NEUTRONIC UNCERTAINTIES ANALYSIS IN OPAL REACTOR

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ABSTRACT

The calculation of the uncertainties of the neutronic parameters is an important factor in the design of a nuclear reactor. These uncertainties impacts in the safety, economic and performance characteristics of the reactor and a good estimation of them for all neutronic parameters is required. There are different sources for the uncertainties of the calculated parameters; we can mention engineering tolerances, nuclear data uncertainties, process uncertainties, modelling criteria and user dependent uncertainties.
INVAP decided to implement uncertainties analysis using Total Monte Carlo Method in the production calculation line (instead of using specific tools for this purposes) to minimize two sources of the mentioned uncertainties: modelling and user uncertainties. This minimization is carried out through the utilization of the INVAP calculation methodology and the proper set of procedures during the whole process of the reactor design and analysis.
INVAP´s calculation line has been used by INVAP and several of its customers for the design, optimization and follow-up of several reactors throughout the world obtaining optimal results, like RA-6, NUR, RA-8, ETRR2, OPAL, CAREM, CNA-II, etc. These codes are also used by nuclear engineering students, master's and doctoral thesis students of the Balseiro Institute, performing a large number of calculations for different reactor types such as MTR, PWR, BWR, PHWR, TRIGA, FBR, ADS and Homogeneous reactors.
On November 2016, OPAL (20MWth multi-purpose open-pool type Research Reactor) reached 10 years of continuous operation by the Australia Nuclear Science and Technology Organization (ANSTO) showing a very good overall performance. In this frame, the uncertainties calculation capability is applied and evaluated in OPAL reactor core for two main reasons: the availability of proper validated models and the intention to apply this new capability on real operating core, which will feedback the INVAP design team about the characteristics of the new evaluated data.
This paper describes the capabilities added to the INVAP calculation line to allow the calculation of the neutronic uncertainties using Total Monte Carlo Method for all design neutronic parameters, in a production environment which minimizes the user and modelling uncertainties. Finally, this new capability is applied in the OPAL reactor core and compared to operational data.

1 Introduction

1.1 The OPAL Research Reactor

Open Pool Australian Light water (OPAL) Research Reactor[1], located at Lucas Heights Australia represents the state-of-art technology in its field. It is a 20MWth multi-purpose open-pool type Research Reactor designed, built and commissioned by INVAP between 2000 and 2006. It has been operated by the Australia Nuclear Science and Technology Organization (ANSTO) since commissioning, showing a high standard of overall performance. On November 2016, OPAL reached 10 years of continuous operation, becoming one of the most reliable and available in its kind worldwide, with an unbeaten record of being fully operational 307 days a year.

Several neutronic parameters were measured during commissioning and they were verified against the design models (models developed during design stage)[2]-[4], also several advanced modelling were done[5]-[7].
High quality experimental data is available[8], which allows reproducing several tests developed more than ten years ago.
The reactor consists of a compact core of 16 LEU (<20%wgt ^{235}U) MTR-type dispersed Uranium-Silicide fuels. The Reactor is cooled and moderated by light water and reflected by heavy water contained in a reflector vessel. The Reactor Shutdown systems are constituted by:

- A fast-actuation First Shutdown System (FSS), comprised by five Hafnium Control Rods (CR), namely four plate-types and a central cross-type rod that is also used as regulating rod.
- An independent, diverse and redundant Second Shutdown System (SSS), comprised by the draining of the heavy water present in the reflector vessel. This drainage is performed by the aim of a piping and heavy water storage tank, where all system is slightly pressurized by Helium gas.

Besides, several irradiation facilities are located in the Reflector Vessel, including a Cold Neutron Source (CNS) with two Cold beams, a thermal neutron source with two beams, a region reserved for a future hot neutron source, a hot neutron beam, 17 vertical irradiation tubes with place for 5 targets each for bulk radioisotope production (such as ^{192}Ir, ^{99}Mo and ^{131}I), 19 pneumatic rigs with 57 target positions for different purposes and 6 neutron transmutation doping (NTD) devices.

