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R&D and Experimental Programs to support the ASTRID Core Assessment in Severe Accidents Conditions

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Abstract – *The ASTRID reactor (Advanced Sodium Technological Reactor for Industrial Demonstration) is a technological demonstrator designed by the CEA with its industrial partners, with very high levels of requirements.*
In the ASTRID project, the safety objectives are to prevent the core melting, in particular by the development of an innovative core (named CFV) with complementary safety prevention devices. To mitigate the consequences of hypothetical core melting situations, specific dispositions or mitigation devices are added to the core and to the reactor. To support those developments, a large R&D program was launched. This R&D program encompasses the development of simulation tools, and, despite an already large existing experimental database, a complementary experimental program to better evaluate the safety margins. In particular, to extend the CABRI and SCARABEE in-pile test programs carried-out by CEA/IPSN, a feasibility study is underway for running a complementary program, named SAIGA, in the Impulse Graphite Reactor (IGR) operated by NNC-RK, with CFV-like heterogeneous fuel pins. In-core mitigation devices qualification and corium-sodium interaction studies will be carried-out in the PLINIUS-2 facility designed by CEA.
The CFV innovative core, the SAIGA objectives, and the first SAIGA feasibility results are presented, as well as the PLINIUS-2 objectives, set-up and preliminary programs.

I. INTRODUCTION

The ASTRID reactor (Advanced Sodium Technological Reactor for Industrial Demonstration) is a technological demonstrator designed by the CEA with its industrial partners, with very high level of requirements[1]. Innovative options have been integrated to enhance the safety, to reduce the capital cost and improve the efficiency, reliability and operability, making the Generation IV SFR an attractive option for electricity production. The WENRA (Western European Nuclear Regulators Association), in line with the continuous improvement of nuclear safety, proposes to extend, at the design stage of the new reactors, 'the design beyond traditional design basis, in the area of core melt prevention and mitigation'; therefore severe accident sequences have to be considered in order to foresee dispositions as a fourth level of defense in depth. It is specified that for accident with core melting, 'design provisions have to be taken so that only limited protective measures in area and time are needed for the public'.

To meet these objectives for ASTRID, a large R&D program was launched in the Severe Accident domain, with a large number of partners. This R&D program covers the approach to define the mitigation devices, the methodology to study the hypothetical severe accident

situations, the development and adaptation of simulation tools, and, despite an already large existing experimental database, a complementary experimental program to reduce uncertainties.

In particular, to extend the CABRI and SCARABEE in-pile test programs carried-out by CEA/IPSN at Cadarache, France, jointly funded by CEA/IPSN and Japan and devoted to fuel pin degradation study under SFR severe accident conditions[2], a feasibility study is underway for running a complementary program in the Impulse Graphite Reactor (IGR) operated by IAE-NNC, with ASTRID-like innovative fuel pins. The feasibility study of this program named SAIGA, standing for 'Severe Accident In-Pile experiments for Generation IV reactor and ASTRID project', is carried-out by CEA and NNC-RK, with the technical support of JAEA to benefit from the feedback of EAGLE1&2 programs in IGR.

For studying the Fuel-Coolant Interaction, assessing the codes models, and qualifying the design of the ASTRID mitigation devices (including in-core mitigation devices), CEA is designing a PLINUS-2 platform dedicated to out-of-pile experiments with large masses of UO_2 and sodium coolant. The program carried out in PLINIUS-2 will enlarge in particular the EAGLE1 & 2 database, programs managed by JAEA and performed by NNC-RK in IGR and out-of-pile facilities[3, 4].

This paper will present:

- The innovative ASTRID CFV core with its mitigation devices,
- The associated R&D issues in the severe accident domain including:
  - The CFV core behavior in severe accident situations, and the associated SAIGA program feasibility,
  - The corium relocation through dedicated mitigation devices out of the CFV core region, the PLINIUS-2 test programs with the associated PLINIUS-2 platform,
  - The fuel coolant interaction (FCI) and the associated PLINIUS-2 test program.

