Pre-conceptual design study of ASTRID core

Frederic Varaine
Atomic Energy and Alternative Energies Commission

Pierre Sciora
Atomic Energy and Alternative Energies Commission

Christophe Venard
Atomic Energy and Alternative Energies Commission

Bruno Fontaine
Atomic Energy and Alternative Energies Commission

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Frédéric VARAINE – Philippe MARSAULT – Marie-Sophie CHENAUD – Bruno BERNARDIN – Alain CONTI
Pierre SCIORA – Christophe VENARD – Bruno FONTAINE – Nicolas DEVICTOR – Laurent MARTIN
Alternative Energies and Atomic Energy Commission
CEA, DEN, DER - 13108 St Paul lez Durance, France
Phone: 33 4 42 25 28 65, Fax: 33 4 42 25 36 35, Email:frederic.varaine@cea.fr

Anne-Claire SCHOLER – Denis VERRIER
AREVA-NP, 10 rue J. Récamier, 69456 Lyon cedex 06, France

Abstract – *In the framework of the ASTRID project at CEA, core design studies are performed at CEA with the AREVA and EDF support. At the stage of the project, pre-conceptual design studies are conducted in accordance with GEN IV reactors criteria, in particularly for safety improvements. An improved safety for a sodium cooled reactor requires revisiting many aspects of the design and is a rather lengthy process in current design approach. Two types of cores are under evaluation, one classical derivated from the SFR V2B and one more challenging called CFV (low void effect core) with a large gain on the sodium void effect.*
The SFR V2b core have the following specifications : a very low burn-up reactivity swing (due to a small cycle reactivity loss) and a reduced sodium void effect with regard to past designs such as the EFR (around 2$ minus). Its performances are an average burn-up of 100 GWd/t, and an internal conversion ratio equal to one given a very good behavior of this core during a control rod withdrawal transient).
The CFV with its specific design offers a negative sodium void worth while maintaining core performances.
In accordance of ASTRID needs for demonstration those cores are 1500 MWth power (600 MWe).
This paper will focus on the CFV pre-conceptual design of the core and S/A, and the performances in terms of safety will be evaluated on different transient scenario like ULOF, in order to assess its intrinsic behavior compared to a more classical design like V2B core.
The gap in term of margin to a severe accident due to a loss of flow initiator underlines the potential capability of this type of core to enhance prevention of severe accident in accordance to safety demonstration.

I. INTRODUCTION

As the prototype of SFR technology, ASTRID (acronym for Advance Sodium Technological Reactor for Industrial Demonstration) has the main objective of demonstrating advances on an industrial scale by qualifying innovative options[1]. It must be possible to extrapolate its characteristics to future industrial high-power SFRs, particularly in terms of safety and operability[2].

The CEA has been given the responsibility by the French Government of the ASTRID Project, which involves:

- Operational management ensured by a project team which is also responsible for the industrial architecture, i.e. it defines the different engineering work packages.
- Managing most of the R&D work and qualification of the options that will be chosen for ASTRID.
- Assessment of studies carried out by industrial partners, in charge of technical work packages, or external engineering companies.
- Direct responsibility of the core design.

The goals for the reactor core design are ambitious. Substantial progress in terms of inherent safety and risk prevention of core meltdown accident are sought for:

- Natural behavior favorable in case of unprotected loss of flow and heat sink transients, with a goal not to reach the sodium boiling in the transient of unprotected blackout ULOSSP (Unprotected Loss Of Station Supply Power) with sufficient margins to make the safety demonstration, robust,

- Natural and favorable behavior during a Control Rod Withdrawal (CRW) with the goal of non-fuel melting in case of a full unprotected withdrawal,

- Best performance in terms of fuel Burn Up and residence time with a zero or positive Breeding gain.

For that, a core stable natural behavior for any disturbance is reached, in association of two independent strong and diversified lines of defense. In case of failure of one or two lines of defense, intrinsic core behavior through natural feedback will bring it to a stable state.

The CFV concept[3] because of its special features allows those possibilities. This concept is selected as reference in the evaluation phase of ASTRID project with assigned objectives outlined above.

II. PERFORMANCE TARGETS

In the preliminary design phases, it is important to link performance goals and objectives of safety.

