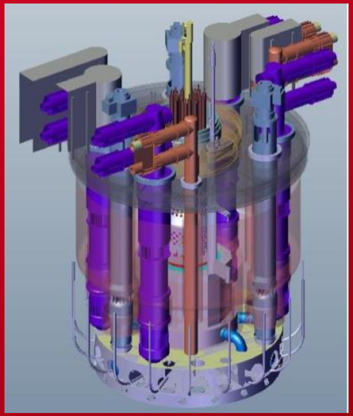
DE LA RECHERCHE À L'INDUSTRIE



Joint ICTP-IAEA Workshop on "Physics and Technology of Innovative Nuclear Energy Systems for Sustainable Development"

Trieste Italy

2022 December 12th – 17th

INTERACTION COOLANT-MATERIALS

Christian Latgé, Scientific Advisor

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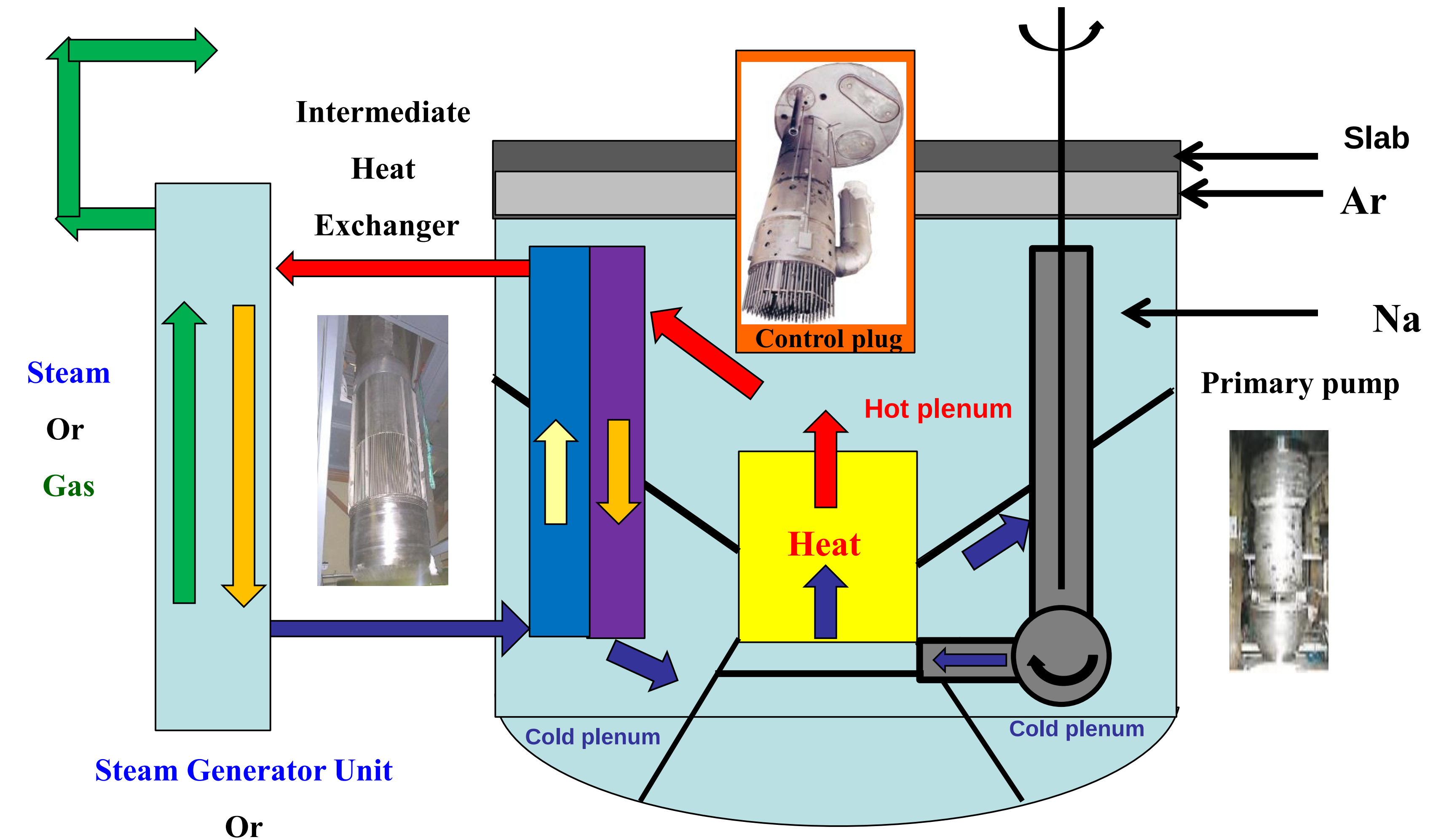
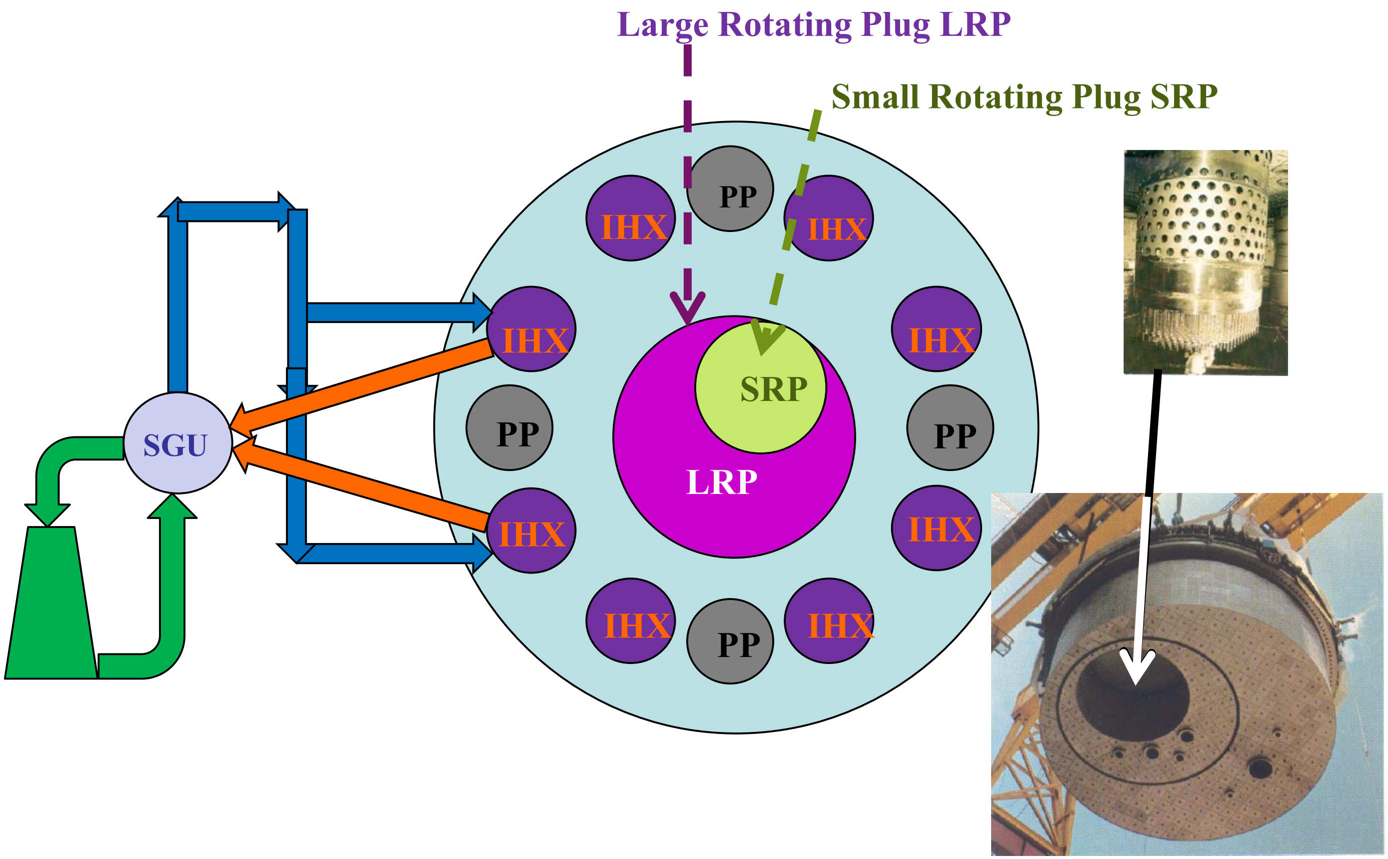
PRIMARY CIRCUIT OF SFR (POOL CONCEPT)