II. THE ASTRID CFV CORE [5]

During the ASTRID conceptual design phase (2013-2015), the choice of an innovative heterogeneous core with a low sodium void effect (named CFV core) was confirmed. It is composed of large-diameter fuel pins, cladded with AIM1 material and with small spacer-wires; the inner core has heterogeneous fuel pins with 2 layers of fissile pellets, and in-between, a layer of fertile pellets. In addition, the length of the pellet stack in the inner core is smaller than in the outer core. The version 3 of the CFV core is also equipped with additional complementary safety devices acting for the protection of the reactor to prevent Severe Accidents in case ULOF or ULOHS (passive hydraulic-suspended shutdown systems acting in case of loss of flow for the first system, and sodium temperature actuated system for the second one). To mitigate the consequences of hypothetical severe accidents, two types of mitigation device are present in the core:

- Control Rod Guide Tube (CRGT) (as used in Monju[6] and JSFR[7] to mitigate the accident) which do not allow direct corium relocation in the ASTRID cold plenum due to the complex supporting core structure of the integrated reactors,
- Dedicated Corium Transfer Tubes (DCS-M-TT, standing for 'Complementary Safety Device for Mitigation – Transfer Tube') in order to evacuate the corium through the diagrid and the strongback toward the core catcher.

This core version (CFV V3), which was finalized in 2014, was used to assess the behavior of the CFV type core under severe accident conditions, before describing the CFV core in the Safety Option Report.

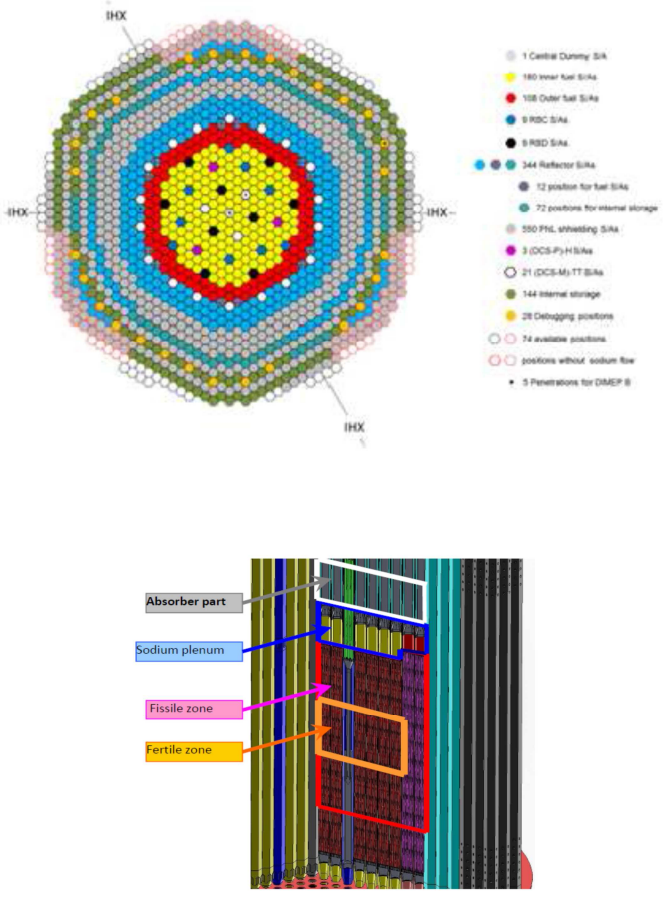


Fig. 1 : ASTRID core main features

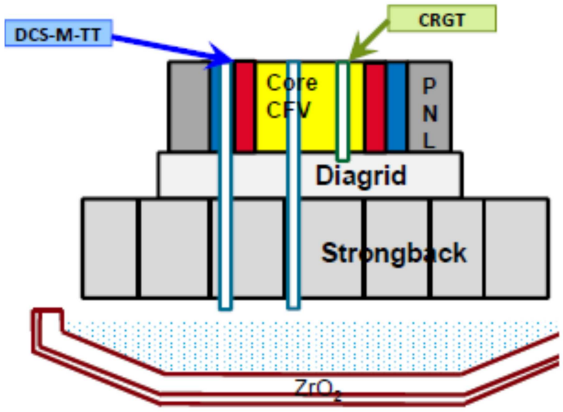


Fig. 2 : ASTRID core, CRGT and Transfer Tubes

III. EVALUATION OF SEVERE ACCIDENT PROGRESSION IN ASTRID

As early as the pre-conceptual and the conceptual phases of ASTRID, the CFV core behavior under severe accident conditions has been evaluated. To enter a severe accident sequence, it is supposed that the scram systems and the severe accident prevention systems have failed. The CFV core transient behavior was studied under the following conditions:

- Unprotected local cooling faults (USAF),
- Unprotected loss of flow (ULOF),
- Unprotected transient overpower (UTOP).

