The SAIGA experimental program to support the ASTRID Core Assessment in Severe Accident Conditions

F. Payot, F. Serre, A. Bassi and C. Suteau

Commissariat à l'Energie Atomique et aux Energies Alternatives

CEA Cadarache - CEA/DEN/CAD/DER Blg 1222, 13108 St Paul lez Durance - FRANCE

E. Batyrbekov, A. Vurim, A. Pakhnits, V. Vityuk

National Nuclear Center of the Republic of Kazakhstan, Krasnoarmeyskaya St., 071100, Kurchatov, VKO, KAZAKHSTAN

E-mail contact of main author: frederic.payot@cea.fr

Abstract. The CEA, together with the NNC, has carried out a feasibility study with regard to conducting an in-pile test program - the future SAIGA program (Severe Accident In-pile experiments for Gen-IV reactors and the Astrid prototype) - on the degradation of an ASTRID-like fuel in the IGR reactor (Impulse Graphite Reactor operated by NNC-RK). The purpose of the SAIGA program is to qualify the SIMMER computer code on the SEASON platform based on tests conducted with axially heterogeneous CFV type ASTRID inner core pins or pin bundles in hypothetical severe accident situations. These tests should be representative, as much as possible, for the phenomena encountered during severe accident sequences considered for ASTRID. The feasibility study aimed to study the generic accident families of loss of coolant and power excursion situations. It is important to point out that the fuel used for these tests can only be a non-irradiated fuel.

The feasibility study focused on tests based on the degradation of one or more fuel pins during Total Instantaneous Blockage (TIB) sequences in a sub-assembly and power excursion (Transient OverPower: TOP) sequences as in SCARABEE and CABRI with homogeneous pins.

For both scenarios, the feasibility study defined the main characteristics of the experimental devices and the operating conditions for the tests to be conducted in the IGR reactor. The purpose of the studies was to assess the capacity of the IGR reactor to provide the necessary neutron flux during all the transients, to demonstrate the capacity to carry out on-line or post-test measurements of the variables of interest, and to assess the cost and schedule for a program of 3 tests incorporating the safety file. Also, the sodium loop feeding the test device and its instrumentation were studied and their feasibility demonstrated.

Key Words: ASTRID, SAIGA, Safety, severe accident

1. Introduction

The ASTRID reactor (Advanced Sodium Technological Reactor for Industrial Demonstration) is a technological integration prototypic of 4^{th} generation SFRs with a sufficient industrial scale to demonstrate the innovative options designed by the CEA with its industrial partners with very high safety requirements. The ASTRID safety level will be equivalent to that of a 3rd generation PWR with the specifications coming out of the lessons learnt from Fukushima accident [1]. To reach these objectives, CEA designed a new innovative heterogeneous core with a low sodium void effect named CFV. A description of this new core is presented in references [2][3]. The ASTRID core design is focused essentially on the low reserve of reactivity in the core and the sodium void effect minimized. For lowering the sodium void worth, the sodium concentration in the core was lowered by using large-diameter fuel pins associated with small-diameter spacer wire (leading to small sodium channels between the pins). To more decrease the sodium void worth, the following items were added: a sodium plenum zone at the top of the fuel assembly with a neutron absorber material zone, a fertile zone near the mid plane of the fuel pins of the inner core, and longer outer fuel pins compared to inner pins (the core has a diabolo shape). The CFV core is a two zone core composed of an inner core with axially heterogeneous subassemblies and an outer core with axially homogeneous subassemblies.

Although complementary safety prevention devices are foreseen in the innovative core, the severe accident sequences were considered by implementing dedicated mitigation devices (e.g. DCS-M-TT) and by benefiting core design elements (e.g. CFV core and CRGT). So, a special attention is given to the possible fuel melting occurrence and subsequent events. In case of severe accidents (design extension conditions), the study and evaluation of the fuel behaviour and associated consequences are needed in view of the establishment of appropriate measures for management of the accident, mitigation and minimization of its consequences [4].

