NURETH14-099

Natural convection test in Phenix reactor and associated CATHARE calculation

D.Pialla[1], D.Tenchine[1], P.Gauthé[2], A.Vasile[2]

[1] CEA, DEN, DER/SSTH, Grenoble, F-17 rue des Martyrs, 38000 Grenoble, France
[2] CEA, DEN, DER/SESI, Cadarache, F-13108 Saint Paul lez Durance, France
david.pialla@cea.fr, denis.tenchine@cea.fr, paul.gauthe@cea.fr, alfredo.vasile@cea.fr

Abstract

The Phenix sodium cooled fast reactor started operation in 1973 and was stopped in 2009. Before the reactor was definitively stopped, final tests were performed, including a natural convection test in the primary circuit. One objective of this natural convection test in Phenix reactor is the qualification of plant dynamic codes as CATHARE code for future safety studies.

The paper firstly describes the Phenix reactor primary circuit. The initial test conditions and the detailed transient scenario are presented. Then, the CATHARE modelling of the Phenix primary circuit is described. The whole transient scenario is calculated, including the nominal state, the steam generators dry out, the scram, the onset of natural convection in the primary circuit and the natural convection phases.

The CATHARE calculations are compared to the Phenix measurements. A particular attention is paid to the significant decrease of the core power before the scram. Then, the evolution of main components inlet and outlet temperatures is compared. The need of coupling a system code with a CFD code to model the 3D behaviour of large pools is pointed out. This work is in progress.

1. Introduction

In the frame of Generation IV deployment, a sodium cooled fast reactor prototype called Astrid is planned for 2020 (ref. [1]). One important safety challenge for Astrid and future reactors is the Decay Heat Removal (DHR) capability. Due to sodium coolant characteristics, passive DHR by natural convection is possible in case of loss of flow situation. However, the reactor design must be adapted to facilitate natural convection and the efficiency and the reliability of natural convection must be checked.

For the Astrid prototype and the future sodium cooled fast reactors, the CATHARE plant dynamics code is used to calculate the thermal hydraulic behaviour of the reactor during transient situations (normal and accidents). Of course, the loss of flow situation and the transition to natural convection is one important case to be calculated by the CATHARE code. The qualification of CATHARE code for natural convection situations requires reactor data for various design and flow conditions. The Phenix plant, which started in operation in 1973, was stopped in 2009. It was previously decided to perform final tests in Phenix reactor before its final shutdown. A natural convection test was selected as it is obviously one important challenge for future reactors. The test was performed in June 2009. This test is used for an international benchmark exercise in the frame of an IAEA Coordinated Research Project

(CRP) and for the THINS (Thermal Hydraulics INnovative System) project on behalf of the 7th European Framework Program.

The paper successively presents Phenix main features, the natural convection test conditions, some test results, CATHARE code main characteristics, the modelling of Phenix reactor and CATHARE results compared to reactor data.

2. Phenix main features

Phenix is a pool type reactor composed of a main primary reactor vessel, three sodium secondary circuits, three tertiary water/steam circuits and one turbine, as depicted in Fig.1. Phenix nominal power was 560 MWt /250 MWe (ref. [2]). However, during the final tests in 2009, one secondary and tertiary circuits were out of operation and the reactor was operating at a lower power of 120MWt for the natural convection test.

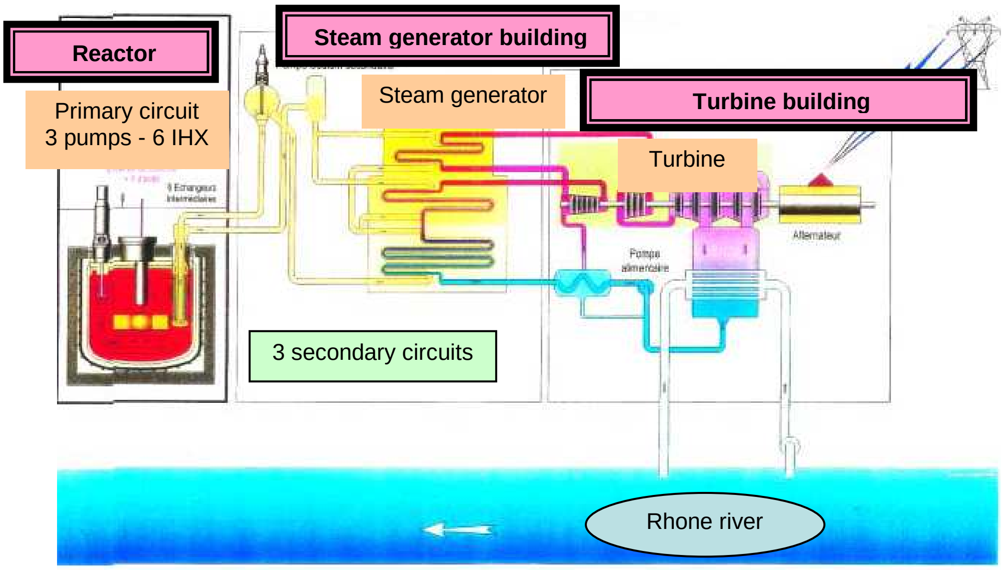

Fig. 1 Phenix power plant

The reactor main vessel shown in Fig. 2 is 11.8 m in diameter and 9.8 m height, filled with 800 tons of sodium (ref. [2]). A safety vessel is surrounding the main vessel in case of sodium leakage. The space between the main vessel and the safety vessel is full of nitrogen. The roof is a 0.06 m depth structure closing the top of the main vessel to ensure a tight barrier. Argon is filling the upper space between the sodium and the roof. A concrete structure is supporting the whole reactor vessel with the internal structures and the primary components. The main internal structures in the reactor vessel are the core support structure, the diagrid structure feeding the core, the above core structure, the handling machine and the internal vessel separating the hot pool and the cold pool. The main components are the three primary pumps and the six intermediate heat exchangers (IHX). The usual nominal core flow rate was 3000 kg/s, but it was reduced to 1280 kg/s for the 120 MWt power state. Only four intermediate heat exchangers were in operation during the final tests, the two other ones being replaced by inactive components called DOTE.

