GEN IV International Forum[SM]

Expertise | Collaboration | Excellence

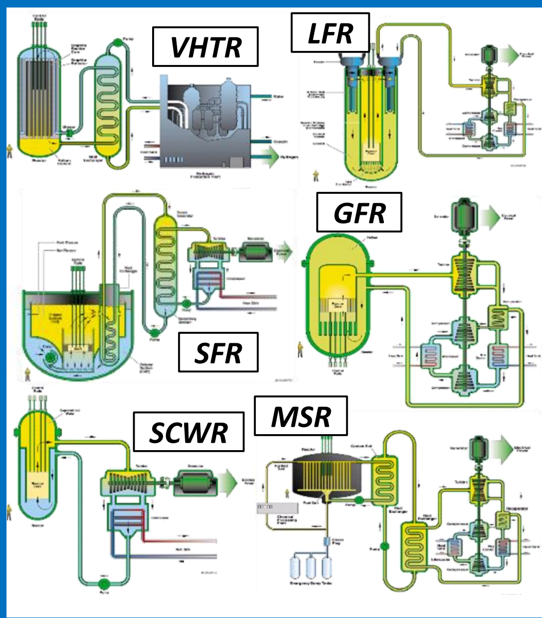
Webinar-GuideBook

GIF Webinar Series

The GIF Education and Training Working Group invites you to participate in monthly webinars presented by speakers from around the world, explaining why GEN IV reactor systems are crucial for the sustainability of the nuclear fuel cycle. Launched in September 2016, the current webinar series includes 35 recordings of lectures already conducted and upcoming presentations tentatively scheduled through 2020.

Mission Statement

The GIF-ETTF serves as a platform to enhance open education and training as well as communication and networking of people and organizations in support of GIF.



Providing Opportunities to learn about advanced reactors

- Identify the stakeholder groups and assess their needs for Generation IV E&T
- Create and maintain a social medium platform to exchange information and ideas on general Gen IV R&D topics as well as related GIF ETTF activities
- Develop and launch webinar series on Gen IV systems and cross-cutting methodologies
- Propose, organize and/or support Gen IV E&T seminars

https://www.gen-4.org/gif/jcms/c_82831/webinars

Members of GIF Education and Training Task Force

Chair: Patricia Paviet
Co-Chair: Konstantin Mikityuk

Members (13) in alphabetical order:

Bucalossi, Andrea	Fratoni, Massimiliano	Harrison, Grace
Hwang, Il Soon	Kulikov, Evgeny	Jun, Sun
Latge, Christian	Liu, Xiaojing	Mihara, Takatsugu
Mikityuk, Konstantin	Mpoza, Nolitha	Nam, Youngmi
Paviet, Patricia		

As of Oct 2019

Webinar list

8. Webinars by winners of the Contest for young generation (EPiC)
   - 8-1. Cement Matrix for Nuclear Waste
   - 8-2. Interactions between Sodium and Fission Products in Case of a Severe Accident in a Sodium-cooled Fast Reactor
   - 8-3. Security Study of Sodium-Gas Heat Exchangers in Frame of Sodium-cooled Fast Reactors

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1-1. ATOMS FOR PEACE
The Next Generation