The safety issue on core reactivity of the SFR reactors have to been widely studied in the past in the unprotected accident situations due to a large positive sodium void worth at the SFR core. The CABRI and SCARABEE programs, conducted by the IRSN (former IPSN), provide a large experimental database on the degradation of pins and bundle of pins during a Core Disruptive Accident of Sodium Cooled Fast Neutron Reactor. Those programs include various transients such as fast, medium and slow Transient Over Power (TOP), Total Instantaneous flow Blockage (TIB) and Unprotected Loss Of Flow (ULOF), at nominal power. With the new innovative heterogeneous ASTRID core based on a low sodium void effect, it will be necessary to enlarge the available experimental data base in order to reduce the uncertainties in possible Severe Accident conditions in the CFV core. Those characteristics can lead to different pin and bundle degradation sequences under severe accident conditions; therefore new in-pile degradation experiments would reduce the uncertainties on core degradation calculations and will extend the experimental data base for code assessments (SIMMER [5] and SAS-SFR).

The CEA, together with the NNC-RK, has carried out a feasibility study with regard to conducting an in-pile test program - the SAIGA program - on the degradation of an ASTRID-like fuel in the IGR reactor. The demonstration of the experimental feasibility in the IGR reactor was studied on the degradation of one or more fuel pins during total instantaneous blockage (TIB) loss-of-coolant sequences in a sub-assembly and power excursion (Transient OverPower: TOP) sequences as in SCARABEE and CABRI with homogeneous pins. The purpose of the SAIGA program is to perform experimental in-pile tests with power transient conditions over ASTRID-like fuel which must be representative of the phenomena encountered during severe accident sequences considered for ASTRID. Uncertainties exist in modeling accidents of reactor ASTRID. Experiments at the IGR reactor will reduce these uncertainties. For this purpose, instrumentation will be introduced in the experimental test device to measure physical data in agreement with the SAIGA specifications and the validation of the calculation codes. The data obtained from the Uranium dioxide fuel in the SAIGA test will be used as input data to validate the SIMMER and SAS-SFR calculation code. Then, the ASTRID reactor calculations will be made with the uranium-plutonium dioxide fuel.

The development of the SIMMER calculation code is in progress. The SIMMER code takes into account energy balance during the melting and freezing of fuel. However, the detailed chemical transformations of fuel are not described in the code. Now, all the chemical interaction between fuel, steel and sodium are out of scope of the SIMMER validation. This task remains for the future.

For the first two in-pile experimental scenarios mentioned above, i.e. TIB and TOP, the feasibility study defined the main characteristics of the experimental devices and the operating conditions to be conducted in the IGR reactor. The purpose of the feasibility study carried out was to assess the capacity of the IGR reactor to provide the necessary neutron flux during all the transients (as IGR core is not cooled, the energy deposited in the driver core during the start-up is limited), to demonstrate the capacity to carry out on-line or post-test measurements of the variables of interest, and to assess the cost and schedule for a program of three tests incorporating the safety file and the post-test examinations. The studies have to demonstrate that the required transients could be created for a sufficient period of time. To do this, the neutron and thermal behaviour of the test device were studied, as was the definition of the IGR reactor core power diagram. The sodium loop feeding the test device and its instrumentation were also studied and their feasibility demonstrated.

The expected results of the SAIGA feasibility study aim to conduct in-pile tests reproducing phenomena similar to those expected in ASTRID mitigation conditions and not full scenarios. A program of three tests has been defined as follow:

- One experimental test to study the behaviour 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,

- Two experimental tests to study the behaviour of a CFV heterogeneous pin under power excursion i.e. Transient OverPower (TOP)) test. For these tests, mild pulses have been studied, as close as possible to CABRI tests.

For both scenarios, a series of tasks have been defined as described after.

2. IGR reactor description

IGR reactor is research Impulse Graphite Reactor on thermal neutrons with homogeneous uranium-graphite core. IGR reactor is a self-quenching reactor by a principle of shutdown of any impulse. Two modes of reactor operation can be used i.e.:

- A mode of quenching neutron flash based on a reactivity insertion into the reactor which exceeds a fraction of delayed neutrons. The flash quenching is due to negative temperature reactivity effect;
- A controlled mode performed by means of control rods compensating negative temperature reactivity effect on specified law. This mode will be used for the TIB and TOP tests.

A central experimental channel is installed along the vertical axis of the core. The SAIGA test device is designed to be located in this experimental central channel.

It is worth nothing that, because of the IGR core reactor characteristics, the maximum energy released in the core must not exceed 5.2GJ corresponding to a temperature of 1373 K which limits the reactor operating capacity.