The CFV core behavior in case of USAF (Fig. 3) shows a slightly quicker onset of cladding degradation compared to SuperPhenix-like core due to the smaller sodium inventory between the pins; however, the hexagonal tube rupture of the blocked fuel subassembly occurs at nearly the same time and the molten mass at this moment is smaller. One may point out that, after clad melting, a larger mass of molten steel is driven by sodium vapor toward the upper structure due to larger vapor velocities in the narrow sodium channels and longer sodium boiling from the fertile layer zone. Some uncertainties exist on the possible plugging of the upper part of the subassembly by refrozen steel deposition; it may affect the corium transfer to neighboring subassemblies in case of leak-tight plug formation (dynamic radial transfer may occur due to pressurization of the degraded upper fissile zone). In addition, there are some uncertainties on the stability of the intermediate fertile layer, despite the relocation and freezing of steel materials in this region (welding effect).

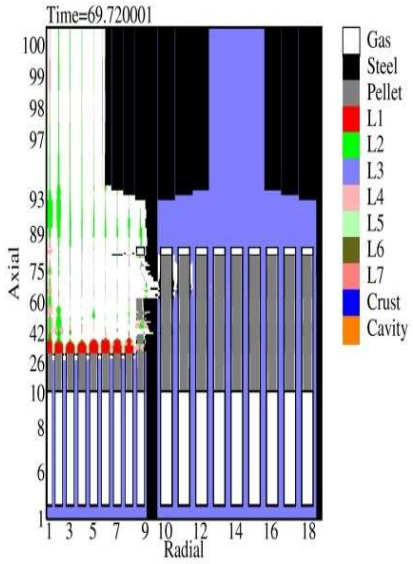


Fig. 3. SIMMER Calculation: Corium propagation under an unprotected local cooling fault situation

Contrary to homogenous core, ULOF transients with the CFV core do not show significant power excursion during the primary phase of the accident[8]. This is due to the negative sodium void worth of the CFV core, which leads to the decrease of the core reactivity with the decrease of the sodium density or the voiding of the upper part of the core up to 39s, with large development of the boiling zone in the external core fuel assemblies. Up to 50s, reactivity oscillations are observed due to hydraulic instabilities (voiding/reflooding) between fuel assemblies (Fig. 4). The power of the core during this period is in the range of 40-80% of the nominal power. After this time, when clads melt and relocate, the reactivity variations are still limited. Later, with fuel relocation, larger reactivity increase is observed due to fuel axial compaction, leading to a mild power excursion at 68s and to the beginning of the transition phase. One can conclude that, the core degradation during the primary phase of the accident is driven by thermal phenomena and not by neutron physics phenomena as in homogenous cores with positive sodium void worth; the associated time scale for CFV core degradation is then rather long.

With two mild power excursions at 68s and 71s, due to fuel degradation and compaction, the CRGT failure and then the DCS-M-TT failure occur, leading to large masses of corium discharges outside the fissile core region through these mitigation devices, and therefore to a large sub-criticality at 72s. At about 80s, around 50% of the core inventory has been relocated through the DCS-M-TT on the core catcher (Fig. 5).

One can notice that only mild fuel coolant interactions occur during this phase in the core region.