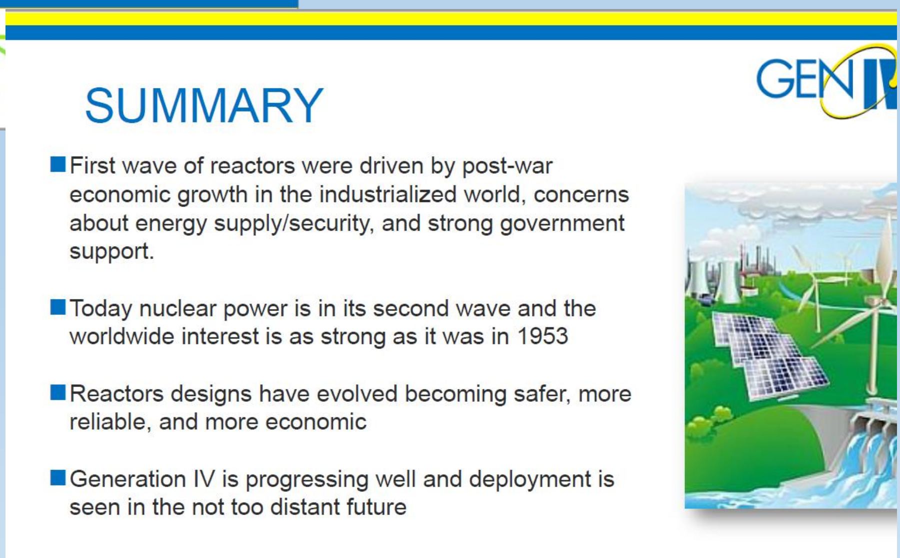
Webcast: 29 September 2016

Summary / Objectives:

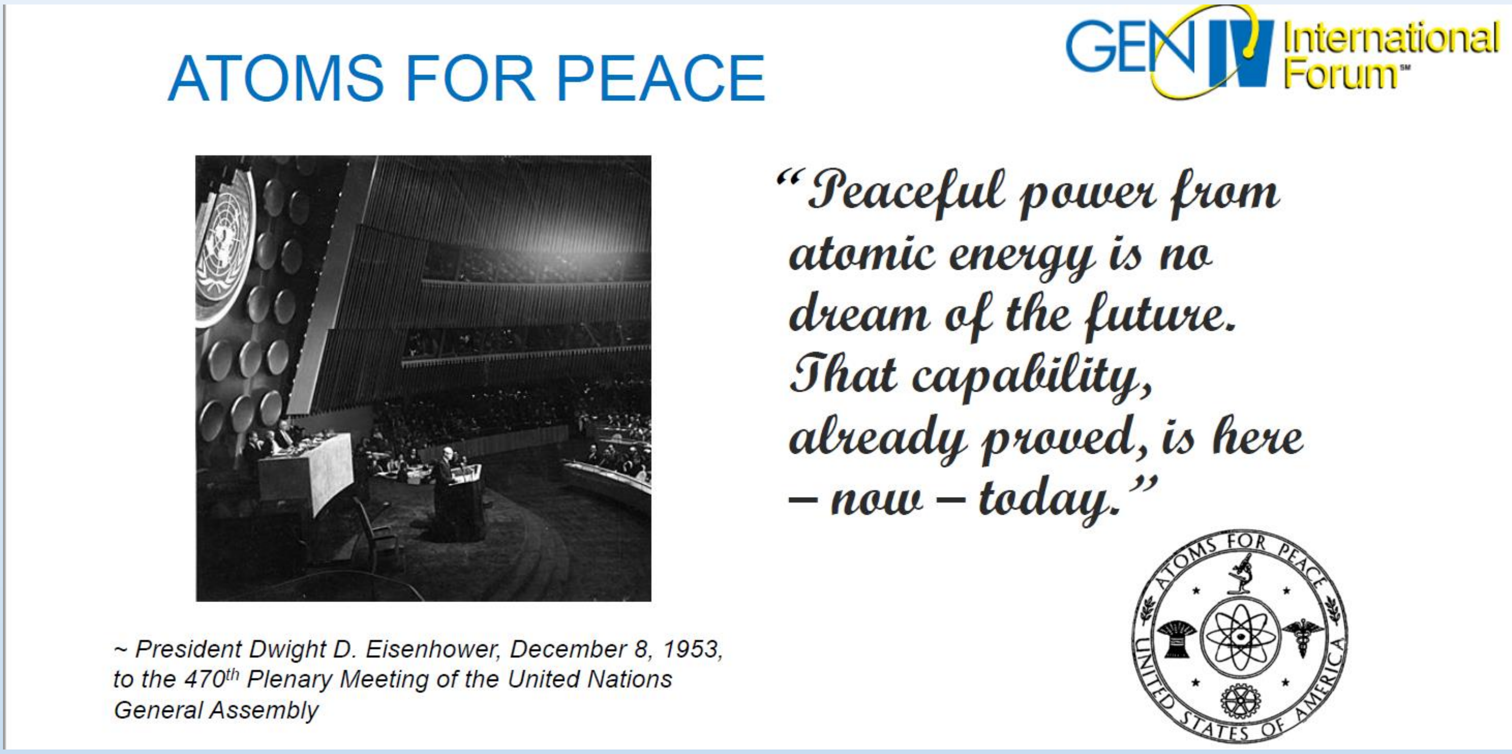
This webinar provides a historical perspective on the Atoms for Peace program, which launched the development of nuclear power around the globe, and describes the current outlook for the development and deployment on the next generation of nuclear power (Generation IV).

