Improvements in simulation tools to be developed within the framework of the ASTRID project

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IMPROVEMENTS IN SIMULATION TOOLS TO BE DEVELOPED WITHIN THE FRAMEWORK OF THE ASTRID PROJECT

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The ASTRID design comprises innovative features compared to past designs. The simulation tools being very important to support the ASTRID design option selection and to assist a robust Safety demonstration, the CEA and its industrial partners have launched a large program for developing a new generation of simulation tools. Within the framework of the ASTRID project, the strategy for simulation is to continuously improve the simulation tools and their verification and validation (V&V). Furthermore, this new generation of tools is implemented for the basic design of ASTRID in compliance with the regulatory and schedule requirements.

Several examples of computation tool developments in the fields of neutronics, fuel behavior, core mechanics, thermal-hydraulics and severe accident analyses are given. The V&V process, described here for the core studies, is also carried out for others domains by the industrial partners.

The approach is closely linked to the realization of the R&D experimental programs, aimed to complete the existing experimental data base and so to validate the new model developments and to decrease the calculation uncertainties.

The development program of new simulation tools is ambitious in order to meet the challenges which arise from the innovative design options implemented in ASTRID and for the will to comply with the objectives of the 4th generation reactors.

I. INTRODUCTION

The ASTRID reactor (Advanced Sodium Technological Reactor for Industrial Demonstration) is a technological demonstrator designed by the CEA together with its industrial partners, subjected to a very high level of requirements.

Innovative options have been introduced in the design with advances on core and sodium-related issues and by taking into account the lessons learnt from the Fukushima accident. These options enhance the safety, improve reliability and operability, making the Generation IV SFR an attractive option for electricity production.

In support to the ASTRID project and to the R&D program with which it is closely linked, numerical simulation is more and more used, with the following objectives:

- to confirm the confidence on the feasibility of the proposed innovations,
- to help to steer R&D priorities in favour of the most promising innovations and needs of the ASTRID project,
- to quantify performances of components,
- to have a better uncertainty assessment, thanks to a multi-physics modelling,
- to limit conservatisms which involve extra costs,
- to help to develop new experimental programs and to size experimental devices.

The final goal is evidently to favor the acceptance of the safety demonstration by the Safety Authorities.

Consequently, the selection process of certain design options and the safety studies of the ASTRID reactor rely on the use of scientific computing tools, some of which require the development of new functionalities to fully address the needs and particularities of this new reactor.

For example, the CFV[a] core concept selected for the ASTRID reactor, aims for a low sodium void effect. This core concept features heterogeneous axial $(U,Pu)O_2$ fuel with a thick fertile zone in the inner core and is characterized by a crucible-shaped core with a sodium plenum above the fissile zone.

This architecture motivates the development of improved tools to describe the new features. The following description is going to be focused on some of the core tools in the field of neutronics, fuel behavior, core mechanics, thermal-hydraulics and severe accidents. The corresponding actions performed by the industrial partners of the project concerning their numerical tools used for the design of ASTRID, are not detailed in the present paper.

All of these developments have to comply with a rigorous approach in order to meet the requirements of the French Regulation. They also have to fit with the schedule and the different milestones of the project.

II. REGULATORY REQUIREMENTS AND THE CORRESPONDING PROCESS

The French Regulation for Basic Nuclear Installations, issued in 2012, requires the safety demonstration to rely on qualified calculation tools for the domains they are used in.

The Qualification level for a calculation tool is the final level following the well-known VVUQ (Verification, Validation and Uncertainties Quantification) process. This level is achieved at the end of a long-term process which involves several steps, depicted on Figure 1, and which will be described hereafter.

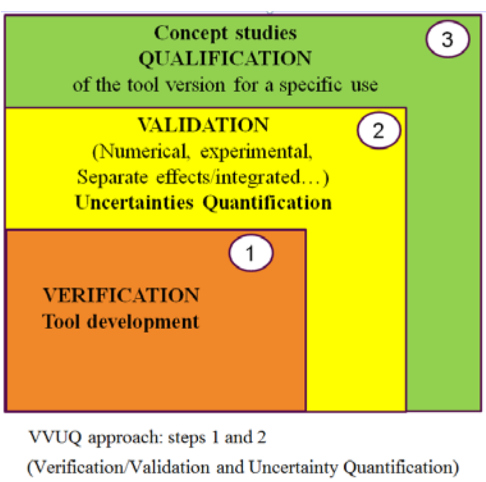

Fig. 1: Different steps leading to the qualification of the version of a tool as part of a perimeter of clearly defined use.

[a] French acronym for "*Cœur à Faible effet de Vide sodium*", *meaning low void effect core.*

Subsequent to the Development, the **Verification** step ensures that the resolution of the equations is correct. In other words, it must be ensured that the calculation tool works as expected (correct digital implementation, correct numerical solution).