It allows the system (in this case the reactor core and the primary circuit) to be optimized as a whole, while leaving the choice to designers on how to use design margins during the design process.

Performance targets and associated potential margins are recalled below, with focusing on the mastering core reactivity in case of incident or accident.

II.A Favorable natural behavior during loss of Flow transients

The behavior of the core during a loss of flow transient is used to evaluate its ability to endure very strong perturbations involving all the risk phenomena.

It was used as an initiator of core meltdown accident in the case of ULOSSP in previous Safety report.

It is assigned to the CFV core to sustain, this transient without boiling with sufficient margin.

In the event that this margin not being large enough to allow a robust demonstration from safety purpose, provisions to ensure stable Sodium boiling[4] will be also considered, by specific laying out of the head the sub-assembly.

II.B Favorable natural behavior during a Control rod Withdrawal

CRW is seen as a potential initiator of general degradation of the core from a local melting, according to the following sequence:

- Increase of linear power in the neighboring fuel sub assemblies of the control rod, reaching the melting of some pellets,
- In case of clad failure, molten fuel ejection in sodium channels and destruction of adjacent pins, creating a blockage and a propagation of the melting zone by default of cooling.
- Compaction of the fuel may implies the drilling of the wrapper tube and spreading of melting to other sub-assemblies that could locally reach criticality

The criterion of "0%"molten fuel" during a protected CRW is maintained as a design goal, , by actuation of the first emergency system, but also in case of total unprotected CRW (which means in this case the failure of two strong lines of defense).

II.C Elimination of Transient of Power

The cores of fast neutron reactors have the characteristic of not being in their most reactive geometry. Given the potential reactivity insertion into play and the kinetics of possible core compaction transient, the risk of significant compaction of the core could reach the prompt-criticality must be "practically eliminated". Considerations of ad hoc arrangements for the ASTRID core (natural strapping, pads on the hexagonal wrapper tubes ensuring the right level of core compactness during earthquakes, for example) are taken into account, allowing the practically elimination of their impact.

Similarly, a large amount of gas flowing through the core or a failure of the supporting structure of the core that could bring up to a large reactivity effect leads to core melting will be also " practically eliminated".

These design provisions will not be detailed further in this article

II.D Improvement of behavior in case of sub-assembly Total Instantaneous Blockage

The objective associated with the analysis of TIB is to demonstrate that the melting of fuel elements does not spread beyond the six adjacent sub-assemblies, taking into account the system of detection and protection (ie control rod shutdown), and then that this type of accident can not cause whole core melting. Therefore it is needed to demonstrate that the geometry of 7 melted sub-assemblies is sub-critical and could be cooling

For the ASTRID core design the goal is to ensure sub - criticality of 7 melted adjacent sub-assemblies, ensuring the performance in terms of consistent detection with this hypothesis. (in this case the TIB must be detected after the melting of the first assembly and before the end of melting of the fuel row of 6 neighbouring fuel sub-assemblies).

The objective to pursue is the early detection of this blockage (before drilling the hexagonal wrapper tube of the blocked sub-assembly) leading ultimately to the melting of the single blocked assembly.

II.E economic Performance

Although the safety improvement goals are the priority, objectives related to the economic system viability associated to industrial SFR competiveness are maintained at the highest level.

For this version of core design with AIM1 austenitic steel pin cladding material, maximal damage rate is around 100 DPA NRT, fuel and absorber residence time is high (~1400 EFPD), and the fuel cycle length is up to 360 EFPD

III. ASTRID CORES

The core design is optimized towards the best natural behavior during transients.

High stability of the core is obtained through stabilizing feedback coefficients in normal operating conditions.

In addition the characteristics of the core are chosen so that the Doppler effect automatically limits any power excursion causing a temperature rise of the core.

Neutronic effects in cases of "disappearance" of the coolant, so-called "void effect" is a subject of recurrent questioning about safety of fast reactors cooled by liquid metal.

A sodium void in some areas of the core (positive void effect), by boiling, decrease of sodium level in the vessel or gas ingress, could lead to power increase. Arrangements can be achieved through design to avoid large sodium boiling, a gas entrainment in the core or the core draining in case of leakage from the main vessel.