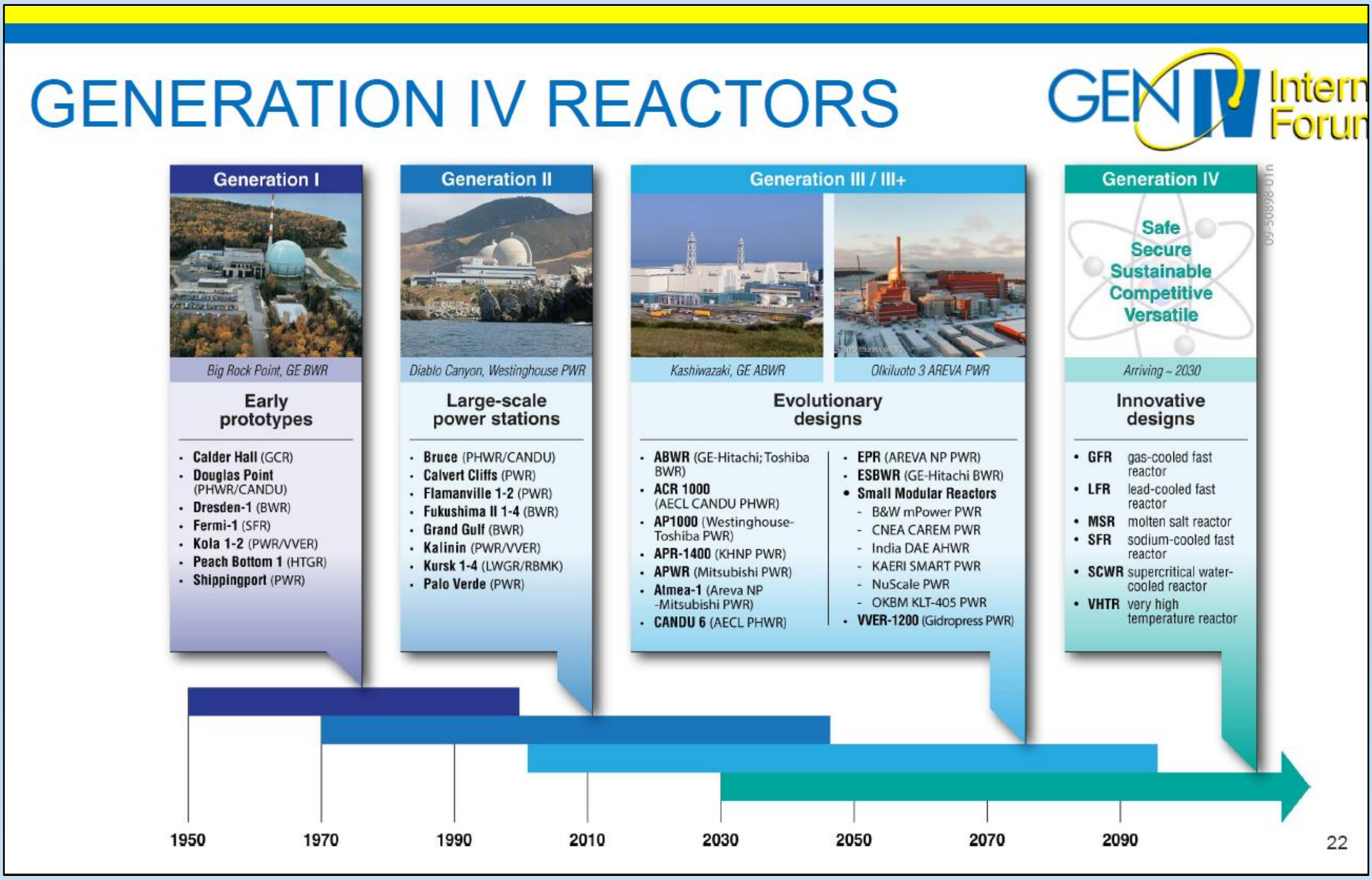
Meet the Presenter:

Dr. John E. Kelly is the Deputy Assistant Secretary for Nuclear Reactor Technologies in the Office of Nuclear Energy, U.S. Department of Energy. He is responsible for the U.S. civilian nuclear reactor research and development portfolio, which includes programs on Small Modular Reactors, Light Water Reactor sustainability, and Generation IV reactors.



For peaceful use of nuclear energy as electric power plants, President Eisenhower's speech as Atoms for peace in 1953 is a symbol of game change. After that early prototypes of power plants (Generation I) have developed into Large-scaled (Gen II) and present Evolutionary designs (Gen III including ABWR,APWR, VVER-1200,SMR). Now that we are developing Gen IV reactors.





https://www.gen-4.org/gif/jcms/c_59461/generation-iv-systems

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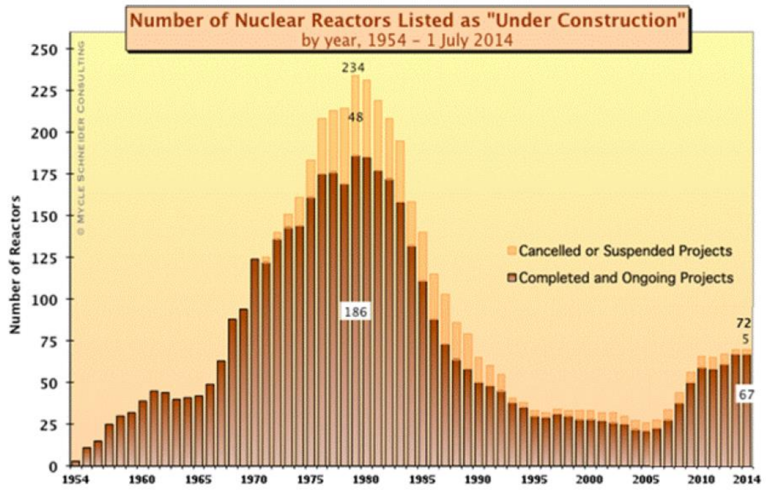
NUCLEAR POWER PLANTS BUILT WORLDWIDE



Source: World Nuclear Report

Two waves of nuclear power plants built, the first 1970s-1980s and the second 2010s. Based on the different drivers.

DRIVERS FOR THE FIRST WAVE OF REACTORS

- **Encouraging drivers**
  - Re-emerging Economies Required Increased Energy in Post World War II Period
  - The Oil Crises of the 1970s
  - Strong Government Backing

- **Neutral drivers**
  - Acid Rain
  - Air Pollution
  - 1971- Inadvertent Climate Modification. Report of the Study of Man's Impact on Climate