Then, the **Validation** of a scientific calculation tool is the process of assessing its predictive ability of real phenomena with regard to the use in the targeted field. It aims to achieve the quantification of uncertainties associated with the calculated quantities.

The validation is to ensure that the mathematical model developed for the calculation of physical phenomena has the ability to represent them properly in an identified domain.

The validation phase involves comparing the results of the simulation tool to experimental data coming from mock-ups and/or reactor operation feedback, as well as to already qualified calculations (benchmarking).

The validation must prove, for example with the use of a coverage matrix, the relevance of the tests and physical phenomena analyzed in relation to the domain of use described. This stage can be achieved with the help of the PIRT (Phenomena Identification and Ranking Table) method which leads to the identification of the major physical phenomena.

The ASTRID simulation tools benefit from a vast experimental data base, relying on the feedback from numerous tests, particularly in the PHENIX and SUPERPHENIX reactors. Nevertheless, the innovative design options of ASTRID involve new needs in terms of R&D programs and motivate the development of new test facilities. Considering the considerable cost of these experiments and the need to widen the set of relevant experiments, international collaboration is required.

Finally, **the Qualification** step is the last stage of the process. During this step, it must be ensured that the field of use of the tool is consistent with its validation domain. The tool must be used in the domain where it is supposed to be valid and the proof of this verification must be provided.

The above described approach must be followed by all ASTRID partners in order to provide safety files, such as required by the French Safety Authorities.

III. SCHEDULE REQUIREMENTS

The schedule for development and verification, validation and uncertainties quantification and qualification of simulation tools is defined by the licensing milestones which are shown on Figure 2.

During the basic design phase, an industrialized version of each tool will be available in order to be used in the studies needed for the preliminary safety report.

A first level of validation and qualification of the simulation tools is required at the end of the Basic Design in order to support the Preliminary Safety Report. This first level will be built on the available validation results and will be completed during the next phases of the project by supplying the necessary validation supplements in due time.

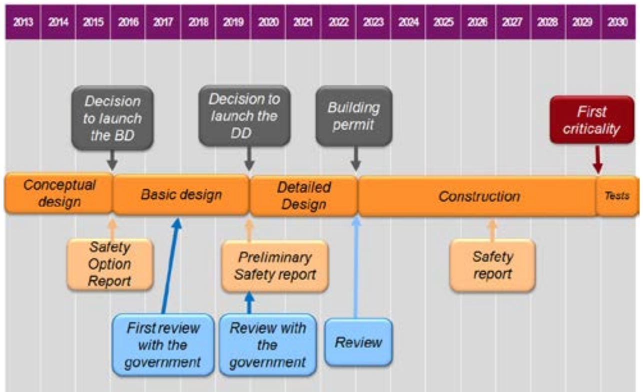

Fig. 2: ASTRID milestones.

IV. COMPUTATIONAL TOOLS

Several examples of computation tool developments in the fields of neutronics, fuel behavior, core mechanics, thermal-hydraulics and severe accident analysis are given in this section.

IV.A. Neutronics Tools

The CFV core concept is focusing on optimizing the economical performances, the fuel consumption and neutronic feedback parameters (reactivity coefficients) so as to obtain an improved natural core behavior during accident conditions.

More specifically, the low sodium void effect is the result of the following design features (see Figure 3):

- The axial heterogeneous concept, incorporating a thick fertile zone within the fuel area.
- The sodium plenum serving as reflector in a normal situation and as neutron leakage passage in case of incidental heating of the sodium. Above the plenum is a highly absorbent zone to capture the leaking neutrons from the core.
- The core geometry with a reduced inner core height (crucible core).

This core design also leads to a lower reactivity loss during cycle compared to previous designs.

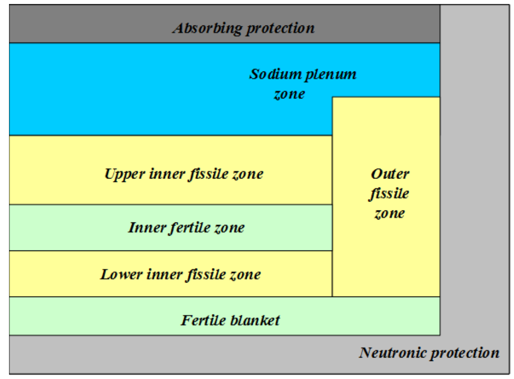

Fig. 3: Core design.

The ASTRID neutronic parameters for design and safety studies will be determined with three different neutronics tools, relying on the new CEA neutronics simulation platform:

- The core physics tool APOLLO3-SFR, based on the new APOLLO3® (Ref.1) deterministic transport code.
- The stochastic code TRIPOLI-4®, used for shielding and criticality application.
- The fuel cycle and depletion tool DARWIN3-SFR, mainly based on the APOLLO3® code and the MENDEL depletion code.

