# Simulation tools and approaches for the compliance with performance-based ECCS cladding acceptance criteria (10 CFR 50.46C)

A. Zoino*, A. Alfonsi**, C. Rabiti**, F. Giannetti*, G. Caruso*

*Sapienza, University of Rome, Italy
** Idaho National Laboratory
angelo@zoino.it, andrea.alfonsi@inl.gov, cristian.rabiti@inl.gov, fabio.giannetti@uniroma1.it, gianfranco.caruso@uniroma1.it

## INTRODUCTION

The Nuclear Regulatory Commission (NRC) is currently considering a revision of the requirements in 10 CFR 50.46C, rules, focused on the ECCS rule in LOCA scenarios[1].

The new approach modifies the analysis strategy in order to take into account also the effects of the burn-up rate. The maximum temperature and the oxidation of the cladding must be casted as function of the fuel exposure in order to find the limiting conditions in the history of the reactor, with its different design and different reloading patterns.

This new analysis requires new tools and capabilities in order to have reasonable computational times and good accuracy, taking in account the dynamic phenomena of multi-physics systems.

In order to perform such analysis, a rigorous Probabilistic Risk Assessment (PRA) strategy needs to be employed.

.

## DESCRIPTION OF THE ACTUAL WORK

This work is a proof of concept for illustrating a new proposed approach that will be required in the next years, in order to face the challenges posed by the new rule.

### Tools and Capabilities

PHISICS (Parallel and Highly Innovative Simulation for the INL Code System) code toolkit [2,3] is being developed at the Idaho National Laboratory. This package is intended to provide a modern analysis tool for reactor physics investigation. It is designed with the mindset to maximize accuracy for a given availability of computational resources and to give state of the art tools to the nuclear engineer. This is obtained by implementing several different algorithms and meshing approaches among which the user will be able to choose, in order to optimize his computational resources and accuracy needs. The software is completely modular in order to simplify the independent development of modules by different teams and future maintenance. The different modules currently available in the PHISICS package are a nodal and semi-structured transport core solver (*INSTANT*), a depletion module (*MRTAU*), a time-dependent solver (*TimeIntegrator*), a cross section interpolation and manipulation framework (*MIXER*), a criticality search module (*CRITICALITY*) and a fuel management and shuffling component (*SHUFFLE*). PHISICS can be run in parallel to takes advantage of multiple computer cores (10 to 100 cores). In addition, the package is coupled with the system safety analysis code RELAP5-3D[4]. Using the coupling between PHISICS and RELAP5-3D is possible to drive a accurate dynamic analysis switching between a steady state and a time-dependent calculation.

The PRA analysis tool of choice is RAVEN[5]. RAVEN is a generic software framework to perform parametric and probabilistic analysis based on the response of complex system codes. RAVEN is capable to agnostically communicate with any system code. This agnosticism is achieved by the implementation of Application Programming Interfaces (APIs). These interfaces are used to allow RAVEN to interact with any code as long as all the parameters that need to be perturbed are accessible by inputs files or via python interfaces. RAVEN is currently coupled with several simulation tools, among which RELAP5-3D (chosen as system code for the LOCA analysis).

### Core Design

The modeling of this reactor is based on the "Benchmark for Evaluation and Validation of Reactor Simulations (BEAVRS)"[6]. BEAVRS is a detailed PWR benchmark containing real plant data for assessing the accuracy of reactor physics simulation tools for the first 2 operational cycles. In Figure 1 and Table I. the radial core layout and the plant key parameters are respectively shown.