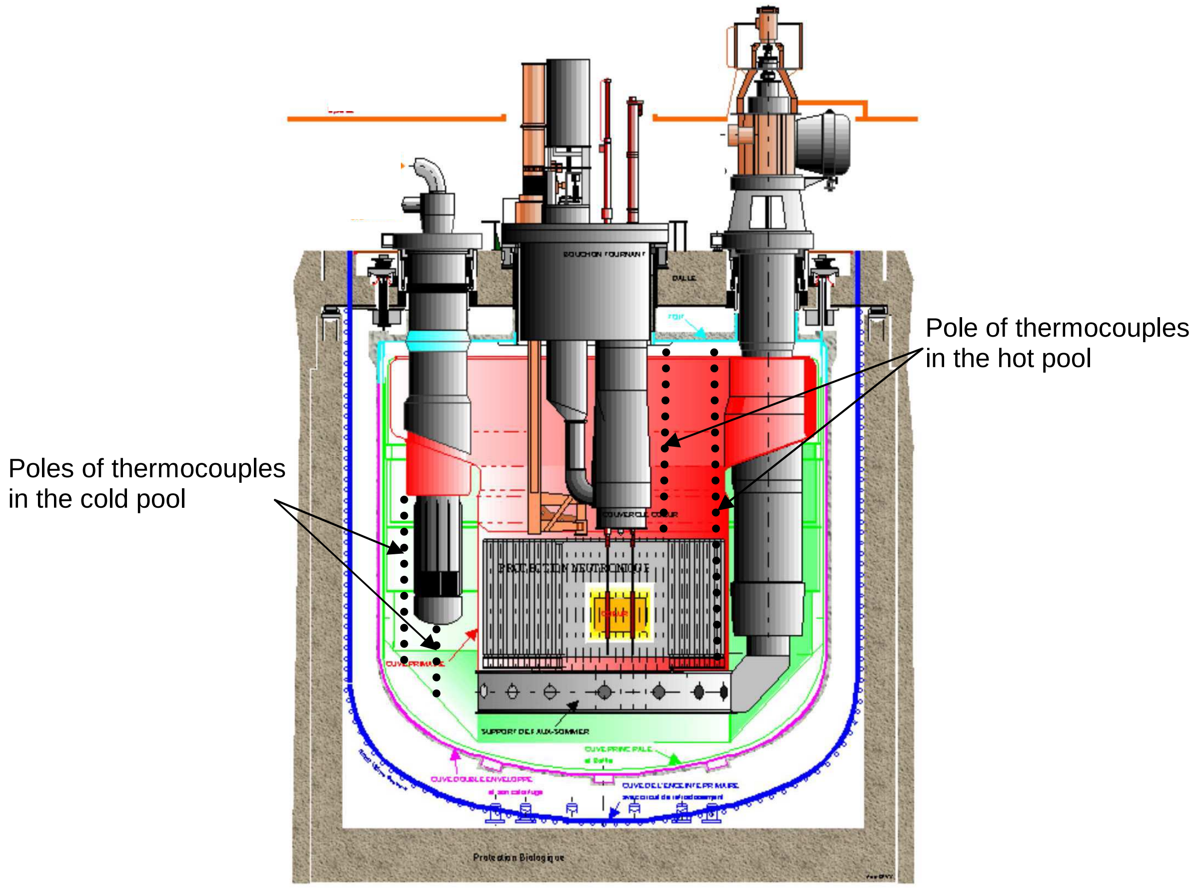

Fig. 2 Phenix primary circuit

The core is composed of three main regions:

- An inner core corresponding to a power of 60 MWt and a flow rate of 50% of the total core mass flow rate.
- An outer core corresponding to a power of 50 MWt and a flow rate of 40% of the total core mass flow rate.
- A fertile zone corresponding to a power of 10 MWt and a flow rate of 10% of the total core mass flow rate.

Reflector and shielding zones are surrounding the main core region, with very low power release and low flow rate. Six main control rods and one safety control rod allow the reactor power control.

The intermediate heat exchangers are made of vertical straight tubes of 5.3 m long and 12 mm of the inlet pipe diameter and 14 mm of the outlet pipe diameter, tied to an upper plate and a lower plate. Primary sodium is flowing in the outer part of the tubes (shell side), with an inlet window in the hot pool and an outlet window in the cold pool. Three pipes are connecting the three primary pumps to the diagrid. A main vessel cooling circuit is arranged from the bottom to the top of the main vessel. A small flow rate of sodium (less than 10 %) is taken at the core inlet to cool the main vessel and it is returned into the cold pool via a weir. At the core outlet, the flow rate through the above core structure is very low. The mean temperatures were approximately 360°C at core inlet and 435°C at core outlet for the 120 MWt power initial state.

Each secondary circuit depicted in Fig. 3 is composed mainly of two intermediate heat exchangers, a steam generator, a secondary pump and an expansion tank. Secondary sodium is

distributed in the tubes of the IHX through an inlet collector. Then, an outlet collector collects the sodium from the tubes and connects the IHX to an outlet pipe. The two outlet pipes from the two IHX of the same circuit join together and sodium is flowing towards the steam generator. The steam generator is composed of three main parts: the evaporator, the heater and the super-heater. The steam generator sodium tubes are nearly 0.2 m in diameter with seven inserted water tubes. Sodium is flowing downwards as water is flowing upwards. The whole steam generator is surrounded by a casing which can be opened if required to ensure an upward natural convection air cooling in case of a steam generator dry out. From the steam generator outlet, secondary sodium is flowing upwards to the expansion tank where is located the secondary pump. Then, sodium is flowing back to the two intermediate heat exchangers through a main pipe which separates in two smaller pipes before reaching the two IHX. The secondary flow rate is 690 kg/s and the temperature's range in the secondary circuit is from 320 °C to 520 °C.
The water/steam tertiary circuit depicted in Fig. 1 is not described as it is not useful for the final tests presented in this paper.

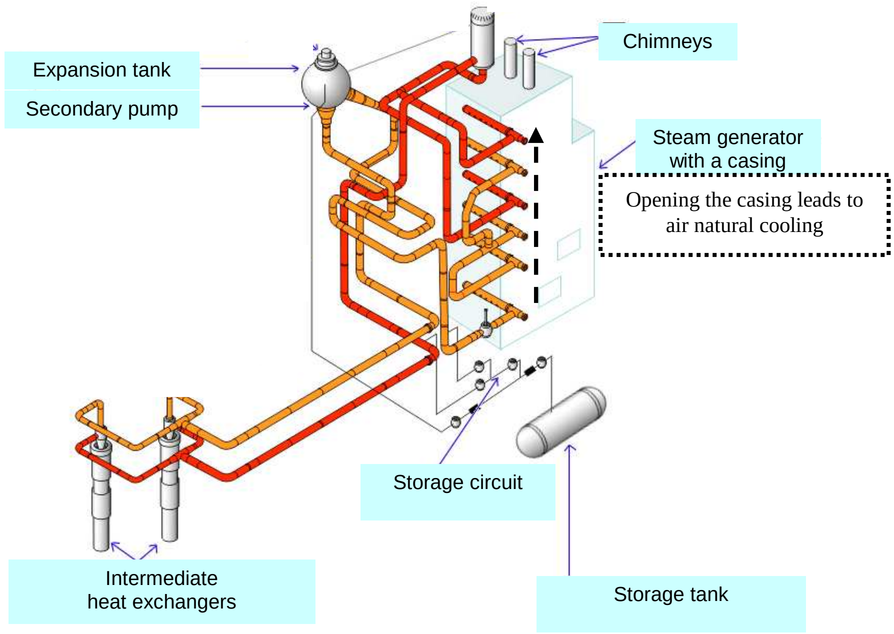

Fig. 3 Phenix secondary circuit

3. Instrumentation in Phenix reactor

The standard instrumentation available in the reactor and a special instrumentation for the final tests were used simultaneously.
The standard instrumentation consists in:

- Primary and secondary pumps speed which can be used to estimate the flow rate via the pumps characteristics.
- Secondary mass flow rate on each secondary loop.
- Core inlet temperature based on the measurement of the temperature at the outlet of the primary pumps.
- Specific pole of thermocouple giving the temperature at the primary pump inlet elevation.

- Fissile subassemblies outlet temperature measured a few centimetres above each fuel subassembly.
- Intermediate heat exchangers inlet and outlet temperatures on the primary side and on the secondary side.
- Steam generator inlet and outlet temperatures on the secondary side and on the tertiary side.
- Wall temperatures on the reactor vessel, on the baffle, on the counter-baffle and on the primary vessel.

A special instrumentation is also used for the final tests as shown in Fig. 2. Four poles of thermocouples are installed, two in the cold pool and two in the hot pool.