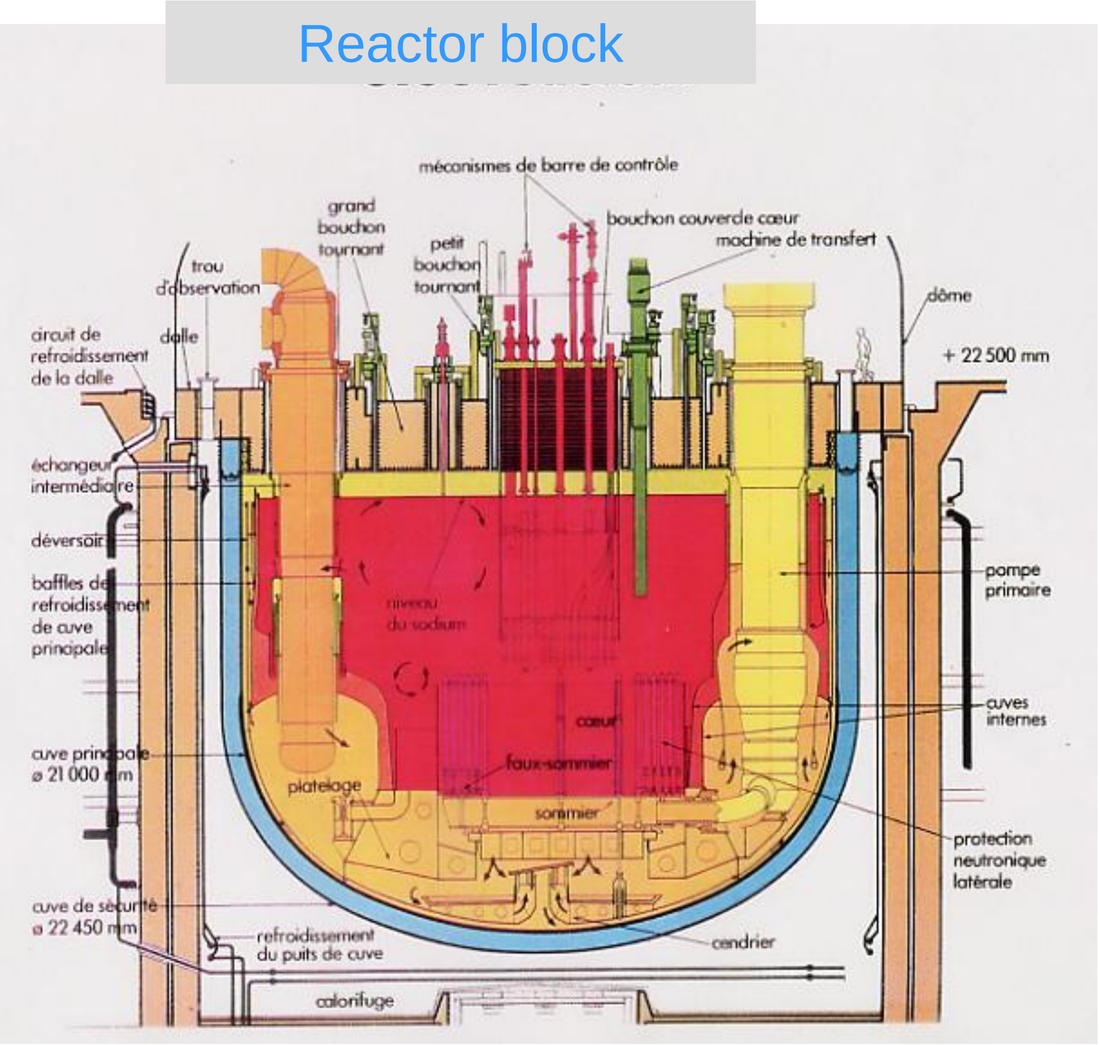


ROTATING PLUGS



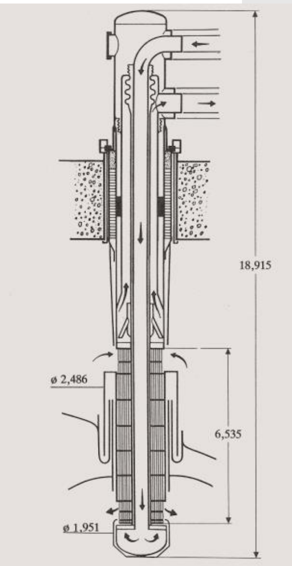
SFR: Superphénix plant brief description 1/2

- 1 200 MWe (3 000 MWth). 2 turbines
- Pool type : ∅21 m primary vessel
- 364 $UPuO_2$ hexagonal fuel elements
- 24 control rods (21 + 3)
- 4 primary mechanical Na pumps
- 8 straigh tube heat exchangers
- 2 integrated Na purification units
- 2 rotating plugs
- Core catcher (severe accident)
- 4 intermediate sodium circuits
- 4 secondary mechanical Na pumps
- 4 helicoil tube steam generator units



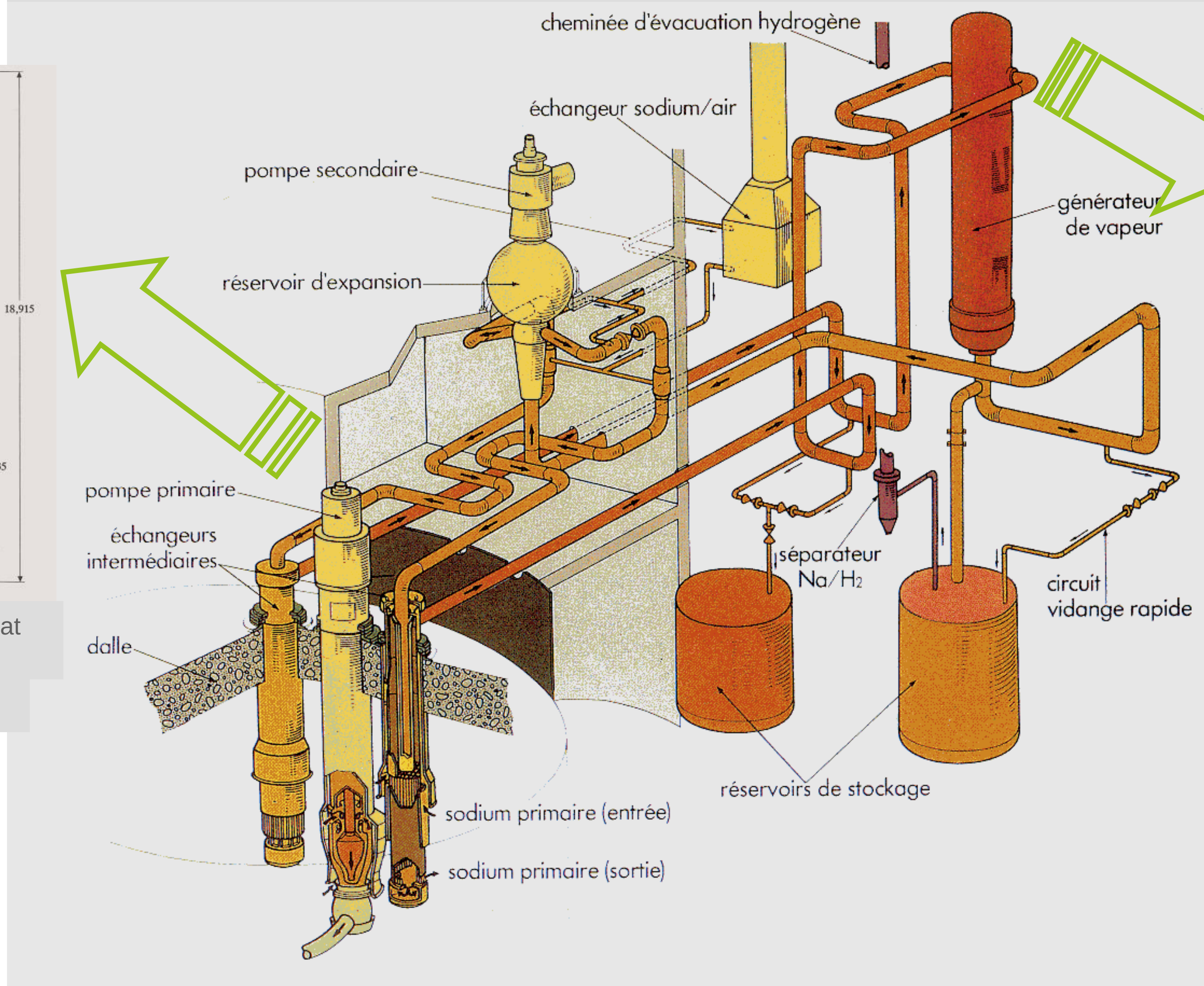
Superphénix plant brief description 2/2

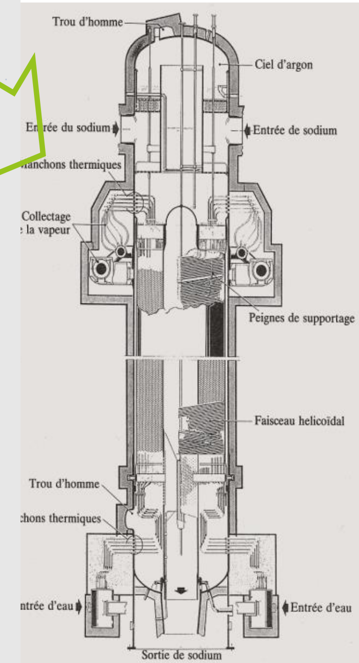
Secondary circuits



Intermediate Heat Exchanger

375 MW





Steam Generator

750 MW

DE LA RECHERCHE À L'INDUSTRIE

THE ASTRID OBJECTIVES

- ➔ **Technological demonstration reactor (*a step before a First Of A Kind)***
- ➔ **Integrating French and international SFRs feedback**
- ➔ **A GEN IV system**

<u>Safety</u> :

- **Level at least equivalent to GEN III systems**
- **Progresses on Na reactors specificities**
- **Integrating FUKUSHIMA accident feedback**
- **Robustness of safety demonstration**

<u>Durability</u>

- **Need of Fast Breeder Reactors and a closed cycle**
- **Pu multi recycling to preserve natural resources**
- **The use of natural depleted uranium in France by FBRs allow producing electricity for few thousands of years**

<u>Operability</u> :

- **Load factor of 80% or more after first "learning" years**
- **Significant progress concerning In Service Inspection & Repair (ISIR)**

<u>Ultimate wastes transmutation</u> :

- **Realization of demonstrations on minor actinides transmutation according to June 28, 2006 French Act on Wastes Management**