This paper is going to be focused on APOLLO3® and APOLLO3-SFR:

APOLLO3® is a modular neutronics calculation system for the deterministic resolution of the neutron transport equation. It provides components to be used as a toolkit. These components cover all the modeling scales found in reactor physics. APOLLO3® provides a modern tool with a new software architecture that is more robust and faster to converge. These developments have been made in order to match the specific issues regarding the CFV core which are a great heterogeneity of the core in the axial direction and an increased neutron leakage.

APOLLO3-SFR is a dedicated application of APOLLO3® code built to predict the main neutronic parameters of the CFV core using the most appropriate solvers available in the code.

New algorithms being capable of taking into account the heterogeneous neutron leakage and a fine energy discretization are needed for the calculation of fuel assemblies, but also to process the transitions between fissile heterogeneous media and subcritical environments. Finally the development of parallel computing is of great interest, in order to relax the constraints of IT resources and achieve the desired information. APOLLO3-SFR also includes the use of the JEFF-3.1.1 nuclear data library.

The APOLLO3-SFR first version is expected at the beginning of 2017.

The extensive feedback from past experiments performed with the French MARSURCA critical mock-up and from the PHENIX and SUPERPHENIX reactors operations contributes to the validation step of the new features. In case of lack of experimental data, the use of the reference Monte Carlo TRIPOLI-4® code is also required.

Nevertheless, the specific configuration of the ASTRID core requires an additional validation compared with the previous reactors. This is the reason for two complementary experimental programs which are in progress: one in the Russian critical mock-up BFS with analytical and parametric approach and the other with a more global approach in MASURCA.

IV.B. Fuel Behavior Tools

The CFV core concept, selected for the ASTRID reactor, features two main innovations evolutions that have consequences on fuel behavior:

- Heterogeneous axial $(U,Pu)O_2$ fuel with a thick fertile zone in the inner core, in order to increase the axial neutron leakage (see Figure 3). This axial heterogeneity creates differential strains on the cladding.
- A pin diameter larger than already tested in the PHENIX and SUPERPHENIX reactors (see Figure 4), coupled with a thin spacer wire. This larger pin together with a thinner spacer wire decreases the sodium volume fraction and consequently leads to a lower sodium void effect. This new design has consequences on the thermal behavior of the fuel and the thermal-hydraulics of the sub-assemblies.

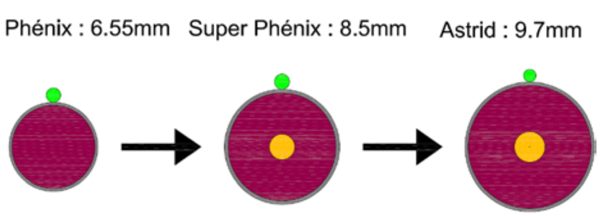

Fig. 4: Pin and spacer wire diameters evolutions.

The GERMINAL code[2] is used to design the fuel pins. This code makes use of the considerable experience gained in the French R & D programs in the RAPSODIE, PHENIX, CABRI and SUPERPHENIX reactors. The implemented models couple the thermomechanical behavior of the pin with the physicochemical changes in pellet and cladding under irradiation. It takes into account the thermal, mechanical, physical and chemical properties of the fuel and its cladding.

The GERMINAL code, which uses a classic 1D½ axisymmetric representation (see Figure 5) of the fuel pin geometry, is continuously improved. The fuel pin is divided into axial slices and the resolution of the physical processes in each slice uses a radial meshing of the slice.

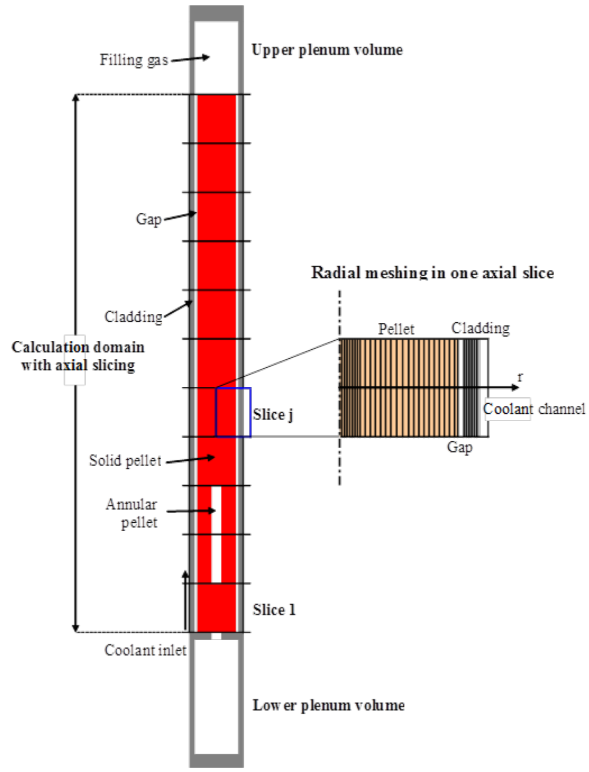

Fig. 5: Calculation model of the fuel pin.

