# COMPUTATIONAL THERMAL HYDRAULIC SCHEMES FOR SFR TRANSIENT STUDIES

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## ABSTRACT

Design and safety studies for Sodium cooled Fast Reactors (SFR) include thermal-hydraulic transient calculations. A wide range of transients must be studied, protected or unprotected, symmetrical or dissymmetrical. Calculations concern the core and primary circuit with or without secondary circuits and with or without connected circuits such as the decay heat removal systems.

The codes used by the CEA for these studies are presented: CATHARE (system code), TRIO_U (Computational Fluid Dynamics code), TRIO_U MC (core sub-assembly code) and TRIO_U MC2 (core sub-assembly code coupled with a 3D model for the inter-wrapper space and the upper plenum).

This paper focuses on core modeling in CATHARE and the use of codes separately, chained or coupled.

The modeling and calculation schemes to be implemented are determined based on the studied physical phenomena, the functionalities of the various codes used for these calculations and the parameters of interest in the different parts of the reactor. They also depend upon the phase of the project (compromise between the required accuracy and the simulation time).

The selected calculation schemes for the following transients are discussed:

- **Loss of supply station power** resulting in the coast down of the primary and secondary coolant pumps and steam generators dry out, without reactor scram: in this case, different schemes can be implemented, that can resort to the coupling of CATHARE with TRIO_U or TRIO_U MC2.
- **Unprotected loss of flow** due to the coast down of all the primary pumps without reactor scram while the secondary coolant pumps remain operational for power removal: the computational scheme is based on CATHARE calculations completed by post-processing with TRIO_U MC.
- **Unprotected Loss of Heat Sinks** due to the coast down of the secondary pumps and steam generator dry out, without reactor scram: the computational scheme is based on CATHARE calculations completed by post-processing with TRIO_U MC.
- **Failure of a pump – grid plate connection pipe** with and without reactor scram: system computation is done with the CATHARE code. It provides boundary conditions for a CFD isothermal hydraulic computation of the grid plate. The values of local clads temperatures are obtained by TRIO_U MC calculations (unsteady version).

Furthermore, the subjects of code validation and uncertainties evaluation are touched upon.

**KEYWORDS**

SFR, transients, thermal hydraulics, computational schemes, CATHARE, TRIO_U / MC

# 1. INTRODUCTION

Sodium cooled Fast Reactors (SFRs) have been developed in France for nearly 50 years. These include first the Rapsodie reactor, then Phenix and finally the Superphenix plant. Nowadays, the so-called ASTRID (Advanced Sodium Technological Reactor for Industrial Demonstration) prototype is undergoing study in France in the framework of the Generation IV deployment. Thermal-hydraulics is recognized as a key scientific subject in the development of SFR. This paper deals with computational thermal hydraulic schemes implemented for transient calculations required to support the design and the safety analysis of SFRs.
The calculation schemes depend upon the phase of the project and the objective of the studies at the moment: selection of design options for the core or the reactor, optimization of a design, safety studies, margins evaluation with probabilistic methods…

Some selected calculation schemes for the following transients are discussed:

- Loss of supply station power resulting in the coast down of the primary and secondary coolant pumps and steam generator dry out, without reactor scram;
- Unprotected loss of flow due to the coast down of all the primary pumps without reactor scram while the secondary coolant pumps remain operational for power removal;
- Unprotected Loss of Heat Sinks due to the coast down of the secondary pumps and steam generator dry out, without reactor scram;
- Failure of a pump – grid plate connection pipe with and without reactor scram.

The objectives of the paper are to:

- present the CEA tools used for these calculations, these tools covering different scales (section 2.) ; important developments of these codes have been realized for the on-going studies at CEA specially in the framework of ASTRID project,
- enlighten some specificities of SFR's core modeling with the CATHARE code since the studied transients are mainly unprotected transients (section 3.),
- discuss the calculations schemes appealing to the various codes (section 4).

The paper aim is to show the approach progressiveness to take into account the constraints of a project. Indeed, even if the current means of calculation allow complex calculations, calculation time can be very long and therefore incompatible with a systematic implementation in phases of a project where design options evolve or with long simulation times.
We can even resort to very simplified calculations in preliminary design phases (example in appendix A).
The implementation of calculation schemes for safety studies cannot be separated from the validation approach for the codes in their domain of use (section 5).
Finally the subject of uncertainties evaluation with probabilistic methods in transients calculations is touched upon in section 6.