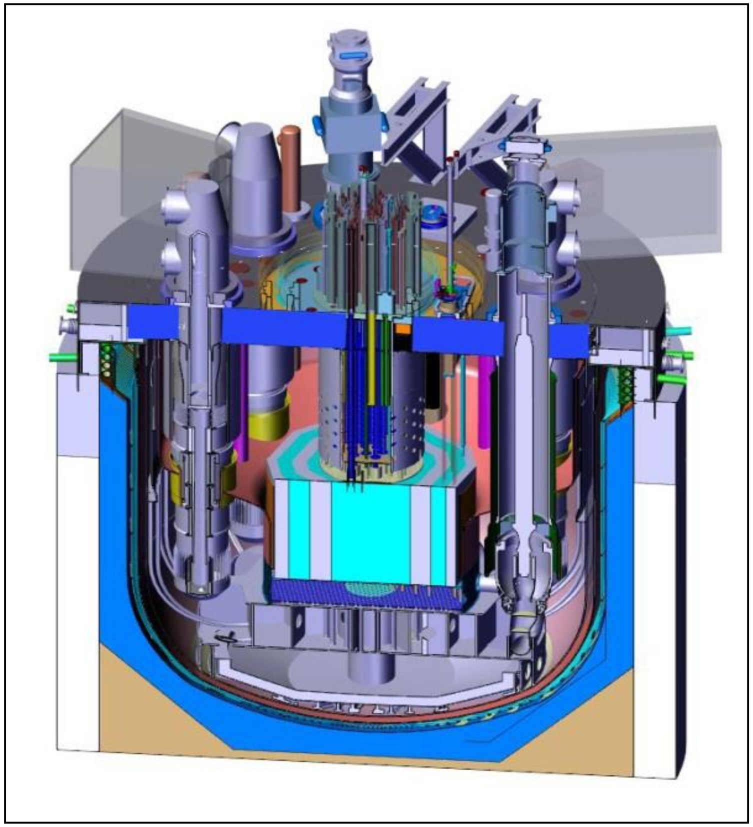
<u>A mastered investment cost</u>

<u>Non proliferation warranty</u>

- ➔ **Irradiation services and options test**

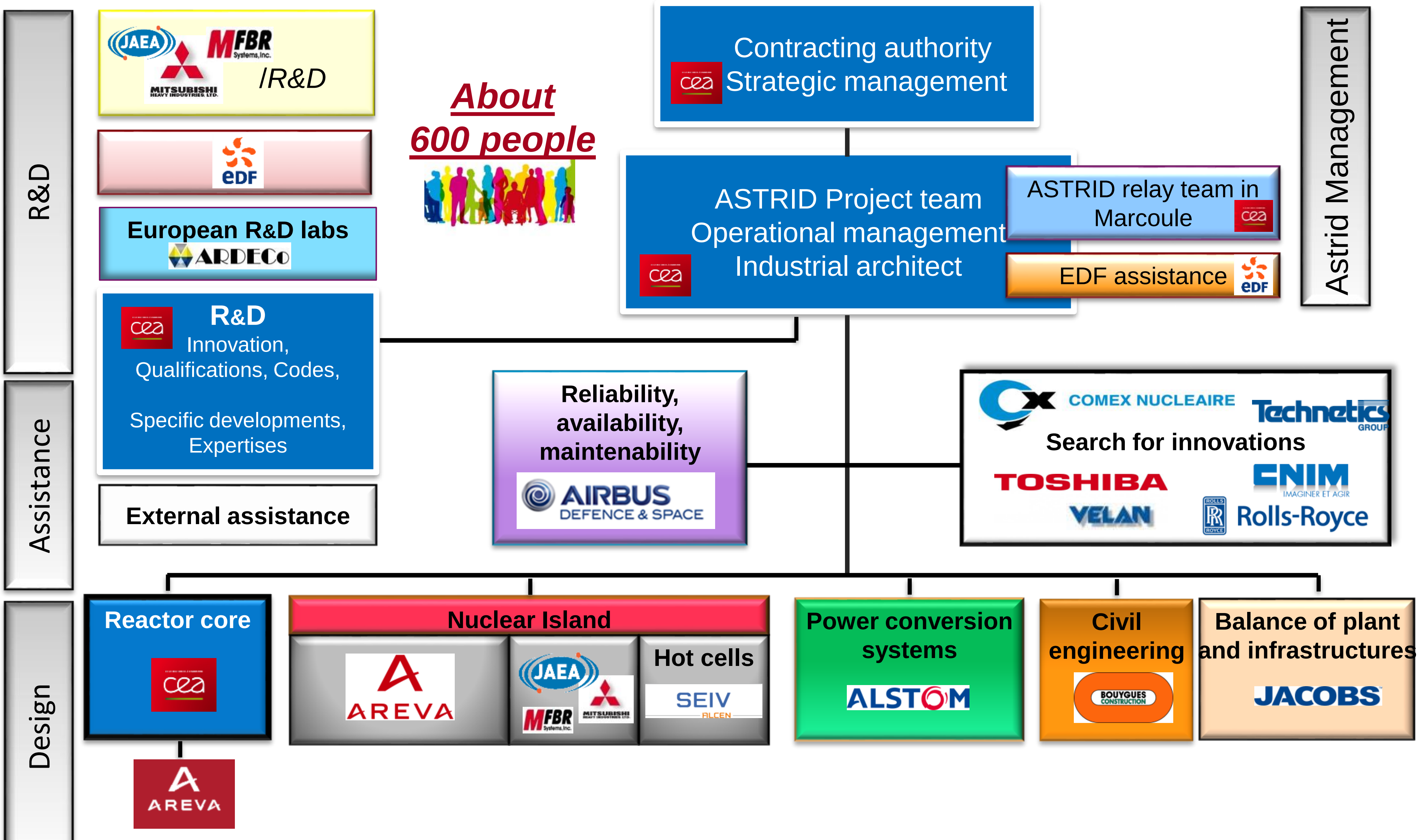
ASTRID MAIN TECHNICAL CHOICES

- **1500 thMW - ~600 eMW**
- **Pool type reactor**
- **With an intermediate sodium circuit**
- **CFV core (low sodium void worth)**
- **Oxide fuel UO_2-PuO_2**
- **Preliminary strategy for severe accidents (internal core catcher…)**
- **Diversified decay heat removal systems**
- **Fuel handling in gas, internal storage**
- **Conical "redan" inner vessel adopted**
- **Reference lay-out (end of conceptual design phase):**
  - ✓ ***3 primary pumps***
  - ✓ ***4 intermediate heat exchangers***
  - ✓ ***4 secondary circuits***
  - ✓ ***5 decay heat removal circuits***



Experimental capabilities : to contribute to the qualification of transmutation, fertile or burner subassemblies

ASTRID PROJECT ORGANISATION



INDUSTRIAL PARTNERS CONTRIBUTIONS

AREVA

AIRBUS DEFENCE & SPACE

EDF

BOUYGUES CONSTRUCTION

TOSHIBA

COMEX NUCLEAIRE

Rolls-Royce

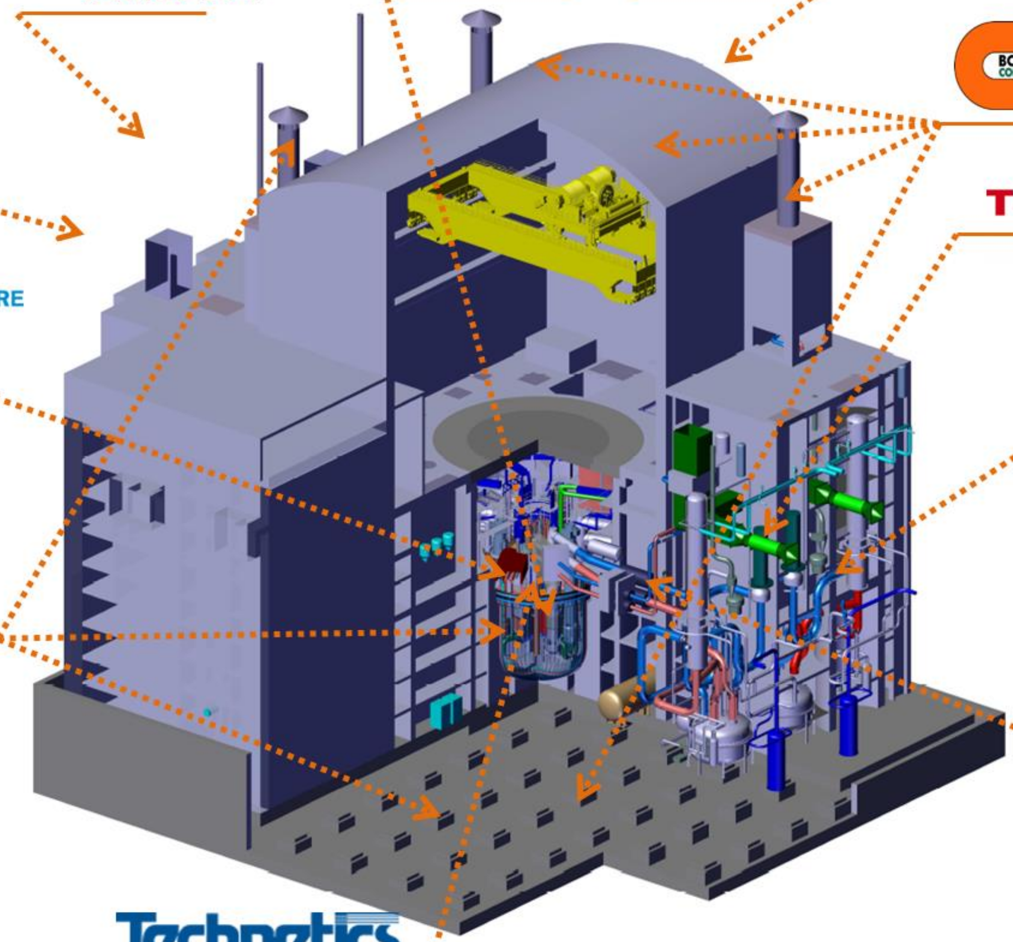
SEIV ALCEN

JAEA

MITSUBISHI HEAVY INDUSTRIES, LTD.

MFBR Systems, Inc.

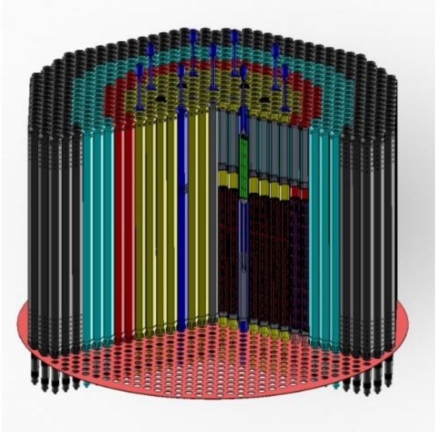
VELAN

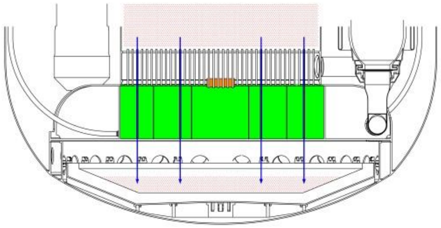


Technetics GROUP

ASTRID MAIN TECHNOLOGICAL INNOVATIONS
NEEDS FOR DEVELOPMENT STUDIES AND EXPERIMENTAL QUALIFICATION

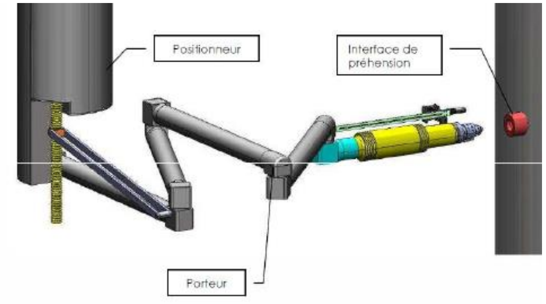
Innovative core design for enhanced safety features (« CFV »),

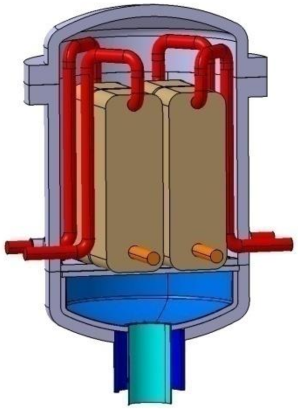




Equipments and dispositions to eliminate large early accidental radiactive release

Equipments and dispositions for performing ISIR capabilities





Open option **: Energy conversion (gaz Brayton cycle)** for practical elimination of accidental Na-water reaction

ASTRID STRUCTURAL MATERIALS AND MAIN OPERATING CONDITIONS

Sodium-cooled Fast Reactor
Pool type
Gas power conversion system or SG

Fixed structures
Design life → 60y

Steam Generators - *345 / 525°C*
Aging Compatibility Na,
Oxidation, Wastage…

800SPH

Hot secondary circuit – 525°C
Creep, Creep-fatigue, Aging…

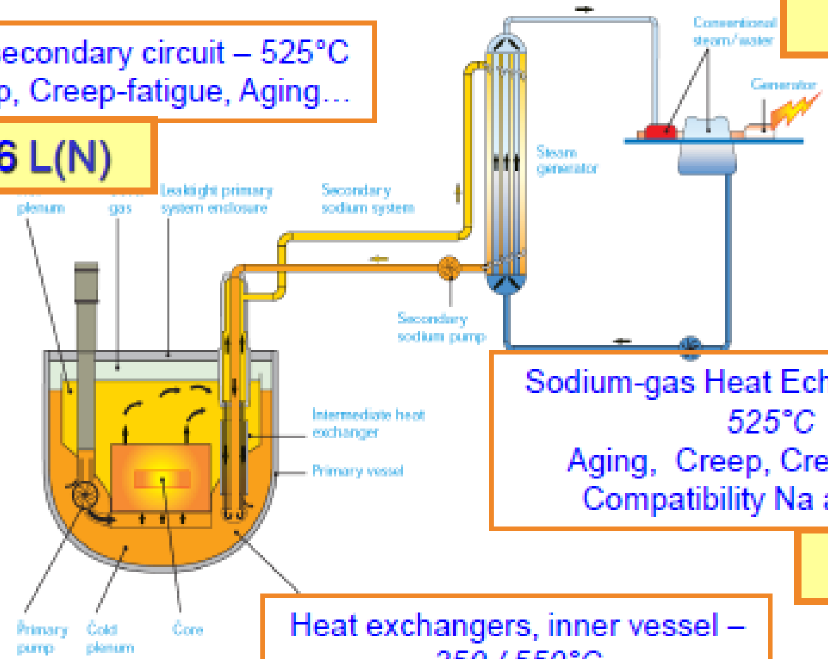
316 L(N)