GERMINAL V2 has recently been improved[2] by implementing the following modelling evolutions (compared to GERMINAL V1):

- Thermal analysis and mechanics are now based on finite elements computations.
- The fuel mechanical behavior model is able to handle elements emptied from material in case of central hole growth.
- The fuel pellet is considered as a continuum. The expansion due to cracking into several fragments during heat-up, and later the free volumes closure between the fragments are described by two imposed strain terms: relocation and accommodation. The resolution of the mechanical equilibrium allows evaluating the interaction between pellet and cladding, which can become significant in off-normal conditions, and also the stresses induced by swelling gradients in the cladding thickness.

- Off-normal conditions are simulated by using a strong coupling of the thermal-hydraulics in the coolant with the thermal analysis of the fuel pin.
- A new fuel depletion scheme has been developed, integrating the creation of the volatile fission products. Their release at high burn-up causes the build-up of JOG ("Joint Oxyde-Gaine") between pellet and cladding. With a refined prediction of the JOG composition, the evaluation of the heat exchange inside the gap at high burn-up, as well as the description of cladding corrosion will be improved.
- The neutronic depletion scheme has also been completed by taking into account the helium production, which is required to simulate fuel pins loaded with minor actinides.

All these evolutions implemented in GERMINAL V2 were validated by simulating fuel elements irradiated in PHENIX or in CABRI reactor, as it is shown on Figure 6. The experimental data base, called BREF, includes samples with various geometries and cladding materials, burn-ups and irradiation histories.

The comparisons of the calculation results with the experimental data show the current abilities and limits of the code, and consequently the remaining working perspectives.

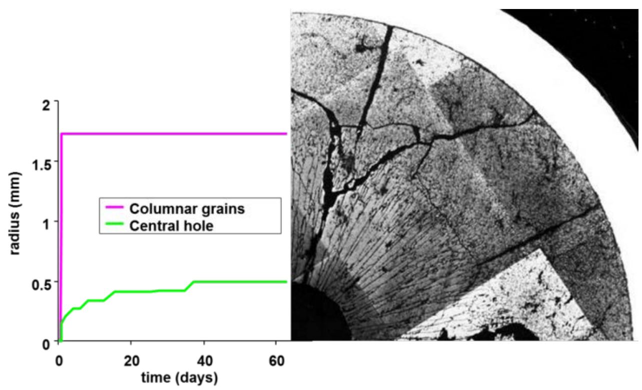

Fig. 6: Fuel restructuring
columnar grains and central hole evolutions
ONYX1 PHÉNIX assembly.

GERMINAL V3 is planned to be released at the end of 2017. The objective of this version is to reduce uncertainties in the calculation of physical quantities used in the design criteria. Work in progress for this new version is focused on:

- Gap closure kinetics, to be revised by interpreting 3D simulations of the pellet fragment behavior.
- Fission gases and helium behavior (pellet swelling and gas release from pellet).
- Internal corrosion of the clad.
- First description of the fission product phases present in fuel pellet, prone to be released and involved in the build-up of JOG. This description is made possible by the coupling between GERMINAL and a thermo-chemistry code.

Besides, additional improvements are planned:

- A more detailed description of the fission product transport within the fuel pellet before release and JOG build-up.
- Failed pins behavior.
- Oxide mechanical behavior modelling with cracking and an improved description of creep for high temperatures and high strain rates.

These new developments will of course require a new validation, as it has been done in the past (see Figure 7).

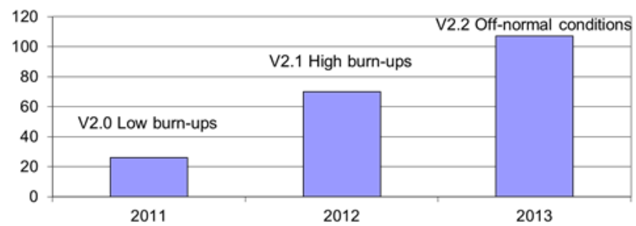

Fig. 7: Evolution of GERMINAL V2 validation base/
Number of samples vs code version.

IV.C. Core Mechanics Tools

In order to evaluate the design choices and their consequences on the mechanical behavior of the core, three new simulation tools have been developed. All of them are relying on the CAST3M computer code.

The mechanical studies must take into account several types of static and dynamic phenomena.

The static mechanical studies are carried out with DOMAJEUR2^{CAST3M} or HARMONIE2^{CAST3M}.