Table I. – Key Plant Parameters

| | |
|---|---|
| N. Fuel assemblies | 193 |
| Loading Pattern | w/o U-235 |
| Region 1 | 1.61 % |
| Region 2 | 2.40 % |
| Region 3 | 3.10% |
| Control Rod | Ag-80%, In-15%,Cd-5% |

| | |
|---|---|
| Burnable Absorber | Borosilicate Glass, 12.5 w/o $B_2O_3$ |
| Power | 3411 MWth |
| Operating Pressure | 15.51 MPa |
| Isothermal Coolant Temperature | 564.82 K |

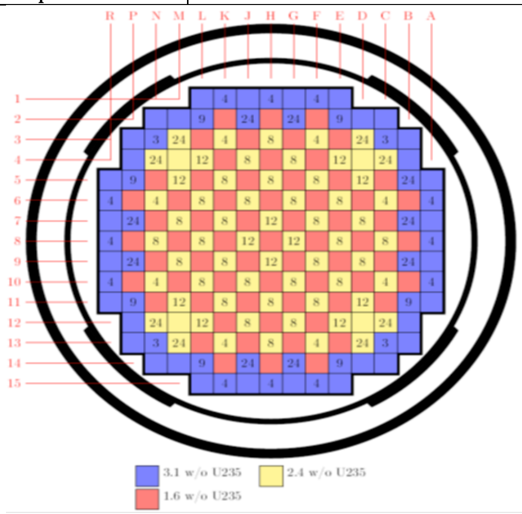



Fig. 1. Reactor Core layout.

The calculation is performed using homogenized cross sections for each assembly, leading to the identification of 29 different cross sections sets for the fuel region and 1 for the radial reflector, composed by the baffle, water between the baffle and the barrel, the barrel and the thermal shield.

In the suite PHISICS/RELAP5-3D, the coupling between the physics is performed through feedback exchange. For this reason, the cross sections sets have been tabulated with respect to several field parameters. For the scope of this work, a N-Dimensional grid of 80 tabulation points has been selected.

**Multi-Cycle Analysis**

In order to assess the compliance of the existing power plants to the new rule, the LOCA accident scenario needs to be initiated in equilibrium cycle conditions (supposedly nowadays reached by all the existing operative reactors in the US). Hence, the reactor evolution needs to be followed for several operational cycles, until reaching the reference equilibrium one. The equilibrium cycle is generally reached after several reloading (~18-20). In this study, we assume that the equilibrium cycle is reached after the 10th reloading.

For the first 10 cycles, the TH model of the reactor has been performed considering the reactor core only (without primary and secondary system). This choice has been taken since the first 10 cycles are used to compute the exposure history of the assemblies but are not active part of the LOCA simulation. For this reason, the primary system is modeled only considering the upper and lower plenum of the core. In order to be as accurate as possible for the determination of the initial conditions in the 11th cycle, the first ten cycles are simulated using a core channel per fuel assembly (193 in total). The radial reflector is modeled as a bypass channel (6% of the mass flow).

The BEAVERS benchmark provides data for the first 2 cycles only (1 reloading pattern); for the sub-sequential $3^{rd}$ through $11^{th}$ cycles, new reloading patterns have been constructed. The BEAVERS reloading pattern is a “high-leakage/low-energy” one. For this work, our goal was to perform analysis on a modern reloading pattern; the first developed 4 cycles’ patterns represent a gradual migration from the “high-leakage/low-energy” to the “low-leakage/high/energy” reloading pattern; the sub-sequential patterns represent the reference final “low-leakage/high/energy” patterns. All the batch enrichments have been computed in order to reach, at the equilibrium, a cycle length of 18 months. The reload patterns are reported in Figure 2.

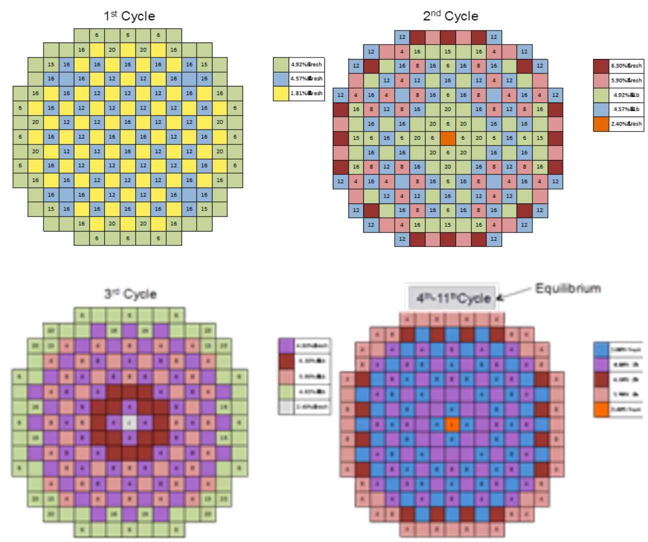



Fig. 2. Loading Pattern.