316 L(N)

Above core structures - *550°C*
, Low irradiation, thermal fatigue, creep…

316 L(N)

Vessel - *400°C*
Negligeable creep



Sodium-gas Heat Echanger - *345 / 525°C*
Aging, Creep, Creep-fatigue
Compatibility Na and N2…

316 DB

Heat exchangers, inner vessel – *350 / 550°C*
Creep, Creep-fatigue, Aging…

316 L(N)

Friction zone uo to *400°C*

coatings

Core support structure, pumps- 400°C (diagrid: low irradiation)

316 L(N)

CORE MATERIALS: FOR 1st CORE *AND NEXT ONES*

Cladding, space wire : AIM1 (optimized 1515Tiε)

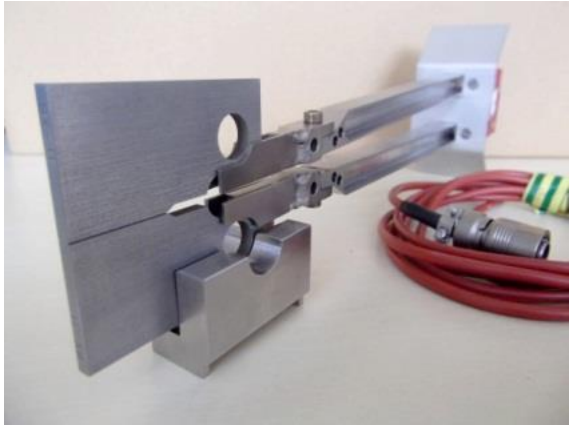
- Complementary exp. data for irradiated material (OLIPHANT-1bis, PAVIX, …)
- Behavior understanding and modeling, confirmation of all mechanical laws …
- Mechanical properties database **up to 1000°C** and modeling,
- Welding process : plug/tube, wire/pin
- + *longer term track : AIM2-120 dpa, ODS 150 dpa …*

Hex-can material : EM10

- Mechanical properties database **up to 900°C**, modeling,
- Exp. data for irradiated materials (BOITIX-9, MATRIX)
- Heterogeneous welding EM10 / 316 (Hex-can/S/A-foot)

Heterogeneous ferritic/austenitic welding

Codification : RAMSES code updating



Tensile test for irradiated materials

REACTOR MATERIAL R&D : SYNTHESIS VIEW

Component expertise on real FBR

operating conditions

- Maintenance Na operation
- Ageing, Creep, Cycling fatigue
- Irradiation

Knowledge R&D

- Welding operation
- Long term corrosion and mechanical damage simulation codes

Robust data for 60 years of operation

- **Safety objectives**
- **Laws, decrees**
- **Standards and codes**

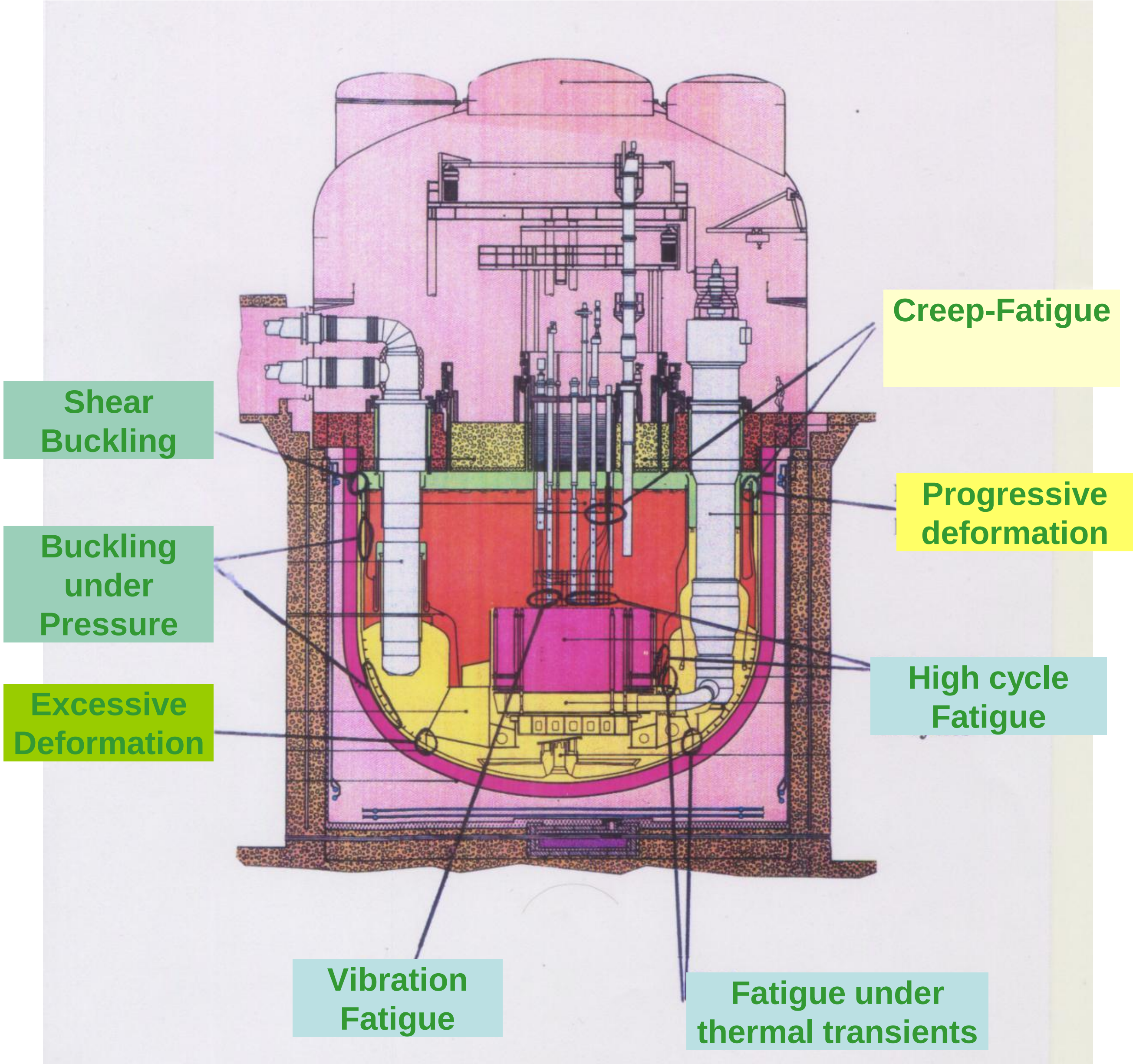
Test program->life-time justification

- Creep and ageing on representative samples (raw materials and welded sample
- Corrosion

Survey program:

- Operating situations counting → design hypotheses conformity
- Uncertainty reductions on material mechanical properties

DAMAGE MODES IN A SFR



CORE MATERIALS

Core components:

- **Fuel cladding (critical component)**
- **Wrapper tube**
- **Absorbers, Reflectors**
- **End-plug, joining…**

→ Which materials ?

ASTRID demonstrator
480-700℃, 110 dpa

- Use of reference materials benefiniting from a large feed-back from the previous French SFRs (Rapsodie, Phénix, SuperPhénix)
- Austenitic steels (cladding), Martensitic steels (wrapper tube), B_4C (absorbers).
- Improving the description of their behavior (swelling, high temperature)
- Qualifying the materials regarding the specificities of ASTRID core

Future SFRs
530-750, 180 dpa

- Use of advanced materials with improved properties
- ODS ferritic/martensitic steels (cladding), Other metallic solutions as V alloys (cladding), SiC/SiC composites (wrapper tube), Innovative absorbers and reflectors.
- R&D to develop/fabricate suitable grades
- Qualifying these materials in ASTRID

MAIN ENVIRONMENTAL EFFECTS ON MATERIALS, WITH LIQUID METAL COOLANTS COOLANTS

Main parameters:

- neutron flux (on fuel cladding, under-core structures (strongback, diagrid…))
- temperature T, T gradients, T cycling, T instabilities & drifts
- Liquid metal chemistry (O, N, C, H, ...)
- local Na velocities and pressures

Involved phenomena:

➔ **On structural materials:**

- generalized corrosion and mass transfer (dissolved & particles)
- Deposition (impact on contamination)
- embrittlement
- desquamation
- Activation....
- Potential Stress corrosion cracking

➔ **On coolant:**

- activation of coolant (^{22}Na, ^{24}Na)
- Na contamination : activated corrosion products, fission products (cesium, tritium…), fuel particles (if open pin rupture)
- introduction of particles

➔ **On cover gas:** contamination (Xe, Kr, …)

Target objective: life duration (requirement: 60 years)

CONTEXT

Materials studies for ASTRID

➡ 316L(N) stainless steels largely used for structural components

➡ Feedback obtained through design and operation of Phenix and Superphenix NPP

Very satisfactory feedback regarding behaviors when in contact with high purity sodium

Nevertheless, utilities have to take care at materials susceptibility to different corrosion mechanisms :

- **Stress Corrosion Cracking (SCC) induced by caustic solution**
- **InterGranular Attack (IGA) induced by acid solution**

These damages are linked with maintenance operation

MAIN ENVIRONMENTAL EFFECTS ON MATERIALS, WITH LIQUID METAL COOLANTS COOLANTS

For Na

Main parameters:

- neutron flux (on fuel cladding, under-core structures (strongback, diagrid…))
- temperature T, T gradients, T cycling, T instabilities & drifts
- Na chemistry (O, N, C, H, ...)
- local Na velocities and pressures

Involved phenomena:

➔ **On structural materials:**

- generalized corrosion and mass transfer (dissolved & particles)
- Deposition (impact on contamination)
- embrittlement
- desquamation
- Activation....
- Potential Stress corrosion cracking (aqueous soda after maintenance)

