Meet the presenter

Mr. Gilles Rodriguez is a senior expert engineer at the CEA/CADARACHE (French Atomic Energy Commission/Cadarache center) and has been in the position of deputy head of the ASTRID Project team since 2016. He graduated from the University of Lyon, France in 1990 with an engineering degree in Chemistry and earned a Master of Science in process engineering from the Polytechnic University of Toulouse, France, in 1991. His areas of expertise include fast reactor technology, liquid metal processes, and process engineering. From 2007 to 2013, he was Project Leader of sodium technology and components, within the CEA SFR project organization. In 2013, Mr. Rodriguez joined the CEA project on Sodium Fast Reactor: ASTRID (ASTRID for Advanced Sodium Technological Reactor for Industrial Demonstration), first as responsible of the ASTRID Nuclear Island.

Email: gilles.rodriguez@cea.fr
Outline

- French nuclear policy and its position regarding the several GENIV systems and related coolants
- The advantages and challenges of the SFR concept
- Overview of the ASTRID Program and its related design (main achievements)
- The lessons learned
The French Multi-annual Energy Plan (MEP) is updated every 5 years. An update will be issued at the end of 2018, after the on-going public debate. The governmental document issued to support the public debate on energy has confirmed the closed fuel cycle strategy, as it allows for Pu management and ensures sustainability of nuclear energy.

Reference of the French roadmap is based on the reprocessing of oxide fuel (hydrometallurgy) and the use of Fast Reactors. Priority is given to Sodium-cooled Fast Reactors (most mature technology). Active survey is performed on other technologies through collaborations.
6 Systems Selected by the GEN IV International Forum

- **Closed fuel cycle**
  - SFR: Sodium-cooled fast reactor
  - GFR: Gas-cooled fast reactor
  - LFR: Lead-cooled fast reactor

- **Open fuel cycle**
  - VHTR: Very high temperature reactor
  - SCWR: Supercritical water-cooled reactor
  - MSR: Molten salt reactor
### 6 Systems Studied Under the Frame of the GEN IV International Forum

<table>
<thead>
<tr>
<th>GIF systems</th>
<th>Canada</th>
<th>China</th>
<th>France</th>
<th>Japan</th>
<th>South Korea</th>
<th>Russia</th>
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<td>CEFR</td>
<td>ASTRID</td>
<td>JSFR</td>
<td>PGSFR</td>
<td>BN-800, 1200, MBIR</td>
<td>(PRISMA), AFR100 TWR (TerraPower )</td>
<td>ESNI/ESFR</td>
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<tr>
<td>VHTR (SA)</td>
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<td><strong>Thermal</strong></td>
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<td>HTR-10</td>
<td>Materials, Hydrogen technology</td>
<td>HTTR</td>
<td>NHDD (H\textsubscript{2}, prod.)</td>
<td>NGNP, Xe -100</td>
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<td>ESNI/ALLEGRO</td>
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<td>SCWR (SA)</td>
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<tr>
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<td>Pressure-tube SCWR</td>
<td>CSR -1000</td>
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<td>SCR2000</td>
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<td></td>
<td>(HPLWR), NUGERIA/SCWR -FQT</td>
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<td>ESNI/ALFRED, MYRRHA</td>
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<td>MSR (MoU)</td>
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<tr>
<td><strong>Fast/thermal</strong></td>
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<td>MOSART</td>
<td></td>
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<td>SAMOFAR</td>
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</table>

#### Situation - Septembre 2016

- **Active contribution**
- **Limited contribution**
- **Observer**
CEA Analysis of Different Coolant Technologies on Gen IV Compatible with Fast Neutrons

- **Sodium is a consensus**
  - High conductivity
  - Liquid from 98°C to 883°C (at 1 bar)
  - Low viscosity
  - Compatible with large variety of steels
  - Industrial fluid
  - Low cost
  - But reactive with air and water, and opaque
  - Need a 2^ary circuit