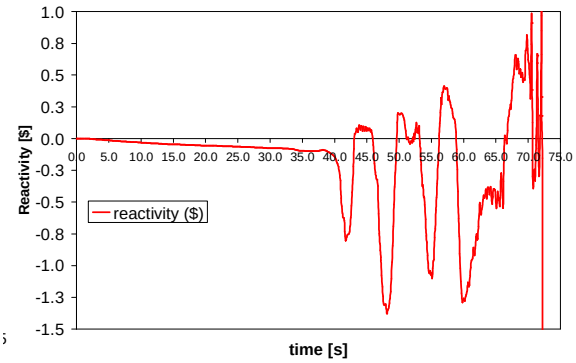


Fig. 4. SIMMER calculation: reactivity evolution

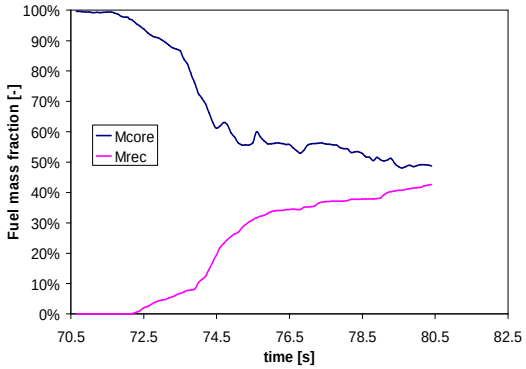


Fig. 5. SIMMER calculation: fuel discharge evolution

Regarding the reactivity insertion ramp situations (UTOP), one doesn't expect very different behavior of the CFV core compared to the homogenous core during the pin degradation phase of the accident. Furthermore no calculation tool is at present available to calculate the behavior of heterogeneous fuel pin in such conditions. Therefore only calculations with simplified models have been performed to estimate the mechanical energy releases during these transients, and the subsequent fuel dispersion under pressure forces when reaching its boiling point (calculation conservative assumptions). In these conditions, the mechanical energy releases in the CFV are lower than in a homogeneous core [9].

IV. NUMERICAL TOOLS AND CODE ASSESSMENT

For the ASTRID core behavior studies under severe accidents conditions, two types of simulation tools are used[10]:

- Mechanistic codes which will be chained or coupled using the SEASON platform to get best-estimate core behavior analysis
- A set of fast-running physico-statistical models which will be implemented in the PROCOR-Na platform, derived from the PROCOR[11] platform used for the LWR safety analyses. Such tools allow quick studies, or to study the scenario variabilities by running large number of calculations with different hypothesis (scenarios) or model parameter values (sensitivity calculations)

Assessment of the mechanistic codes is of prime importance to have a good confidence in the produced results. Physico-statistical models are benchmarked against the mechanistic code.

For mechanistic CFV core degradation analyses under unprotected loss of flow conditions, mainly the SIMMER-III and SIMMER-IV[12, 13] codes are used. They have been widely assessed against the SCARABEE-N tests devoted to the study of local flow blockages, degradation of the blocked fuel subassembly, and propagation of the degradation to the neighboring fuel assemblies. The SCARABEE-N Program[2] performed by the CEA/IPSN (France) with the support of EDF (France), AEA (UK), KfK (Germany), PNC (Japan) and JRC (European Community), was carried-out with homogeneous SuperPhenix-like fuel (with small diameter).

It was agreed with JAEA to enlarge the SIMMER capabilities to the calculations of the primary phase of the severe accident and to be able to calculate consequences of power excursion of fuel pins (including heterogeneous fuel pins). The new version will be named SIMMER-V[14]. Such capabilities of simulations for homogeneous oxide-fueled pins is devoted to SAS-SFR which was largely assessed against the CABRI programs[2] (CABRI-1, CABRI-2, CABRI-FAST, and CABRI RAFT) performed by the CEA/IPSN (France) with the support of PNC (Japan) and other partners depending on the program. In total, 54 CABRI tests have been performed with Phenix or SuperPhenix homogeneous fuel pins. It was then decided by JAEA and CEA to implement SAS-SFR[15]-like models, extensively assessed against CABRI program, in the SIMMER-V code.
Regarding the assessment of models involved in corium discharge through dedicated duct to extract corium from the core region for mitigating the consequences of severe accidents, SIMMER-III and SIMMER-IV were assessed against the GEYSER[16] experiments performed by CEA (Molten UO_2 or boiling steel released in a tube or in a pin bundle), and are presently assessed by JAEA with the support of CEA against EAGLE-1 and EAGLE-2 programs[3, 4] managed by JAEA and carried-out by NNC-RK. EAGLE project is aiming at acquiring data of core-material relocation to demonstrate effectiveness of the FAIDUS (Fuel Assembly with Inner-Duct Structure) for exclusion of recriticality issue by controlled core-material relocations.

Up to now, the studies of corium-sodium interaction for ASTRID are performed using simple models or using SIMMER with parametric models. Fuel-Coolant Interactions in case of sodium is quite different from fuel-water interaction. With Sodium, past experiments have shown that the FCI phenomenology evolves with the sodium temperature[17]. In subcooled sodium, it is difficult to obtain a stable vapor film around the corium particles, thermal effects lead to immediate heavy fragmentation of corium. Such a phenomenological feature cannot allow the premixing phase, which is governed by fluid dynamic instability, as with Water. When sodium heats up and its temperature reaches levels close to saturation, large masses of corium and sodium can be pre-mixed, pressure rise times and released mechanical energies are more akin to FCI with Water, which can lead to challenging situations. Therefore, to evaluate such situations, a mechanistic code dealing with the whole processes is needed. CEA started recently the development of a mechanistic code named SCONE[18] (Software for COrium Na interaction Evaluation) using the DUNE[19] project numerical library. SCONE will be used to perform reactor case calculations and to produce parametric simple models to be implemented in the SIMMER code.