# 2. CEA SIMULATION TOOLS FOR SFR THERMAL-HYDRAULIC SAFETY STUDIES

The wide range of physical phenomena that may influence the progress of an accidental transient in a SFR, leads to the proposal of a "multi-scale" approach. Thus, three codes are used in the study of the primary circuit behavior during accidental transients:

- CATHARE to model on a global scale, called "system", all the components of the primary system (core, plena, pumps, exchangers) and loops (secondary circuit, decay heat removal systems …),
- The TRIO_U, CFD code (Computational Fluid Dynamics) that allows researchers to represent the refined dynamics of the sodium flows, for example in large plena;
- TRIO_U MC that models the thermal hydraulics in the core on a "sub-channel" scale. When coupled with a TRIO_U modeling of the inter-wrapper space and of the hot plenum, it is referred as TRIO_U MC2.

According to the transient, these three codes can be used:

- chained if the local effects (calculated by TRIO_U or TRIO_U MC2) do not affect the overall dynamics of the transient (calculated by CATHARE). Then it makes sense to use the results of a standalone CATHARE calculation as boundary conditions for TRIO_U or TRIO_U MC2 calculations;
- coupled if we expect feedback from local effects on the overall dynamics. In this case, a CATHARE calculation of the whole circuit is conducted in parallel with TRIO_U or TRIO_U MC2 calculations and the results are used to correct the CATHARE calculation at each time step by means of a specific interface.

### 2.1 CATHARE code [1]

The CATHARE code is a system code and was initially developed in the 1980s for Pressurized Water Reactors. The code has been successively adapted to other reactors such as boiling water reactors, experimental reactors, super critical light water reactors, gas cooled reactors and recently liquid metal cooled reactors. The CATHARE code is now the French reference code for SFR application.

The major differences between the applications lie first in the equations of the state of the fluid and on the closure laws for the interfacial and wall transfers.

Sodium has been introduced in CATHARE as a new fluid and specific friction and heat transfer correlations have been implemented.

Major developments were related to point kinetics to take into account specific reactivity feedback in SFRs (section 3)

For electromagnetic pumps, which are an interesting option in SFRs, two models are available.

### 2.2 TRIO_U code [2]

In pool-type reactors where the cold and hot plena are of great dimensions, both in height and diameter, transient regimes can induce considerable change in the plena thermal hydraulic behavior. Buoyancy forces play a significant role in many transient situations, especially when the primary flow rate is reduced in decay heat removal situations. In the hot pool and the cold pool, the expected phenomena are thermal stratification or particular re-circulations.

The system codes generally using a 0D approach for plena, implying homogeneous mixing, present some limitations when 3D phenomena become significant due to non-symmetrical situations or local buoyancy effects.

So, an important challenge has involved the coupling of system and CFD codes to take into account 3D effects on the global system behavior during transient situations

The so-called TRIO U Computational Fluid Dynamic (CFD) code developed at CEA has been progressively adapted to SFR concerns.

Two strategies exist to achieve the coupling of the CATHARE and TRIO_U codes. In the method by decomposition of domain, the system code does not calculate the domain calculated by the CFD code. Both codes exchange the data via boundary conditions. This method can lead to numerical convergence difficulties.

In the overlapping method, the system code calculates the whole domain and the CFD code allows to make a zoom on plena. This solution has been retained because it is numerically stable and flexible.

The exchanged data at the boundaries are enthalpies, mass flowrates and momentum sources.

An example of coupled CATHARE- TRIO_U modeling is presented in Figure 1.

The CFD domain only represents the liquid domain. A specific domain represents the gas-liquid interface where the sodium free level variations are calculated.

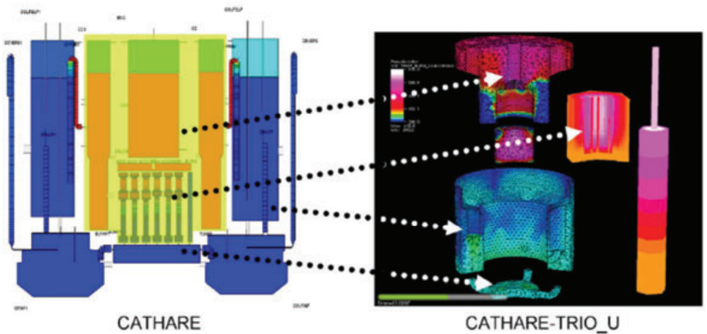


**Figure 1 : Coupled CATHARE-TRIO U modeling**

### 2.3 TRIO_U MC application [3]

Core design and safety studies require the calculation of the local pin cladding temperature, in order to ensure that design and safety criteria are met. To that end, a complete-core sub-channel model has been developed at CEA in the framework of the TRIO_U code. Known as TRIO_U MC ("Core Model"), this code features both a fast, marching-type resolution method for forced/mixed convection steady-states (for design and optimization studies) and a 6-equation, staggered-grid semi-implicit method suitable for single and two-phase transient analysis. An example of steady-state calculation is shown in Figure 2.
In sub-channel codes, the effects of spacer wires on bundle pressure drop and sub-channel mixing must be taken into account by means of correlations : those currently used in TRIO_U MC were derived by Cheng and Todreas [4].