- **Lead is a variant**
  - No reactivity with air and water
  - Good coolant
  - But corrosive, toxic, very dense, opaque (and solid...), high temperature for maintenance (400°C), high pressure (pumping effort, seismic behavior), risk of vapor explosion in primary circuit (secondary circuit)

- **Helium is an alternative**
  - No temperature constraints
  - No phase change
  - Inert
  - Transparent
  - But low density, high pressure => challenge for a DHR architecture with passive systems
  - Helium is not such an abundant material on earth
The French Gen IV Program

- French nuclear policy towards a more sustainable nuclear energy ⇒ it requires a cycle based on fast neutrons reactor
- Possibility for a deployment of commercial fast neutrons reactors in the second part of the century ⇒ sodium cooled fast reactor based on a maturity level analysis
  - About 450 years of operation in the world with SFR
  - No implementation of Lead FR, Fast MSR, GFR
- French Fast Reactor program:
  - Priority is given to SFR (reactor and cycle) via the ASTRID project ⇒ ASTRID will help to validate breakthroughs on cycle and SFR
  - Active survey on other GenIV fast and thermal neutrons systems through
    - Contribution to projects from EURATOM/H2020, IAEA, OECD/NEA
    - Contribution to GenIV International Forum (systems, working group, task forces…)
    - Specific cooperation frames (for GFR associate member of V4G4, and for MSR cooperation with CNRS)

<table>
<thead>
<tr>
<th>Reactor (Country)</th>
<th>Thermal Power</th>
<th>First Criticality</th>
<th>Final Shutdown</th>
<th>Operational period (years)</th>
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<tbody>
<tr>
<td>EBR-I (USA)</td>
<td>1.4</td>
<td>1951</td>
<td>1963</td>
<td>12</td>
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<td>BR-5/BR-10 (Russia)</td>
<td>8</td>
<td>1958</td>
<td>2002</td>
<td>44</td>
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<td>DFR (UK)</td>
<td>60</td>
<td>1959</td>
<td>1977</td>
<td>18</td>
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<td>1961</td>
<td>1991</td>
<td>30</td>
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<td>EFFBR (USA)</td>
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<td>1963</td>
<td>1972</td>
<td>9</td>
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<td>Rapsodie (France)</td>
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<td>1967</td>
<td>1983</td>
<td>16</td>
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<td>BOR-60 (Russia)</td>
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<td>1968</td>
<td>1997</td>
<td>30</td>
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<td>1969</td>
<td>1972</td>
<td>3</td>
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<td>BN-350 (Kazakhstan)</td>
<td>750</td>
<td>1972</td>
<td>1999</td>
<td>27</td>
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<td>Phenix (France)</td>
<td>563</td>
<td>1973</td>
<td>2009</td>
<td>36</td>
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<tr>
<td>PFR (UK)</td>
<td>650</td>
<td>1974</td>
<td>1994</td>
<td>20</td>
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<td>JRR-2 (Japan)</td>
<td>50–750</td>
<td>1977</td>
<td>1997</td>
<td>21</td>
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<td>KNK-II (Germany)</td>
<td>55</td>
<td>1977</td>
<td>1991</td>
<td>14</td>
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<td>FFTF (USA)</td>
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<td>1980</td>
<td>1993</td>
<td>13</td>
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<td>BN-600 (Russia)</td>
<td>1470</td>
<td>1980</td>
<td>1990</td>
<td>38</td>
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<td>SuperPhenix (France)</td>
<td>3000</td>
<td>1985</td>
<td>1997</td>
<td>12</td>
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<td>PBTR (India)</td>
<td>40</td>
<td>1985</td>
<td></td>
<td>33</td>
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<tr>
<td>MAMU (Japan)</td>
<td>714</td>
<td>1994</td>
<td>2016</td>
<td>22</td>
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<tr>
<td>BN 400 (Korea)</td>
<td>2000</td>
<td>2014</td>
<td></td>
<td>5</td>
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<tr>
<td>CEFR (China)</td>
<td>65</td>
<td>2010</td>
<td></td>
<td>4</td>
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<tr>
<td>PFBR (India)</td>
<td>1250</td>
<td>Under</td>
<td></td>
<td>450</td>
</tr>
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</table>