4. Natural convection test

The final natural convection test in Phenix reactor was performed on June 22-23, 2009.
The test scenario is described hereafter:

- The reactor is stabilized at a power of 120 MWt and a core inlet temperature of 360°C, with the three primary pumps in operation but one secondary circuit not operating.
- Manual dry out of the two steam generators to reduce the temperature difference between the primary and secondary sodium at the inlet of the intermediate heat exchanger (reference time 0). This part of the transient is consistent with an Unprotected Loss of Heat Sink transient (ULOHS transient).
- Manual scram 460 s later when this difference of temperature is lower than 15 °C
- Manual trip on the three primary pumps 8 s after the scram, with pumps speed decreasing to zero on their own inertia.
- Secondary pumps speed decreases automatically to 110 rpm in about one minute after the scram.
- Onset and development of natural convection in the primary circuit.

The natural convection test in the primary circuit is divided into three phases:

- Phase 1: loss of heat sink, scram and trip of primary pump.
- Phase 2: no significant heat sink in the secondary circuits, except the heat losses along the piping and through the casing of the steam generator.
- Phase 3: natural convection with an efficient heat sink by opening the casing at the bottom and at the top of the steam generator, which induces efficient air natural circulation in the casing.

The duration of phase 1 is about 7 minutes, the duration of the phase 2 is about 3 hours and the duration of phase 3 is about 4 hours.
The core power for the beginning of the test (from t=0s to t=500s) is given in Fig 4. The primary pump inlet temperature and core outlet temperatures are depicted respectively in Fig. 5 and Fig. 6
The first main important physical phenomena is the core power decrease from 120 MWt to about 50 MWt before the scram as shown in Fig. 4. This significant decrease of power is induced by the temperature increase at the primary pump inlet temperature (Fig. 5), and therefore at the core inlet, after the steam generators dry out. Several reactivity feedback effects are concerned with this core inlet temperature increase, such as the thermal expansion of the diagrid and the relative thermal expansion of the control rods. The well-known negative reactivity effect induced by the increase of temperature at the core inlet is clearly confirmed by the Phenix test. This is an important safety feature of sodium cooled fast reactors.

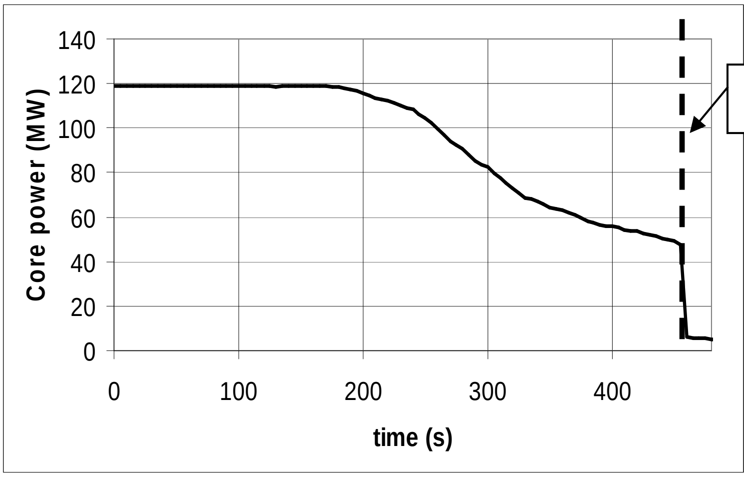

Fig. 4 : Natural convection test: core power

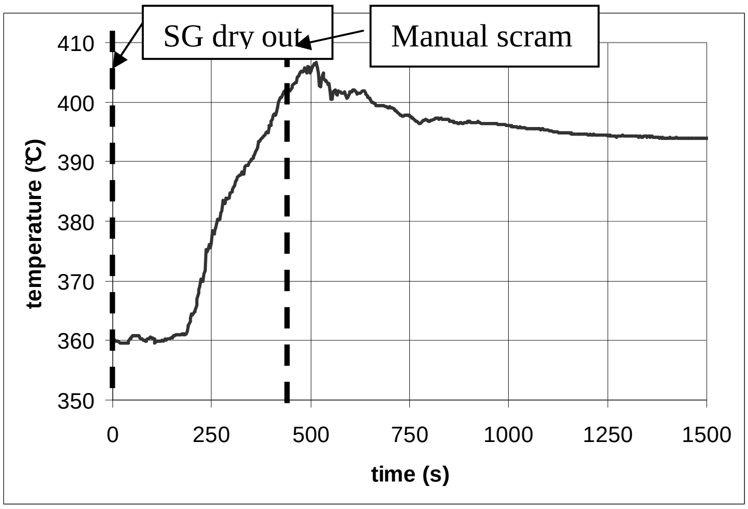

Fig. 5 : Pump inlet temperature

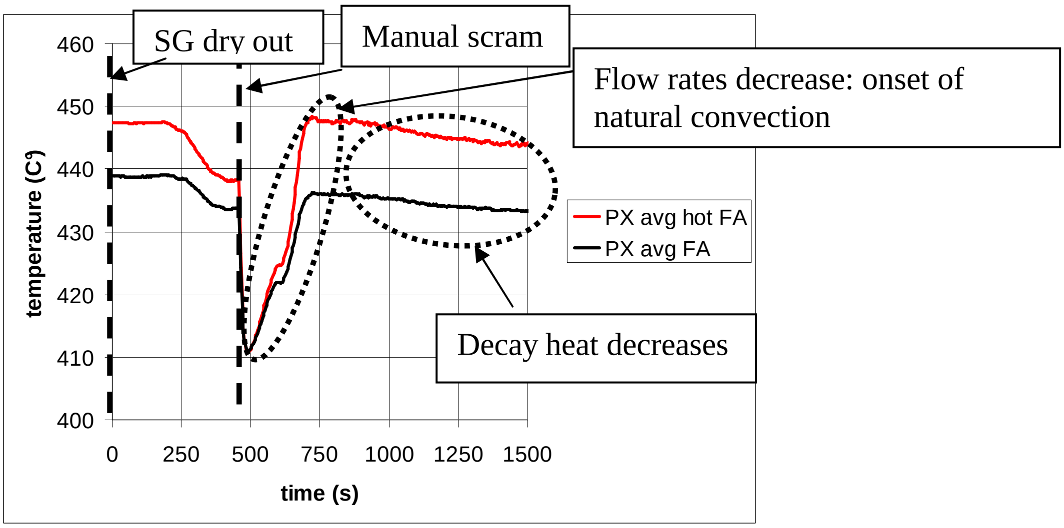

Fig. 6 : Core outlet temperature (average hot fuel assembly and average fuel assembly)

Just before the scram, the pump inlet temperature has increased from 360 °C to 405 °C and the core outlet temperatures has decreased of nearly 7°C due to the power decrease. Thirty seconds after the scram, the core power is lower than 0.5 MWt. The three primary pumps are tripped 8 seconds after the scram and the pump speed decreases to half-speed in 30 seconds and reaches zero within two minutes. The core outlet temperatures decrease sharply at the scram to 410 °C within one minute. Then, the outlet temperatures increases to 448 °C for the hottest fuel assemblies and 437°C for the others fuel assemblies in about four minutes corresponding to the onset of natural convection. About five minutes after the scram, a slight decrease of the core outlet temperature occurs, due to the core power decrease.

In the same period, the pump inlet temperature decreases slowly due to the thermal inertia of the hot and the cold plenum; the minimal value is reaches 395°C about 15 minutes after the beginning of the transient.