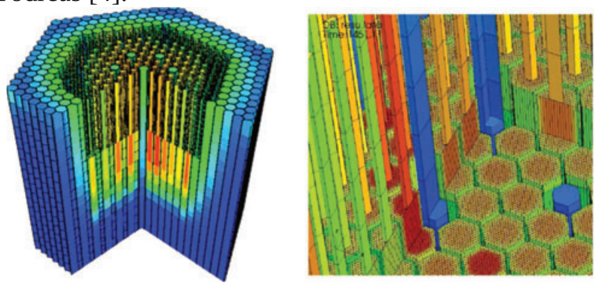

**Figure 2 : Whole core calculation with the sub-channel modeling (TRIO_U MC)**

### 2.4 TRIO_U MC2 model [3]

Flows within the inter-wrapper gaps between subassemblies only account for a small fraction of heat removal from the core in the nominal state: nevertheless, they can have a strong effect on hexcan temperatures, which can in turn affect the mechanical behavior of the core. Moreover, heat removal by inter-wrapper flows plays a stronger role in natural convection, especially when decay heat removal is performed by dipped heat exchangers in the hot plenum.
In order to describe this flow, a model coupling a TRIO_U MC sub-channel description of the subassemblies with a CFD TRIO_U description of the inter-wrapper region and hot pool has been developed : TRIO_U MC2 ("Core-Hot Pool Model"). MC2 features an implicit thermal coupling at the hexcan boundary in order to avoid the time step stability conditions entailed by explicit coupling schemes. A hybrid description of the above core structure (porous-body representation with explicit meshing of the rod sheaths) was developed in order to obtain an optimal compromise between accuracy and computational cost. The mesh currently in use consists in approx. $4.10^5$ meshes for the 400 $m^3$ hot pool (including the above core structure): an example calculation in natural convection is shown in Figure 7. The accuracy of this mesh was studied by comparison with more detailed calculations (with up to $3.10^6$ meshes).

## 3. CORE MODELING CHALLENGES IN THE CATHARE CODE

This paper focuses on unprotected transients. In case of an unprotected transient i.e. without scram, the core modeling is important for the neutronic calculation of the core power evolution.

The calculation of the evolution of the core power is performed by the point kinetics model of CATHARE. Specific reactivity feedbacks in SFR are considered in the CATHARE code:

- Doppler effect induced by fuel temperature variation,
- void effect induced by sodium density change,
- fuel axial expansion,
- thermal expansion of core structures (clads and wrappers),
- control rods insertion in the core due to relative thermal expansion of different structures,
- modification of the compactness of the core (radial expansion) related to the grid plate expansion.

These reactivity feedback effects are taken into account in the CATHARE code through reactivity coefficients based on 3D neutronic computations.

Fuel modeling in CATHARE in order to properly compute the fuel axial expansion and the Doppler effect is discussed in [5].

The two last points are discussed in the following 3.1 and 3.2 parts.

### 3.1 Control rods insertion in the core due to relative thermal expansion of different structures.

The insertion of the rods in the core is the result of the differential expansion of the primary vessel, the core support structure, the structures under the fissile zone, the fissile column, the absorbers and the rods driveline (Figure 3).
These expansions are computed by the CATHARE code from data given by the user (such as the expansion time constant of the control rods tips and of the drive mechanism), except the primary vessel and the core support structure expansions.

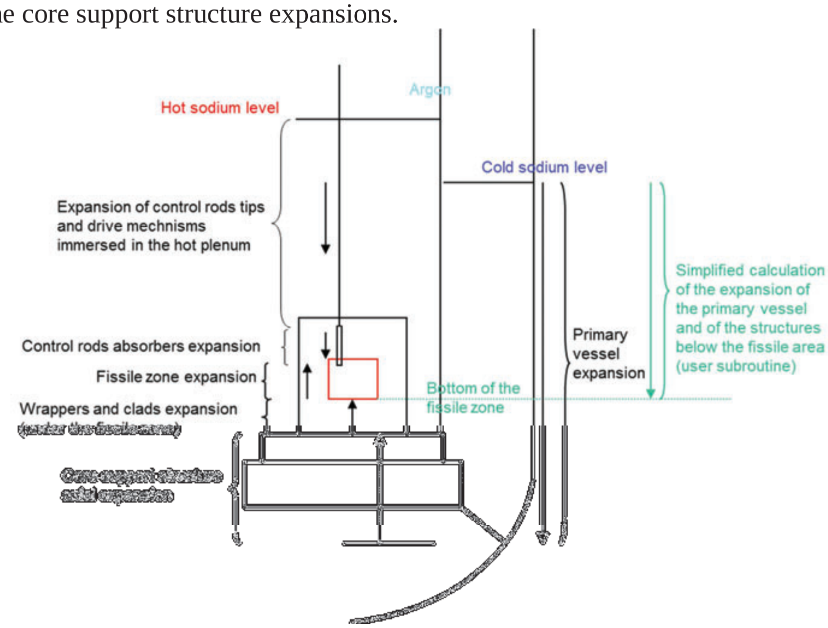


**Figure 3: Representation of differential expansion of the primary vessel, the core support, the structures under the fissile zone, the fissile column, the absorbers, the rods tips and mechanisms.**