In addition, in order to maintain the reactor “critical” ($k_{eff}$ = 1.0), the boron concentration is automatically adjusted by the PHISICS code; when its value fells below 5 ppm, a new cycle is automatically initiated (automatic multi-cycle capability).

**LOCA Analysis**

As already mentioned, the new approach for the analysis of LOCA scenarios determines the need to consider a detailed burn-up calculation, which strongly impacts the cladding oxidation phenomena. In order to reduce the time of calculation all the power is remapped from 193 assemblies to 6 channels. The 6 channels represent:

- 3 are representative of the 3 different batches (Fresh Fuel, once burned, twice burned);

- 3 represent the pins, in the above zones, with the highest peaking factors.

Moreover, the mode is extended adding the primary system, a 4 loops, for the LOCA analysis. As an example, in Figure 3 the assembly-wise (radial) integrated power and peaking factors are reported for the Begin, Middle and End of cycle (10th); in addition, Figure 4 shows the detailed fuel exposure (burn-up) for the same points in time.

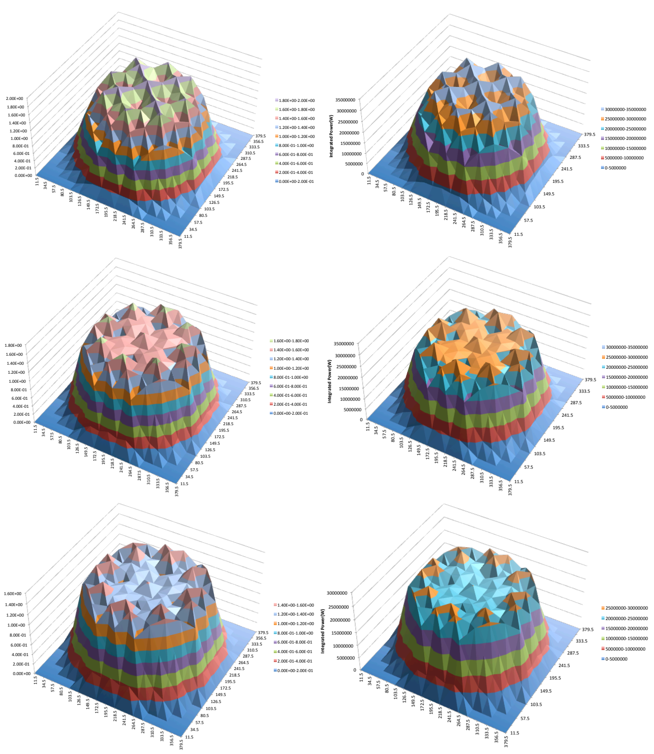


Fig. 3. Power (left) and Assembly Peaking Factor (right) for BOC, MOC and EOC (respectively from the up to the bottom).

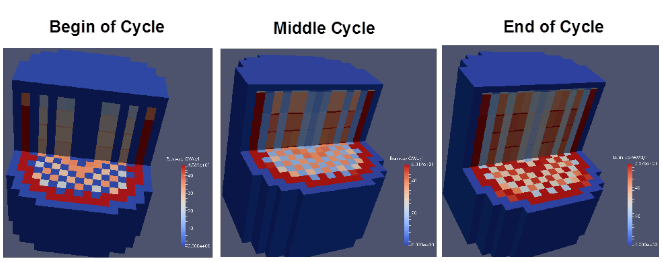



Fig. 4. Burn Up for BOC, MOC and EOC.

The above reported three points in time (different burn-up levels) have been used as initial boundary conditions to analyze the machinery for performing 3 examples of LBLOCA analysis with RELAP5-3D. Figures 5 and 6 show the results of the analysis. As it can be inferred in Figure 5, the core status at BOC, MOC and EOC does not determine challenging conditions for the LOCA analysis.