3. TIB test

The TIB test of the SAIGA program aims to assess both the degradation events of a 37 heterogeneous-pin sub-assembly under loss of cooling conditions and get data on:

- The course of events from the loss of flow to hexcan failure. Core damage timing should be identified together with the formation of a molten pool (or two) in the fissile zone (s) in large-diameter small-spacing-wire pin subassembly ;
- The influence of the internal fertile layer and the narrow Na channels in the bundle on the cladding and fuel material relocation.

Design of the in-pile device for the TIB scenario

The TIB scenario is divided into two phases. The first phase is a period of time for reaching thermal equilibrium conditions, as described after. The second phase corresponds to the degradation transient after stopping the sodium flow rate in the pin bundle. To achieve this TIB scenario, the experimental test device (see figure 1) is consisted of:

- A hexagonal container with 37 heterogeneous fuel pins fabricated from Kazakhstani pellets, diameter 6.9 mm (outer diameter of cladding). The two fissile areas consist of solid annular pellets and the fertile area at the centre of the pins is made up of solid full pellets of depleted uranium;
- A 1 mm diameter spacing wire, wound round each pin with a 180 mm pitch;
- Sodium circulating in the hexagonal wrapper. The sodium flow conditions must be representative of the nominal conditions in ASTRID, before the blockage occurs.

The possibility to design a Sodium Loop Circuit has been considered (see figure hereafter) within the framework of studies in order to conduct in-pile experiments at the IGR reactor. The Sodium Loop Circuit will have to ensure the circulation of heated liquid sodium through experimental device installed in the central experimental channel of IGR reactor.

A Kazakhstani Fuel coming from the BN350 reactor (named BN350 fuel in this paper) would be used to avoid any international fuel transportation. So, the fissile fuel pellets should be elaborated from the BN-350 pellets which present a ^{235}U enrichment rate equal to 17%. The outer diameter of fuel pellets is equal to 5.9 mm with a central hole equal to 1.5 mm.

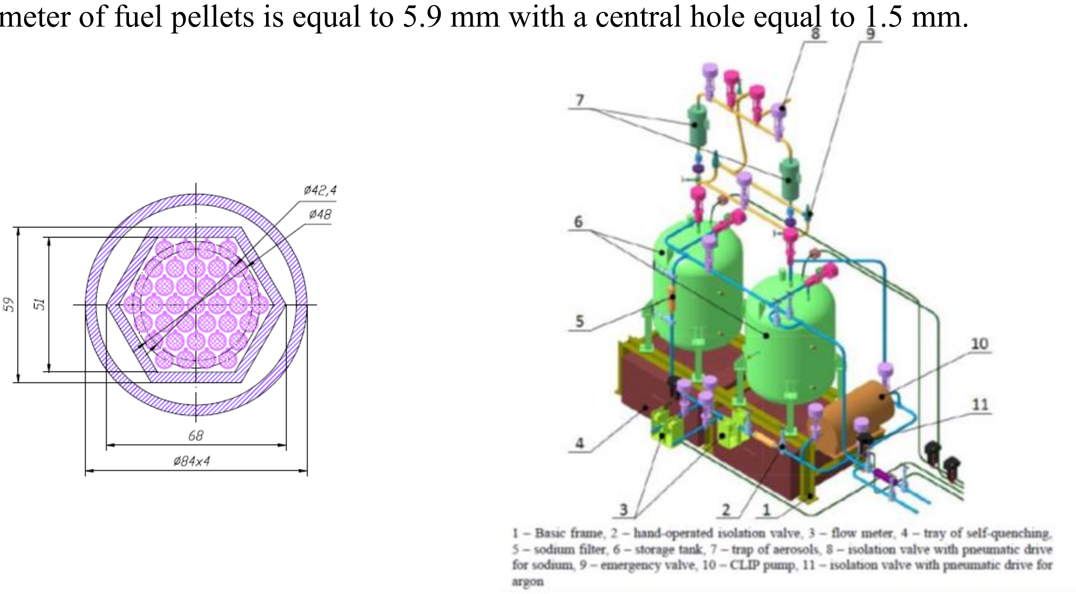


Figure 1: illustration of the bundle test section (on the left) and the sodium loop circuit (on the right)