In order to take into account the primary vessel and the core support structure expansions, heat transfers from the hot plenum to the reactor vessel through the 'redan' and the vessel cooling system walls are described in the CATHARE data set. The expansion of the primary vessel is calculated in a user subroutine and at each time step, the position of the rods in the core is corrected according to the evolution of the length of the vessel from the level corresponding to the bottom of the fissile zone up to cold sodium level. Under the level corresponding to the bottom of the fissile zone, the reactor vessel expansion and the core support structure expansion are supposed to occur at the same rate and magnitude and therefore cancel.

On Figure 4, we see that it is important to take into account the core support structure and the vessel expansion. In the precise case of an unprotected loss of heat sink, the reactivity feed-back resulting from control rods insertion in the core due to relative thermal expansion of different structures is negative as long as the core outlet increase of temperature is more important than the core intlet increase of temperature. This feed-back becomes positive after; the vessel is 'cooled' by sodium at the temperature of the sodium entering the core.

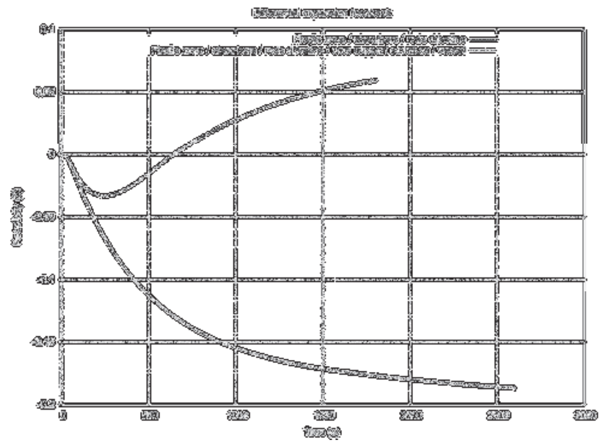

**Figure 4: Reactivity feed-back resulting from control rods insertion in the core due to relative thermal expansion of different structures (with and without taking into account the core support structure and the vessel expansions) for an academic unprotected loss of heat sink in a SFR core**

### 3.2 Core radial expansion related to the grid plate expansion

For unprotected transients during which the core inlet temperature significantly increases, the feedback due to the core radial expansion is important (Figure 5 - diagrid curve).
The grid plate radial expansion time constant is a user input data in the CATHARE code; it can be defined by a thermal-mechanical study of the grid plate structure (response to a temperature rise step of the sodium at the grid plate inlet) with a code for fluid and structural mechanics.

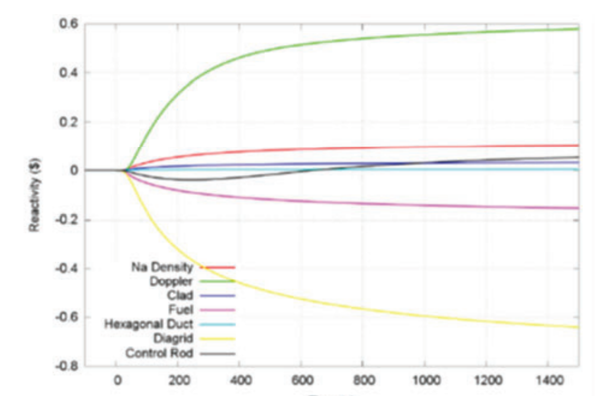


**Figure 5: Example of feedback effects for an academic unprotected loss of heat sink in a SFR core**

## 4. COMPUTATIONAL THERMAL HYDRAULIC SCHEMES

Core thermal-hydraulic studies at nominal operating conditions are described in [3].

For thermal hydraulic transient calculations required to support the safety analysis of SFRs, the modeling and calculation schemes to be implemented are determined based on the studied physical phenomena, functionalities of the various codes used for these calculations and the parameters of interest in the different parts of the reactor.

Key issues for the choice of computational thermal hydraulics schemes are:

- phenomena identification (related to types of transients),
- scenario specific phenomena,
- initial conditions (including ranges),
- imposed sequence of events (including list of assumptions),
- boundary conditions (including ranges),
- sequence of main events,
- range of variation of output quantities.