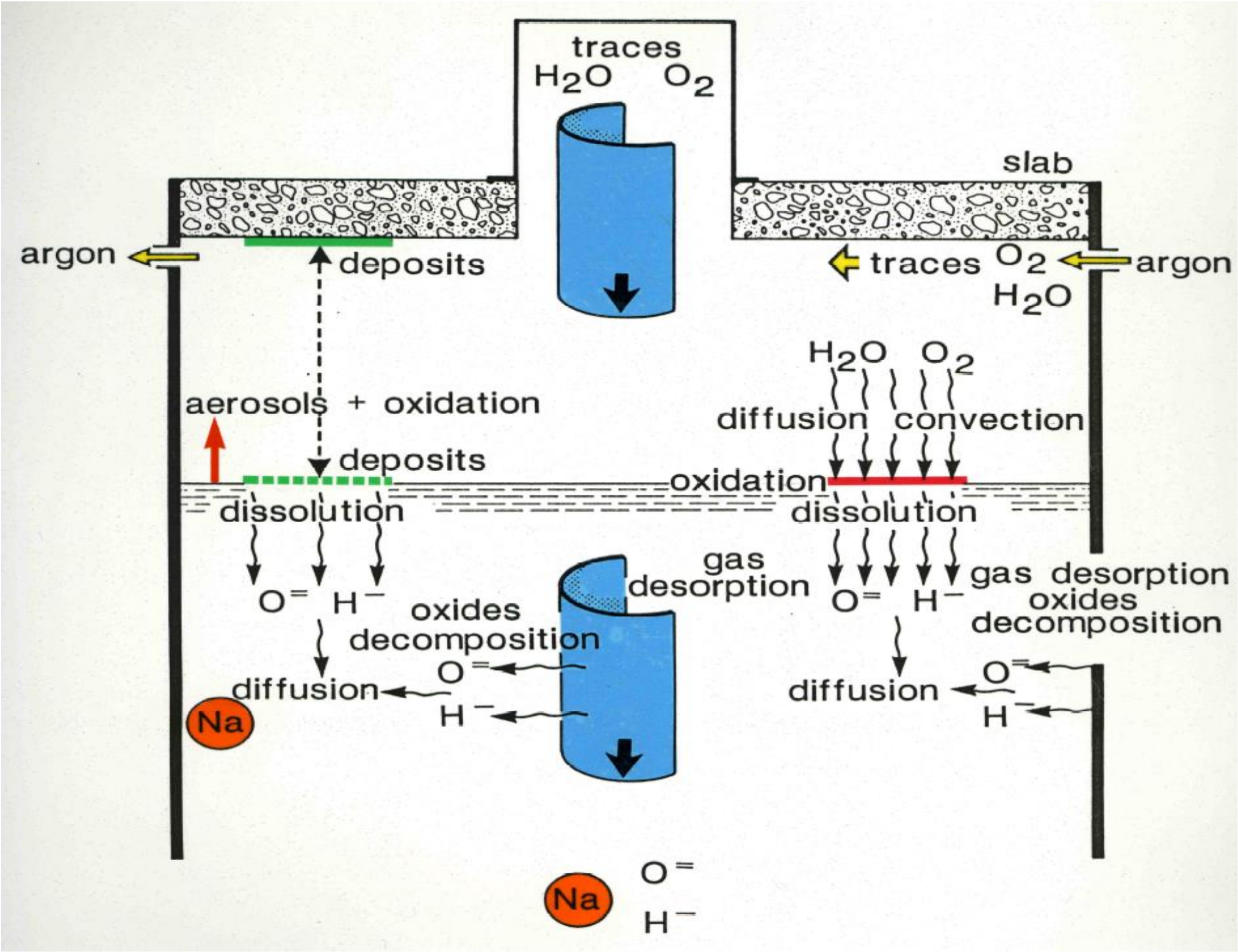
➔ **On coolant:**

- activation of coolant (^{22}Na, ^{24}Na)
- Na contamination : activated corrosion products, fission products (cesium, tritium…), fuel particles (if open pin rupture)
- introduction of particles (no Na_2O but around 10 µm $NaCrO_2$)

➔ **On cover gas:** contamination (Xe, Kr, Cs…)

Target objective: life duration (requirement: 60 years) (Impact on particles?)

POTENTIAL POLLUTION IN PRIMARY VESSEL



C. Cabet, J.L. Courouau, C. Latge, L. Martinelli, J. Guidez
« Reactor chemistry of next generation nuclear systems and structural materials behavior"
Conférence ICAPP Juan-Les-Pins France, May 2019

Background:

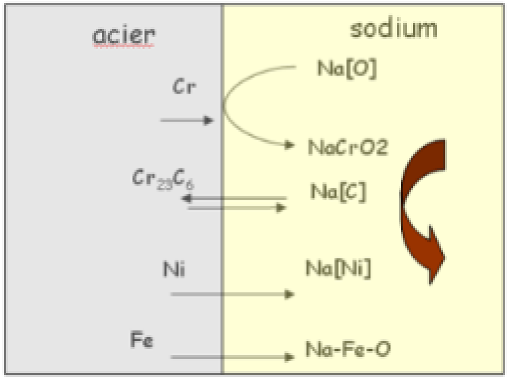
Very good compatibility of steels with pure sodium ([O]< 5 ppm) for steels used with operating conditions of the existing reactors.

Nevertheless, new needs for ASTRID and SFR

- Life duration for structures: 60 years (316LN…)

- New materials: ODS, advanced austenitic steels,….

Na	Normal conditions	Transients
1	370-550°C - max 650°C 8-12 m/s - [O] < 5 µg/g	850°C (s- min) [O] = 15 µg/g/ 100 h
2	300-550°C - some m/s [O] < 5 µg/g	[O] = 200 µg/g/ 2000h



Corrosion: homogeneous phenomena but several mechanisms: dissolution, oxidation, intergranular diffusion (C, O, H, B), mass transfer

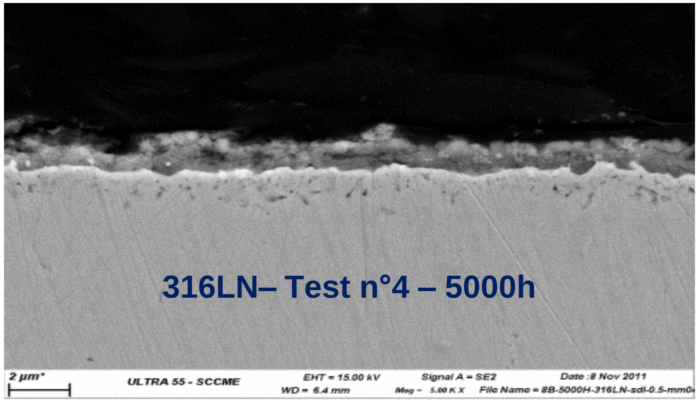
Parameters: température, duration, hydrodynamics, [O], activities, minor alloy compounds (ie Mo), microstructure, neutron flux, ΔT (IHX)…,

Consequences:
- mainly release of activated corrosion products,
- réduction of thickness (to a less ext

Basic research to improve the knowledge:
- ternary oxides behaviour (Na_4FeO_3 …),
- effect of solvation,
- diffusivities, …

Up to now Semi-empirical modeling:(Baqué – Thorley)
➔ Development of new corrosion models
➔ Development of a new transfer model (OSCAR-Na)

CORROSION IN NA

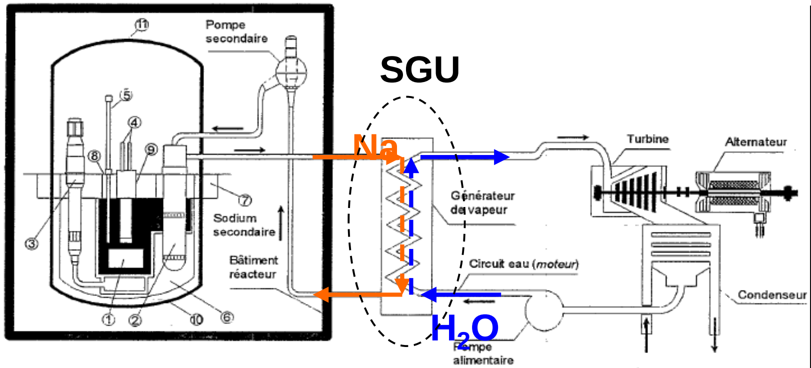


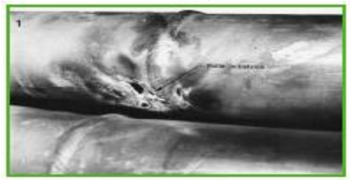
- Kinetics available up to 5000 h at 550°C for [O] < 10 µg/g
- Ferritic steels more sensitive to oxidation and carburization than austenitic steels
- 9Cr steels have a similar behaviour

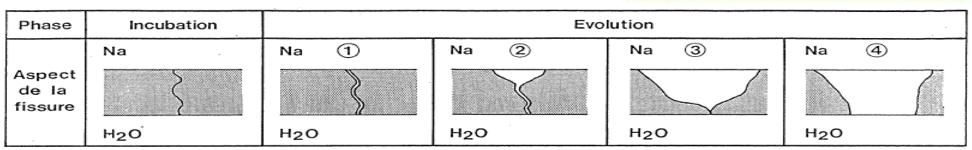
CORRONa facility (CEA-DPC)

JL Courouau et all “*Corrosion by oxidation and carburization in liquid sodium at 550°C of austenitic steels for sodium fast reactors”* FR13 Paris March 2013

Sodium-water reaction







No leak ***Micro leak*** ***small leak*** ***evolution***

ORIGINS : **Normal operation of steam generator induces damage of heat exchange tubes**

tube corrosion : **mainly in welding zones, inducing leaks due to cracking**

thermal chocks : **when under-saturated water is injected at super heater inlet (Phenix), inducing thermal fatigue,when fluctuation of heat exchange conditions**

✓ ***impossible tube expansion:*** **buckling, inducing differential expansion with envelope**

✓ ***tube bundle vibrations :*** **hydraulic effect of sodium flow, inducing tube wear**

Na-H2O : a violent and exothermal chemical reaction

Main reaction

$Na + H_2O \rightarrow NaOH + ½ H_2$ + 162 kJ/water mole *(at 500 ° C)*

Complete, quasi-instantaneous and non-reversible reaction

Many secondary reactions

2Na + NaOH ←2 1→ [O2-]Na + [H-]Na ↔ Na2O + NaH

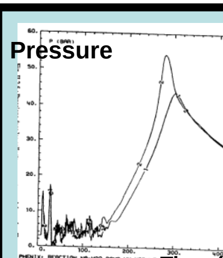
Equilibrium reaction depending on sodium temperature and hydrogen dissolved and hydrogen partial pressure equilibrium

Above about 300 ° C, and with sodium in excess,

hydroxide is decomposed in sodium oxide and hydride (reaction → 1)

Above 410 ° C, reaction (→2) occurs only if PH2 reach Pequilibrium in cover gas; The experimental conditions doesn't satisfy this condition; Thus the decomposition of NaOH is total.

