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A FUEL CYCLE AND CORE DESIGN ANALYSIS METHOD FOR NEW CLADDING ACCEPTANCE CRITERIA USING PHISICS, RAVEN AND RELAP5-3D

Andrea Alfonsi
Idaho National Laboratory
Idaho Falls, ID, 83402

George L Mesina
Idaho National Laboratory
Idaho Falls, ID, 83402

Angelo Zoino
University "La Sapienza"
Rome, 00135, Italy

Cristian Rabiti
Idaho National Laboratory
Idaho Falls, ID, 83402

ABSTRACT

The Nuclear Regulatory Commission (NRC) has considered revising the 10 CFR 50.46C rule [1] for analyzing reactor accident scenarios to take the effects of burn-up rate into account. Both maximum temperature and oxidation of the cladding must be cast as functions of fuel exposure in order to find limiting conditions, making safety margins dynamic limits that evolve with the operation and reloading of the reactor. In order to perform such new analysis in a reasonable computational time with good accuracy, INL (Idaho National Laboratory) has developed new multi-physics tools by combining existing codes and adding new capabilities. The PHISICS (Parallel Highly Innovative Simulation INL Code System) toolkit [2,3] for neutronic and reactor physics is coupled with RELAP5-3D [4] (Reactor Excursion and Leak Analysis Program) for the LOCA (Loss of Coolant Accident) analysis and RAVEN [5] for the PRA (Probabilistic Risk Assessment) and margin characterization analysis.

In order to perform this analysis, the sequence of RELAP5-3D input models had to get executed in a sequence of multiple input decks, each of them had to restart and slightly modify the previous model (in this case, on the neutronic side only) This new RELAP5-3D multi-deck processing capability has application to parameter studies and uncertainty quantification. The combined RAVEN/PHISICS/RELAP5-3D tool is used to analyze a typical PWR (Pressurized Water Reactor).

INTRODUCTION

The nuclear power industry is continually improving its designs, safety equipment, processes, and analysis methods. The NRC is considering a revision of the requirements in 10 CFR 50.46C rule, focused on the operation of the ECCS (Emergency Core Coolant System) in LOCA scenarios [1]. Novel analysis strategies will be required to account for the effects of fuel burn-up rate. It is necessary to cast the maximum temperature and oxidation of the cladding as functions of the fuel exposure in order to find the limiting conditions of the reactor, with its different design and different reloading patterns.

This revision requires the development of new tools and capabilities to calculate the dynamic phenomena of the multi-physics system to the required accuracy in a reasonable amount of time. To perform such analysis, a rigorous Probabilistic Risk Assessment (PRA) strategy must be employed.

The PHISICS code toolkit [2,3] is being developed at INL to provide state of the art analysis tools to nuclear engineers. It implements many choices of algorithms and meshing schemes for optimizing accuracy needs on available computational resources. Analysis tools 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 tool *SHUFFLE*. The tools are developed as independent modules in a pluggable fashion in order to simplify maintenance and development. PHISICS can be run in parallel to takes advantage of multiple computer cores (workstations and high-performance computing systems).

The package is directly coupled with the system safety analysis code RELAP5-3D [4] through a Fortran 95 interfacing module that contains communication subroutines that translate physical quantities into the native form of the receiving code. Using the coupling between PHISICS and RELAP5-3D, it is possible to drive an accurate dynamic analysis switching between steady state, quasi-equilibrium, and time-dependent calculations.

The PRA analysis tool of choice is RAVEN [5], a generic software framework that performs parametric and probabilistic analysis based on the response of complex system codes. RAVEN can communicate with any system code through its Application Programming Interfaces (API) as long as all the parameters that must be perturbed are accessible by input files or python interfaces. Currently, RAVEN is coupled to several simulation codes, including RELAP5-3D.

NOMENCLATURE

BEAVRS	Benchmark for Evaluation And Validation of Reactor Simulations
ECCS	Emergency Core Coolant System
NRC	Nuclear Regulatory Commission
ppm	parts per million
PWR	Pressurized Water Reactor
RELAP5-3D	Reactor Excursion and Leak Analysis Program

MULTI-RESTART RELAP5-3D INPUT

In order to assess the compliance with the newly proposed rule, a three stages analysis is needed:

1) The nuclear power plant operation needs to be simulated until reaching the equilibrium cycle;
2) In order to determine challenging conditions for the LOCA scenarios (high power peaking factor), an operation maneuver needs to be considered and simulated;
3) The LOCA scenario can be simulated.

The PRA analysis (see Fig. 1) is characterized by:

- Sampling of the time at which the maneuver will 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.