The choice of computational thermal hydraulics schemes also depends upon the objectives of the characterization of the transient e.g. directed at a single component or a single phenomenon or addressing the entire plant and upon the simulation time. It can also be different according to the phase of a project. The calculation schemes presented below are implemented within the framework of conceptual design studies.

### 4.1 Unprotected loss of flow sequence due to total loss of supply station power resulting in the coast down of the primary and secondary coolant pumps and steam generator dry out, without reactor scram

The establishment of natural convection in the primary circuit during this transient gives rise to several complex phenomena, which cannot be represented in CATHARE. The thermal stratification in the plena in particular can only be represented with a CFD code.

In studies for previous reactors or projects, for example with the DYN2B system code, plena were subdivided into several sub-volumes and flow distribution between these sub-volumes was determined according to the Richardson parameter.

The appeal to calculation schemes implying the coupling of CATHARE and TRIO_U that are time consuming cannot be systematic in phases of a project where the design options evolve and where numerous calculations are realized to estimate these options. An innovative methodology has been developed to determine the need for CATHARE/TRIO_U coupling.
Two CATHARE calculations are performed:

- a calculation with a 0D approach for the hot plenum where the sodium coming out from the core is instantaneously mixed with the sodium of the hot plenum (homogeneous mixing in the volumes modelled with CATHARE),
- a calculation with a 1D without mixing approach called ‘piston’ model where the sodium coming out from the core is directed to the heat exchanger inlet through a 1D element without mixing with the sodium of the hot plenum

In Figure 6, we see the results for the primary intermediate heat exchanger inlet temperature in case of an unprotected loss of flow transient with three different modeling schemes for the hot plenum : CATHARE 0 model, CATHARE 1D without mixing model called ‘piston model’ and coupled CATHARE/TRIO_U model.
The results of the standalone CATHARE calculation show that because of the total mixing in the hot plenum, the temperature at the intermediate heat exchangers inlet is underestimated. This tends to favor the primary natural convection flow, changing thermal center in the exchanger.

The results of the calculation with the 'piston model' show that very hot sodium is injected at the intermediate heat exchangers inlet, what disadvantages the establishment of the primary natural convection. In this example, the great difference of results obtained with CATHARE 0D and 'piston' models led us to resort to a CATHARE / TRIO_U coupling.
The coupled calculation results are positioned between both extreme hypotheses. This study has proved the utility of the calculation coupled to take into account the 3D phenomena in the collectors for this transient study.

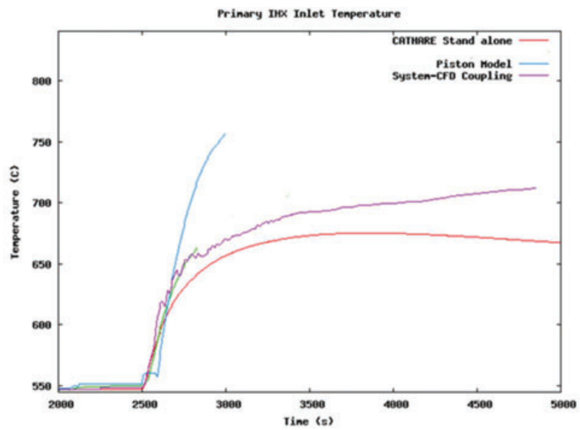


**Figure 6: Primary intermediate heat exchanger inlet temperature in case of a loss of flow transient considering different modeling schemes**

When the primary flow is low (less than 5% of the nominal flow), the cooling of the core can be partially ensured by re-circulations between subassemblies and in the inter-wrapper. These re-circulations, which come in addition to the natural convection in the primary circuit, can be represented only in TRIO_U MC2. The recourse to coupling CATHARE with TRIO_U MC2 is considered only if these re-circulations may affect the temperatures in the core, the evolution of the neutronic power and the flow of natural convection.

Thus the following adaptive approach is adopted:

(1) performance of CATHARE calculations using two different models for the plena (0D and 1D without mixing),

(2) comparison of the results to estimate the sensitivity of the transient on the effects of stratification,

(3) if the evolution of primary flow and the core power of both calculations differs noticeably (as in the example of Figure 6), a coupled CATHARE / TRIO_U calculation can be performed to take into account the effect on the global dynamics of the 3D phenomena in the plena;

(4) performance of a TRIO_U MC2 calculation in post-processing, using the results of flow and core power calculated in (1) or in (3), to determine the local temperatures and the re-circulations in the core and the inter-wrapper,

(5) comparison of the temperatures and the pressures obtained by the system calculation and by TRIO_U MC2 on the common domain to estimate the need for recourse to coupling CATHARE and TRIO_U MC 2.

(6) CATHARE / TRIO_U MC 2 coupled calculation if the effect of re-circulations seems significant enough to modify the primary flow.

The maps of core temperatures calculated by TRIO_U MC2 in (4) or (6) allow us to obtain the local clad temperatures during the transient.