1.2 INVAP Neutronic Calculation Methodology

INVAP designs and builds research reactors with very demanding requirements following a mature calculation methodology[9]. These demanding requirements lead to a requirement to continuously improve the design and analysis, which need better prediction capabilities to reduce the design margins due to the numerical and engineering uncertainties. INVAP uses its own-developed calculation line[10](see Figure 1) to predict the behaviour of the reactor to be built. The detailed models developed under this framework must be with a consistent level of detail from all the engineering variables, where the analysts play a very important role in the development of accurate models, which are used to simulate the system.
INVAP´s calculation line has been used by INVAP and several of its customers for the design, optimization and follow-up of several reactors throughout the world obtaining optimal results, like RA-6, NUR, RA-8, ETRR2, OPAL, CAREM, CNA-II, etc. These codes are also used by nuclear engineering students, master’s and doctoral thesis students of the Balseiro Institute, performing a large number of calculations for different reactor types such as MTR, PWR, BWR, PHWR, TRIGA, FBR, ADS and Homogeneous reactors.
Under the continuous improvement program, INVAP decided to implement uncertainties analysis using Total Monte Carlo Method (TMC)[11] in the production calculation line (instead of using specific tools for this purposes) to minimize two sources of uncertainties: Modelling and User effect uncertainties. This minimization is carried out through the utilization of the INVAP calculation methodology and the proper set of procedures during the whole process of the reactor design and analysis.
The other two sources of uncertainties, namely Nuclear Data and engineering data uncertainties are properly managed by the calculation line, through the proper modification of the working library and the main calculation codes: CONDOR[12] cell-level code and CITVAP[10] core-level code.
The modification of the production calculation line allows to properly calculate uncertainties in any of engineering parameter, as for example: Reactivity, Power Peaking Factor, Control Rod worth, Feedback Coefficients, Irradiation fluxes, Kinetic Parameters, Fuel Management, etc.