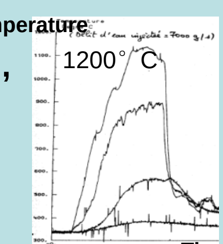
Reaction rates depend on temperature



Effects: chemical,

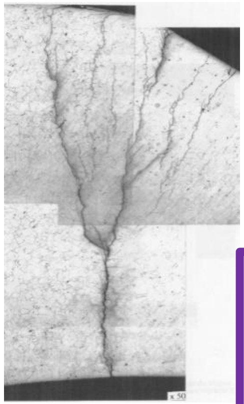
mechanical,

thermal



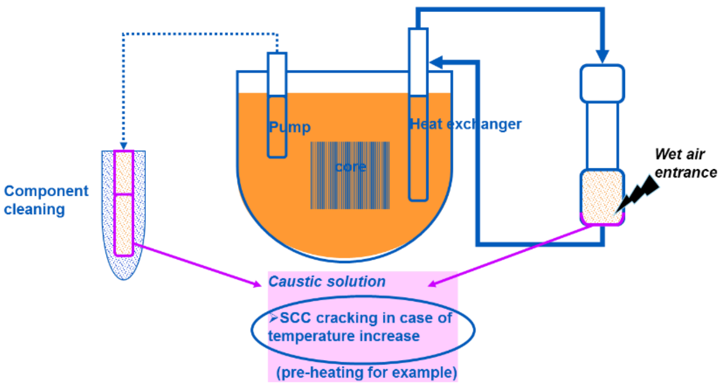
Risk of Stress Corrosion Cracking after repair

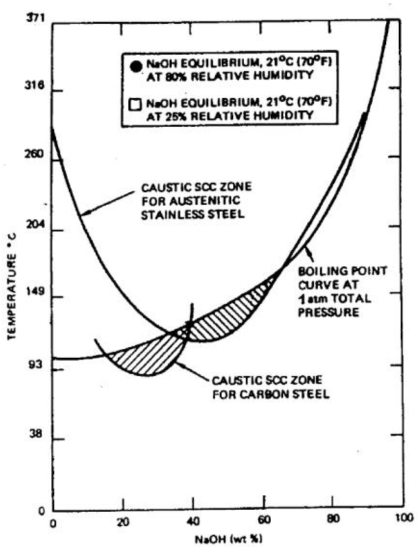
- **Very localized corrosion with small amount of aqueous NaOH**
- **Corrosion Process characterized by transgranular cracks (austenitic steels)** (Can be intergranular under low stresses)
- **Very fast phenomena**



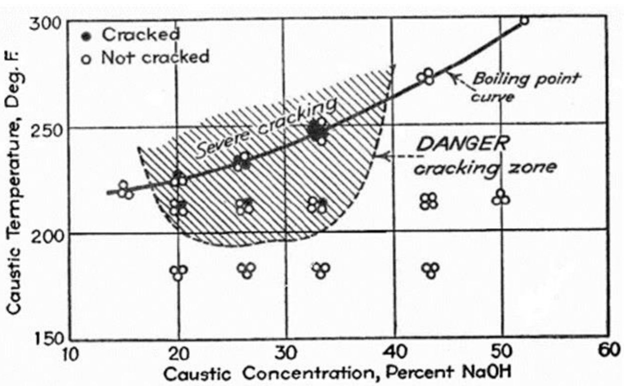
Phénix : support de palier de guidage du clapet

→ A specific procedure to restart is applied since more than 20 years to avoid such phenomena





Hoffman diagram 1976



Caustic users will do well to keep out of the zone above 180 deg. F. and between the concentrations, 15 and 43 percent.

DOMAIN OF SCC for SAE 1020 steel
(Immersion during 30 days of U-Bend)
0,2%C 0,3-0,6% Mn

M. Blat-Yrieix et al EDF Feedback from Stainless Steels Corrosion related Issues during Maintenance Operation in SFR : SCC in caustic solution and Intergranular corrosion by Acid SolutionFR17 – Yekaterinburg 26—29 June 2017

Stress Corrosion Cracking induced by caustic solution

Maintenance operation

↓ Room temperature

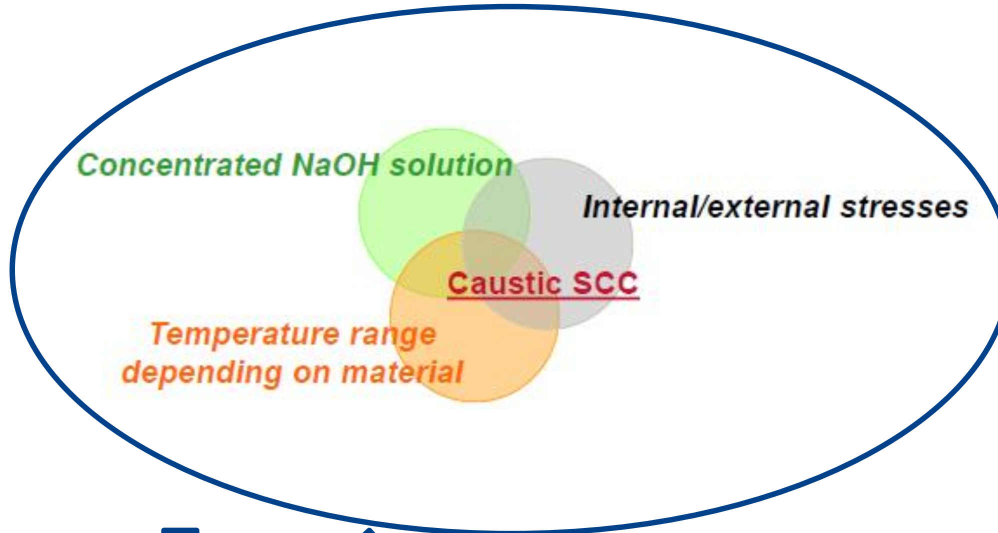
Component where Na was drained

↓

Sodium residue in contact with moist air

↓

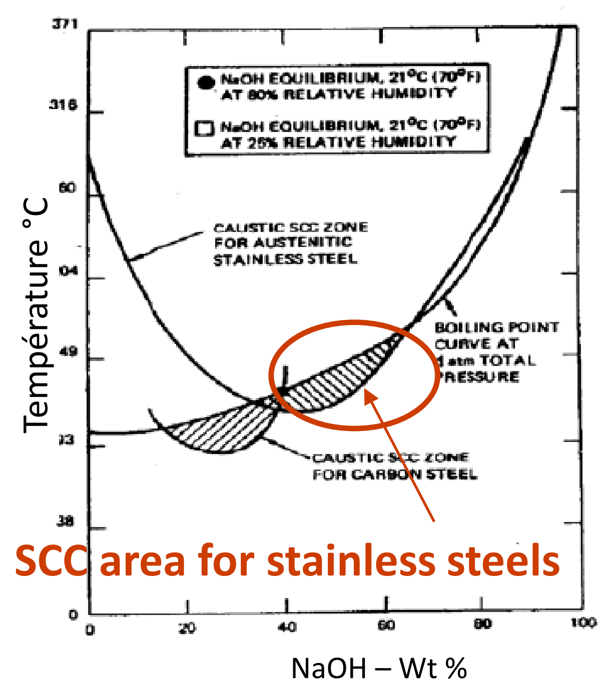
Aqueous sodium hydroxide formation



According to Hoffman diagram :

- SCC sensibility is dependent of
  - Temperature
  - NaOH concentration

- **SCC sensibility's area for stainless steels** :
  < 120 – 200 > °C
  < 30 to 80> % caustic solution concentration



Hoffman diagram 1976

Stress Corrosion Cracking induced by caustic solution

- **Improvement to manage SCC in such environment**

  Component temperature and/or existence aqueous sodium solution :

  - **Recommendations on design rules and operating procedures**
    - Avoid aqueous hydroxide formation by **design and draining options** : suppression of blind hole, design improvement to eliminate liquid metal retention
    - **Procedure** dealing with particular attention when crossing the temperature threshold of the SCC risk area : efficient cleaning procedure, use of dry inert gaz….
    - **Washing procedure** : take care for thick sodium retention elimination (for instance mechanical elimination)

- **For PX : all removable components dismounted, cleaned, inspected, repaired …. and satisfactory re-used in reactor**

  Feasibility demonstration : high valuable feedback for SFR operation

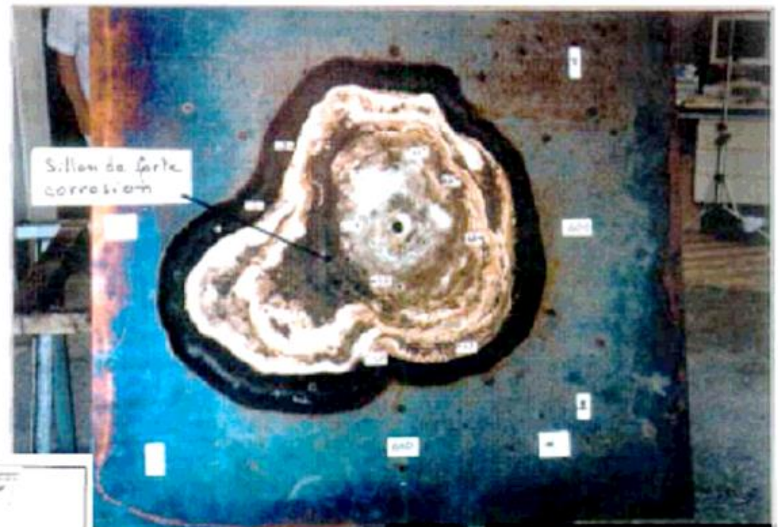
CORROSION PHENOMENA DUE TO NA INTERACTION WITH INSULATING MATERIAL

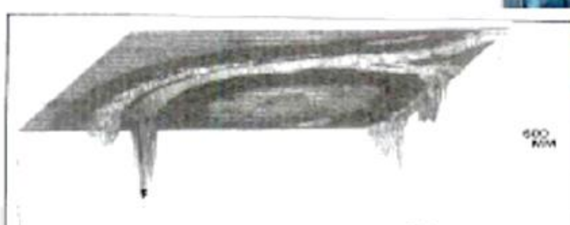
In order to avoid this phenomenon, new technology developped at CEA.