The ASTRID reactor core will be developed with the aim of minimizing the positive void effect in order to improve safety.

III.A CFV concept

The CFV concept[4] (see Fig. 1) elaborated by CEA consists of the optimized combination of characteristics such as described thereafter,

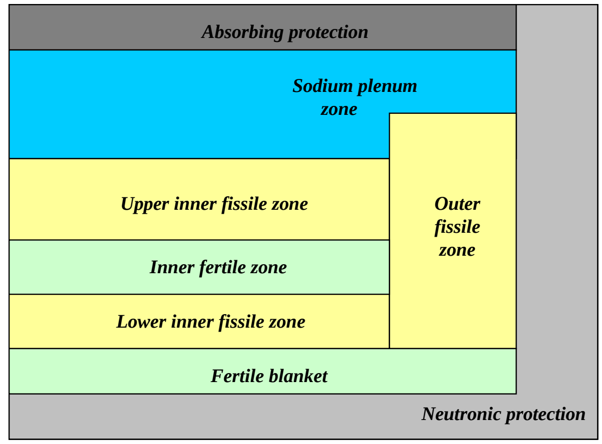

Fig. 1. CFV core geometry.

To reduce the sodium void worth by increasing the leakages, different configurations can be considered. Based on the sodium temperature distribution, the most effective leakages will be at the top of the core, where the margin to boiling is minimum. The use of a sodium plenum zone seems essential. Unfortunately, it is not sufficient in spite of a large thickness of this zone. It is necessary to maximize the leakages in case of voided sodium case. It means: increase the flux at the upper boundary of the core between nominal and voided conditions and decrease the flux at the upper boundary, in order to improve the effectiveness of the plenum zone.

The use of an internal blanket in specific core geometry allows an increase of the flux level at the upper surface. Of course, thickness, position and radius of the blanket have to be adjusted. The same is needed for the core height.

The use of an absorbing zone in upper shielding prevents neutrons to return to the core during voiding. The sodium plenum zone thickness is optimized for reducing the void effect taking into account shielding needs during nominal operation.

IV. CORE PERFORMANCES

Performances presented here are those of an intermediate version of the current phase of the project so called CFV V1. In parallel and to have elements of comparison, a version of the large power V2B[5] core (previous homogeneous core developed by CEA in 2009) adapted to ASTRID power level and vessel constraint was also studied.

The main performances are presented in the following Table I, and the cores layout are shown in Fig.2 and Fig.3 (for this last scheme, the core layout presented here is issued from calculation tools, but the external shape will be similar of the CFV one).

TABLE I

CFV and V2B core characteristics

Core design	CFV	V2B 1500
Thermal Power	1500 MW	1500 MW
Elect. Power	600 MW	600 MW
Fuel Residence Time	1440 EFPD	1560 EFPD
Fuel Cycle Length	360 EFPD	390 EFPD
Batch	4	4
S/A Pitch	17,5 cm	16,8 cm
Nb fuel elements C1/C2	177 / 114	144 / 144
Pin diameter	9,7 mm	10,73 mm
Pins/Assembly	217	169
CSD/DSD Nb	12 / 6	18 / 6
Pvol (Fiss.+ int. fertile)	226 W/cm3	194 W/cm3
Pvol (Fissile))	258 W/cm3	194 W/cm3
$\Delta\rho$ (cycle)	- 1550 pcm	- 875
$\Delta\rho$ (efpd)	- 4,3 pcm	- 2,2 pcm
Fuel burnup C1/C2	105/ 69 GWj/t	76 / 67 GWj/t
Pu enrichment C1/C2	23,5 / 20 %	13,9 / 17,6 %
DPA max	113	108
Void effect EOC	-0,5 $	+5,1 $
Beff	364 pcm	373 pcm
Breeding Gain	-0,02	-0,05
Pu inventory (HN)	4,9 t	5,3 t
Max linear power rate	483 W/cm	407 W/cm
Core pressure drop	2,6 b	>3b
Total core flow rate	7990 ks/s	7990 kg/s
Inlet core temperature	400°C	400°C
Outlet core temperature	550°C	550°C
Fissile zone diameter	340 cm	312 cm