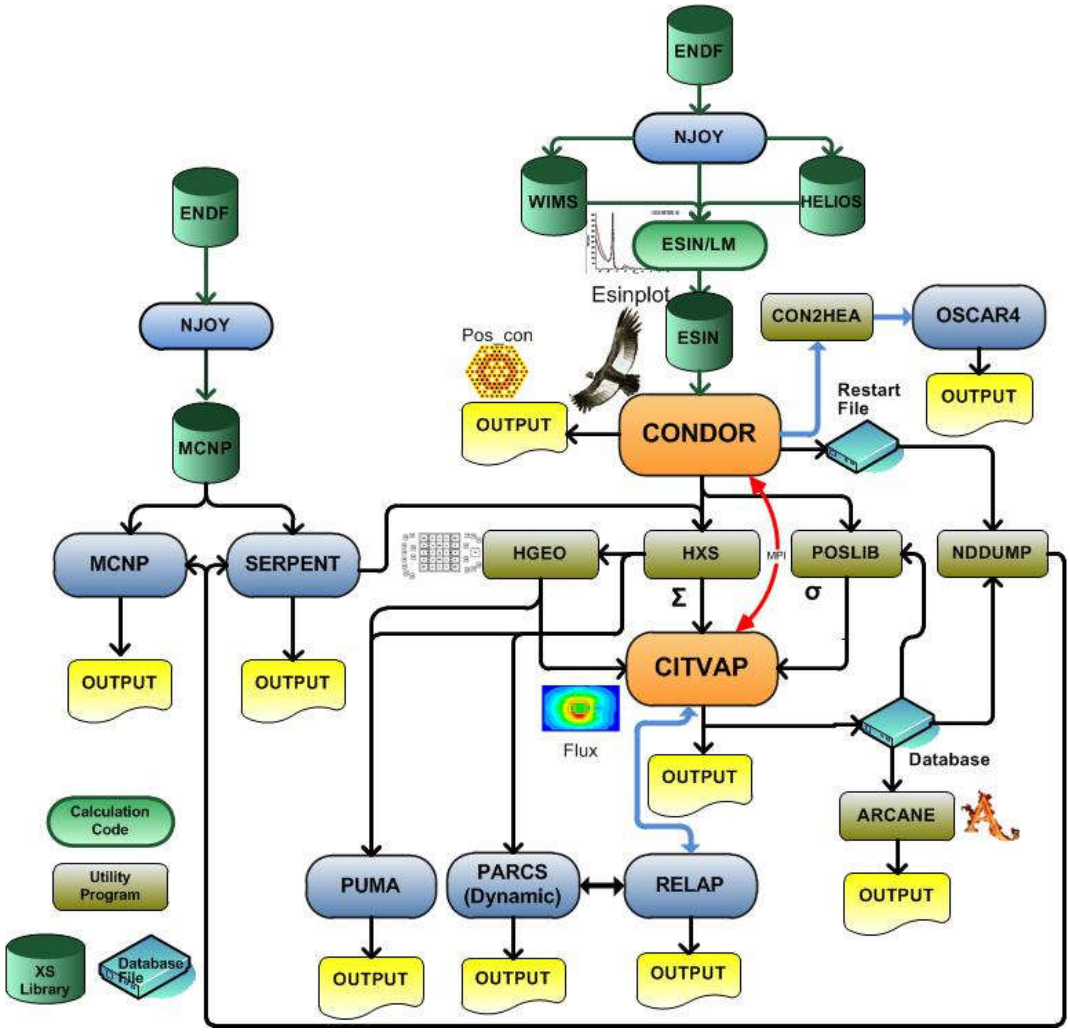


Figure 1 Neutronic Calculation Line

2 Uncertainties Calculations

The calculation of the uncertainties of the neutronic parameters constitutes an important factor in the design of a nuclear reactor. These uncertainties impacts in the safety, economic and performance characteristics of the reactor and a good estimation of them for all neutronic parameters is needed. As was already mentioned, there are different sources for the uncertainties of the calculated parameters; where can be mentioned the engineering tolerances, nuclear data uncertainties, process uncertainties, modelling criteria and user effect uncertainties. The last two are minimized using the procedures and methodology of the production calculation line, and the first two are described in the following sections.

INVAP is planning to use uncertainties analysis in a production way during the design stage of Research Reactor. For this reason the present work makes a preliminary analysis of some uncertainties to properly understand how to get, process, use and analyse the engineering uncertainties and finally feedback the whole design process with these engineering design parameters uncertainties.

In the current work some engineering and nuclear data uncertainties were taken into account (see next subsections), while no process uncertainties were considered for simplicity (thermal-hydraulic state of the reactor, control rod position, irradiation facilities loading, etc.).

This methodology can be used also for sensitivity analysis, but this point is not described in this paper, because will be used as an important analysis tool during the design stage.

2.1 Calculation line modifications

Two codes were mainly modified to manage the TMC method for uncertainties calculations: CONDOR and CITVAP (see Figure 1).Both codes were updated to have the capability to generate random numbers with Uniform and Normal distribution and include them in the input definitions. Both distributions options counts with lower and upper limit for the generation of the random numbers, and for the Normal distribution option the average value and its standard deviation is required. This random number generating capability and the mathematical operations capability of the codes during input processing allows managing any engineering uncertainty (namely dimensions and compositions). As an example the uncertainty in compositions can be managed maintaining the sum of weight factors (which must be 100%), or the fuel meat tolerance in dimensions for a MTR fuel can be modelled, where the sum of the claddings and meat thickness can be maintained (must be the plate thickness).

2.2 Nuclear data Uncertainties

The uncertainties of the nuclear data were generated using the TALYS system [13], where the isotopes taken into account were downloaded from the site [14].

The nuclear cross section library generated included the reference and 200 random isotopes per isotope with uncertainties treatment. At the current stage this analysis only take into account the following isotopes H, Al, ^{234}U, ^{235}U, ^{236}U and ^{238}U, which are only used in the Fuel Assembly model for simplicity.Problems were found for available data in the ^{113}Cd, which lead to avoid the analysis of this isotope in the present work.

2.3 Engineering Uncertainties

The engineering uncertainties were properly defined between the tolerances of each parameter. Manufacturing values (average value and its standard deviation) were used when enough available data were found. The standard deviation of each of the variables was defined using the criteria given in [15]. A significant difference can be found between design uncertainties (nominal value and tolerances), and manufacturing uncertainties (average value and standard deviation). The former will be used in a design stage, and the second one in a comparison against measured values.

In the current work, only engineering uncertainties in the Fuel Assembly were taken into account. The Fuel Assembly model (see Figure 2) was done in symmetry 1, to take into account the uncertainties in each Fuel Plate (composition and geometry), Cadmium wire (composition and geometry) and Frame (geometry).