For very long simulation times (several days of simulation), we can be brought to return to CATHARE standalone calculations after the priming of natural convection in the primary circuit, when the decay heat removal systems in the hot plenum have reached stabilized working conditions and when the evolution of the temperatures in the reactor block is very slow.

> Please note: For the loss of supply station power with scram, the thermal hydraulic computational challenges are the same as those for this unprotected transient.

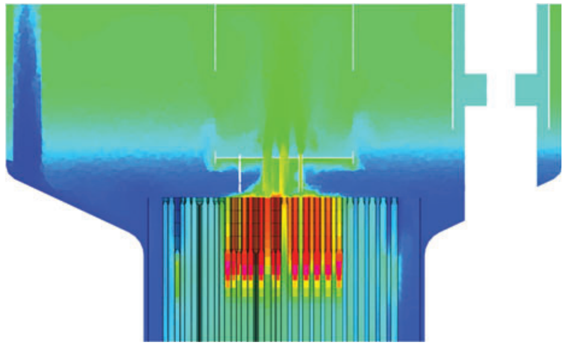

**Figure 7: Example of Trio_U MC2 calculation of natural-convection decay heat removal by passive dipped heat exchangers in the hot pool (on the left)**

### 4.2 Unprotected loss of flow sequence due to the primary pump coast down without reactor scram while the secondary circuits remain operational for power removal

Compared to the previous transient, in this sequence,

- With the secondary circuits continuing to evacuate power, the outlet temperature of the intermediate heat exchangers remains nearly constant during the transient. The evolution of the flow of natural convection in the primary circuit should be less influenced by the stratification in the hot plenum than in the previous case. Thus, the recourse to coupling CATHARE and TRIO_U does not seem necessary in this case.
- With the secondary circuits continuing to operate, we expect that the flow of natural convection in the primary circuit is rather high and is relatively little impacted by re-circulations between assemblies and with the inter-wrapper. The recourse to coupling CATHARE and TRIO_U MC2 does not seem necessary in this case.

The study of this transient is based on a CATHARE calculation completed by post-processing using TRIO_U MC.

### 4.3 Unprotected loss of heat sink due to secondary pump coast down and steam generator dry out, without reactor scram

This accident leads to the increase of temperatures in all parts of the reactor.
With the primary pumps continuing to operate during the transient, a standalone CATHARE calculation allows us to determine the global dynamics (primary flow, core power, temperatures) of the primary circuit during this transient. The coupling with TRIO_U for the plena is not necessary.
A parameter of interest for this transient is the isothermal temperature in the primary circuit at which the fission power is extinguished (without inserting the absorbers) thanks to reactivity feedback.
The TRIO_U MC is used in post-processing to determine local sodium temperatures in the core and clad temperatures.

### 4.4 Failure of a pump – grid plate connection pipe with and without reactor scram

This accident leads to a brutal decrease of sodium flow in the core especially in the part of the core near the broken pipe. The computational challenge is to determine the asymmetry of the core flow supply. Although it noticeably affects the progress of the transient, the asymmetry of the core flow supply caused by the rupture of a pump – grid plate connection pipe has a purely hydraulic character.

We can thus calculate this asymmetry independent of the study of the transient itself, by performing successively:

- a CATHARE calculation to determine the evolution of the sodium flow and pressure conditions in the pump – grid plate connection at the grid plate inlet and at the core inlet;
  In CATHARE, the rupture directive is used combined with a source directive in the cold plenum, the two models being connected by a hydraulic link.
- isothermal hydraulic TRIO_U calculations of the grid plate (constant temperature at the nominal value of 400°C). The objective is to determine the pressure profile in the grid plate and deduct flow distribution map in the core. For example, the CFD calculation of a grid plate structure of 7.6 m of diameter with about 1500 places for subassemblies requires a meshing of approximately 70 million meshes. The evaluation of the asymmetry of the core flow supply is realized from the calculation of steady states before and after the pipe rupture.

We can then carry out a new CATHARE calculation of the transient, correcting the flows in the various subassemblies according to this flow distribution map.

Please note: for Superphenix, the core flow supply asymmetries had been deduced from experimental results.

At last, TRIO_U MC (in its unsteady version because the transient is fast) is used in post-processing to estimate the clad temperatures during the transient.