Total all fast reactors: 450
Projects of SFRs Worldwide

France: ASTRID  
Fuel: MOX-RNR

Russia: BN-800, MBIR, BN 1200  
Fuel: UOX, MOX-RNR, Nitride

China: CEFR, CFR600, CCFR1000  
Fuel MOX-RNR

USA: designs of SMR, Terrapower, PRISM, VTR  
Collaboration under discussion with the USA

India: PFBR, 2xFBR (500 MWe), MFDR, MFBR  
Fuel: MOX, metallic

South Korea: PGSFR (150MWe, 2028)  
Fuel: metallic

Japan: prototype JSFR (stand-by)
SFR Design (Basic Principle)
Favorable Features of SFR

- The whole primary circuit is contained in the main vessel, including the core, the intermediate heat exchangers and the primary pumps.
- The primary system is not pressurized.
- The intermediate (or secondary) system transfers the energy to steam generators, thus providing for an extra containment between the primary circuit and the environment.
- Large boiling margin of sodium (>300 K)
- The large quantity of primary coolant provides for a high thermal inertia in case of loss of main heat sink.
- Good natural convection and circulation features allow to design passive, diversified decay heat removal systems.
- Power control by single rod position, no xenon effect, no need of soluble neutron poison.
- Collective dose on a pool type SFR is very low compared to PWR.
## Improvements in SFR Design

<table>
<thead>
<tr>
<th>Feedback of previous SFRs</th>
<th>R&amp;D directions</th>
<th>ASTRID Orientations</th>
</tr>
</thead>
</table>
| **Core Sodium voiding reactivity**  
  → Safety | Optimization of core design to improve natural behavior during abnormal transients.  
  Exploration of heterogeneous cores | CFV core: innovative approach, negative overall sodium voiding reactivity  
  Better natural behavior of the core, for instance in case of loss of flow (e.g. due to loss of supply power). Avoid neutronic power excursion |
| **Sodium-Water interaction**  
  → Safety - Availability | Robust steam generators  
  *2 options are studied*  
  Gas Power Conversion System (nitrogen in place of steam/water), *that will allow to physically avoid the sodium-water reaction* | Limitation of total released energy in case of sodium-water interaction, and integrity of the envelop of the steam generator and the secondary loop.  
  Design studies conducted by General Electrics. No show stopper.  
  Design studies on sodium-gas exchanges conducted with FRAMATOME. |
## Improvements in SFR Design

<table>
<thead>
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<th>Feedback of previous SFRs</th>
<th>R&amp;D directions</th>
<th>ASTRID Orientations</th>
</tr>
</thead>
<tbody>
<tr>
<td>Sodium fire → Safety</td>
<td>Innovative Sodium leak detection systems</td>
<td>Improving detection (Patent of detection system integrated in the heat insulator)</td>
</tr>
<tr>
<td></td>
<td>R&amp;D on Sodium aerosols</td>
<td>Close containment (inert gas + restriction of available oxygen)</td>
</tr>
<tr>
<td>Severe accidents → Safety</td>
<td>Core catcher</td>
<td>Core catcher</td>
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<td></td>
<td>Research on corium and sodium-corium interaction</td>
<td>Transfer tubes</td>
</tr>
<tr>
<td>Decay heat removal → Safety</td>
<td>Reactor vessel auxiliary cooling system (scaling rules)</td>
<td>Combination of proved Decay Heat Removal systems and Vessel Natural Air draft cooling with the objective to practically eliminate the long term loss of the function</td>
</tr>
<tr>
<td>In-Service Inspection and Repair</td>
<td>ISI&amp;R taken into account from the design stage and Simplification of primary system design</td>
<td>Under-sodium viewing: improvement of Signal processing and sensor technologies (ultrasound at high temperature, fission chambers, Optical Fibers, ...)</td>
</tr>
<tr>
<td></td>
<td></td>
<td>Remote handling for inspection or repair</td>
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</table>
The ASTRID Program
(Advanced Sodium Technological Reactor for Industrial Demonstration)

- ASTRID is a **technological demonstrator** and is not a First of a Kind of a commercial reactor.