In the analysis, RAVEN calls RELAP5-3D combined with PHISICS on a single thread for each parameter selection of the PRA. However, each parameter selection is comprised of a vector of values, some in the RELAP5-3D model, some in the PHISICS model. Moreover, some of the parameters affect the RELAP5-3D steady-state model while others are applied in the ensuing LOCA analysis. Also, to achieve a reasonable runtime, the number of flow channels in the core model is reduced for the LOCA analysis.

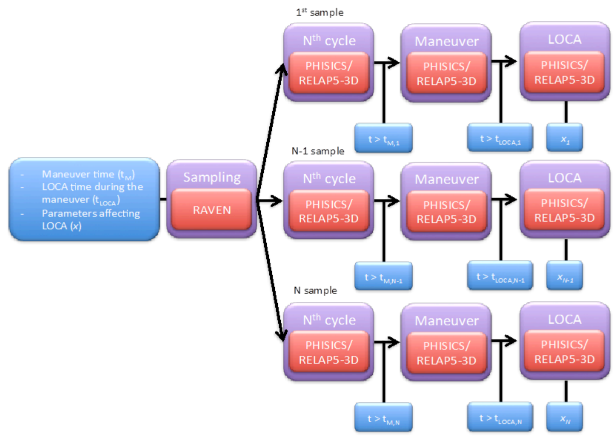

Figure 1. Current PRA strategy scheme

To achieve maximum accuracy in the first two stages, the core is modeled with one flow channel per fuel assembly by RELAP5-3D. In the third stage (LOCA), the number of flow channels in the core is reduced while the rest of the primary and secondary systems are added.

The diagram of the PRA analysis is given in Fig. 1. It indicates that the output of the first RELAP5-3D/PHISICS run is used as input to the second, and the second to the third. Moreover, the second and the third runs each must modify the preceding model. RELAP5-3D multi-case input is inadequate because, even though the required changes to one model can be made by the next case, the code reinitializes everything to initial conditions before running each and every case.

The RELAP5-3D restart process also allows modification to the input model, however it does not reinitialize as multi-case does, but rather continues from a user-selected restart time. The need to run all three RELAP5-3D/PHISICS calculations in sequence on a single thread of a parallel RAVEN job placed new requirements on RELAP5-3D, namely that three separate runs, a base case and two restarts, would not suffice. To run all three from a single RELAP5-3D input file created the new requirement that one input deck of the file must be able to restart a preceding one. Though the ability to run multiple input decks in a single input RELAP5-3D file has long existed, the capability to restart an earlier deck did not.

Two changes enabled this capability. First, because at the conclusion of the first deck's processing, the restart file sits at its end-of-file, immediately repositioning to its beginning allowed one restart. However, when the first restart run finished input processing, the code wrote a new restart dump with all the modified model information on the restart file. Because this restart dump had the same timestep and cumulative time as the restarted dump it was restarting, the second and subsequent multi-deck restarts failed. Therefore, a change was made to allow only one restart dump at each value of cumulative time. It is carried out by overwriting the first such restart dump with the newer one.

This multi-deck feature passes RELAP5-3D sequential verification testing [6, 7]. It enables many new forms of parameter studies. The study presented here is one example.

CORE DESIGN

The reference plant chosen for the purpose of this project is a typical PWR. The model is based on a detailed PWR benchmark BEAVRS (Benchmark for Evaluation And Validation of Reactor Simulations) [8], having real plant data for assessing the accuracy of reactor physics simulation tools for the first 2 operational cycles. In Figure 2 and Table I, the radial core layout and the plant key parameters are shown respectively.

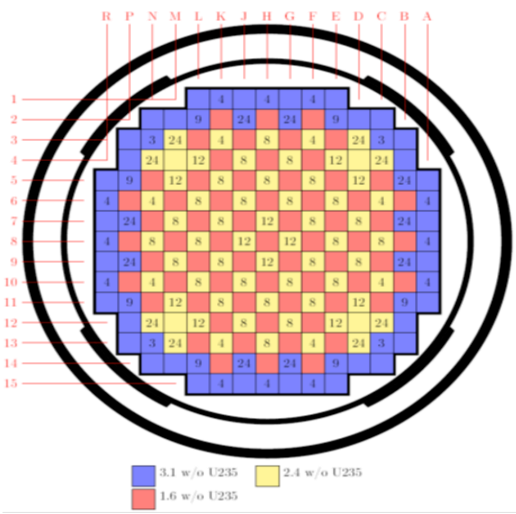

Figure 2. Reactor Core layout

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.

For 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 108 tabulation points has been selected.