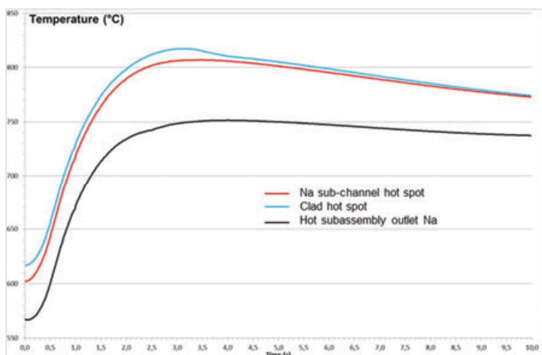


**Figure 8: example of results obtained with TRIO_U MC post-processing of a transient calculation (academic unprotected rupture of a pump-grid plate connection in a SFR reactor)**

On these curves, we see that the difference between a subassembly mean outlet temperature and the maximum clad temperature in this subassembly is round 70°C at the temperature peak. This difference is mainly due to radial sodium temperature profile inside the subassembly: the pins at the center of the subassembly are less cooled than the pins at the periphery (hydraulic cross sections are different).

## 5. VALIDATION

The above mentioned codes are subjected to a validation plan covering their domain of use in the selected computational schemes for the above studies.

The approach includes the following stages for each phase of the transients:

(1) Identification of the relevant physical phenomena;
(2) Determination of the models and the laws likely to influence the phenomena identified in (1);
(3) Estimation, for each code, of the uncertainties associated with the laws and the models identified in (2);
(4) Identification from (3) of the laws and models to be validated;
(5) Comparison of these needs in validation with the existing experimental database

(6) Identification of additional needs in validation, for the needs which are not covered by the existing experimental database and according to the targeted levels of uncertainties;
(7) Sizing of experiments to meet these needs.

The available experimental database is very substantial including the results of the analytical experiments, component measurements and reactor data in particular coming from ultimate tests performed in the Phenix sodium cooled fast reactor. Indeed, numerous experiments have been conducted at CEA for SPX and in the framework of EFR studies. Some of the most relevant past, current or planned experiments are quoted below.
Tests conducted in the CORMORAN set up provided suitable results for the validation of codes to be used in the assessment of stratification phenomena [6].
PX ultimate tests include a natural convection test in the primary circuit [7] and an asymmetrical test.
TRIO_U validation data base includes MONJU experiments [8].
In TALL-3D, specific experiments have been conducted to validate the CFD and system codes coupling approach [9] and are used to validate the TRIO_U and CATHARE codes coupling.
PLANDTL experiments conducted in Japan [10] are of interest for the validation of TRIO_U MC2.
New experiments are under construction in France with the ASTRID configuration; the first mock-up at 1/6 scale (called MICAS), connected to the PLATEAU loop (water facility), is dedicated to the study the flow regime of the hot plenum [11].

## 6. UNCERTAINTIES EVALUATION [12]

For prevention situations (resulting from the accidents described in section 4 for example), a BEPU approach can be implemented to estimate margins to safety criteria.

This approach can be summarized in three stages:
(a) The uncertainty space of input parameters (defined by their uncertainty ranges) is sampled at random according to the combined probability distribution of the uncertain parameters, and code calculations are performed by sampled sets of parameters.
(b) The number of code calculations is determined by the requirement to estimate a tolerance and confidence interval for the quantity of interest. Wilks' formula can be used to determine the number of calculations required to obtain the uncertainty bands. A bootstrap methodology can also be used.
(c) Statistical evaluations are performed to determine the sensitivities of input parameter uncertainties on the uncertainties of key results (parameter importance analysis).

This methodology has been applied for some transient studies to evaluate the uncertainty associated with a safety parameter.
In the example hereafter (Figure 9), numerous system code calculations with CATHARE have been performed to evaluate the uncertainty associated with the margin to sodium boiling in the core in case of an unprotected loss of supply station power (short term calculations).

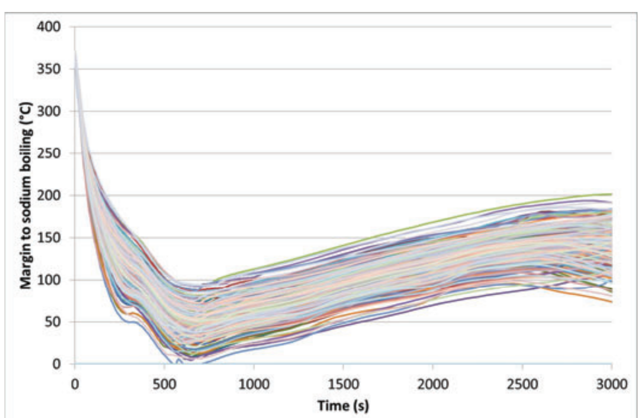


**Figure 9 : Example of evolution of the margin to sodium boiling during a loss of flow transient without scram for different data set of input parameters sampled in their uncertainty space**

This approach can also be applied for the core thermal-hydraulic calculation where some input data and associated uncertainties can result from system calculations; a new methodology to quantify the uncertainties associated to functional random variables is studied at CEA.

The methodology can be different according to the computational thermal-hydraulic schemes. The propagation of uncertainties in CFD or coupled system-CFD transient computations is a large field of investigation and begins at CEA.