In longer term, as there is no real heat sink, except the reactor thermal inertia and the heat losses in the secondary circuits, the temperatures have a slow evolution (refer to Fig.7 and Fig. 8 between t=1500s to t=10 500s). This global reactor vessel behaviour lasts about three hours, until the end of the second phase.

Now, let us look at the long term behaviour corresponding to the end of the second phase. For the primary pump inlet temperature, after reaching the minimal value (cold shock effect), the temperature increase slowly to 400°C (refer to Fig. 7). Regarding to the core outlet temperatures, there are a slight decreases of nearly 10°C during this three hours: there is still a significant radial gradient of temperature at the fuel assemblies outlet.

At t=10500s, the third phase begins: the natural circulation of air in the casing generates significant heat losses which cool efficiently the secondary sodium. The IHX secondary inlet temperature decreases from 400 °C to 340 °C in about four hours. Consequently, the pump inlet temperature depicted in Fig. 7 decreases from 400 °C to 355 °C within the same time, as the core outlet temperatures presented in Fig. 8 decreases from 425 °C and 435°C, respectively to 390°C and 395°C.

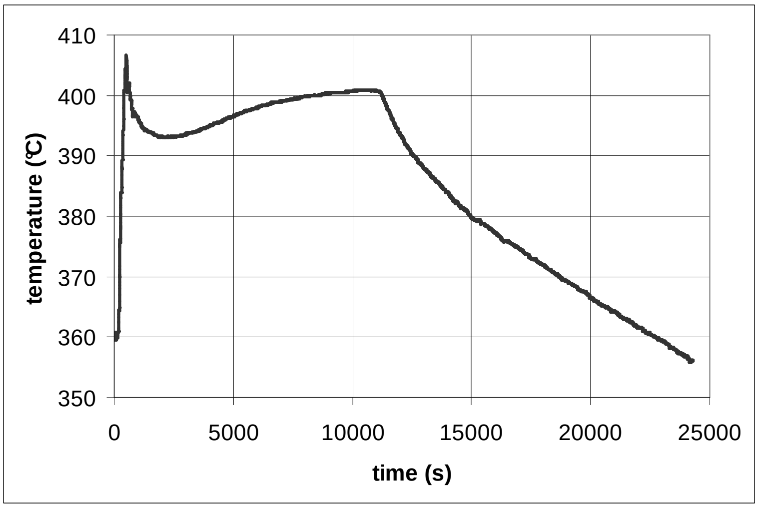

Fig. 7 : Pump inlet temperature (long term)

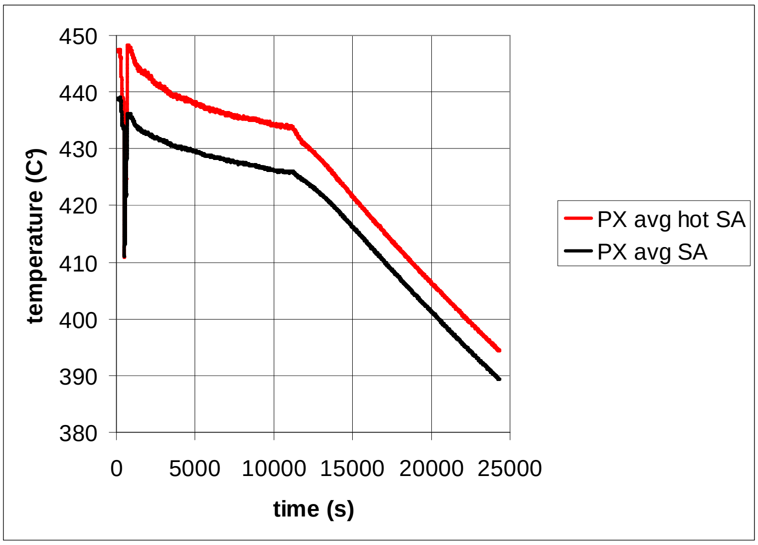

Fig. 8 : Core outlet temperatures (long term)

Remarks:
- The analyses of measurements from the pole of thermocouples in the hot and cold pools are still in progress. It will be presented in a future paper.
- We have checked that during the transient, there was no asymmetrical behaviour in the hot and cold pools. The fact that only two secondary loops are in operation during the test does not produce any significant asymmetrical behaviour.

5. CATHARE modelling

The CATHARE system code has been developed in collaboration between CEA, EDF, IRSN and AREVA-NP for more than 30 years. CATHARE is the reference code in France for the Pressurized Water Reactor (PWR) safety analysis. It has also been used for others light water reactor concepts (BWR, VVER) and for experimental reactors. In the frame of the Generation IV, CATHARE code has already integrated new developments in order to calculate Gas Cooled Reactors (GCR), Super Critical Light Water Reactors (SCLWR), Heavy Metal Reactor (lead and lead-bismuth coolant reactor), space engine applications (hydrogen and oxygen) and Sodium cooled Fast Reactors (SFR), ref. [3], [4]).
For Sodium cooled Fast Reactor calculations, several developments were added to the standard version of CATHARE code (ref. [3] and [4]), including thermodynamic and transport properties of sodium phases (liquid and vapor), specific heat transfer correlations adapted to sodium liquid metal, specific pressure drop correlations for fuel pin regions and updates of the neutronics point kinetics model (by taking into account specific neutronic feedback effects due to temperature variation, such as fuel, clad, wrapper tube, diagrid and control rod relative expansion effects). These specific options are used for Phenix calculation. The core power can be computed by the neutronics point kinetics model or issued as input data. Several SuperPhenix tests have already been used for CATHARE qualifications in forced convection regimes. The calculation of the natural convection test in Phenix reactor is a key milestone regarding to the whole qualification process of CATHARE code for SFR calculations. CATHARE qualification will include the calculation of other Phenix final tests performed in 2009 (ref. [5]). Moreover, Monju transient tests are also calculated to qualify the

code on a loop type reactor (ref. [4]), complementary to the French pool type reactors. The CATHARE modeling of Phenix reactor vessel is presented in Fig. 9.