The following list shows the variables taken into account in the uncertainties analysis:

- Geometry:
  - Fixed Values: Fuel Assembly Pitch, Active Height, and Fuel Plate Pitch.
  - Fuel Assembly Geometry (most of them with asymmetric tolerances): External FA width and height, frame width,
  - Internal and External Fuel Plate Geometry (most of them with asymmetric tolerances): Plate thickness, Cladding thickness, width and thickness of slot plate, plate width, plate border (distance between plate and meat).
  - Dimensions by balance: meat thickness, water channels, and meat width.
  - Cd Wires: Radius and Cd wire slot width.
- Materials composition:
  - Uranium Loading: Standard, Type 1 and Type 2 Fuel Assemblies.
  - Silicon Loading.
  - Aluminium Loading
  - Impurities (Equivalent Boron)
  - Enrichment (^{234}U, ^{235}U, ^{236}U). ^{238}U by balance.

For all the parameters analysed (it means all parameters were sampled simultaneously in each case), 200 random cases were used, getting the average value, its standard deviation, and also the minimum and maximum values. The nominal case was also evaluated and the difference between nominal value and the average value was also calculated.

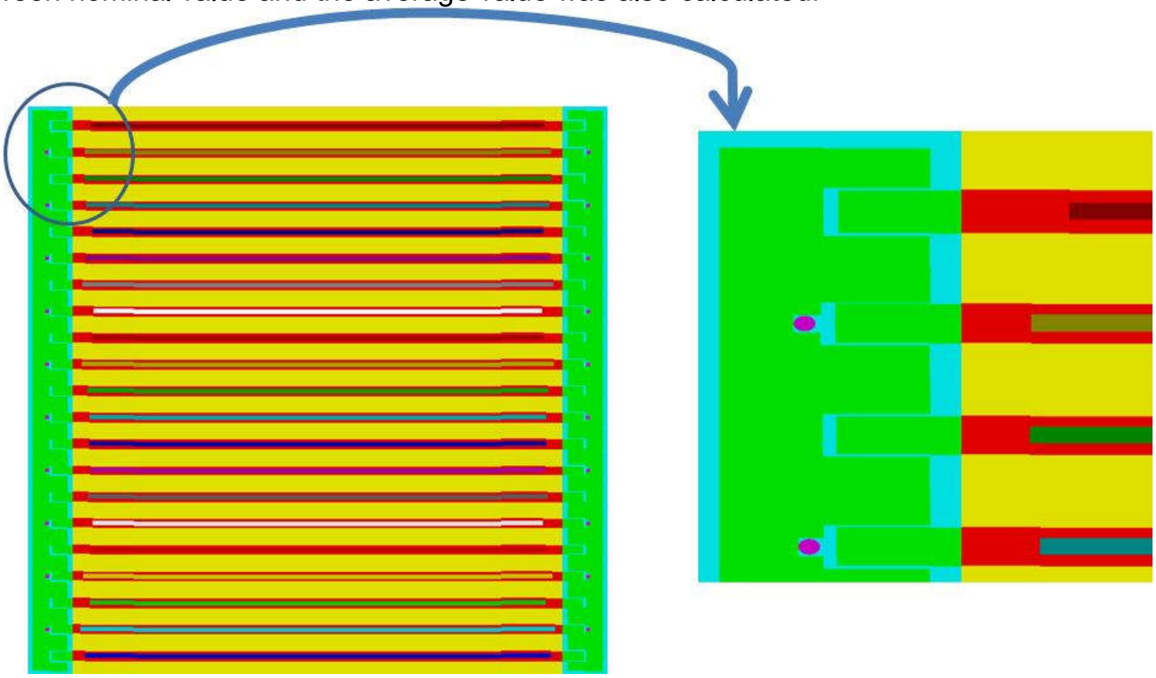


Figure 2 Fuel Assembly Model – detail of engineering uncertainty modelling

3 Results

High quality data is available for OPAL reactor which was obtained from measured parameters during commissioning[8].

Using the engineering and Nuclear Data uncertainties described in Section 2 a series of 200 random cases where analysed for each critical stage measured during OPAL commissioning. The results were processed as a whole, obtaining the min, max and mean value and standard deviation as stated in TMC methodology [11]. The following sections presents the most relevant results found up to date.