- Based on the feedback experiences of past Sodium-cooled Fast Reactors, ASTRID has the objective to demonstrate at a scale allowing the industrial extrapolation, the relevancy and performances of innovations, in particular in the fields of **safety and operability**.

- ASTRID with the related R&D facilities will allow:
  - to test and qualify innovative safety design options towards the commercial reactor,
  - to qualify different fuels (transmutation, plutonium burner, …),
  - to obtain the necessary data to justify a useful lifetime of 60 years for future SFR,
  - to confirm performances of innovative components and systems in order to optimize the design of future commercial reactors from a technical and economical points of view,
  - to establish a reference for the SFR cost assessment (building and operation).
The technology of ASTRID allows to have a very resilient design to external events (earthquake, flooding, loss of power, airplane crash...)

Based on the feedback experiences of past Sodium-cooled Fast Reactors operated in the world, examples of innovations:
- Mitigation devices (core catcher...)
- Larger in-service inspection capabilities
- Core with an improved intrinsic behavior
- Gas power conversion system

Industrial partners
Leaders in nuclear and high-tech
ASTRID: Partnerships Organization

EDF R&D, PSI, Sweden (KTH, Chalmers, Uppsala), HZDR, KIT, ENEA, JRC/ITU, NNL, CIEMAT ...

CNRS (NEEDS), Universities
### CEA Experimental Platforms

In addition to available foreign experimental platform, CEA is carrying out particular investments

<table>
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<th>Needs</th>
<th>CEA Platform</th>
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<tr>
<td>Neutronic qualification of innovative core</td>
<td><strong>MASURCA</strong> : <strong>under reconsideration / International reactors</strong></td>
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</tbody>
</table>
| Analytical water tests, TH code validation (gas entrainment, hot pool flows), Component qualification (ISIR) | **GISEH** : Acronym for “Group Installations in Surrogate coolant for Hydraulics, thermal-hydraulics, mechanics, fluid-structure interaction”  
  **In operation**                                                  |
| Small Na loop (<3 m³ Na) TH code validation and Component and technological qualification (under Na viewing) | **PIAPURUS** : Acronym for “Parc of small Installation of R&d for Utilization of Sodium  
  Corrosion test, heat exchanger test, instrumentation...  
  **In operation**                                                 |
| Large Na loop (<100 m³ Na) Component qualification (close to scale 1 prototypes) | **CHEOPS** : Sodium-gas heat exchanger or steam generator mock-up, subassembly thermal-hydraulics, Control rods, passive shutdown system qualification, sodium fuel handling, ...  
  **Under reconsideration / Jp platform**                      |
| Severe Accidents corium behaviour, Qualification of mitigation device (core catcher...) | **PLINIUS-2** : experimental studies of corium-sodium-interaction and core catcher (100-500 kg of UO2), analytical test  
  **Under design. Decision to build**                           |
Use of Digital in ASTRID Project…

- Design and operation of innovative systems
- Assessment of performances
- Management of a large & complex project (13 ind. partners, up to over 500 participants)
… Model Complex Phenomena to Consolidate Demonstrations

- Numerical simulation is required to support design studies and safety analysis
- Multi-physics and multiscale modelling

Modelling of an accidental scenario with loss of the primary pump of ASTRID reactor, without control rod protection (ULOIF). Natural convection is leading to a stabilised sodium flow in the core allowing core assemblies cooling and avoiding a severe accident situation (no core or fuel melting). CEA/Saclay studies
... Management of a Large Complex Project

A lot of partners

Organization based on systems engineering approach

Plant Life Management

3D numerical model and configuration management
- Model from models provided by 13 partners
- 15,000 interface data
- 20 Gigas of data and 200,000 objects in the CAO 3D mockup
Advantages From the Use of Virtual Reality

Promising deployment of the use of Virtual Reality

- Design
  - Large data treatment
  - Fusion of 3D PDMS / CATIA models

- Review of CAO mock-up
  - Technical review realized inside the 3D meeting room

- Design-to-Maintenance
  - Integration of maintenance and handling operations at the early stage of design

- Kinematics studies
  - Validation of some complex kinematics

Example of a kinematic of handling processes inside the ASTRID hot cell
Main Achievements for 2015, and After…

- A synthesis file was sent to the government mid 2015:
  - Strategy leading to the choice of Gen IV sodium cooled fast reactor and closed fuel cycle.