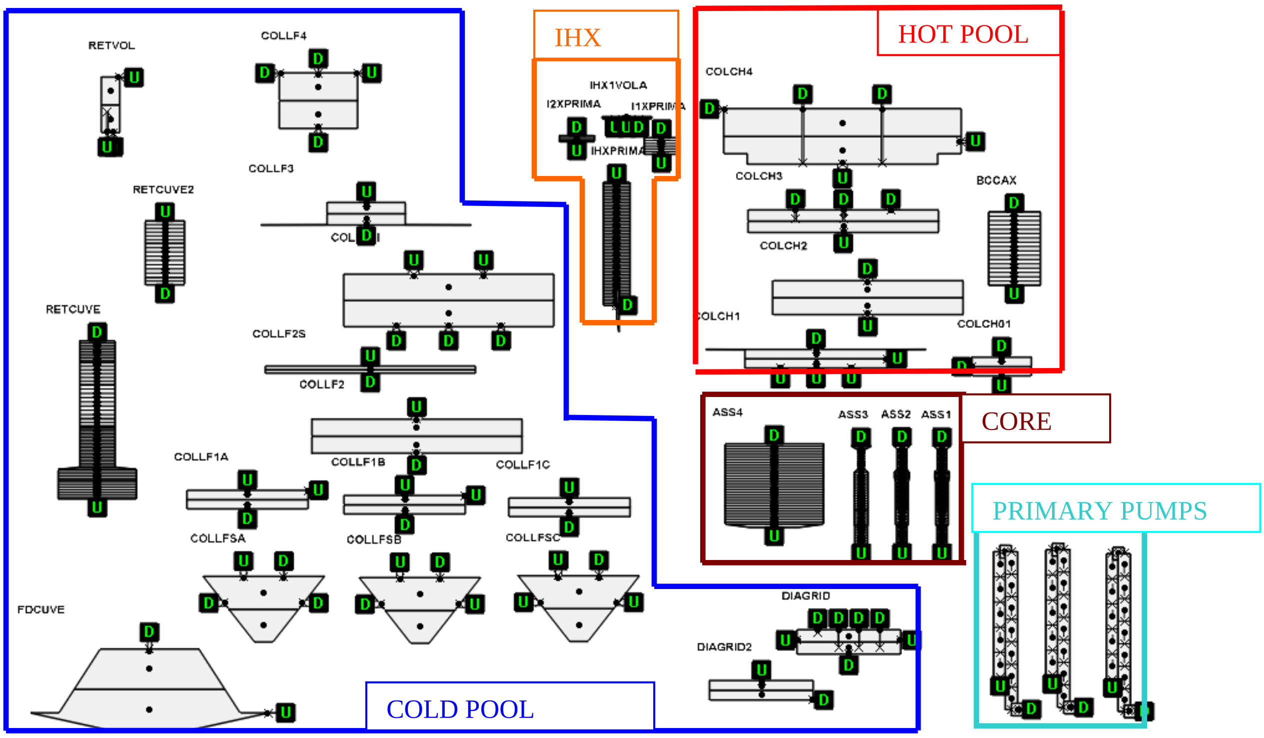

Fig. 9 : CATHARE modelling of Phenix primary circuit

The CATHARE input deck is briefly described hereafter:

- The core is modeled with four parallel 1D pipes, one for the inner core, one for the outer core, one for the blanket and one for the reflector and shielding region. A specific friction coefficient law (Pontier's law) is used in the fuel pin zones for the fuel assemblies.
- The core outlet region is modeled by a 0D module with two connections to take into account the hydraulic path of the hot sodium through the Upper Core Structure and the direct path to the IHX primary inlet window.
- The hot pool is modeled with several connected 0D volumes to roughly evaluate thermal stratification during the transient conditions; heat transfer with the cold pool through the "redan" structure is taken into account.
- The pressure of the cover gas is regulated.
- The hydraulic path trough the Upper Core Structure is modeled by a 1D axial connecting the hot core outlet region to the upper part of the hot pool.
- The IHX is modeled with two 1D pipes, one simulating the primary side (shell side) and the other one the secondary side (tube side), with heat transfer between both sides.
- The cold pool is modeled with several connected 0D volumes to be able to evaluate thermal stratification; the cold pool is divided into three sectors to deal with asymmetrical conditions.
- The reactor vessel cooling system is modeled by a 1D pipe, a 0D volume and a 1D pipe.
- The diagrid, the strong back and the lower plenum are modeled with 0D volumes.
- The primary pumps are modeled by three 1D pipes and pump characteristics are input data.

With the use of 0D volumes at the top of the hot pool, the cold pool and the reactor vessel cooling system, the code computes the three free levels in the upper part of the reactor vessel.
The secondary circuits of the reactor are not modeled for the present calculation, except the secondary side of the IHX modeled by a 1D pipe. Boundary conditions from the secondary circuits are input data given at the IHX secondary inlet point, namely secondary flow rate and temperature as a function of time (therefore the heat losses on the secondary loops and the heat exchange when the steam generator casing are open is not part of the CATHARE calculation). The other boundary conditions are the core power evolution and the primary pumps speed evolution after the scram.

6. CATHARE results

6.1 Nominal state 120MW

The following table gives the comparison between Phenix values and CATHARE code calculation for the nominal state (*the input data are in italic*).

Parameter	**Phenix data**	**CATHARE results**
Core power	*120 MWt*	*120 MWt*
Primary pump speed	*350 rpm*	*350 rpm*
Core inlet temperature	358°C	359.4°C
Primary IHX inlet temperature	435.4°C	431.4°C
Primary pump inlet temperature	*360.0°C*	*360.0°C*
Secondary IHX mass flow rate	375.4 kg/s	374.0 kg/s
Secondary IHX inlet temperature	*307.0°C*	*307.0°C*
Secondary IHX outlet temperature	430.5°C	430.4°C
Hot pool level	*1.88m*	*1.88m*
Reactor cooling system level	1.65m	1.65m

Table 1 : 120 MWt initial state: PHENIX data and CATHARE results

We have checked that the differences between plant data and code predictions remain below the sensors' uncertainties. Therefore, we can point out that there is a good agreement between Phenix data and CATHARE code predictions for the nominal state. The same comparison has been done successfully with the full power nominal state.

6.2 Natural convection transient

6.2.1. Phase 1: Unprotected part of the transient

The beginning of the transient consists in an unprotected loss of the heat sink (ULOHS). In the section 4, we have already explained in detail the reactivity feedbacks conducting to the core power sharp decrease. Fig. 10 shows the evolution of the core power for two CATHARE calculations:

- CATHARE calculation with a simplified input deck ('CATHARE small' i.e. scope of modelling reduced to diagrid, core channels, one mixing area at core outlet and neutronics point kinetic model with temporal evolutions of primary pump mass flow rates and temperatures), the code prediction is consistent with the Phenix data
- CATHARE calculation with the extended input deck ('CATHARE full' i.e. modelling of the whole reactor vessel, the simplified secondary sides and neutronics point kinetics model), the decrease of core power predicted by the code is slightly underestimated (60MWt instead of 50MWt), due to an inaccurate prediction of the

core inlet temperature rise. In fact, with the main assumptions of system code, the increase of inlet core temperature is delayed: the real thermal inertia between IHX primary outlet and core inlet is not precisely described.

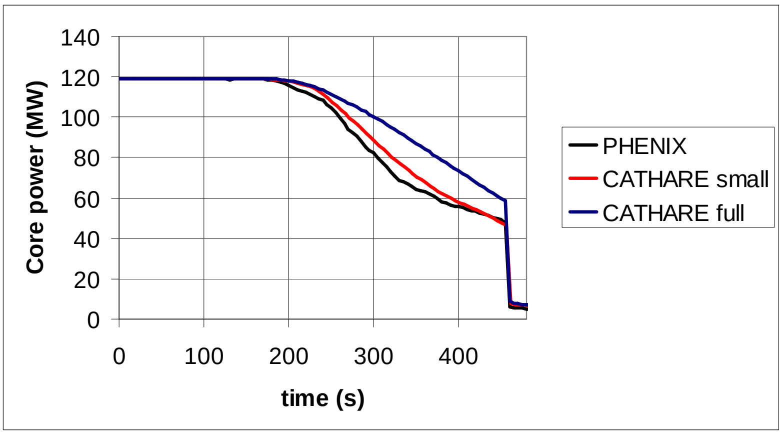

Fig. 10: Unprotected part of the transient: core power before scram

Nevertheless, the figure shows that the neutronics point kinetics model in CATHARE can predict properly the feed-back effects during an ULOHS transient. It also shows the difficulty for system code with a simplified model to take into account properly thermal inertia in a complicated configuration.

