and experimental configurations are often similar.
- Secondly, a work-in-progress is undertaken on both sensitivities and propagated uncertainties of this sequence. Moreover, investigations are also on experimental data to determine associated uncertainties and relative items.

ACKNOWLEDGMENTS

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