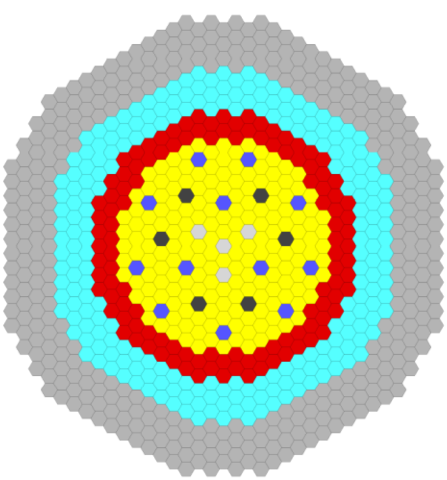

Fig. 2. CFV core layout.

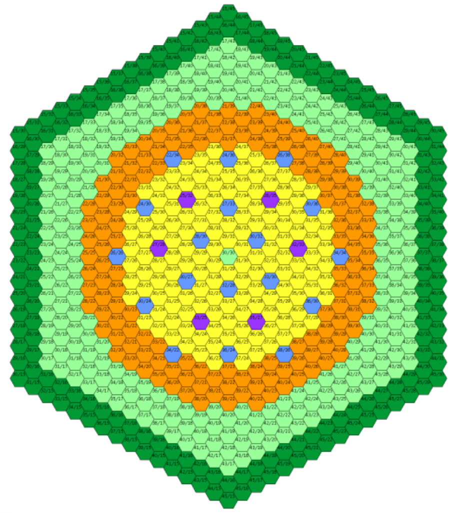

Fig. 3. V2B core layout.

Neutronic and thermal-hydraulic core calculations were performed with CEA's reference code system ERANOS[6,7] and CATHARE[8].

The design of core elements and thermo-mechanical behavior are not detailed here, but have been evaluated at all levels: pin (also fuel behavior in normal and accidental case), sub-assemblies, and core.

In the same time some improvements in terms of methodology or modeling are performed, in thermal hydraulic/mechanical behavior, neutronic calculation[9] or fuel and absorber design.

IV.A Behavior during thermal hydraulic transients.

ASTRID reactor is pool type with cylindrical-conical shape, called "redan" to separate the hot and cold plenum with 3 mechanical primary pumps (PP) with a an halving time of about 20s and four intermediate heat exchangers (IHX). The Fig.4 shows the scheme of the vessel.

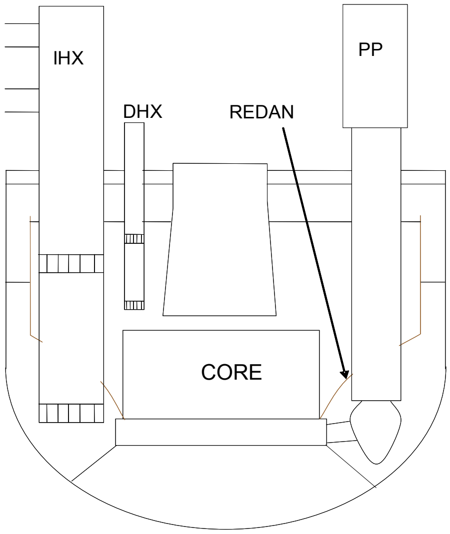

Fig. 4. Synoptic scheme of ASTRID primary vessel.

For the CFV core the total flow of the core is around 8000 kg/s and the core pressure drop is 2,6 bars.

The baseline for studying the dynamic behavior of the core is a situation of unprotected blackout transient without scram and without starting up of ultimate emergency systems or decay heat removal so called ULOSSP

In addition to this scenario, the natural behavior (without rod drop) of the reactor has been evaluated in the following scenarios:

- ULOF (Unprotected Loss of Flow): loss of primary pumps without scram, secondary pumps remaining at nominal flow.
- ULOHS (Unprotected Lost Of Heat Sink): Stop of secondary pumps in 100s (inertia estimated for these studies) without scram, the primary pumps still remaining in normal operation.

The behavior of the CFV V1 during ULOSSP is presented below. There is a temperature peak at 300s with a margin to Na boiling in the range of 50 ° C (see Fig. 5) and a shutdown of neutronic power at 3000s (see Fig. 6) without any intervention or emergency system startup.