6.2.2. Global transient

For the long term transient, the core power is used as input data. The figures Fig.11 to Fig. 15 give some comparison between Phenix plant data and CATHARE computation, for short and long terms, for the primary pump inlet temperature (Fig.11 and Fig.12), for the core outlet temperatures (Fig.13 and Fig.14) and for the IHX primary inlet and outlet temperatures (Fig.15 and Fig.16). The total core mass flow rate is calculated by CATHARE, but there is no data available for Phenix (Fig.17).
Through the comparison between experimental data and CATHARE results, we can point out:

- First phase (unprotected loss of heat sink): the main event during this phase is the heating of the lower part of the reactor vessel, due to the dry out of the Steam Generator. At the core outlet, we can observe first a cold shock (t=500s, scram effect) and a hot shock (t=750s, pump trip effect). CATHARE code slightly under-estimates the primary pump inlet temperature heating due to the modelling of the hydraulic path between IHX primary outlet and primary pump inlet (see Fig. 11): the maximal temperature after scram is 405°C for Phenix data and 400°C for CATHARE calculation. Regarding to the core outlet temperature, in the CATHARE calculation the average temperature uses a full mixing hypothesis, whereas in the Phenix test, there is a significant radial gradient of temperature between the hottest and the coldest fuel assemblies.
- Second phase (natural convection on secondary loop heat losses): during this phase, due to the relatively low core ΔT at initial state (only 80°C, whereas it could go up to 150°C at full power), the driven force for natural convection is rather low. During the onset of natural convection in the core, CATHARE predicts some downward flow in the blanket and in the reflector and shielding channels. In the long term, there is a homogenisation of the temperature field in the reactor vessel. The comparison for core outlet temperature (Fig. 13) is a challenging one in the CATHARE calculation, due to the full mixing hypothesis at the core outlet region, the thermal inertia is much higher and the cold shock occurs with a

longer delay (t=700s for Phenix data and t=1150s for CATHARE calculation). The CATHARE calculation can not properly predict the sharp decrease of temperature occurring at the elevation of the IHX primary outlet window refer to Fig. 15), when the scram and the primary pumps trip occur.. This fall of temperature at the IHX outlet window could be explained by the sodium buoyancy during the primary pumps trip and the fact that the sensor becomes surrounded by a colder sodium plug: analysis are in progress to check this assumption. Nevertheless, the prediction of IHX inlet temperature in the primary side is consistent with Phenix data (Fig. 15).

- Third phase (natural convection with efficient heat sink): during this phase, as long as, the SG is an efficient heat sink, there is an efficient cooling of the reactor vessel. The natural convection mass flow rate increases drastically (+50%). In the CATHARE calculation, due to the modelling choice, the effect of the more efficient heat sink occurs later than in Phenix test (Fig. 12: around t=10 500s for Phenix data and t=14 000s for CATHARE calculation). In long term, regarding the temperature cooling rate the CATHARE predictions are consistent with Phenix data for the primary pump inlet and IHX primary outlet temperatures (Fig. 14 and Fig. 16), whereas CATHARE code could underestimate the cooling rate at IHX primary inlet (Fig. 16).

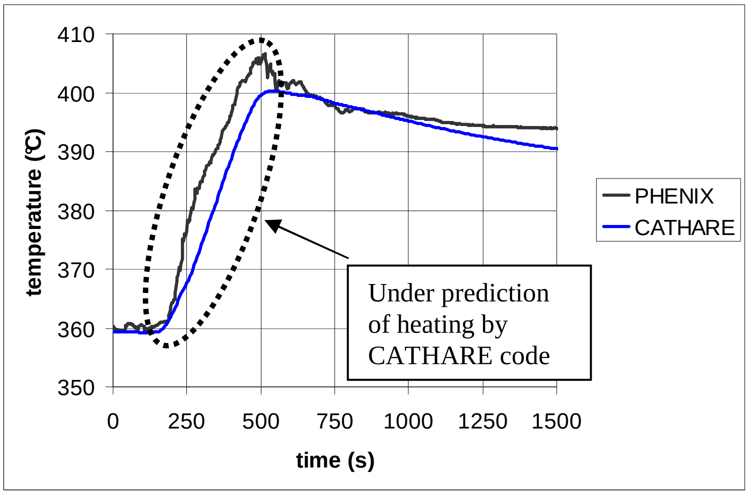

Fig. 11: Primary pump inlet temperature (short term)

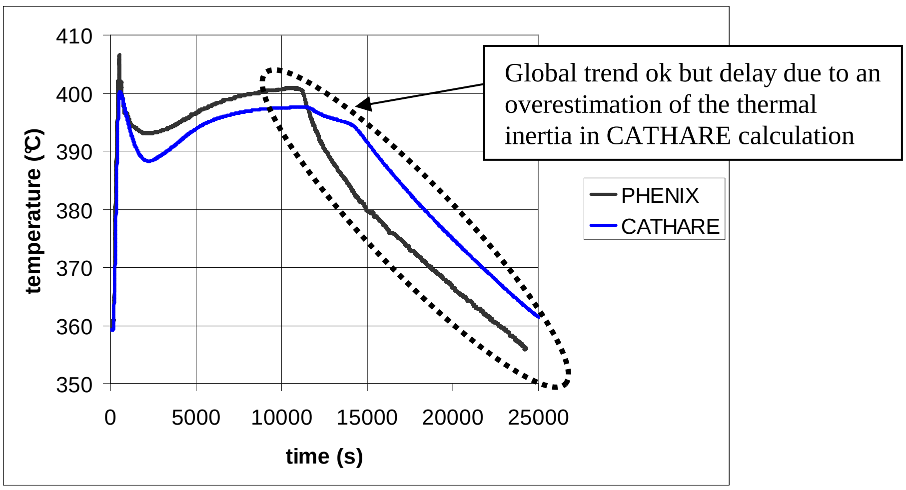

Fig. 12 : Primary pump inlet temperature (long term)

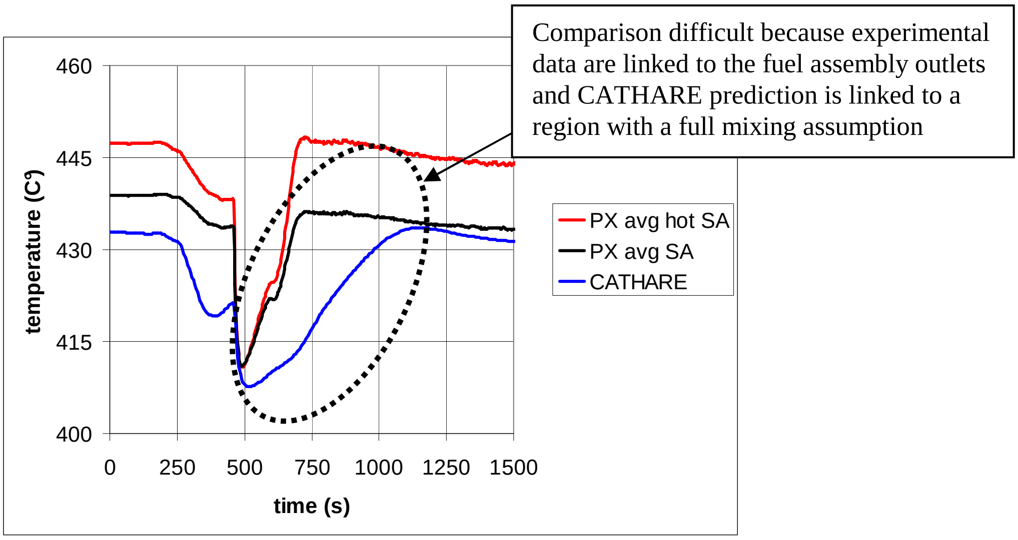

Fig. 13 : Core outlet temperatures (short term and long term)