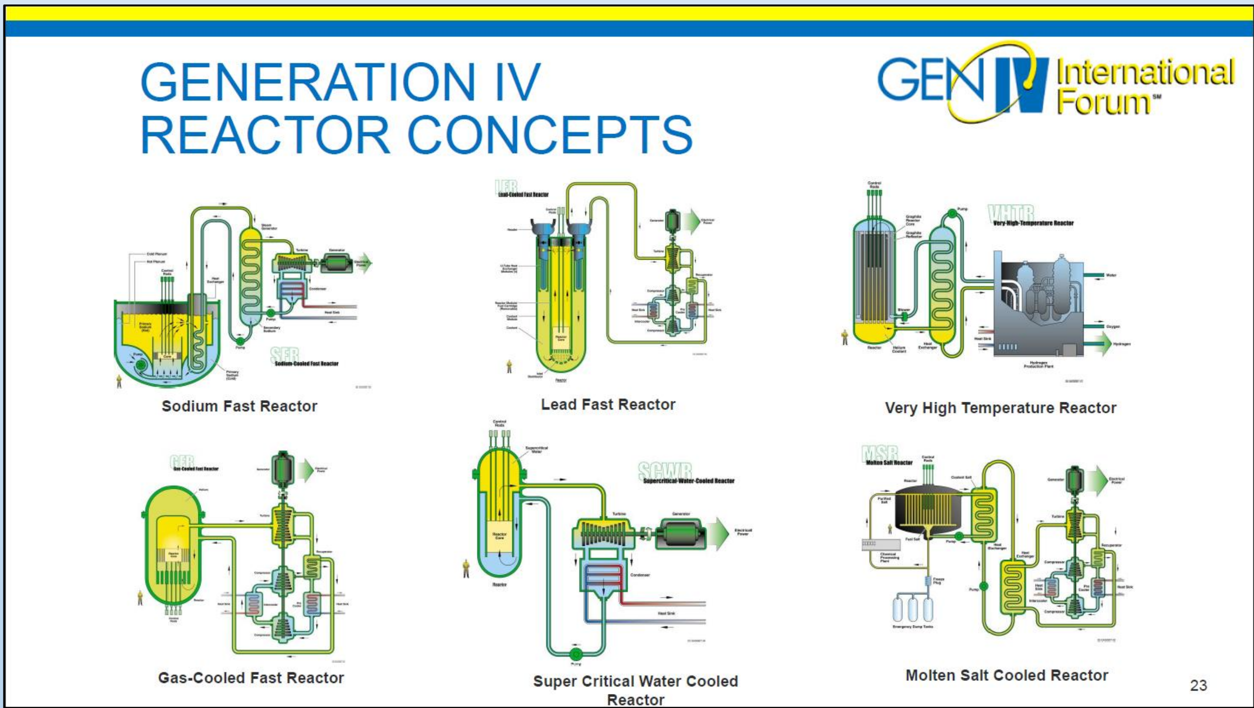
- **Discouraging drivers**
  - High Interest Rates
  - Fear of Radiation
  - Fear of Nuclear Weapons
  - Three Mile Island Accident
  - Chernobyl Accident
  - Waste Management Impasse

CURRENT DRIVERS FOR NUCLEAR POWER

- Energy security
  - Nuclear shelters countries from imports of costly fossil fuels
  - Replacing retired nuclear or coal generation plants
- Economic incentives
  - Nations rich in fossil fuel would prefer to export those resources and use nuclear for domestic electricity production
- Environmental protection
  - Replacing coal with nuclear can alleviate air pollution problems
- Climate change concerns
  - Nuclear is the “emission-free” base load generation technology
  - Dry condenser cooling possible with small modular reactors when water usage is restricted

GIF has led international collaborative efforts to develop next generation nuclear energy systems that can help meet the world's future energy needs. Generation IV designs will use fuel more efficiently, reduce waste production, be economically competitive, and meet stringent standards of safety and proliferation resistance.

With these goals in mind, some 100 experts evaluated 130 reactor concepts before GIF selected six reactor technologies for further research and development. These include the: Gas-cooled Fast Reactor (GFR), Lead-cooled Fast Reactor (LFR), Molten Salt Reactor (MSR), Supercritical Water-cooled Reactor (SCWR), Sodium-cooled Fast Reactor (SFR) and Very High Temperature Reactor (VHTR).



COLLABORATIONS

Generation IV Systems	Canada	China	France	Japan	Korea	Russia	Switzerland	U.S.A.	EU
Sodium-cooled Fast Reactor (SFR)		●	●	●	●	●		●	●
Very-high Temperature Gas cooled Reactor (VHTR)		●	●	●	●		●	●	●
Gas-cooled Fast Reactor (GFR)			●	●					●
Supercritical-water cooled Reactor (SCWR)	●	●		●		●			●
Lead-cooled Fast Reactor (LFR)				●	●	●			●
Molten Salt Reactor (MSR)			●			●	●		●

● *Participating member, signatory of a System Arrangement as of July 2016*

25

As of 2016, for the latest see the below site.
https://www.gen-4.org/gif/jcms/c_9342/framework-agreement

1-2. Introduction to Nuclear Reactor Design

Summary / Objectives:

Why is a 4th generation of nuclear reactors needed? And what are the most promising reactor technologies? The GIF initiative has led to reconsider some of the options adopted in the past and stimulated the investigation of new tracks for long term sustainable nuclear energy. To grasp the rationale for selecting Generation IV reactor systems, and their main characteristics, requires some basic knowledge in the fundamentals of nuclear reactor design. What is behind the terms “criticality,” “breeding,” and “fast or thermal neutrons”? How to select the coolant, moderator, neutron spectrum, fuel materials and composition and to choose the ad hoc combinations to design nuclear reactors in line with Generation IV criteria, in particular sustainability? This is the objective of this rather technical webinar targeting civil society stakeholders.