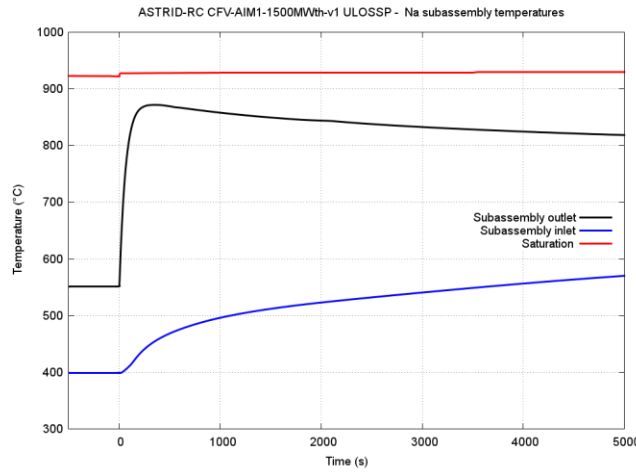

Fig. 5. Na temperature during ULOSPP (CFV Core)

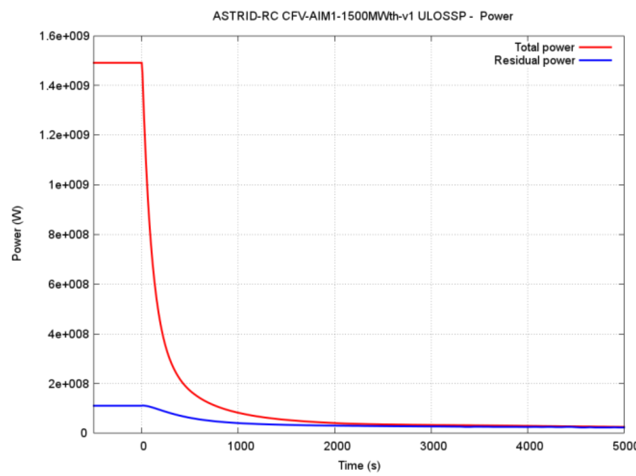

Fig. 6. Core Power during ULOSPP (CFV Core)

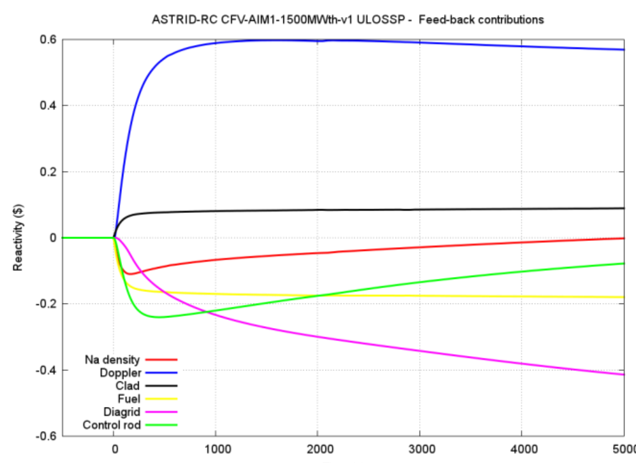

Fig. 7. Feed-Back coefficients during ULOSPP (CFV Core)

This good behavior is obtained thanks to the contribution of the negative effect of sodium density (red curve in Fig. 7) that is positive in a standard core.

Fig 8 shows a comparison between CFV and V2B CFV core for outlet temperature S/A during ULOSSP in order to underline behavior discrepancies: a standard core like V2B reaches Sodium boiling in less than 100 s.

This is explained .by the fundamental difference in feedback coefficient value of Sodium density that is shown in Fig. 9: in case of homogeneous core like V2B this value is positive; in case of CFV it is negative.

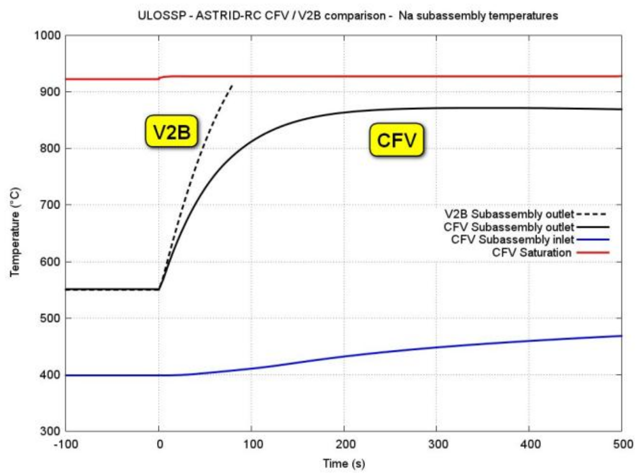

Fig. 8. Comparison between CFV and V2B on ULOSSP transient.