3.1 Critical Core Calculations

During commissioning 74 critical cores were measured[8]. Table 1 shows uncertainties analysis of the calculated reactivity and power peaking factor (PPF) for the 74 critical cores measured during the commissioning:

Case	Exp. Data	Nom. [pcm]	Avg	Min	Max	Std.	Nom vs Avg
Reactivity [pcm]							
Avg (74)	-	172.3	178.1	-111.2	385.7	84.4	5.8
Max Value	-	295.2	303.5	34.8	499.0	34.8	8.3
Min Value	-	-140.8	-133.3	-446.9	115.7	92.1	7.5
PPF [-]							
Avg (74)	NA	2.595	2.574	2.510	2.623	0.021	-0.021
Max Value	NA	2.833	2.812	2.713	2.833	0.037	-0.021
Min Value	NA	2.305	2.296	2.242	2.353	0.023	-0.009

Table 1: Critical Core Uncertainties Analysis.

The Figure 3 show in details the 200 cases calculated reactivityfor each of the 74 critical cores (left) , and a their statistical analysis (right) and the Figure 4 show in details the 200 cases calculated power peaking factor for each of the 74 critical cores (left) , and a their statistical analysis (right).

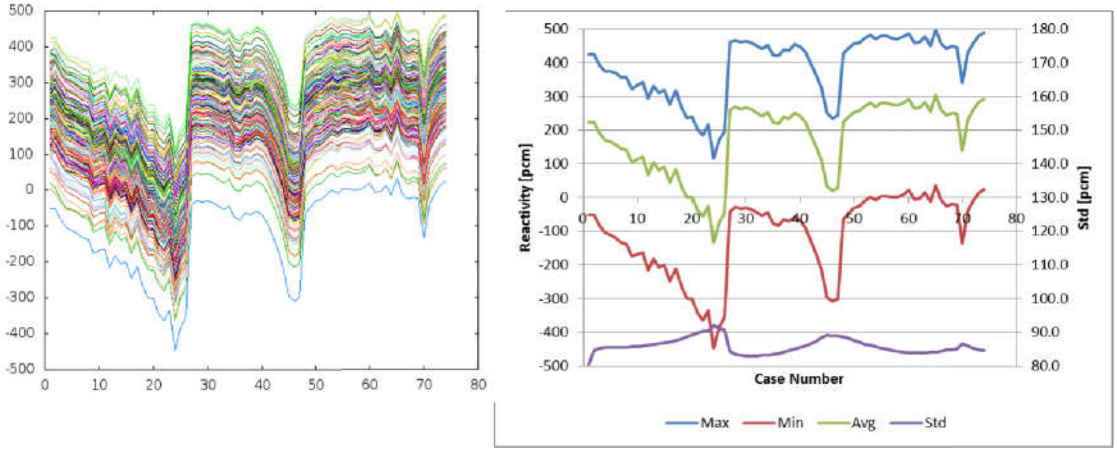


Figure 3 Critical Core Uncertainties Analysis.

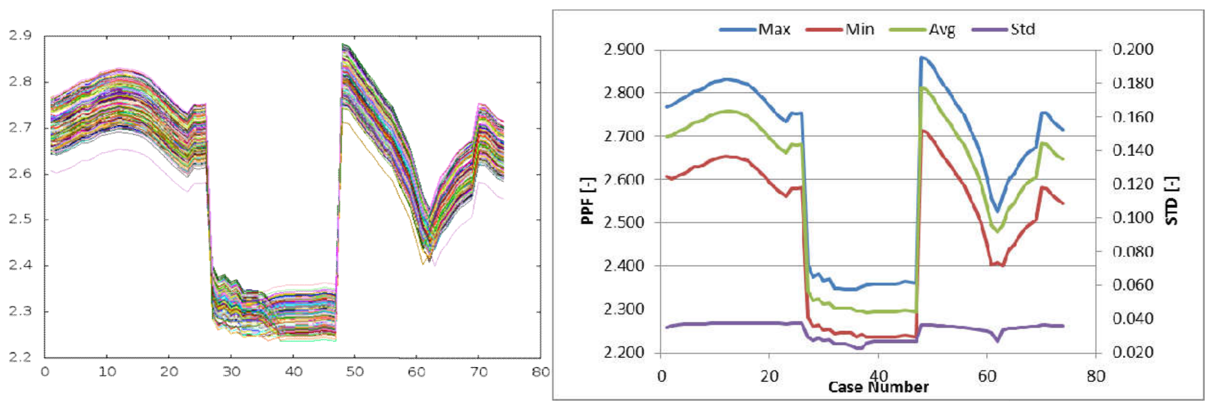


Figure 4 Power Peaking Factor Core Uncertainties Analysis.