V. SAIGA PROGRAM FEASIBILITY FOR CFV ASSESSMENT

In order to reduce uncertainties on SIMMER simulations on degradation of heterogeneous fuel pins and subassemblies of the CFV core in hypothetical accident situations with fuel pin degradation, CEA is studying with NNC-RK the feasibility of an in-pile experiment program SAIGA in the IGR reactor[4].
To take into account the generic accident families of loss of coolant situations, and power excursion situations in the SAIGA program, the feasibility study has been defined as follow:

- One test to study the behavior of a CFV heterogeneous pin sub-assembly under a loss of cooling situation: to catch the CFV design effects on such type of accident, a sub-assembly Total Instantaneous flow Blockage (TIB) was chosen, as studied in the SCARABEE-N Program.
- To study the behavior of a CFV heterogeneous pin under power excursion (TOP), 2 tests similar to CABRI tests have been chosen: one with a fast power pulse transient, and one with a mild pulse.

The results of the proposed SAIGA program would widen to heterogeneous fuel pin behavior, the CABRI and SCARABEE experimental database, programs performed with homogeneous fuel pins as used in Phenix or SuperPhenix reactors.

The objective of a SAIGA TIB test would be to assess the behavior of a 37 heterogeneous-pin subassembly, as in SCARABEE tests (lowering the side effect), under loss of cooling conditions, and to get data on:

- Core damage timing in small-spacing-wired pin subassembly,
- Material relocation in small-sodium channel subassembly (with possible material entrainment by sodium vapors),
- Plug formation in small-spacing-wired pin subassembly,
- Plug tightness and plug mechanical strength under pressure build-up in small-spacing-wired pin subassembly,
- Occurrence of fuel/sodium interactions and pressure transients
- Impact of pressure spikes on upper stainless-steel plug during FCI,
- Formation of a molten pool in the fissile zone in small-spacing-wired pin subassembly,
- Pressure build-up in the subassembly,
- Impact of the presence of 2 fissile zones and 2 molten pools in the CFV subassembly,
- Behavior of the fertile layer of the CFV heterogeneous pins.

The test would be stopped after the rupture of the hexagonal tube.

The objectives of the SAIGA TOP tests with heterogeneous pins would be to study:

- Plenum gas behavior;
- Fuel axial expansion versus Pellet-Cladding Mechanical Interactions (cavity pressure);
- Clad failure(s);
- Dry-out and cladding melting (if any);
- Broken or Molten clad motion;
- Effect of the fertile layer and of the 2 fissile zones.

The main requirements for SAIGA program will be:

- To be able to perform 3 tests before end of the basic design phase (2019)
- To avoid any international fuel transportation (the use of Kazakhstani Fuel is then required: BN350 and VVER fuel pellets are available)
- To use of austenitic cladding for TOP Tests
- To have highly instrumented tests
- To include the Post-Test Examinations in the SAIGA program.

The feasibility study for the 3 tests includes the predesign of the test section with the experimental instrumentation, the predesign of a sodium loop to feed the experimental device, neutronic calculations to estimate the power deposition in the experimental fuel using BN350 or VVER-1000 fuel (17% and 4.4% enrichments) and the coupling factor with the driver-core, calculation of the temperature field in the experimental test device, pretest calculations of the fuel degradation to determine the

experimental power deposition evolution, the core power diagram and associated control rod motion, licensing requirements, possible tests schedule (which will take into account the JAEA EAGLE-3 program schedule) and preliminary cost estimation.

Test devices are designed for BN350 fuel pins and VVER fuel pins (the VVER pin diameter being closer to the ASTRID one).