DOMAJEUR2^{CAST3M} can describe the 3D pin bundle thermal-mechanical behavior and its interaction with the wrapper tube. For the different needs associated with the design and dimensioning, three calculation schemes have been developed.

- The first is a fast analysis scheme which allows, with a short calculation time (about one minute), to assess the risk of interaction between the pins, the spacer wires and the wrapper tube as it has already been observed (see Figure 8). It also assesses the mean severity of the interaction at all the axial positions.
- The second one, an industrial scheme, can assess in a reasonable computational time the pin damages and the deformation of a whole hexagonal wrapper tube,

taking into account all the major physical phenomena (creep, swelling, expansion, contacts…).

- The third one is the best-estimate scheme consisting of a geometrically accurate – thus costly – finite element model limited to a small number of rod pins which serves as a reference for development and the validation of the simplified schemes above.

The development of DOMAJEUR2^{CAST3M} began in 2012 and the VVUQ process is in progress so as to reach at the end of 2017 the 2.2 version to perform the Preliminary Safety Report studies. The validation plan is based on the available CEA data.

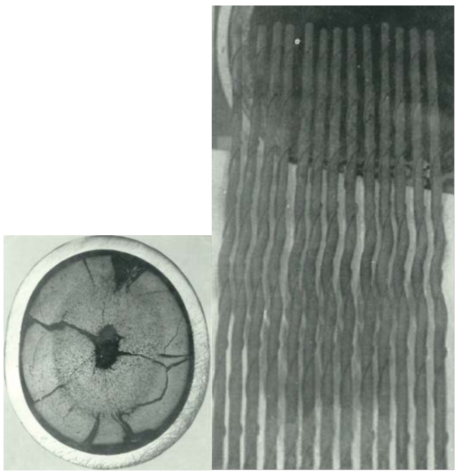

Fig. 8: Pin deformation.

HARMONIE2^{CAST3M} can describe the 3D core thermal-mechanical behavior and its interaction with the core restraint structures for various irradiation conditions, from normal to accidental ones. The core behavior is analyzed during nominal conditions, fuel handling operations and unprotected transients (i.e. with complete failure of all automatic shutdown systems)[3].

This tool describes the core assembly duct bowing due to thermal expansion, irradiation swelling and duct-to-duct interaction which occurs at the spacer pads during the reactor lifetime. As in the case of DOMAJEUR2^{CAST3M}, several calculation schemes can be used, depending on the accuracy of the results sought or the available computational resources.

The current version V2.0 of HARMONIE2^{CAST3M} is the result of several improvements (compared with the previous versions HARMONIE V2 and HARMONIE1^{CAST3M}), as for example the support grid simulation, the swelling behavior of the duct, taking into account individually each subassembly, the introduction of an irregular grid and the plastic deformation of spacer pads and subassembly nozzles.

HARMONIE2^{CAST3M} V3.0 is planned for the beginning of 2017. It will be fully validated with CEA available data and will be able to face all the needs required for ASTRID: thermal and irradiation creeping of the subassembly spacer pads, distribution of deformation on the six faces of the duct due to pressing-in of one or more spacer pads.

BASILIQ^{CAST3M} is used to understand the dynamic behavior of the core and to carry out the design and safety studies. Two main movements have been considered so far: global horizontal movements following a seismic excitation and flowering – *i.e.* radial expansion – of the core (see Figure 9). The dynamic behavior of the core is strongly influenced by the presence of sodium (which leads to complex interactions between the structures in the whole core) and by the contacts between subassemblies. Plasticity and friction also have to be considered to correctly reproduce the dynamic response of the core. Two modeling levels are available in BASILIQ^{CAST3M}: the main one features a homogenization technique which drastically reduce the size of the numerical system and leads to an execution time of about one hour for a transient on a full core, whereas the other can be used as a best estimate calculation to provide more detailed results on a specific reduced area of the core[4].

BASILIQ^{CAST3M} is in continuous improvement in order to meet in 2017 (with BASILIQ^{CAST3M} V3) the needs of ASTRID. The validation plan will use available CEA data like the flowering test performed at PHENIX (2013), as well as the results from collaboration with IGCAR.

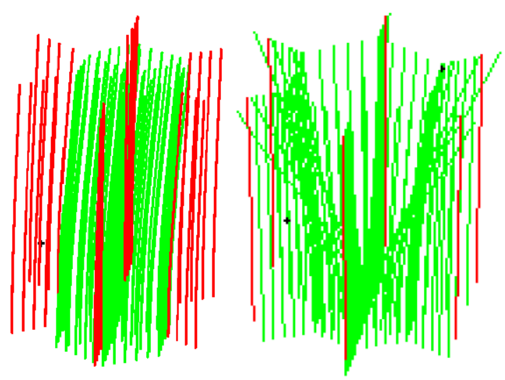

Fig. 9: Global (left) and flowering (right) movement of a core bundle.