## 7. CONCLUSION AND PERSPECTIVES

Computational schemes concerning four transient calculations required to support the safety analysis of SFRs are presented in this paper. It is reminded that the approach for the determination of a computational scheme for a given study is adaptive and is based on a physical pre-analysis as well as on the consideration of some constraints.

These schemes involve three CEA thermal-hydraulic codes at different scales: CATHARE (system code), TRIO_U (Computational Fluid Dynamics code), TRIO_U MC (core subassembly code).

For the study of a total loss of supply station power without reactor scram, different schemes can be implemented that resort to the coupling of CATHARE with TRIO_U or TRIO_U MC2.

The primary pumps coast down and the loss of heat sink without scram transient studies are based on CATHARE calculations completed by post-processing with TRIO_U MC.

For the rupture of a pump - grid plate connection pipe, system computation is done with the CATHARE code. The system computation provides boundary conditions for a CFD isothermal hydraulic computation of the grid plate. The objective is to determine the asymmetries of the core flow supply. The values of local sodium temperatures and clad temperatures are obtained by TRIO_U MC calculations (unsteady version).

The codes are subjected to a validation plan covering their domain of use in the selected computational schemes for the above studies.

The perspectives of continuing this work are the following:

- the continuation of the implementation and the evaluation of the thermal-hydraulic computational schemes presented in this paper for various scenarios of the studied transients and different configurations of reactors,
- the continuation of defining the code validation plan,
- the continuation of determining the specific thermal-hydraulic computational schemes for other transients such as asymmetrical transients, transients of power by control rod withdrawal or global reactivity variation, local core accidents….
- the continuation of defining methods of uncertainties propagation in complex calculations schemes involving the chaining and/or the coupling of codes.

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## APPENDIX A

The implementation of calculation schemes such as described above cannot be systematic in particular in the phases of projects where we have to estimate design orientations. The example below concerns the design approach of a core with regard to its behavior during unprotected transients.

The reactivity evolution during a transient can be expressed by:

$$\int_{Te,n}^{Te} k.\,dTe + \int_{\Delta T,n}^{\Delta T} g.\,d\Delta T + \int_{P,n}^{P} h.\,dP \;+\; \rho_{\mathrm{ext}} = 0$$

Where:

'k' is a global neutronic coefficient (i.e. grouping elementary neutronic coefficients) related to sodium core inlet temperature variations

'g' is a global neutronic coefficient (i.e. grouping elementary neutronic coefficients) related to sodium temperature elevation (through the core) variations

'h' is a global neutronic coefficient (i.e. grouping elementary neutronic coefficients) related to core power variations

For example, in the case of an unprotected loss of heat sink transient (ULOHS), by using these coefficient (supposed constant), we can approximate the isothermal temperature in the primary circuit at which the fission power is extinguished without inserting the absorbers) thanks to reactivity feedback:

$$T_{end} = Tec_0 + (\Delta Tc_0 *g + P_0*h) / k = 400 + (150\ g + 100\ h) / k$$

with $Tec_0$ : sodium core inlet temperature at initial operating conditions
$\Delta Tc_0$ : sodium temperature elevation through the core at normal operating conditions
$P_0$ : core power (in %) at normal operating conditions

The system computation gives the dynamics of the transient.

In [13], it is shown that using simplified models integrating global reactivity coefficients, it is possible to determine some constraints concerning these coefficients taken separately or combined in order to improve the natural behavior of a core in response to unprotected accident initiators. This analysis is based on the study of reactors of various powers with oxide or metal-fueled cores.

In [14], the same approach is applied for the design of a carbide-fueled core to meet some safety objectives related to enhanced natural behavior during accidental transients. The simplified quasi-static transient modeling provides the possibility to run sensitivity analyses for all the investigated parameters. This is illustrated in Figure 10 , where core parameters are evolving independently two by two, to highlight the theoretical domains where all the safety criteria are met (white region). This multi-parameter sensitivity analysis provides the designer with the shape of the hypervolume where all the safety objectives are reached.

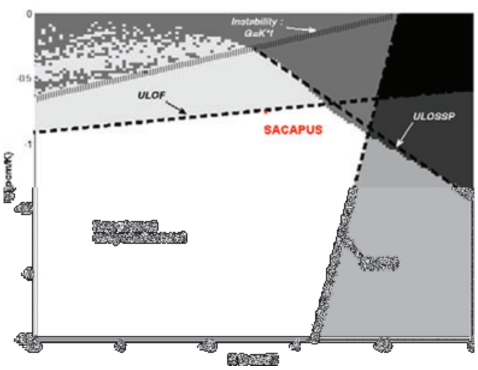


**Figure 10 : Projection, for an academic core, of the estimated domain in the k/g plane (given value for h) where the criteria of transients are met**