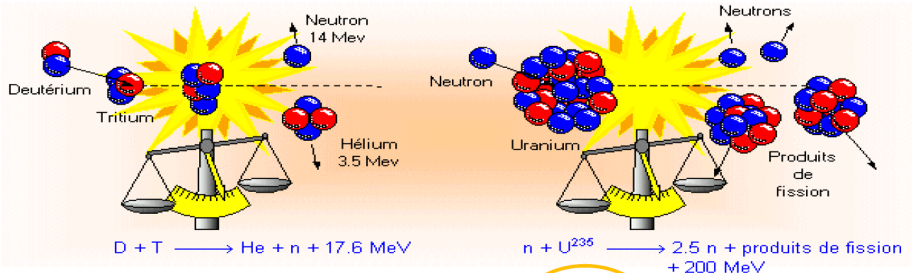
Meet the Presenter:

Dr. Claude Renault has been working at CEA for more than 30 years in R&D and E&T. He is a senior expert at CEA and professor. In 2010, he joined the INSTN where he is currently the International Project Leader. His expertise and teaching experience mainly cover thermal-hydraulics, design and operation of nuclear reactors, including the different families of reactors in particular the concepts of 4th generation. Claude Renault came to CEA in 1984 in the development team of CATHARE, the reference CEA-EDF-AREVA-IRSN computer code for the simulation of accidental transients in Pressurized Water Reactors (PWR). He was subsequently responsible, at national and international level, for several R&D projects in the areas of severe accidents (ASTEC) and nuclear fuel behavior (PLEIADES).
Between 2001 and 2009, he was heavily involved in R&D programs devoted to future nuclear reactors. He intervened at the Directorate of Nuclear Energy (CEA/DEN) in the definition and monitoring of research programs on the different concepts of 4th generation reactors. He chaired the Steering Committee of the Molten Salt Reactor in Generation IV.

Why Generation IV, especially fast reactors?

Fission, fusion, fossil fuel burning?

The potential of nuclear energy is fantastic!



Combustion of 1 ton of fossil oil:	0.5 MWd	(42 GJ)
Total fission of 1 g of ^{235}U:	1 MWd	(83 GJ)
Total fusion of 1 g of fuel (D,T):	4 MWd	(330 GJ)

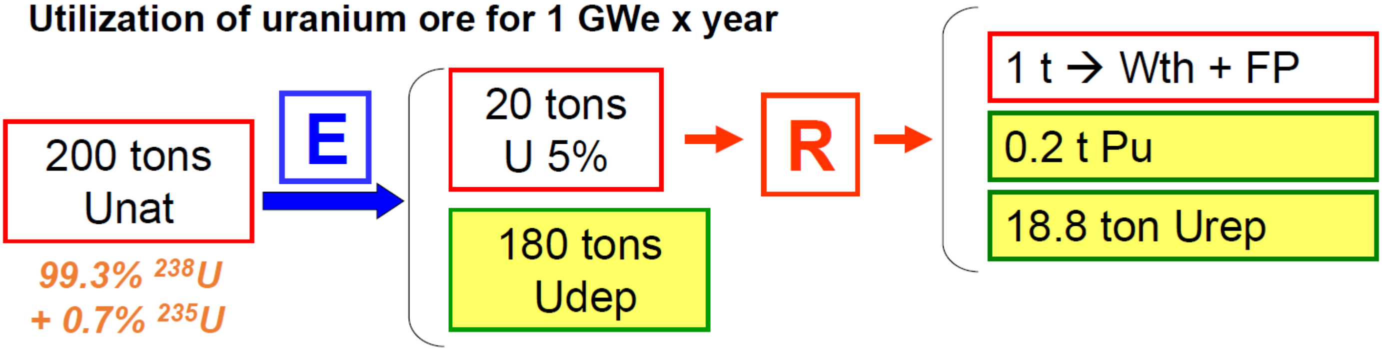
6

2,000,000 times energy from fission than fossil energy like coal, oil, gas.