Essai BPR 03 Coupe de la zone de réaction sodium-calorifuge

Sodium leak + thermal insulation (silica)

Corroded area at pipe external surface



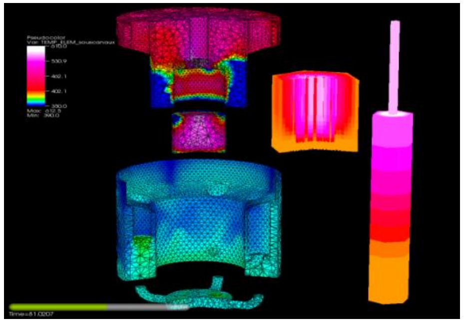


Corroded surface measurement

Table 3 X-ray Results

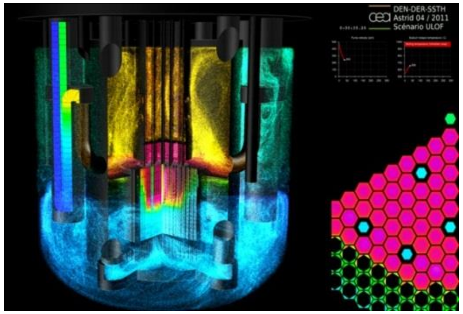
Sampling Positions	XRD Results
Green product near surface	Na_4SiO_4, $NaAlO_2$
Black product on surface	$NaFeO_2$, NaOH
White product near surface	Na_2O_2, Na_2O, NaOH, (SiO_2)

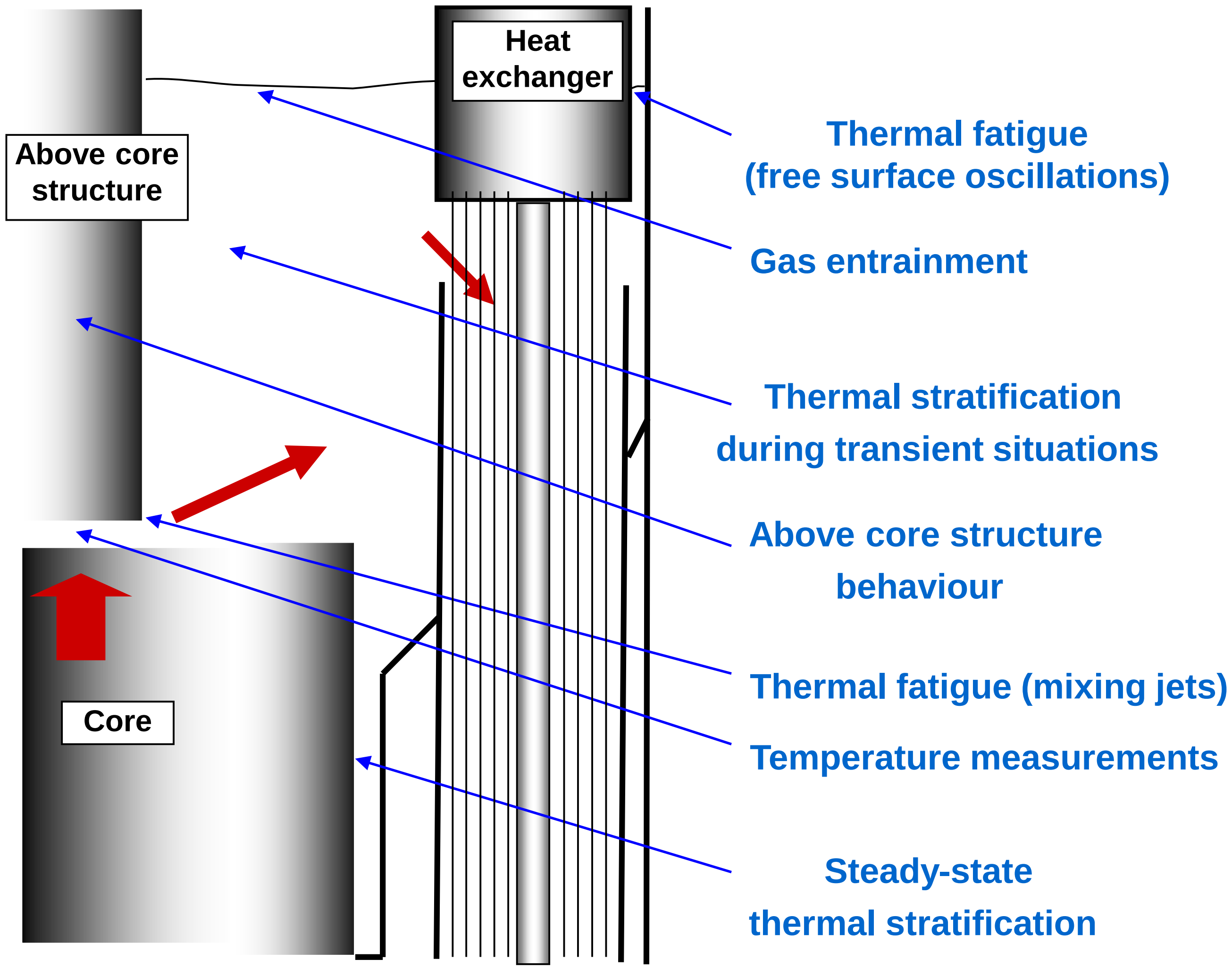
Hydraulic and Thermal Hydraulic challenges in hot pool



Asessment of thermal stresses on the structures in:
- steady-state
- transient situations

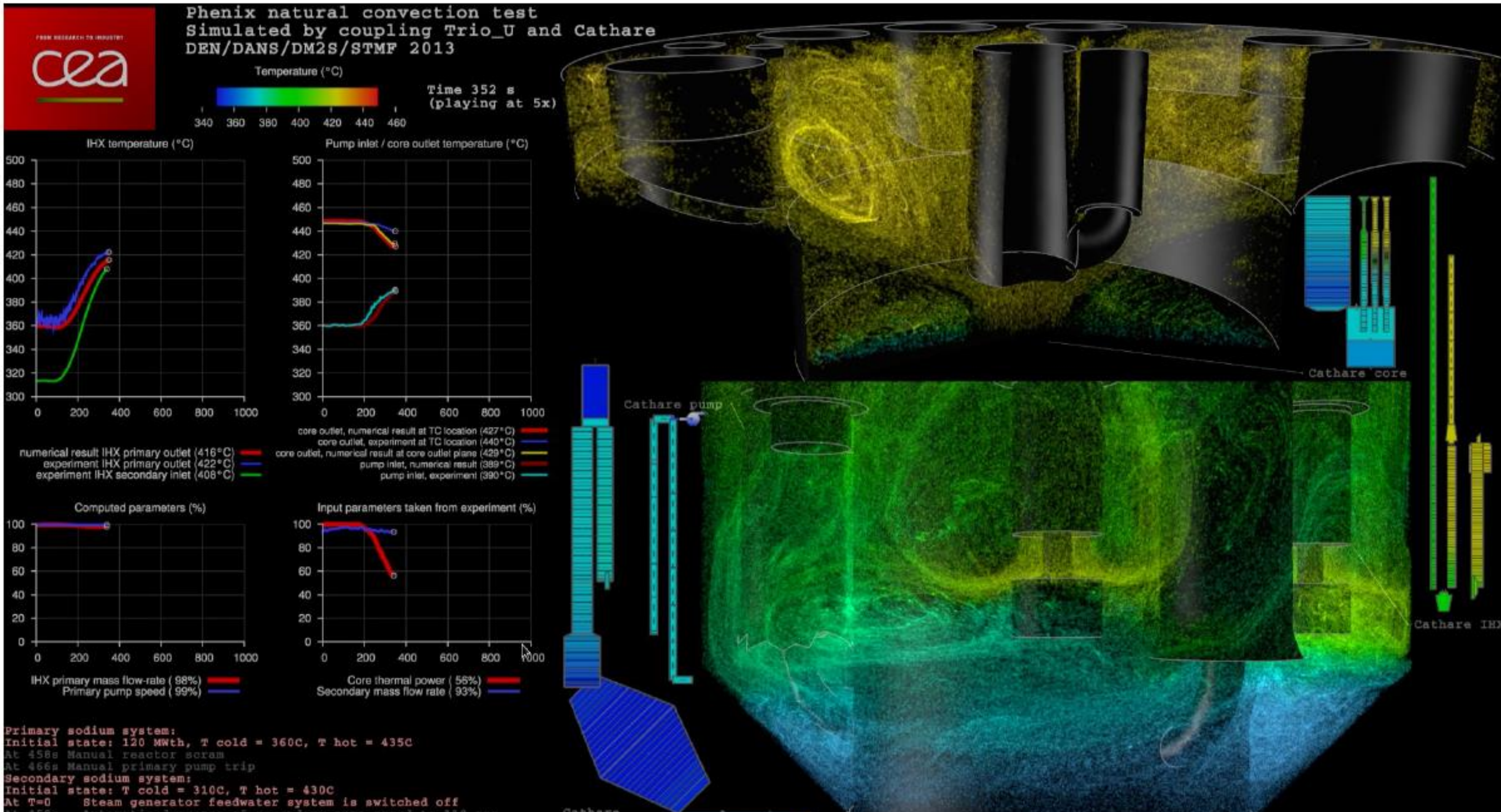
by computation thanks to optimized system code coupled with CFD and, if required with mockups.





D. Tenchine, « Some thermal hydraulic challenges in sodium cooled fast reactors. », NED 240 (2010) 1195–1217.

SYSTEM THERMAL-HYDRAULICS



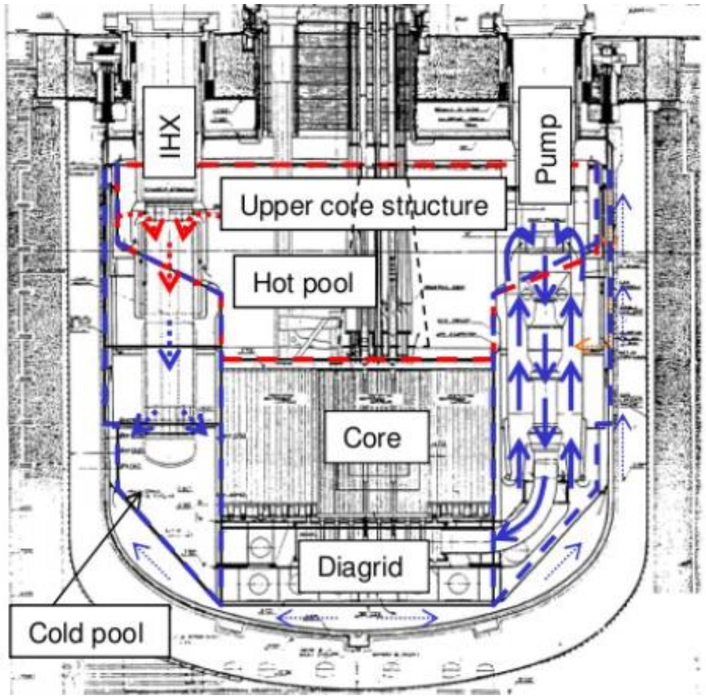
Thermal-hydraulic studies relevant for material analysis

- reactor steady-state :
  justify thermo-mechanical criteria for
  - 4-year design-life : subassemblies
  - 60-year design life : most primary internals (inner vessel, diagrid, strongback, core catcher...)
  - somewhere in between : large components $\rightarrow$ IHXes, pumps(, UCS)
- planned transients :
  - reactor maintenance, shutdown, scram

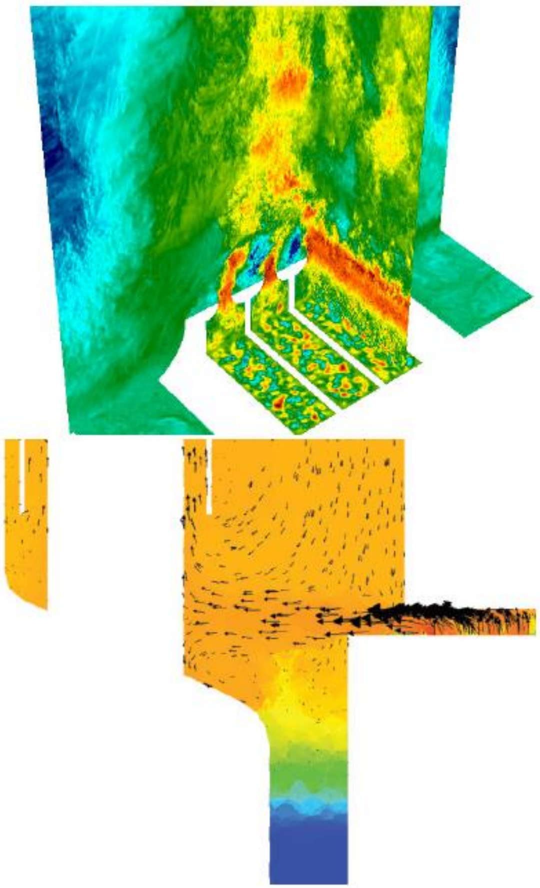
  $\Rightarrow$ same design life goals as above
- accidental transients $\rightarrow$ short-term behavior of the cladding, hexcans...