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

MULTI-CYCLE ANALYSIS

In order to assess the compliance of the existing power plants to the new NRC rule, the LOCA accident scenario needs to be initiated in equilibrium cycle conditions, something reached at all operating US nuclear power plants nowadays. 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, it is assumed that the equilibrium cycle is reached after the 10th reloading.

For the first 10 cycles, the TH model contains only the reactor core (without primary and secondary system) 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, as it can be seen in Fig. 3. To achieve the greatest accuracy 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).

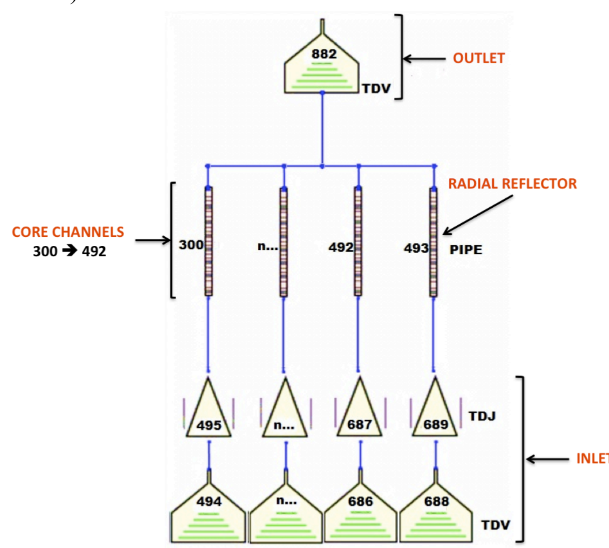

Figure 3. Reactor Core RELAP5-3D nodalization

Thus the RELAP5-3D model runs initially to steady state with a core-only model then runs an operational transient with feedback to and from PHISICS to the end of the 10^{th} cycle. These are the first two stages of each thread in Fig. 1.

The BEAVERS benchmark provides data for the first 2 cycles only (1 reloading pattern). For cycles 3 through 11, new reloading patterns have been constructed. The BEAVERS reloading is a "high-leakage/low-energy" pattern. The goal here is to perform analysis on a modern reloading pattern; the first developed 4 cycle patterns represent a gradual migration from "high-leakage/low-energy" to "low-leakage/high/energy" reloading patterns. 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 Fig. 4.

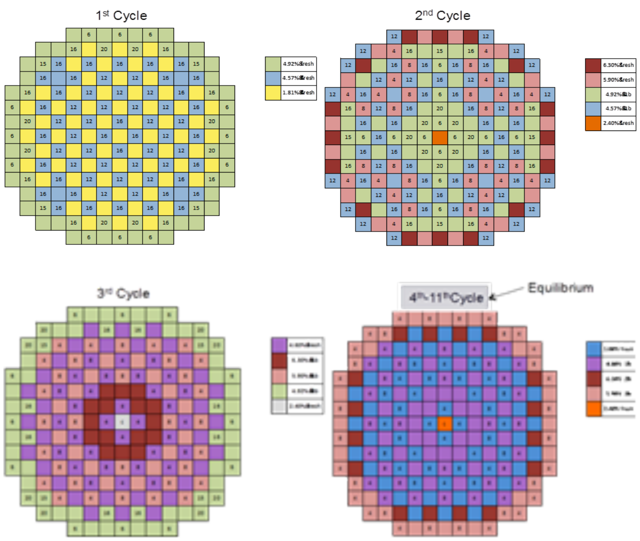

Figure 4. Loading Pattern

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

The new approach for the analysis of LOCA scenarios requires 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 different batches (Fresh Fuel, once-burned, twice-burned)
- 3 pins, in the above zones, with the highest peaking factors

This change in the RELAP5-3D model is accomplished in through restart input and represents stage three in each thread shown in Fig. 1.

The mode is extended by adding the 4 loop primary system for the LOCA analysis [9]. As an example, Fig. 5 reports the assembly-wise radial integrated power and peaking factors for the BOC (beginning of cycle), MOC (middle of cycle), and EOC (end of cycle) at the 10th cycle. Fig. 6 shows the detailed fuel exposure (burn-up) for the same points in time.

At these 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.