IV.D. Thermal-Hydraulics Tools

Thermal-hydraulic simulations are needed in order to support design option choices, as well as to perform the transient calculations required in support of the reactor safety analysis.

For the latter case, several transients have to be described: inadvertent control rod withdrawal, pump over speed, loss of primary flow, loss of heat sink...

These transients give rise to a wide range of physical phenomena: for some transients, local effects may influence the global reactor behavior. In order to reproduce these effects, CEA has adopted a "multi-scale" approach concerning the TH analysis (Fig. 10). Three codes are involved in the modelling of the primary circuit behavior[5,6]:

- The CATHARE system thermal-hydraulics (STH) code can represent on a coarse scale all the components of the primary system (core, plena, pumps, exchangers) and the associated loops (secondary circuit, Decay Heat Removal systems, …).
- The TrioCFD code is used to model 3D sodium flows within large plena at the fine scale of CFD (Computational Fluid Dynamics).
- Finally, a sub-channel code, TrioMC, has been developed in order to model the flows within subassemblies at an intermediate modelling scale between the coarse STH scale and the fine CFD scale. This code can be coupled with a TrioCFD model of the gaps between the hex-cans: this coupling is referred to as TrioMC2.

Depending on the transient considered, these three codes can be used:

- In sequence ("chaining") if the local effects (calculated by TrioCFD or TrioMC) do not affect the overall dynamics of the transient (calculated by CATHARE). In this case, the results of a standalone CATHARE calculation can be used as a boundary condition for TrioCFD or TrioMC calculations.
- Simultaneously ("code coupling") if feedback from local effects on the overall dynamics is expected. In this case, a CATHARE calculation of the whole circuit is conducted in parallel with the TrioCFD or TrioMC calculations: at each time steps, the results of the detailed calculations are used to improve the predictions of the coarse STH models.

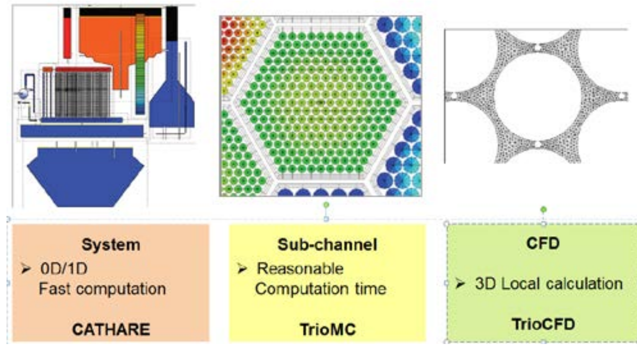

Fig. 10: Thermal-hydraulic simulation tools/ available modelling scales.

In the following, the development and validation of CATHARE3 (the next version of CATHARE) for ASTRID applications will be detailed. Several developments were undertaken:

- Architectural improvements (such as dynamic memory allocation, the removal of fixed-size tables and the conversion from FORTRAN to C++) were implemented.
- Several ASTRID-specific component models are being implemented in order to model, primary pump performance (including cavitation regimes), electromagnetic pump behavior and heat exchanger performance.
- The main two-phase correlations used in CATHARE, which were originally established for water flows, are being reviewed and updated in order to improve the modelling of two-phase sodium flows, in particular by benchmarking them against the correlations used existing sodium boiling codes such as SABENA (JAEA) and TRACE (PSI).
- Finally, CATHARE 3 is the focal point of many code couplings considered for ASTRID, both with the thermal-hydraulic codes TrioCFD and TrioMC and with fuel thermal mechanics (GERMINAL) and 3D neutronics (APOLLO3). Most of these couplings still require developments in CATHARE.

Besides these major developments, several minor ones are planned in CATHARE 3. For example, thanks to the CATHARE 3 module allowing adding new fluids, an external vessel heat exchanger with featuring oil-based coolant could be simulated. The developments are planned to be implemented progressively till 2024, with a first version having the same features as CATHARE 2 expected at the end of 2016 and an intermediate version which will include two-phase system behavior improvements expected at the end of 2019.

The validation of CATHARE 3 relies on several sources:

- The behavior of each component (sub-assembly, heat exchanger, pump…) can be validated by means of component mock-ups and analytical experiments. It has to be underlined that for some components, such as pumps or heat exchangers, the validation data required will be provided by the component's qualification program.
- The integral behavior of the system has to be validated with full-scale reactor tests. This validation will use extensive feedback collected from the operation of PHENIX and SUPERPHENIX, as well as from reactors operated abroad. In this field, code - to-code benchmarks also increase confidence in the simulation results.

Finally, it has to be kept in mind that several new validation needs are conditioned to the realization of new mock-ups or to the availability of manufacturer data.

IV.E. Severe Accidents Tools

In the ASTRID project, the safety objective is to prevent the core melting, in particular by the development of an innovative core with complementary safety prevention devices, and to enhance the reactor resistance to severe accidents by design. To mitigate the consequences of hypothetical core melting situations, specific dispositions or mitigation devices will be added to the core and to the reactor.