With only one ^{235}U enrichment rate (i.e. 17%) in the fissile zones inside the bundle, the first neutronic and thermal results show that the radial power distribution between the central pin and the peripheral ring in the 37-pins bundle is relatively « hollowed ». To reach the radial flat power distribution, it was decided to use fuel pellets with variable ^{235}U enrichments in the two fissile zones in order to be further representative for the SFR degradation behaviour [3]. Few fuel pellets (with 7.8 diameter) were elaborated, using the standard BN350 fuel with a ^{235}U enrichment level of 17% and UO_2 powder with a ^{235}U enrichment level 4.45%, to ensure the manufacturing process feasibility. A bundle with variable enrichment had already been used for some tests (TIB type tests) in the SCARABEE program. Expected results of this SAIGA study aimed to demonstrate the feasibility for the fuel pellet production with variable ^{235}U enrichments. The pellet pilot batches were produced and were provided by UMP Kazakh JSCwith various ^{235}U enrichments lower than 17%, as determined by the neutronic study.

From the results of neutron calculations (presented in the following paragraph), the ^{235}U enrichment rates in the bundle pin rings (with a BN-350 fuel pellet diameter equal to 5.9mm) were determined for each pin ring i.e. 17% for the central pin and the first ring, 12.7% for the second ring and 8.6% for the peripheral ring.

Neutronic study for the TIB scenario

Neutronic study was based on numerical simulations of IGR core with the experimental device using MCNP5 with the neutron constants ENDF/B-VI library.

From this calculation tool, total power deposited in the experimental device was evaluated to be 522 kW which reproduces the same average sodium heating in the experimental bundle as in ASTRID. The axial power distribution along the bundle was studied to be roughly similar, on average, to that of the ASTRID power distribution i.e. ~90 W/g(UO_2) for the upper fissile layer, ~74 W/g(UO_2) for the lower fissile layer and a very small amount for the central fertile zone. Besides, the power of the IGR reactor was estimated. From the maximum energy release of the IGR reactor (i.e. 5.2GJ), the maximum operating time at the estimated power level of the IGR reactor was evaluated.

Figure hereafter shows the calculation result of radial power distribution in some pins of the bundle which was radially divided into 11 parts so that each part area is equal to other. The radial power distribution between the center of the pellet and the peripheral zone of the pellet is lower than 11%. This value is quite acceptable to be representative for a SFR accident scenario.

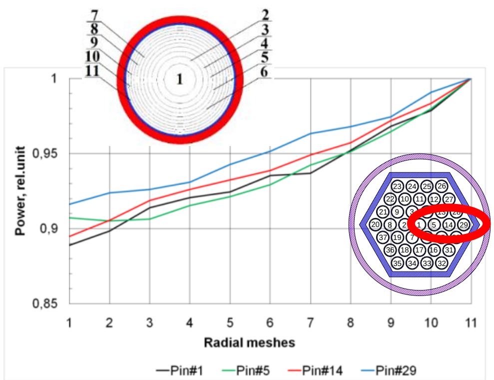


Figure 2: Radial power distributions in some pins (on the right)

Thermal study for the TIB scenario

Thermal study was carried out using a 3-D mathematical model. The calculation results aim to determine both the nuclear heating time required to reach the thermal steady-state conditions and the temperature field in the bundle.

Taking into account the predetermined energy release in the pin bundle and required sodium temperatures (T_{inlet}=673K and T_{outlet}=823K similar to mean temperatures at the entrance and at the exit of the ASTRID reactor, respectively), the sodium flow rate throughout the bundle was evaluated.

The calculation results show that the maximum temperature inside the pins under the specified energy parameters does not exceed ~2850 K when the steady state phase is reached (see figure below). The sodium temperature in-between pins reaches ~820 K at the top of the bundle during the steady state phase.

Beside, time to reach the steady-state conditions in the pin bundle before stopping the sodium flow was calculated to be ~20 seconds, as displayed in figure 3.