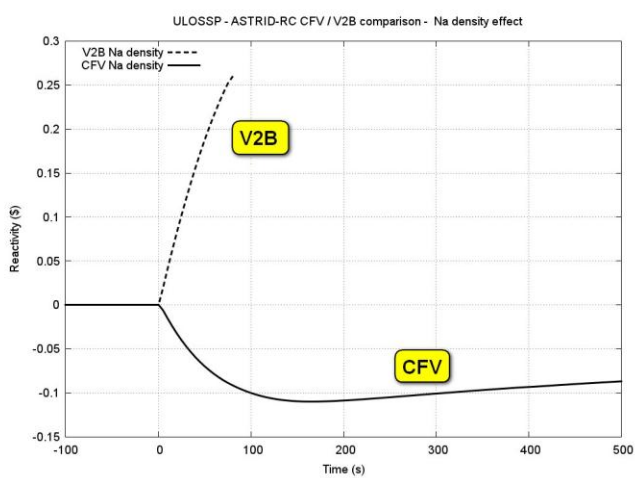

Fig. 9. Feed-Back coefficients during ULOSPP (CFV and V2B Cores)

Similarly, CFV shows a better behavior than V2B for ULOHS and ULOF transients. (See Fig. 10 below for ULOHS case)

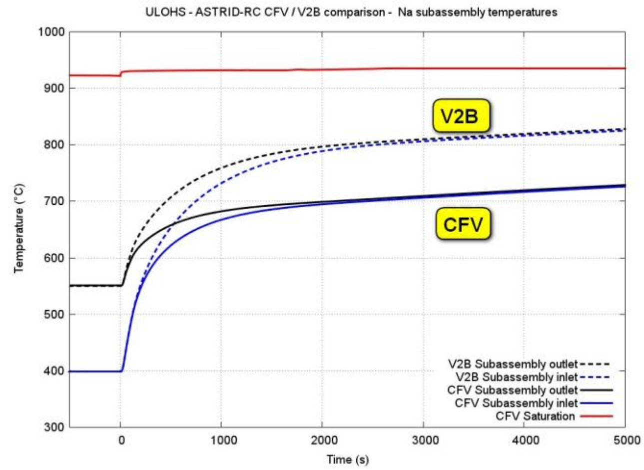

Fig. 10. Comparison between CFV and V2B on ULOHS transient.

In the case of CFV during ULOF, boiling is reached because, at around 800s, feedback coefficient resulting from the control rod expansion due to the main vessel expansion, decreases, leading to a relative movement between control rods and the core, which is extracted from it. (Fig. 11, green curve).

The contribution of the stabilizing effect of sodium density remains also in this transient.

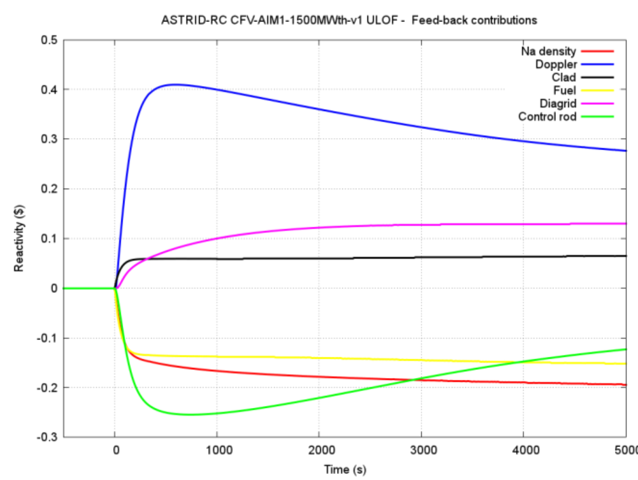

Fig. 11. Feed-Back coefficients during ULOF (CFV Core)

Table II below, summarizes all results concerning transient behavior

TABLE II

CFV and V2B transient performances

Core design	CFV	V2B 1500
ULOSSP	55°C of marging	Na boiling ~100s
ULOF	Na boiling ~3500s	Na boiling ~100s
ULOHS	Temp. of neutronic shutdown 700°C	Temp. of neutronic shutdown 800°C

IV.B Core behavior during a control rod withdrawal.