First important results showed:

- A highly instrumented test device can be designed to fit the experimental bundle with 37 heterogeneous pins (Fig. 6)

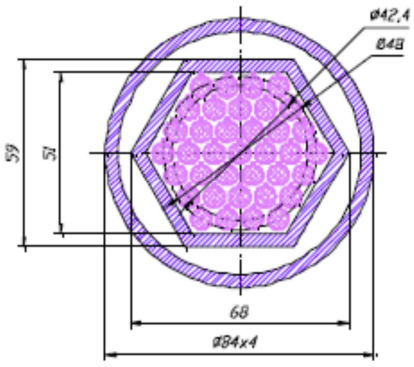


Fig. 6. SAIGA Test Device: Radial Cut

- A sodium loop can be implemented in IGR to feed the experimental test device (Fig. 7)

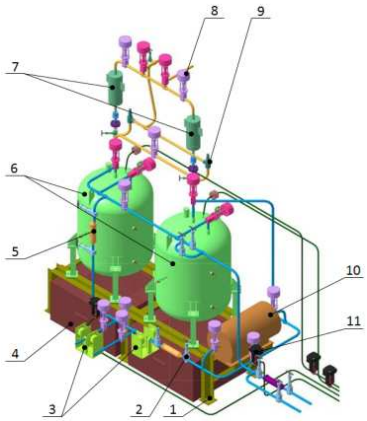


Fig. 7. SAIGA Sodium Loop

- the coupling factor with VVER fuel is not large enough to perform a TIB test with a large corium pool formation (power deposition in the IGR driver core is limited to 5.2GJ); however TIB tests can be performed with thinner BN350 fuel pins: in this case, at the end of the TIB transient, the IGR core deposition is in the range of 3.2GJ.
- TOP tests on BN350 fuel can be performed in IGR (NNC-RK performed calculations and start-up for the demonstration: Fig. 8)

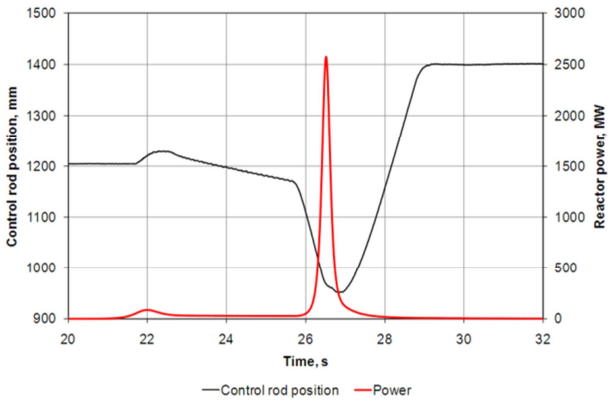


Fig. 8. TOP diagram with Pmax/P0=93

Since in IGR core, as in SCARABEE, the radial power deposition profile is not flat in the experimental pin bundle (with higher power deposition in the external fuel pins due to the thermal neutron flux of the driver core moderated with graphite) and, since the associated temperature profile is also not flat, two additional fuel pin arrangements and associated test devices have been studied to get flatter temperature profile in the experimental bundle (with higher sodium flow in the external ring or with peripheral fuel pins to harden the neutron flux in the experimental bundle).

Since no satisfactory pin arrangement was found with BN350 fuel to get a flat temperature profile in the experimental bundle, and consequently a pin degradation spreading from the inner part of the bundle toward the outer part as in an actual SFR fuel assembly, it was decided to study the feasibility to produce fuel pellets with different enrichments from the BN350 pellets for the TIB experimental bundle, and to run TOP test on a single pin. This study is underway at Ulba Metallurgical Plant and NNC-RK; first pellet batches with LWR diameter and various enrichments have been produced by UMP to demonstrate the feasibility of pellets production from BN350 pellets for TIB SAIGA tests.

With enrichments of fuel pins of 8.6%, 12.7% and 17% from the outer ring to the inner rings and same pellet diameter as BN350 (ϕ=5,9mm), a flat temperature profile can be obtained in the test bundle (Fig. 9).

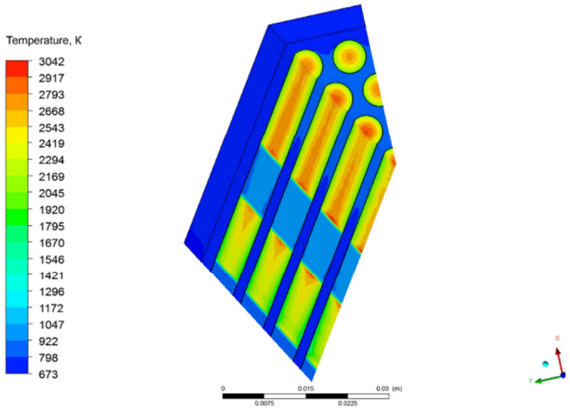


Fig. 9. SAIGA Temperature Field in the experimental bundle with different enriched fuel pins

In these conditions, the pool formation and then the onset of hexagonal tube failure are obtained 21s after the beginning of the TIB transient according to SIMMER calculations (Fig. 10).