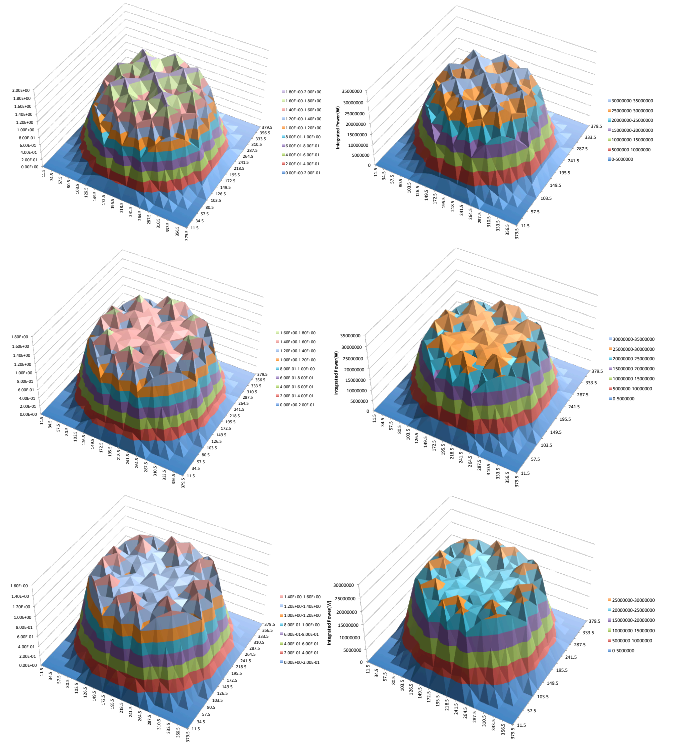
Figure 5. Power (left) and Assembly Peaking Factor (right) for BOC (top), MOC and EOC

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. As already mentioned, for the scope of this work, the maneuver that has been considered is a load-following operation of the reactor.

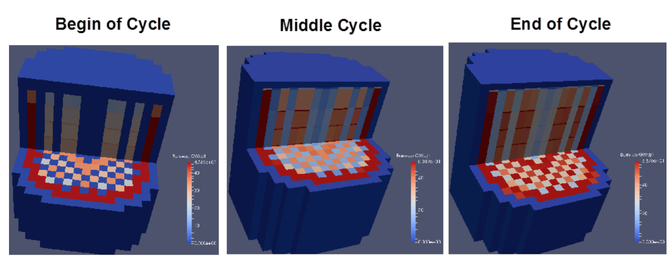

Figure 6. Burnup for BOC, MOC and EOC

Figs. 7 and 8 show the results of the analysis. As it can be inferred in Fig. 8 the core status at BOC, MOC and EOC does not determine challenging conditions for the LOCA analysis.

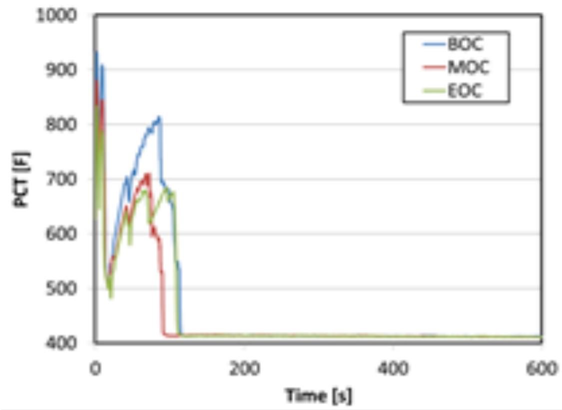

Figure 7. Peak clad temperature during the LBLOCA scenario initiated at BOC, MOC and EOC [9].

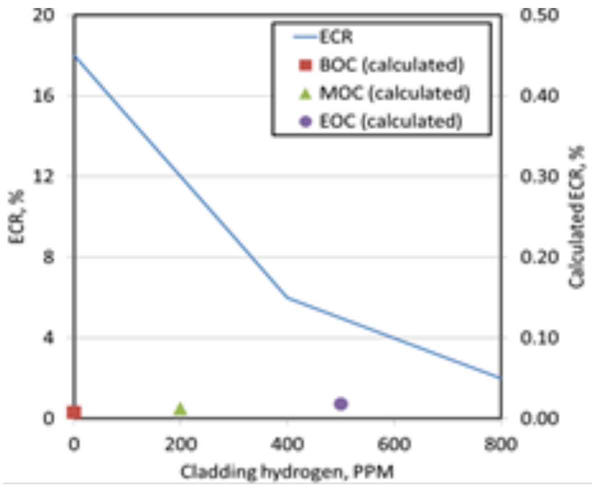

Figure 8. Maximum local oxidation rate during the LBLOCA scenario initiated at BOC, MOC and EOC [9].

PRA Strategy

In order to assess the compliance of the operating nuclear power plants to the new rule, a rigorous Probabilistic Risk Assessment (PRA) is carried out. 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 must be considered in such analyses 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 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 [10,11] 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.

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