The Compliance with the criterion is based on the comparison for each adjacent assembly to the control rod between the linear power (Plin) at the end of CRW and the melting fuel linear power (Pfus) calculated by GERMINAL code[10]. The linear power during the CRW transient is calculated as follows Eq. (1):

$$\overline{P_{lin}}(t) = P_{lin_o}\left[1+k'_i.\Delta\rho(t)\right].\left[1+b_0.\Delta\rho(t)\right] \quad (1)$$

The Neutronic code ERANOS[6] calculates the following parameters:

- Plin0, initial linear power of the considered fuel sub-assembly (see Fig. 12),
- k'i , relative variation of linear power density per unit of inserted reactivity,
- Δρ total worth reactivity of the control rod between its initial position in the core and the parking position at the end of the withdrawal.

The coefficient b0 is calculated by the system code CATHARE[8] and represents the relative variation of the total power of the core per unit of reactivity inserted.

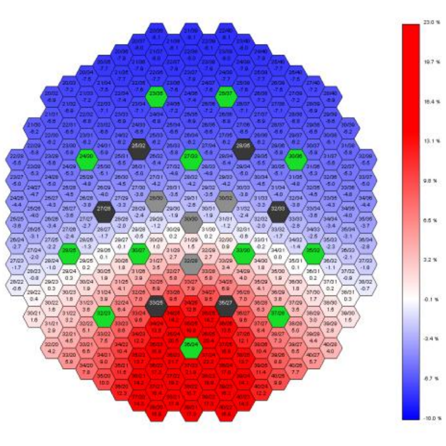

Fig. 12. Power shape after a complete Control Rod Withdrawal

All the CRW that can occur on the CFV V1 can be detected by two devoted independent systems of core detection which stop the reactor by scram.

The first system prompted is the core temperature monitoring; the second is the neutron detection
The case of a total control rod withdrawal corresponds to the failure of two strong lines of defense; the outer rods do not comply with the criterion of no melting in this specific case, but could be reached by improving the methodology combined with appropriate design features.

V. CONCLUSIONS

Pre-conceptual design studies of ASTRID core are ongoing. A first version of the CFV core called CFV V1 had been design, and shows the pre-feasibility of this concept at this stage.

Improvements in CFV core compared to previous SFR core (like SPX or EFR in France) or homogeneous standard SFR core exhibit improvements in terms of performance and natural behavior.

Feedback coefficients are optimized (in particular sodium void worth) to allow better natural behavior in unprotected loss of flow transient (ULOx):

- no sodium boiling with margins (short term behavior),
- Temperature of neutronic shutdown reduced (long term behavior) compared to SPX core.

These characteristics allow the possibility for the CFV core to eliminate energetic severe accident scenarios due to this event category.

The overall negative sodium void worth gives margin in case of global core boiling or sodium draining. This new specificity compared with standard core (void coefficients> $ 5) has been also extrapolated to large industrial core[2].

The low core pressure drop is also favorable for fast loss of flow transient or in case of natural convection.

The weak loss of reactivity of the core during irradiation cycle is also favorable for the CRW accident.

Studies will be conducted until the end of 2012, and an improved version of the ASTRID CFV core will be designed in order to fulfill all the CRW criteria, increase the margin for all unprotected loss of flow transient, and keep high economic performances.

ACKNOWLEDGMENTS

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NOMENCLATURE

CFV: “Coeur à Faible Vidange” (Low Sodium Void Core)
CRW: Control Rod Withdrawal
IHX: Intermediate Heat Exchanger
TIB: Totally Instantaneous Blockage
ULOSSP: Unprotected Loss Of Station Supply Power
ULOF: Unprotected Loss Of Flow
ULOHS: Unprotected Loss Of Heat Sink

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