3.2 Void Feedback Coefficient

During commissioning an experiment was carried out to measure the void feedback coefficient of the core. Table 2 shows uncertainties analysis of the calculated void feedback coefficient measured during the commissioning:

Exp. Data	Nom. [pcm]	Avg	Min	Max	Std.	Nom vs Avg
-806 pcm (-223 pcm/%)	-786.7	-792.0	-811.1	-768.0	7.6	-5.3

Table 2: Void Feedback coefficient Uncertainties Analysis

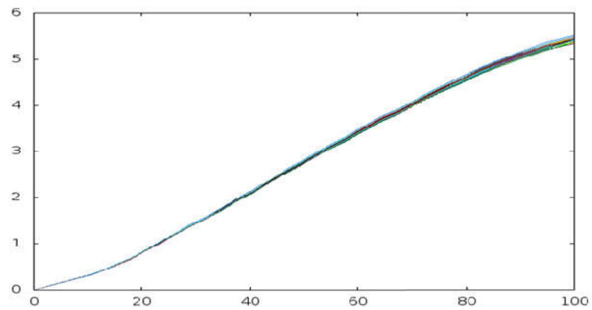
3.3 Control Rod Calibration

During commissioning the control rod number 2 was calibrated. Table 3 shows uncertainties analysis of the calculated control rod calibration measured during the commissioning:

Case	Exp. Data	Nom. [pcm]	Avg	Min	Max	Std.	Nom vs Avg
Total Control Rod Worth [pcm]							
CRWorth	5.697	5.384	5.425	5.340	5.509	0.032	0.041

Table 3: Control rod calibration Uncertainties Analysis.

The Figure 5 show in details the 200 cases calculated calibration of the control rod cores (left), and a statistical analysis (right).



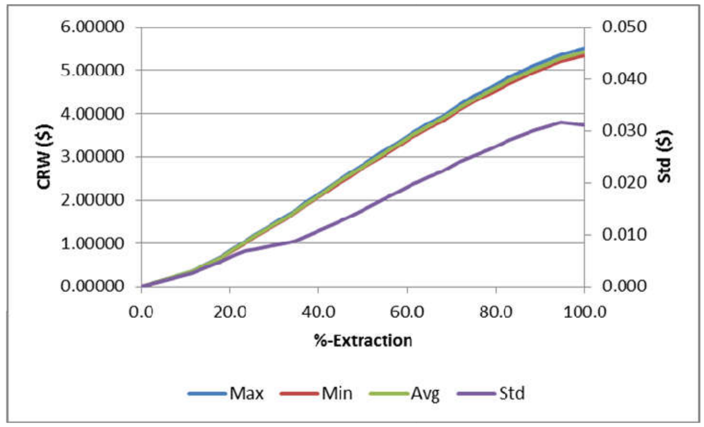


Figure 5 Control rod Calibration Uncertainties Analysis.

4 Conclusions

A TCM method for uncertainties calculation in the INVAP calculation line and methodology was implemented. Currently, this method allows us to take into account any engineering uncertainties and Nuclear Data uncertainties. This method can be also used to perform sensitivity analysis.

The utilization of the INVAP methodology minimizes some of the user dependent uncertainties, which currently are not properly numerically evaluated.

As a summary the preliminary analysis shows that the engineering uncertainties are in the order of the of the calculation capabilities of the deterministic calculation line, where modelling uncertainties are present.

A deeper analysis is required for a proper evaluation of this methodology and its implementation of the whole design process, such as:

- Comparison between tolerances and manufacturing standard deviations.
- Addition of extra isotopes to deal with Nuclear Data uncertainties.
- Improvement in the process of uncertainties,
- Inclusion of user dependent models

Finally, in a next development stage some process uncertainties will be included in the analysis. For example: Thermal-hydraulic parameters, movable irradiation facilities, control rod position, etc.

5 Acknowledgements

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