- Synthesis file summarizing the conceptual design phase (2010-2015) provided in December 2015:
  - Scope statement, with technological choices (including conversion system), issued from Conceptual Design.
  - Workplan for Basic Design, with associated R&D infrastructures.

- Authorization at the end of 2015 from the government to proceed until the end of 2019 (Basic design phase).
ASTRID Main Technical Choices

- 1500 thMW - ~600 eMW
- Pool type reactor
- With an intermediate sodium circuit
- CFV core (low sodium void worth)
- Oxide fuel UO2-PuO2
- Preliminary strategy for severe accidents (internal core catcher, no large mechanical energy release, …)
- Redundant and diversified decay heat removal systems
- Fuel handling in sodium + combination of internal storage and small external storage (to increase of the availability rate)
Option with Steam Water Power Conversion System
Option with Gas Power Conversion System
ASTRID Design: the Fuel Handling Route
Progress on constructibility operation and Balance of Plant
ASTRID - Balance Of Plant
Lessons Learned

- Nuclear energy is a well proven source of large baseload electricity, with no GHG emissions. It will remain one of the pillars of the future French low carbon energy mix.
- The closed fuel cycle associated with FNR will lead to drastic improvement in U resources management, and important reduction in footprint and radiotoxicity of final wastes.
- French program on Generation IV is based on:
  - ASTRID program
  - Basic design phase on-going (2016-2019)
  - Schedule and organization for next phases are under preparation with French government and industrial partners
  - One option could be to review the power of the demonstration reactor. In that option, a large part of works performed from 2010 will be reused (design processes, methodologies (V&V&Q for instance), new generation numerical tools, PLM, lot of innovative design options (maturity level is known…). Efficiency for next works will then be improved and current on-going R&D program is mostly relevant.
  - An active survey on other GenIV fast and thermal neutrons system
Lessons Learned

- SFR is a mature technology because many SFR reactors built from the 50’s to the 70’s were then operated. But the gap to achieve a GenIV concept is significant because GenIV is requesting improvements mainly in safety, operational and economics aspects; and it is impacting the related design.

- Even if mature, the SFR technology is not obvious and in that field knowledge preservation and transmission to the coming young generation is also a key challenge if you want to keep this key technology available for decades. Thus the use of sodium as coolant – as for the other liquid metal or Helium coolants – needs courses, practice and skills.

- Innovation is the way to design new reactors. It needs to get a close relationships between industry and design teams in one hand and R&D teams on the other hand. The role of the ASTRID Team project is to make them run together.

- SFR reactor design cannot be achieved without international collaboration, mainly to mutualize technological platforms and infrastructures. It is a win-win cost savings approach.
Thank you for your attention
<table>
<thead>
<tr>
<th>Date</th>
<th>Topic</th>
<th>Speaker</th>
<th>Institution</th>
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<tbody>
<tr>
<td>22 August 2018</td>
<td>BREST-300 Lead-Cooled Fast Reactor</td>
<td>Dr. Valery Rachkov</td>
<td>IPPE, Russia</td>
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<tr>
<td>26 September 2018</td>
<td>Advanced Lead Fast reactor European Demonstrator – ALFRED Project</td>
<td>Dr. Alessandro Alemberti</td>
<td>Ansaldo Nucleare, Italy</td>
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<tr>
<td>24 October 2018</td>
<td>Safety of Gen IV Reactors</td>
<td>Dr. Luca Ammirabile</td>
<td>EU</td>
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