This is due to the fact that the LOCA scenarios for the assessment of the safety margins are generally performed considering the reactor right after a maneuver that can initiate, for example, a Xenon transient. For the scope of this work, the maneuver that has been considered is a load-following operation of the reactor.

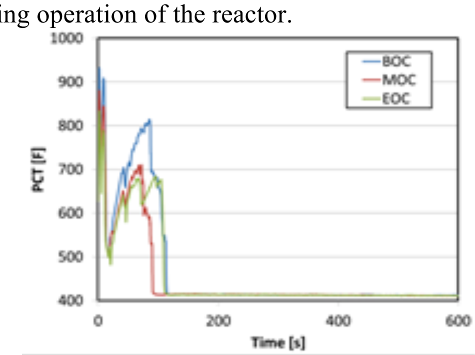
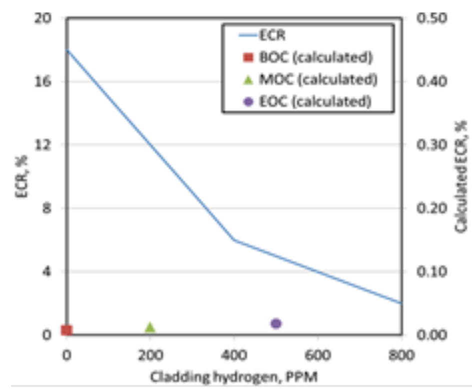



Fig. 5. Peak clad temperature during the LBLOCA scenario




initiated at BOC, MOC and EOC.

Fig. 6. Maximum local oxidation rate during the LBLOCA scenario initiated at BOC, MOC and EOC.

**PRA Strategy**

In order to assess the compliance of the operating nuclear power plants to the new rule, a rigorous Probabilistic Risk Assessment (PRA) needs to be employed. The new safety margins are related to the cladding oxidation ratio as function of the burn-up level reached by the assemblies when the LOCA scenario is initiated. This means that the limits cannot be seen as static thresholds but must be considered in a dynamic environment, since they evolve during the operation of the reactor.

Another aspect that needs to be considered in such analysis is the presence of several uncertainties associated with the key parameters of the plant that, depending on their value, can lead to completely different accident scenarios.

From a practical point of view, the goal of the PRA analysis of LOCA events can be summarized as follows:

- Computation of the probability of exceeding the proposed 50.46c safety margins for cladding oxidation

- Sensitivity analysis on the uncertain parameters that can influence the LOCA scenario and sub-sequential ranking
- Identification of the uncertain parameters' margins through the research of the reliability (or limit) surface

In order to assess the probability of exceeding the burn-up dependent limit, a sampling of the parameters affected by uncertainties is needed. This kind of analysis is characterized by high level of complexity, like the computation time of the simulation codes, high dimensionality, cause the uncertain parameters to take in consideration, and a high discontinuity create by the presence of safety systems that can suddenly start operating. The approach that is going to be used (currently) to perform such analysis is based on the well-known Monte Carlo technique.

The PRA analysis, that is going to be performed (see Fig.7), will characterized by:

- Sampling of time at which the maneuver is going to be initiated;
- Sampling of time at which the LOCA scenario begins (within the maneuver or after);
- Sampling of all the other uncertain parameters that affect the LOCA scenario.

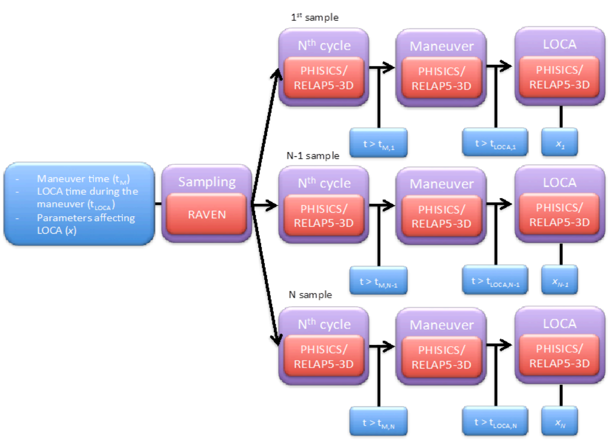


Fig. 7. Current PRA strategy scheme.