Why is a new generation of nuclear reactors needed?

Open cycle in LWRs

Utilization of uranium ore for 1 GWe x year



In PWRs, about 5% of the initial uranium set in reactor (enriched U) is consumed for electricity production (fuel technological limits)

This represents only 0.5-0.6% of the initial natural uranium

Breeder reactors (FNRs) need only 1 ton U238 (Udep & Urep) that is converted into plutonium and burned in situ *(regeneration → breeding of fissile fuel)*

24

200 tons U for 1GWe electricity in PWRs, **1 ton U238** in FNRs.
EBR-1 , 1951 USA Idaho: Uranium metal fuel and NaK primary coolant, Fast neutron power reactor.
(BORAX-III, 1955 Thermal neutron power reactor for BWR type.)

What is the condition for self-sustained reaction?

A necessary condition for criticality is that the reproduction factor η is significantly larger than 1

$$k = \frac{\bar{\nu}\dfrac{\Sigma_f}{\Sigma_a}}{1 + \dfrac{AR_{other} + LR}{AR_{fuel}}}$$

Reproduction factor η for uranium fuel (fissile fraction e):

Fissile fraction e	0.71 % (U nat)	3 %	10 %	15 %	100 %
For fast neutrons	0.10	0.35	0.85	1.07	1.88
For « thermal » neutrons	1.33	1.84	2.00	2.02	2.07

The chain reaction is not possible with natural uranium and fast neutrons.

Therefore 2 solutions:

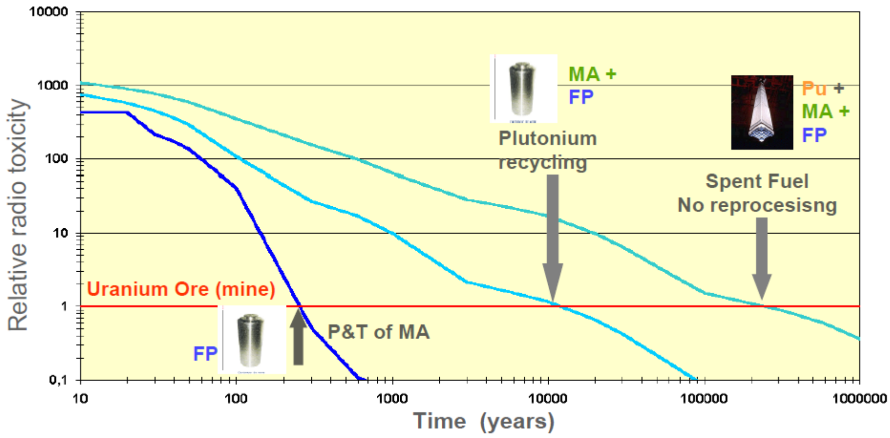
- **to slow down neutrons (criticality possible whatever the fissile content, Unat possible for strict neutron economy)**
  ➔ ***Thermal Neutrons Reactors, TNR*** **(PWR, BWR, CANDU,…)**
- **to use fast neutrons and subsequently increase the fissile fraction in the fuel**
  ➔ ***Fast Neutrons Reactors, FNR***

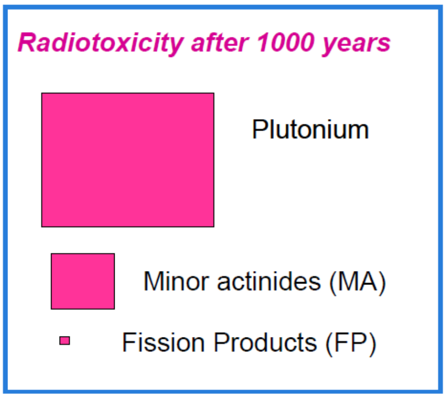
16

Adequate fissile fraction for thermal neutron reactors and fast neutron reactors.