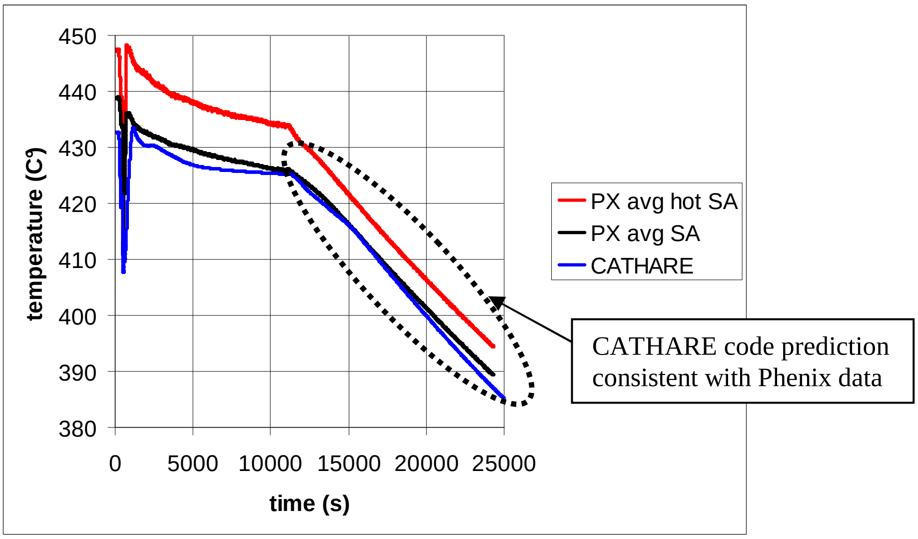

Fig. 14 : Core outlet temperatures (long term)

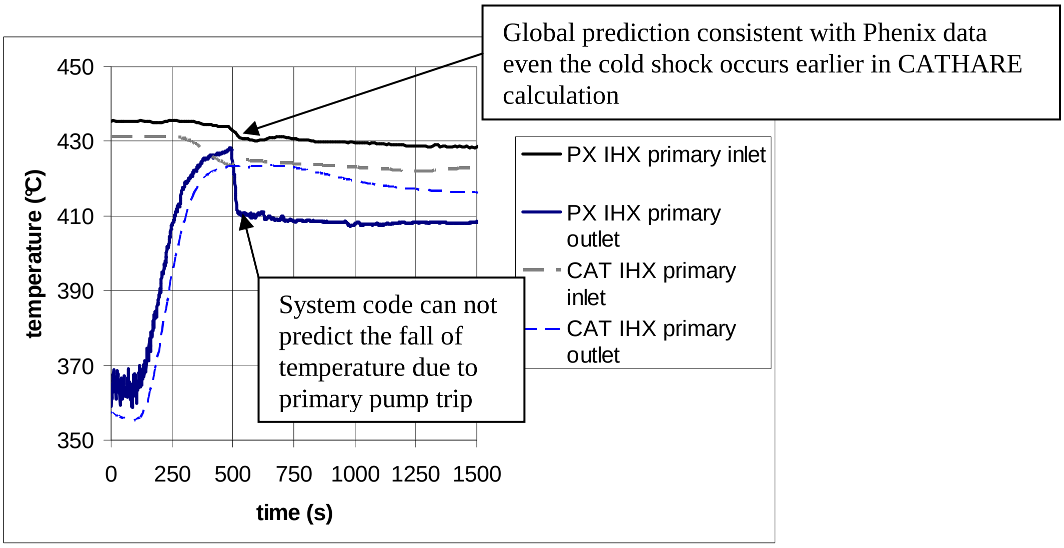

Fig. 15 : IHX inlet and outlet temperatures (short term)

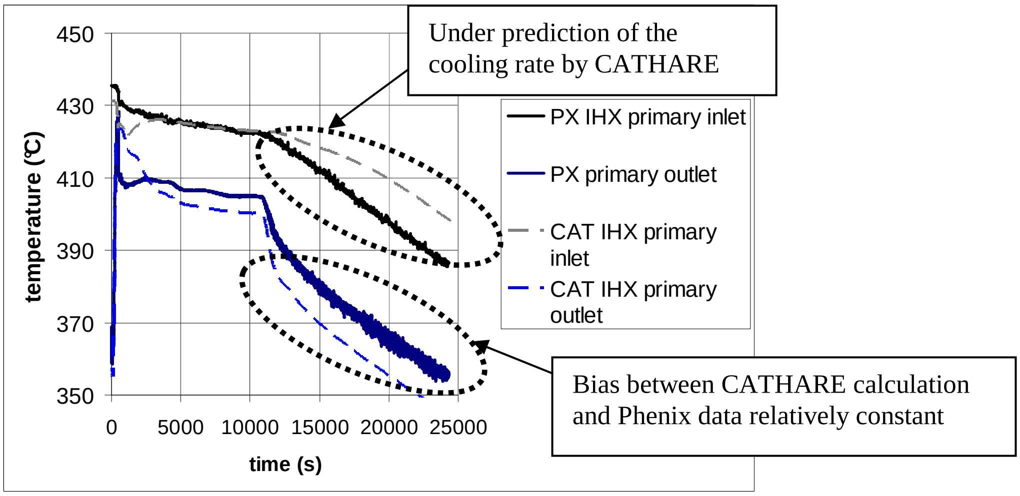

Fig. 16 : IHX inlet and outlet temperatures (long term)

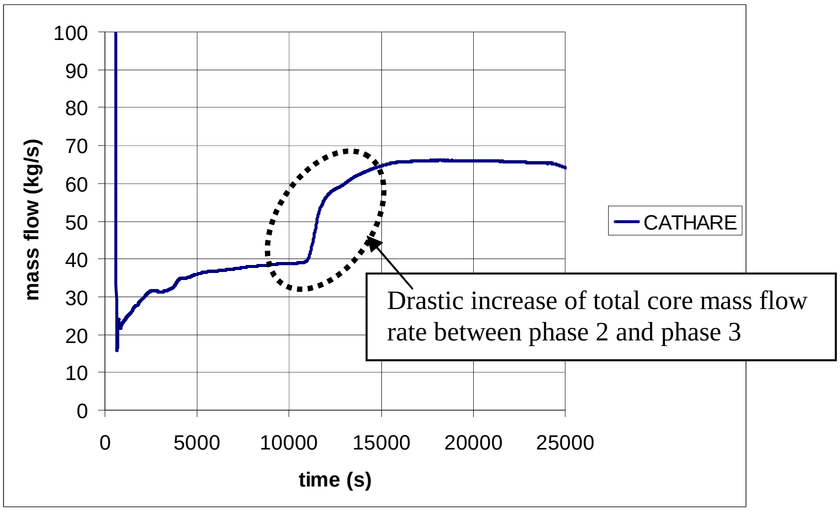

Fig. 17 : Core total mass flow rate (whole transient)

6.3 Discussion and sensitivity tests

The CATHARE code has shown it capacity to represent the different phases of this complex natural convection transient, with some identified discrepancies. Several sensitivity calculations have been made in order to reduce the discrepancies between code prediction and Phenix data:

- Modification of the friction law for fuel pin area (use of Rehme's correlation instead of Pontier's law)
- Modification of lateral heat exchange (increase or decrease of lateral heat exchange between hot an cold pools and cold pool and reactor vessel cooling system) :
    - In the second phase (natural convection on secondary loops heat losses), the increase or decrease of the lateral heat exchanges plays a key role for the natural convection total mass flow rate in the core.
    - In the third phase (natural convection with efficient heat sink), the increase or decrease of lateral heat exchanges has low impact on the total mass flow rate and the cooling rate.
- Modification of modelling of the cold plenum (with a direct path from the IHX outlet window to the primary pump suction head within the same module):

- In the first phase, the primary pump inlet temperature increase is slightly better predicted.
- In the second phase, the slow warm up of the lower part of the cold plenum is under-predicted
- In the third phase, the global cooling of the reactor vessel remains nearly within the same order compare to the reference CATHARE calculation.