Reference tool for primary natural circulation situations**:**
CATHARE +coupling with TRIO_U-MC2 / TRIO_U

The **validation of the coupled model** against available experimental data is in progress ; first results show a reasonable agreement ;
Extra developments are foreseen to further improve the model (ex: refined model with recirculation within the core, …)



Phenomena of interest



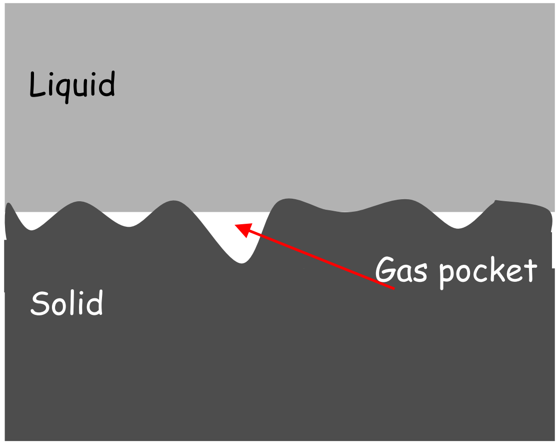
Hot pool

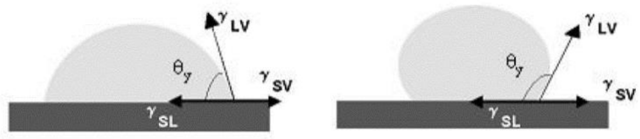
- jet mixing at the core outlet :
  fuel S/A (570°) vs CR S/A (450°)
  → thermal striping on UCS
- core outlet jet behavior :
  flat (high flow) vs bent (low flow)
  → affects hot pool stratification
  → thermal load on inner vessel
- thermal shocks during transients
  (-150° in 40s during scram)
- free level fluctuations :
  at steady-state → thermal striping
  during transients → thermal shocks

WETTING PHENOMENA

Wetting phenomena, which depend of gas adsorption, structural material oxidation,… are key interface phenomena between the coolant and the structural material. Therefore it is considered as a key factor with regards the following items:

- **accuracy of measurements** for some instrumentation devices such as ultra-sonic based traducers, electro-magnetic flow-meters, electro-chemical cells,…
- **interactions between structural material and liquid metal**: corrosion, embrittlement, stress corrosion cracking….
- **mass transfer** such as activated corrosion products, tritium,…
- **thermal exchanges in Heat Exchangers**, liquid metal targets,…
- **Technology developments, cleaning of residual layer,…**





➔ **Due to non-significant material embrittlement in Na, there is no necessity to foresee coatings to prevent wetting and its deleterious consequences.**

(except to prevent from wearing & fretting effects)

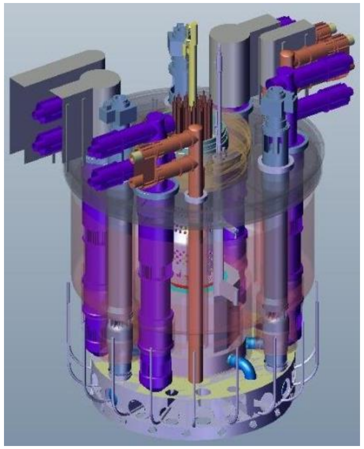
➔ **Na: a strong reducer: a very good wetting is obtained, even at low temperature *(ie T=180°C)* thanks to the possibility to reduce oxygen content down to a very low value (< 3ppm)**

ISI REQUIREMENTS

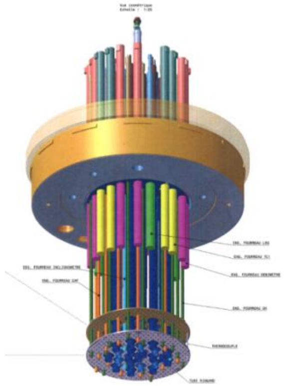
- To satisfy the requirements of this 4th generation in terms of safety, reliability, availability and energy savings, **SFRs will need to achieve a higher level of performance than that of previous fast reactors.**
- **In-Service Inspection and Repair** must contribute to this increase of the safety and availability levels:

  - continuous monitoring of the operating parameters during reactor operation (including core monitoring and protection against abnormal events)

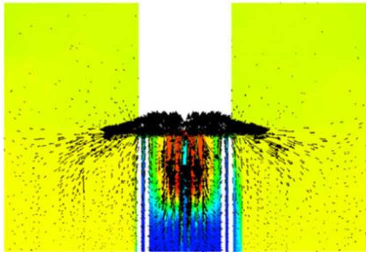
  - periodical inspection of structures, welds



- **Limitation by design of the areas to be inspected:** few and shorter welds, design margins, structures redundancy, slow evolution of defects, possible access in the reactor block for inspection...
- **Requirements for implementation of instrumentation and related systems taking into account "environmental" conditions** (temperature, Na velocity, radiation, presence of Na aerosols…)



➔ ACS*, a key component/system, with hard "environmental" conditions, inducing permanent demonstration of its reliability and availability (core reactivity control,…)



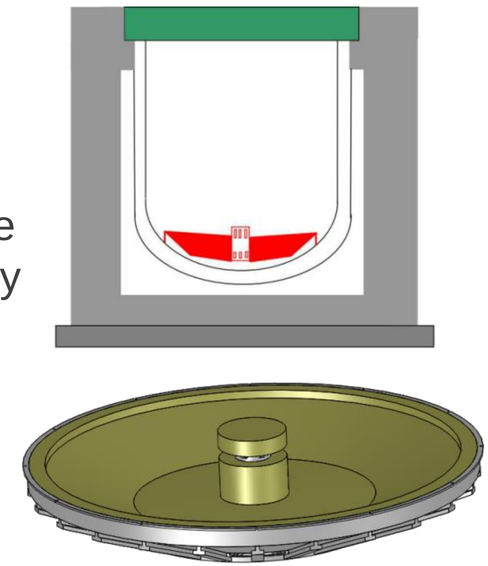
* Above Core Structure (ACS)

Primary circuit / Core catcher

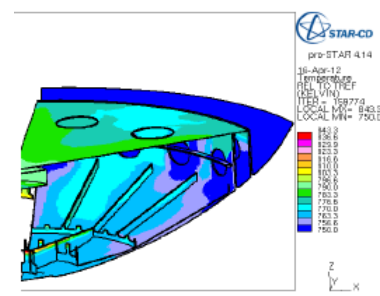
➔ Reference option: In-vessel core catcher

- **Design**
  - ✓ Located inside the main vessel below the core support structure
  - ✓ ZrO_2 covered for corium jet impingement protection (preliminary design)
  - ✓ Cooled by sodium natural convection

- **Advantages**
  - ✓ Protection of the main vessel (Integrity)
  - ✓ Well adapted to pool type reactor (large tray)
  - ✓ Could use in-vessel DHR if available



In-vessel core catcher design



In-vessel core catcher cooling studies

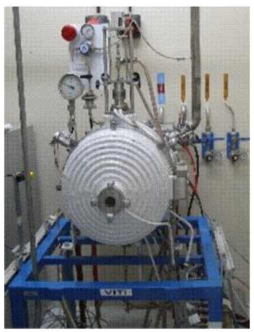
SACRIFICIAL MATERIALS FOR CORE CATCHER

SFR project: Enhanced safety requested

➔ Necessity to master the risk of a core meltdown accident

➔ Use of sacrificial materials to control the reactivity in the reactor and to prevent the recriticality*

- **Sacrificial materials**
  - Absorber materials: able to absorb neutrons coming from the nuclear chain reaction, ex: boron carbide B_4C in passive mitigation systems
  - Diluents: materials used in the core catcher, able to dilute the mixture of molten fuel and molten structures (=corium)
- **Objectives:**
  - To select the sacrificial materials for the SFR core catcher **(in VITI facility)**
  - To understand the behavior of the absorber material B_4C during interaction with corium, from chemical and thermodynamic point of view

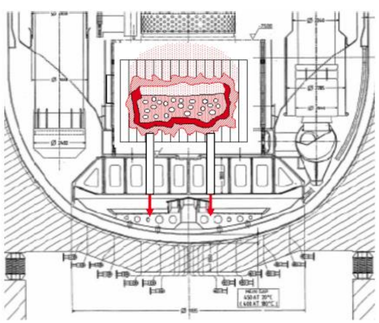


General view of VITI facility

Induction furnace

\- Synthesis of 2 types of materials based on HfO_2 and $LaAlO_x$ and tests of compatibility in Na, in **CORRONa**

➔ Studied mixture (≈ 10 g) placed in a protected graphite or tungsten crucible
➔ Inductive heating until **2600°C**
➔ Adjustable atmosphere (2 bars, under argon ≈ 5 L/min)
➔ Measurements during the experiments:
Melting temperature ➔2 pyrometers
Quantity of the formed gas ➔ measure of pressure and gas flows
➔ Sample analysis after the experiments: identification of different formed products



**Recriticality: occurs when corium forms a critical mass producing a sustained nuclear chain reaction ➔ generation of heat leading rapidly to melting and boiling*

TOPICAL AREAS AND OBJECTIVES OF THE OECD EXPERT GROUP ON LIQUID METALS

OECD Expert Group: Two tracks: Na & Pb &Pb-Bi

➔ **Environmental conditions and factors that affect materials behaviour relevant for the structural integrity of confinement barriers and components.** These include the impact on mechanical properties from the environment such as irradiation effects and liquid metal embrittlement as well as environmental assisted property effects like corrosion.
➔*The objective is to address the* ***environmental effects relevant for construction standards via a fundamental understanding of materials behaviour (corrosion and mechanical properties in the liquid metals and under irradiation****).*

➔ **Coolant and cover gas issues. Here the focus is placed on issues relevant for radiological impact assessment, operation and handling.** Topics to be addressed are the chemistry, radio-chemistry and physics of the coolant, its interaction with the cover gas, the impact of irradiation, the influence of corrosion, etc.
➔*The objective is to answer* ***key technical issues to address radiological impact, operation, handling and inspection as relevant for licensing (reactor operation, dismantling).***

➔**Thermal-hydraulics for Heavy Liquid metals.** Thermal hydraulic behaviour of the coolant is a **crucial factor** in the sense that it essentially determines a large part of the environmental conditions for materials and the cooling such as the **flow distribution and mixing, temperatures, erosion rates, operation of components**, etc ➔*The objective is to collect experimental data for* ***correlations relevant for heat exchange, pressure drops, vibrations, mass transfer, etc. in order to assess and improve knowledge of the environmental conditions for materials and the coolant*** **behaviour.** (some points ie heat exchange and pressure drops in the frame of NAPRO for SFR; out of scope of OECD mandate)

Thank you for your attention