Severe accident situations have to be taken into account in the core design and require specific simulation tools to describe the core degradation and its consequences. The CEA strategy relies on two sets of calculation codes: a reference mechanistic set of codes and a set of physical models with statistics and probabilistic capabilities[7].

Concerning the core, the reference mechanistic set of calculation codes includes SIMMER embedded in the SEASON (SEvere Accident SimulatiON) Platform and SCONE (Software for COrium Na interaction Evaluation) which are going to be described in more detail hereafter.

The reference mechanistic set of calculation codes previously used to simulate core degradation and corium progression towards the core catcher was composed of SAS-SFR and SIMMER-III or IV (SIMMER-III is a 2D simulation whereas SIMMER IV is a 3D one : both of them are JAEA property).

Thanks to the collaboration between JAEA and CEA[8], the development of SIMMER-V is in progress in order to take into account the innovative ASTRID design and to cope the SAS-SFR/SIMMER limitations, as for example the significant non-continuity of the simulation when switching form SAS-SFR to SIMMER at the end of the primary phase due to different models used in the 2 codes. The present study includes the result of "Technical development program on a commercialized FBR plant" entrusted to Japan Atomic Energy Agency by Ministry of Economy, Trade and Industry of Japan (METI).

SIMMER-V will be able to model heterogeneous fuel pins behavior during the primary phase, as well as thermal and thermomechanical behavior (with radial heat transfer) in reactor structures; it will be able to describe:

- Core degradation (see on Figure 11) with continuity between the different phases of degradation (the primary phase of the core degradation until the first failure of a hexagonal wrapper tube, the transition phase (propagation of the corium pool), the secondary phase (behavior of the corium pool) and the relocation phase. To simulate the primary phase, a new pin degradation module including in-pin fuel motion capabilities adapted to axially heterogeneous pin is under development in SIMMMER-V.
- Core melt progression towards the core catcher through the dedicated transfer tubes implemented in the core to allow downward corium dispersion for lowering the core reactivity.

SIMMER-V will be coupled with a fuel irradiation code (for example GERMINAL) and a primary loop thermo-hydraulics code (for example CATHARE) by means of a new platform named SEASON. SEASON will also provide an interface between SIMMER-V and an external neutronics code, such as APOLLO-3®. Due to the integration of existing calculation codes, the SEASON platform will benefit of the progress and validation of each code.

The partnership between JAEA and CEA began in 2014 and is planned to last until 2020 (Ref.8). The working tasks are in progress and will include in 2016 a first version of SIMMER-V and then a new version every year with improved physical models. The first version of SEASON including SIMMER-V will be ready in 2017.

SIMMER-V will be validated on the base of the existing experimental database (CEA programs CABRI and SCARABEE, and the JAEA programs EAGLE1&2) and on expected future programs. CEA plans to perform experimental programs in the PLINIUS-2 (Platform for Increase of Nuclear Industry and Utility Safety) experimental facility which is under design. This platform will contribute to the validation of the simulation of the corium flow in the transfer tubes.

Furthermore, ASTRID’s innovative heterogeneous core design requires new experimental validations. CEA targets to improve the validation of SIMMER for pin and bundle degradation. To do so, CEA is defining and studying the feasibility with NNC-RK (National Nuclear Center-Republic of Kazakhstan) of the in-pile experiment program SAIGA (Severe Accident In-pile experiments for

Generation IV reactors and ASTRID project) in the IGR reactor, which had been used for the EAGLE experimental program. The goal is to increase the knowledge about the degradation of axially heterogeneous pins that contain two fissile zones separated by a fertile zone.

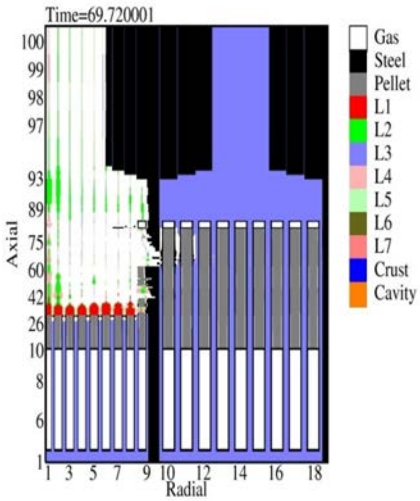

Fig.11: SIMMER Calculation: Corium propagation from a Total Instantaneous Blockage scenario (TIB). L1 to L7 are liquid phases: L1/fuel, L2 /steel, L3/Na

In order to describe the Fuel/corium and Coolant Interaction (FCI), CEA recently started the development of the SCONE code. SCONE will be used to simulate all types of interaction, such as ejection of corium out of a ruptured pin following a power transient, in case corium propagation to neighboring sub-assemblies following an instantaneous total assembly blockage, during corium progression through the transfer tubes, and corium interaction in the cold plenum during corium discharge on the core catcher…

The main phenomena to be simulated have been identified.
For some (fragmentation of the corium drops, flow map), it is possible to find models in the literature. For others (sodium boiling systems and the related heat transfer), research work led to progress but has to be continued and experiences are needed.
A first version of SCONE will be ready in 2016, but all the developments to be implemented will not be included at this time.