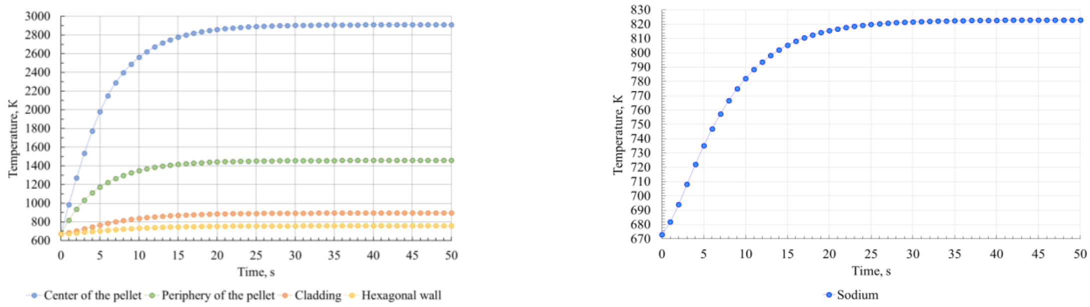


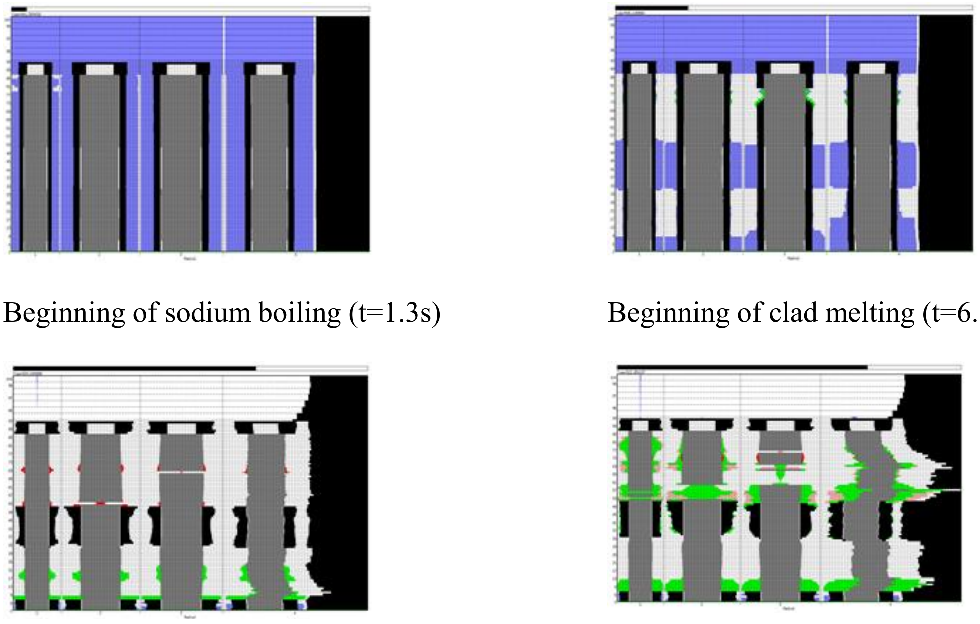
Figure 3 : Variation in temperatures of fuel, cladding, Hexagonal tube (on the left) and sodium (on the right) at the upper part of the bundle to reach the steady state phase

Study of the bundle degradation for the TIB scenario

The SIMMER III calculation code was used both to simulate the bundle degradation behaviour and to determine the duration of the transient at nominal power. Figure 4 shows four illustrations giving the key events during the TIB degradation transient i.e.

- The first Na boiling from the central pin in the upper fissile part before total dry-out in the fissile parts at 1.3s after stopping of the sodium flow rate;
- The first melting of the cladding and fuel in the upper fissile part at 6.15s;
- A degradation which spreads from the 3rd ring followed quickly by 2nd and 1st together and finally the 4th ring;
- A hexagonal can wall failure at 21.85s when a partial boiling pool is created in the upper fissile part.

It can be seen that, due to small sodium channels between the pins, large masses of molten clad is swept along the pins by sodium vapors toward the top and the bottom of the pins. When arriving at the level of the coldest pin areas, the stainless steel freezes. One of main objectives of this SAIGA TIB test is to observe the molten steel relocalization. As a consequence for the reactor case, the refrozen steel deposition can impacted the phenomenology of the corium propagation towards the neighboring sub-assemblies.



Beginning of sodium boiling (t=1.3s) Beginning of clad melting (t=6.15s : Ring 3)

Beginning of fuel melting (t=20.55s : Ring 3) Can wall breakup (t=21.85s)

Figure 4: Illustrations of the pin bundle degradation with a flat radial power distribution for the TIB scenario by using the SIMMER III code

As a conclusion, these calculation results obtained from this test device with a variable enrichment rate shows that the TIB degradation phenomenology can be considered as representative for the SFR behaviour.

Also, the full degradation sequence (steady state and TIB transient phases) was evaluated to be ~41s i.e. ~20s for the steady state phase (from the thermal study) and ~21s for the transient phase (from the present SIMMER calculations). From the maximum operating time of the IGR reactor determined earlier, it can be deduced that the IGR reactor has enough capacity to produce TIB sequence.