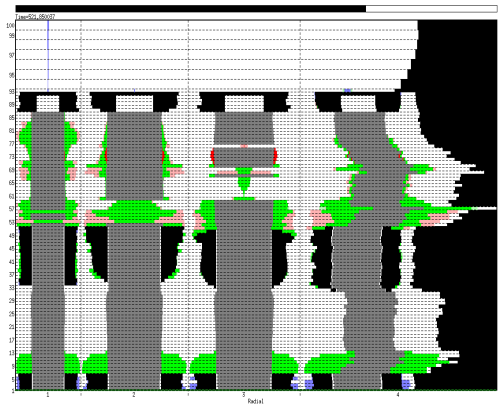


Fig. 10.SIMMER calculation of the TIB scenario with enriched fuel pins

Since 20 seconds are necessary to obtain the thermal steady state before initiating the TIB, the required transient time (41s) is achievable in IGR (limited to 52.5s to reach 5.2GJ in such conditions). Next step of the feasibility study is to check whether a TIB test could be performed with pellet having the same diameter as those in ASTRID (ϕ=8.45mm).

VI. PLINIUS-2 PROGRAM FOR CFV ASSESMENT

The EAGLE1&2 Programs gives experimental data to assess code models for corium transfer in 'ducts' similar to the CRGTs or the DCS-M-TTs and give information on fuel coolant interactions in such configuration. FCI is the driver-force of the corium discharge in the 'ducts' but also may restrain the corium discharge and also jeopardize the core structure (including the discharge tube).

Since the EAGLE1&2 in-pile test number was limited, used duct design different than the ASTRID DCS-M-TT (length, diameter, …), and the out-of-pile tests were performed with corium simulant (Al_2O_3), to reduce the ASTRID calculation uncertainties and strengthen the demonstration of the DCS-M-TT efficiency, CEA studies the possibility to build a new facility (PLINIUS-2), able to cope with large masses of prototypic corium at high temperature in sodium environment. CEA will benefit from its experience gained by operating the PLINIUS facility and running corium experiments at high temperatures.

The objective of the PLINIUS-2 facility for SFRs will be:

- To study corium melt relocation through dedicated tubes and to validate the DCS-M-TT concept in sodium environment,
- To extend the knowledge on fuel coolant interaction in sodium environment,
- To study corium/core catcher sacrificial material interactions and qualify the core catcher design for ASTRID.

The first 2 issues concern the CFV Core Assessment and the mechanistic codes SIMMER and SCONE used to calculated its behavior in Severe Accident conditions. It must be pointed out that PLINIUS-2 will be also used for LWR R&D needs.

The PLINIUS-2 building[20, 21] will be composed of 3 corium experimental halls in the basement (Fig. 11):

- A corium-material interaction and mitigation hall,
- A corium-sodium interaction hall,
- A corium-water interaction hall.

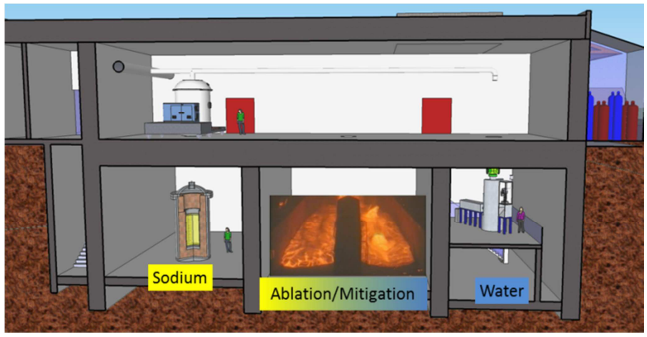


Fig. 11. PLINIUS-2 Building

A furnace with a large corium capacity (about 500 kg) using a cold crucible and an induction technology will be available for the 3 halls. A preheating of the corium by thermite reaction will be possible. Corium will be transferred through dedicated transfer devices to the test sections located in the experimental halls.

A test section is foreseen to carry-out experiments in which the corium is delivered below the sodium free surface though corium duct. The design of this test section is

derived from the one used for EAGLE out-of-pile experiments used to study the JSFR corium ducts with simulant and smaller scale. This PLINIUS-2 Test Section will be used to qualify the ASTRID DCS-M-TT device.