The uncertain parameters that will be considered for the analysis are:

- Reactor decay heat power multiplier
- Accumulator pressure multiplier
- Accumulator liquid volume
- Accumulator temperature
- Sub-cooled multiplier for critical flow
- Two-phase multiplier for critical flow
- Superheated vapor multiplier for critical flow
- Fuel thermal conductivity multiplier
- Average temperature
- Film boiling heat transfer coefficient multiplier

## FINAL REMARKS

As near future PRA strategy, in order to overcome the computation burden of the Monte Carlo method, a Hybrid Dynamic Event Tree (HDET) methodology[7,8] will be used.

The exploration of the system response using the Monte-Carlo (and, in the future the HDET) will ultimately lead to the knowledge of several possible outcomes of the LOCA accident scenario (in terms of PCT and corresponding burn-up and oxidation) with their corresponding probability. A post-processing function, build within RAVEN, will allow combining this information to assess what is the final probability to exceed the new limits.

After this preliminary analysis is completed it will be possible to perform sub-sequential investigation where the computation of sensitivity coefficient will allow to establish what are the most relevant uncertainties effecting the success/failure probability.

Finally using the RAVEN feature to utilize artificial intelligence accelerated search of reliability surface, it will be possible to use the HDET methodology to determine region of the input space that either leads to a positive/negative final outcome of the LOCA accident.

## REFERENCES


1. R. Borchard and M. Johnson, "10 CFR 50.46c Rulemaking: Request to Defer Draft Guidance and Extension Request for Final Rule and Final Guidance," U.S. Nuclear Regulatory Commission, Washington, DC, March 4, 2013
2. C. Rabiti, Y. Wang, G. Palmiotti, A. Alfonsi, A. Epiney, H. Hiruta, T. Grimmet, PHISICS: Parallel and Highly Innovative Simulation for INL Code System, INL/EXT-11-21384, Novembre 2011.
3. A. Alfonsi, C. Rabiti, A. Epiney, Y. Wang, and J. Cogliati, "PHISICS Toolkit: Multi-Reactor Transmutation Analysis Utility – MRTAU", Proc. PHYSOR 2012 Advances in Reactor Physics Linking Research, Industry, and Education, on CD-ROM, American Nuclear Society, LaGrange Park, Ill (2012), Knoxville, Tenn., April 15-20, 2012.
4. The RELAP5-3D© Code Development Team, "RELAP5-3D© Code Manual Volume I: Code Structure, System Models, and Solution Methods," Idaho National Laboratory Report, INEEL-EXT-98-00834, Rev. 4, June, 2012.
5. A. Alfonsi, C. Rabiti, D. Mandelli, J. Cogliati, and R. Kinoshita, "Raven as a tool for dynamic probabilistic risk assessment: Software overview," in Proceeding of M&C2013 International Topical Meeting on Mathematics and Computation, CD-ROM, American Nuclear Society, LaGrange Park, IL (2013).
6. N. Horelik, B. Herman, B. Forget, K. Smith, "Benchmark for Evaluation And Validation for Reactor Simulations (BEAVRS) v1.01," Proc. Int. Conf.

Mathematics and Computational Methods Applied to Nuc. Sci. & Eng., Sun Valley, Idaho, 2013.

7. A. Alfonsi, C. Rabiti, D. Mandelli, J. Cogliati, R. Kinoshita, A. Naviglio, "Dynamic Event Tree Analysis Through RAVEN", International Topical Meeting on Probabilistic Safety Assessment and Analysis (PSA 2013), September 22-26, Columbia, SC, USA, (2013).

8. A. Alfonsi, C. Rabiti, D. Mandelli, J. Cogliati, and R. Kinoshita, "RAVEN: Development of the adaptive dynamic event tree approach," Tech. Rep. INL/MIS-14-33246, Idaho National Laboratory (INL), (2014).