4. TOP test

The goal of the experiments is to evaluate the degradation of a single pin with emphasis on the influence of the internal fertile layer during significant power increase.

Design of the in-pile device for the TOP scenario

As the TIB scenario, two phases have to be distinguished i.e. period of time for reaching thermal equilibrium conditions followed by the degradation transient under the effect of the power pulse. The course of the TOP scenario was established for a test device consisting of:

- One pin (fabricated with BN350 pellets and solid fertile pellets). At the extremities of the CFV fuel pin, upper and lower plenums are presents which will be representative for the ASTRID pin design. The outer diameter of the pins studied was 6.9 mm with a U^{235} enrichment level of 17%;
- Circulation of sodium around the pin (with a hydraulic diameter of 14 mm).

The neutronic and thermal studies for the TOP scenario

As previously for the TIB scenario, the neutronic study was performed by means of MCNP5 tool with the neutron constants ENDF/B-VI library. This study was established to evaluate the neutronic data during the steady state phase before the power pulse.

The axial power distribution (in W/g) along the pin during the steady-state phase was studied to be near to that of ASTRID, as for the TIB SAIGA test. For that, the total power deposited in the single pin experimental device was evaluated to be 14 kW. With the respect of the coupling factor between the test device and the IGR reactor, the power of the IGR reactor was estimated to be 28.8 MW.

Despite the thermal flux in IGR, the in-pin radial power distribution is relatively flat i.e. ~22% of difference between the center and external of the pin. This result is consistent with that presented in figure 2. Therefore, no pre-pulse is required, unlike CABRI tests (where a niobium tube was used for the test device to get high coupling factor between the driver-core and the experimental pin), to heat-up the center of the pellets before the performance of the main pulse.

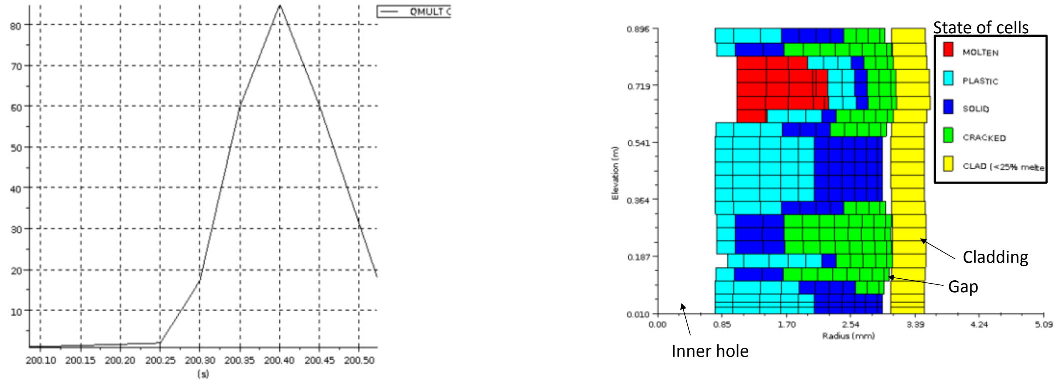
With these neutronic heating conditions, the sodium flow rate was estimated to reach the difference of 150°C between the entrance and the exist of the single pin.,

Thermal study during steady state phase was carried out as previously, the maximum fuel temperature in test section reaches 2744 K in the upper fissile zone. Besides, the sodium temperature at the outlet varies from ~815 K in the near wall zone of sodium channel to ~ 840 K in the fuel-cladding gap.

An important result is that the thermal steady-state conditions are reached after ~20s. This result is consistent with the test feasibility of the TOP scenario.

Study of the bundle degradation for the TOP scenario

SAS-SFR calculation code was used to simulate the pin degradation behaviour. The pulse shape (P_{max}/P_0=85 with a total duration=300 ms and a width at $P_{max}/2$=150 ms) is shown in Figure 6 together with an illustration showing the calculation results at the end of the pulse (i.e. 270 ms over 300 ms) during the TOP degradation transient. From these calculation results, it can be seen the pin failure at t=270 ms and the fuel melting at the upper part of the fissile zone. This calculation shows that the fuel melting starts from inner cell of the pellets towards outer cells. Also, the cladding fails, under thermohydraulic and neutron conditions similar to those in ASTRID. As a conclusion, on the basis of a rather flat in-pin radial power distribution, the TOP phenomenology simulated by the SAS-SFR calculation seems to be representative for SFR behaviour.