The design of test sections for corium-sodium interaction studies reproduces the major outlines of the FARO-TERMOS[22] test section (Fig. 12).

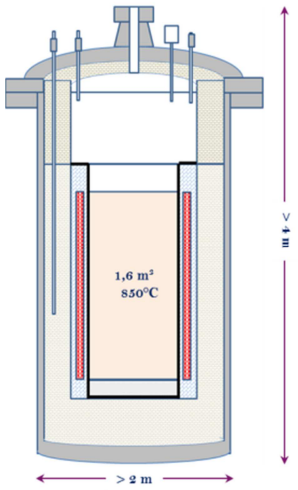


Fig. 12. Sketch of the PLINIUS-2 FCI test section

It must be pointed out that X-Ray device could be used in each experimental hall, with energy reaching 25MeV.

A 4th hall will be available for separate effect tests; Tests will be performed with smaller masses of corium (starting from 5g), and with simulants.

The PLINUS-2 facility will be available for experiments in 2020, for confirmation of the ASTRID design choices.
PLINIUS-2 programs are right now under definition to support the ASTRID safety demonstration and qualify the design of the ASTRID mitigation devices (DCS-M-TT and core catcher). For issues which concern the CFV core, the programs will be:

- The PLINIUS-2 TR program (TR standing for 'Transfer') dedicated to the DCS-M-TT qualification and the assessment of the related SIMMER models; the opening of the corium tube will be studied as well as the corium transfer in the corium tube. These tests will enlarge the GEYSER database and EAGLE database with prototypic corium and large corium masses (up to about 500kg), different corium compositions (Oxide uranium-iron-B_4C), with representative geometry of the DCS-M-TT in sodium environment,
- The PLINIUS-2 DROP program dedicated to the study of hot corium droplet fragmentation for SCONE assessment; the program is composed of Analytical Tests with X-Ray instrumentation to track the droplet fragmentation
- The PLINIUS-2 FR program (FR standing for 'Fragmentation') dedicated to corium jet fragmentation at the outlet of the DCS-M-TT to enlarge the MELT[23] and SOFI[24] database; X-Ray instrumentation will be used to track the fragmentation. The impact of sodium temperature, corium velocity and DCS-M-TT diameter will be studied.
- The PLINIUS-2 EXPLO program (EXPLO standing for 'Explosion') to improve the knowledge on the sodium vapor explosion and on the mechanical dynamic loads of structure in the vicinity of the explosion. Large masses of corium will be used with models of the DCS-M-TT. The impact of sodium temperature, corium velocity and DCS-M-TT diameter will be studied.

For information, programs dedicated to the corium jet impact on the sacrificial material of the core catcher (PLINIUS-2 IMPACT), and on the corium-sacrificial materials interactions (PLINIUS-2 CERAM), for core catcher and models qualification will be also carried-out in PLINIUS-2.

VII. CONCLUSIONS

To support the development of the ASTRID demonstrator and the innovative CFV core, a large R&D program has been set up by the CEA. This program has 4 folds:

- Study of severe accident scenarios in ASTRID
- Definition of mitigation devices for mitigating the severe accident consequences
- Developments and assessment of calculation tools for supporting the ASTRID design option choices and for a robust safety demonstration in the field of severe accidents
- Definition of experimental programs to complete the existing experimental database for qualification of ASTRID options, for new model assessment for decreasing the calculation uncertainties.

This R&D program is set up according the safety approach defined for ASTRID, and taking into account the safety objectives and the ASTRID design options. This R&D program is defined and carried out with the ASTRID industrial partners (mainly AREVA, EDF, and MFBR) and with R&D foreign organizations (mainly JAEA, KIT, NNC-RK and IGCAR).

In particular, for reducing the SIMMER calculations uncertainties on the heterogeneous fuel pin degradation under severe accident conditions, a feasibility study of the SAIGA program in IGR is ongoing; it is carried out by CEA and NNC-RK, with the technical support of JAEA to

benefit from the feedback of EAGLE1&2 programs in IGR.

Furthermore, to enlarge the experimental database, the PLINIUS-2 facility is under study at CEA to assess:

- The SIMMER-V models of corium discharge through the dedicated corium transfer tube implemented in the CFV core,
- The SCONE models of corium-sodium interaction in the core, in the transfer tubes and out of the transfer tube in the reactor cold plenum,
- To qualify the corium transfer tube,

Preliminary PLINIUS-2 programs meeting above objectives are proposed.

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