SCONE will be validated on the existing experimental data as the Fuel Coolant Interaction large scale experimental program at JRC ISPRA, FARO-TERMOS.

Nevertheless, the presence of specially designed corium discharge ducts (Transfer Tubes) between the core region and the lower plenum increases the probability of very hot corium interacting with the sodium in the lower plenum, and progressively increasing the lower plenum sodium temperature. Moreover locally in the vicinity of the corium jet, high sodium temperatures might be reached. Experiments with low sodium subcooling are thus necessary, as well as tests in which corium is introduced inside the sodium pool by means of discharge channels. The CEA is currently designing the PLINIUS-2 platform (see above) to also perform in particular these types of experiments.

More data are also necessary in order to dispose of a satisfactory heat transfer model for sodium film boiling which is necessary for SCONE. The feasibility of an experiment called SERUA (Sodium boiling Experiment Rig for Understanding of fuel-coolant interAction) is being studied.

V. CONCLUSIONS

A new generation of tools is currently under development at CEA for the Basic Design phase of the ASTRID project, in compliance with the regulatory and schedule requirements and in order to support the ASTRID design option choices and a robust Safety demonstration.

These new developments lead to the definition of experimental programs to complete the existing experimental data base and so to validate the new model developments and to decrease the calculation uncertainties. As far as possible, existing facilities are used, either in France or abroad in the framework of international collaborations.

The approach described here in the field of the core studies is also carried out for others domains by the industrial partners.

The development program of new simulation tools is ambitious in order to meet the challenges which arise from the innovative design options implemented in ASTRID and for the will to comply with the objectives of the 4th generation reactors.

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REFERENCES

1. B. ROQUE, P. ARCHIER, P. BOURDOT, C. DE SAINT JEAN, F. GABRIEL, JM. PALAU, V. PASCAL, D. SCHNEIDER, G. RIMPAULT, JF. VIDAL, "APOLLO3 roadmap for a new generation of simulation tools devoted to the neutronic core calculation of the ASTRID prototype", Proc. Int. Conf. on Fast reactors and related fuel cycles, FR13, Paris, France, May 2013.

2. M. LAINET, V. BOUINEAU, T. HELFER, M. PELLETIER, "Recent modelling improvements in fuel performance code GERMINAL for SFR oxide fuel pins". IAEA-CN-199/241, International Conference on Fast Reactors and Related Fuel Cycles (FR13), March 2013, Paris, France.

3. P. LAMAGNÈRE, S. DARDOUR, R. DOS SANTOS, A. FREIERMUTH, M. MARQUÉS, N. PÉROT AND N. SCHMIDT, « Multi-objective optimization of sub-assemblies design towards static mechanical equilibrium of sodium-cooled fast reactor core ». SMiRT-23, Manchester, United Kingdom - August 10-14, 2015.

4. D. BROC, J. CARDOLACCIA, L. MARTIN, "Physical and numerical methods for the dynamic behavior of the fact reactor cores. ASME Pressure Vessel &Piping Division Conference. Anaheim, California, USA, July 20-24, 2014.

5. A. CONTI, A. GERSCHENFELD, Y.GORSSE, " Numerical analysis of core thermal-hydraulic for sodium cooled fast reactors ", 16th International Topical Meeting on Nuclear Thermal-Hydraulics (NURETH-16), Chicago, USA, paper 12980 (2015).

6. M-S. CHENAUD, S. LI, M. ANDERHUBER, L. MATTEO, A. GERSHENFELD, "Computational thermal hydraulic shemes for SFR transient studies", 16th International Topical Meeting on Nuclear Thermal-Hydraulics (NURETH-16), Chicago, USA, paper 13084 (2015).

7. F. SERRE, F. BERTRAND, C. JOURNEAU, C. SUTEAU, D. VERWAERDE, D. SCHMITT, B. FARGES, « Status of the French R&D program on the Severe Accident Issue to Develop GENIV SFRs" ICAPP 2015, May 03-06, 2015 - Nice (France), paper 15373.

8. J. ROUAULT, P. LE COZ, J-C. GARNIER, J-M. HAMY, H. HAYAFUNE, T. IITUKA, H. MOCHIDA, « Japan-France collaboration on the ASTRID program and sodium fast reactor » ICAPP 2015, May 03-06, 2015 - Nice (France), paper 15440.