Up to now, no key parameter has emerged from these sensitivity tests. The key point seems to be the difficulties for system code, with basic assumptions, to represent the complex and 3D phenomena involved during transients in large plena, such as hot and cold pools. In such geometry, the local buoyancy effect plays a key role: depending on the transient phases, the hydraulic paths can be different.

These system code limitations are well known by the thermal-hydraulics code users. A new track to solve these limitations could be to couple system code to CFD code, such as CATHARE – Trio-U coupling in progress at CEA.

7. Conclusion

Before Phenix reactor was definitively stopped in 2009, final tests were performed to provide valuable data for the development of future sodium cooled fast reactors as the so-called Astrid prototype in France (ref. [5]). A special instrumentation depicted in the paper was installed in the reactor before the tests, especially new poles of thermocouples in the hot and cold pools. The paper deals with the natural convection test in the primary circuit. The natural convection test consisted in a dry out of the steam generators (unprotected loss of heat sink) followed by a scram a few minutes later, immediately followed by a trip of the three primary pumps. The test duration after the pump trip was about seven hours. The test is used for the qualification of the CATHARE system code developed at CEA in collaboration with AREVA, EDF and IRSN.

The natural convection test has shown the significant effect of the increase of temperature due to the steam generator dry out on the core reactivity, with a decrease of the core power from 120 MWt to about 50 MWt before the scram and the pump trip. After the primary pumps trip, natural convection is developed within about five minutes. The first phase without significant heat sink except thermal inertia and secondary heat losses lasted about 3 hours and it shows a slight decrease of the core outlet temperature. The second phase with natural air cooling in the steam generator casing lasted about 4 hours and it showed a higher decrease of the core outlet temperature.

The analyses of measurements from the pole of thermocouples in the hot and cold pools are still in progress. It will be presented in a future paper. We have checked that no asymmetrical behaviour in the hot and cold pools occurs, although only two secondary loops are in operation during the test. A specific asymmetrical test (trip on one secondary pump, ref. [5]) has been identified to check the capability of CATHARE code to deal with asymmetrical transients.

The CATHARE code can predict the initial state of the Phenix reactor. The point kinetics model is able to predict properly the neutronics feedbacks and the core power decrease during the beginning of a ULOHS transient. But within the whole system calculation, the thermal inertia between IHX outlet and primary pump inlet, which is quite complex due to the specific cold pool geometry in Phenix reactor, is underestimated. As a consequence of the underestimation, the core power is over-predicted (+20% at scram). Such kind of system code limitations are well known by the users for transient analysis.

If the global prediction of CATHARE code is consistent with the Phenix data, we can point out the difficulties for a system code to take into account the modification of hydraulic paths during the transient in complex geometry such as cold and hot pools.
This kind of physical problems, where the 3D phenomena are significant, requires more complex codes than system codes: the coupling of system code and CFD code (CATHARE and Trio-U) is in progress at CEA.

Unity

°C	Celsius degree
kg/s	Mass flow rate : kilogram par second
m	Meter
rpm	Rotation speed: round par minute
MWe	Electrical Mega Watt
MWt	Core thermal Mega Watt

Glossary

avg	average
AREVA-NP	Areva Nuclear Power
BWR	Boiling Water Reactor
CEA	Commissariat à l'Energie Atomique et aux Energies Alternatives
DHR	Decay Heat Removal
EDF	Electricité de France
FA	Fuel Assembly
HLMR	Heavy Metal Liquid reactor
CAT	CATHARE
CEA	Commissariat à l'Energie Nucléaire et aux Energies Alternatives
CRP	Common Research Project
GCFR	Gas Cooled Fast Reactor
IAEA	International Atomic Energy Agency
JAEA	Japanese Atomic Energy Agency
IHX	Intermediate Heat eXchanger
IRSN	Institut de Radioprotection et de Sûreté Nucléaire
PCRD	Programme-Cadre de Recherche et de Développement (Framework programs)
PWR	Pressurized Water Reactor
PX	Phenix
SCRAM	Safety Control Rod Axe Man
SEPTEN	Service Etudes et Projets Thermiques et Nucléaires
SG	Steam Generator
SFR	Sodium Fast Reactor
SPX	SuperPhenix
SCLWR	Super Critical Light Water Reactor
THINS	Thermal Hydraulics for Innovative Nuclear Systems
ULOHS	Unprotected Loss of Heat Sink
ULOF	Unprotected Loss of Flow
VVER	Vodo-Vodní Energetický Reaktor

Acknowledgment

The authors would like to acknowledge all the Phenix reactor staff which prepared and performed the final tests, including this natural convection test. Another acknowledgement is for the IAEA, to have retained this transient for a Common Research Project (CRP). And finally, the last acknowledgement is for the THINS project in the frame of the 7th PCRD for their support and financial funding in the test analysis.

References

[1] J.L. CARBONNIER (CEA) – J.-M. DELBECQ (EDF) – R. ASSEDO (AREVA NP), “French Program for Sodium Fast Reactors”, Proceedings of the ICCAP 2007 conference, Nice, France, May 2007

[2] Laurent MARTIN, Didier DALL’AVAL, Joël GUIDEZ (CEA) – Jean GUIHARD, Jean-François SAUVAGE (EDF), Revue Générale du Nucléaire, ‘THE PHENIX REACTOR – Assessment of 35 Year’ Operation’, ‘Phenix: 35 Years’ Operation 1974-2009’, REVUE GENERALE NUCLEAIRE, Janvier-Février 2009

[3] Geneviève GEFRRAYE et all (CEA), ‘CATHARE_2v25_2: a single version of code system for multi-purpose applications ‘, Proceeding of the NURETH 13, Kanazawa, Japan, 2009

[4] David PIALLA, Denis TENCHINE, Alain BERJON (CEA) – Fumiaki YAMADA (JAEA), ‘MONJU calculations with French CATHARE system code’, Proceedings of the Annual Meeting of the Atomic Energy Society of Japan, Tokyo, Japan, 2009

[5] A. VASILE, B. FONTAINE. M. VANIER, P. GAUTHE, V. PASCAL, G. PRULHIERE, P. JAECKI (CEA Cadarache) - D. TENCHINE (CEA Grenoble) - L. MARTIN (Centrale Phenix – CEA Marcoule) – France - J.F. SAUVAGE (EDF / SEPTEN) - R. DUPRAZ, A. WOAYE-HUNE (AREVA NP), “The PHENIX final tests”, Proceedings of the ENC2010, Barcelona, Spain, May 30 to June 2, 2010