a) The pulse shape b) Results at the end of the pulse

Figure 5 : The pulse shape and SAS-SFR calculation results

Results of diagram neutronics study for TOP scenario

The IGR core power diagram and the rod position were determined for a single pin test design (see figure hereafter). For this exercise, the power plateau is reached after the beginning of the IGR operation (~12 s) and after ~40 more seconds, during the steady state power, the overpower is initiated. The IGR reactor power increases from 28.8MW to reach about 2400MW. The P_{max}/P_0 is about 85 with a width at $P_{max}/2$ ~ 150 ms. It is worth noting that the maximum IGR temperature is evaluated to be 720K in the IGR core which is lower than the safety temperature limit i.e. 1373K.

Finally, in order to show the feasibility of such a scenario in the IGR reactor, a start-up was carried-out with empty central experimental channel which is displayed in figure below.

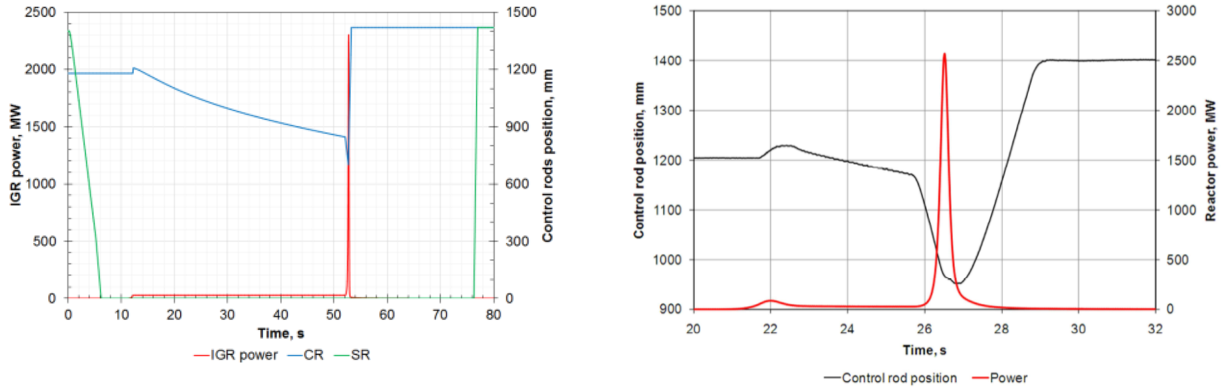


Figure 6 : IGR Start-up calculation for TOP scenario with a single pin (on the left) : Diagram performed in the IGR reactor with pulse P_{max}/P_0~93 (on the right)

For this TOP scenario, the feasibility of the required power transients was demonstrated by performing start-ups in the IGR reactor, and the results of SAS-SFR calculations show that the type of pulse generated by the reactor after establishment of thermal equilibrium ensures degradation conditions similar to those in ASTRID.

5. CONCLUSIONS

A preliminary SAIGA program has been proposed with TIB and TOP tests. The feasibility of a ‘loss of cooling type’ test and of ‘power excursion’ tests was demonstrated to study the degradation behaviour of a heterogeneous fuel bundle in ASTRID severe accident conditions.

For both first two in-pile experimental test scenarios TOP and TIB, the feasibility was demonstrated, as the reactor is capable of producing the required neutron flux for a period higher than the total duration of the test (i.e. establishment of thermal equilibrium and degradation transient) and for thermohydraulic and neutron conditions similar to those in ASTRID. The feasibility study defined the main characteristics of the experimental devices and the operating conditions to be conducted in the IGR reactor.

Discussions are in progress between CEA and NNC-RK to define, from these feasibility study results, three in-pile experimental tests SAIGA to perform in IGR in conditions close to those expected in ASTRID. These test definitions could be as followed:

- Test 1 : a type-TOP scenario with type-ASTRID 7-pin bundle undergoing a power excursion. The main phenomenology to study concerns the ejection and relocation of fuel in a narrow hydraulic channel (CFV type) with a heterogeneous fuel during a power excursion.
- Test 2 : a loss of flow on a CFV-type fuel sub-assembly at ASTRID nominal power
- Test 3 : a corium propagation outside the ASTRID sub-assembly with a corium discharge area. This latter test is considered as optional service according to the feasibility study carried at the beginning of the program.

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