Why Fast Neutron Reactors? The waste management issue

- Plutonium is the major contributor to the long term radiotoxicity of spent fuel → ***Plutonium recycling***
- After plutonium, MA (Am, Cm, Np) have the major impact to the long term radiotoxicity → ***MA transmutation***





The ratio fission/capture is favourable to MA fission with fast neutrons

27

Comparison of radiotoxicity in spent fuel after 1000 years.
Pu for recycling, MA for transmutation.

Nuclear reactors today

General characteristics of nuclear reactors in operation

Reactor type	Fuel type	Moderator	Coolant	Core power density (MW/m^3)	Pressure (bar)	Temperature (°C)	Efficiency (%)
UNGG	**Unat**	C	CO_2	1	41	400	30
Magnox							
PHWR		D_2O	D_2O	12	130	300	30
LWGR	**U 1-2%**	C	H_2O	2	70	284	31
AGR		C	CO_2	3	40	645	40
BWR	**U 3-5%**	H_2O	H_2O	50	72	288	37
PWR				100	155	330	35
FBR (FNR)	**Pu 20-30%**	-	Na	500	1	550	40

21

	SFR	LFR	GFR	VHTR	SCWR	MSR	*PWR*
Neutron spectrum (T/F)	F	F	F	T	T/F?	T/F	*T*
Moderator	[illegible]	[illegible]	[illegible]	graphite	H_2O (or D_2O)	graphite (or none)	*H_2O*
Coolant	Na	Pb (or Pb-Bi)	He	He	H_2O	molten salt	*H_2O*
Fuel type	MOX (pins)	nitride (pins)	carbide	carbide (particles)	UOX, MOX	liquid fuel (U, Pu, Th)	*UOX, MOX*
Core outlet t° (°C)	550	500	850	> 900	550	700	*330*
Primary pressure (MPa)	0.1	0.3-0.4	7	5-8	25	0.1-0.2	*15.5*
Core power density (MW/m^3)	240	140	100	4-6	100	20-300	*100*

The values given in the table are fairly indicative!

The design of Gen IV systems is ongoing (R&D development work) 36

Comparison of core power density and plant parameters.

GIF and a new generation of nuclear systems

Nuclear is a CO_2-free option for sustainable energy

New requirements for sustainable nuclear energy

Search innovative solutions for:

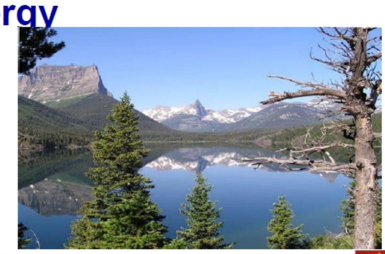
- Waste minimisation
- Natural resources conservation
- *Proliferation resistance*

Perform continuous progress on:

- Competitiveness
- Safety and reliability

Develop the potential for new applications:

hydrogen, syn-fuels, desalinated water, process heat



➔ ***Systems marketable from 2040 onwards***

33

2-1. Safety of Generation IV Reactors

Summary / Objectives:

Excellence in safety and reliability is among the goals identified in the technology roadmap for Generation IV nuclear reactors. This webinar will give an overview of the activities of the GIF Risk and Safety Working Group done in support of the six Generation IV nuclear energy systems towards the fulfilment of this goal. Topics include a presentation of the safety philosophy for Generation IV systems, the current safety framework for advanced reactors, and the methodology developed by the group for the safety assessment of Generation IV designs. Other ongoing activities between the group and the designers of Generation IV systems will be also highlighted.

Meet the Presenter:

Dr. Luca Ammirabile works at the European Commission (EC), Joint Research Centre in Petten, the Netherlands, where he is Group Leader of the NUclear Reactor Accident Modelling (NURAM) team of the Nuclear Reactor Safety and Emergency Preparedness Unit. His group deals with Nuclear Reactor Safety assessment for current and innovative reactors, focusing on the safety issues related to the prevention and mitigation of Severe Accident conditions and Source Term estimation. His current research activities are core thermal-hydraulic analyses, deterministic code application and development, and safety assessment of advanced reactors. Since 2014, he has been co-chairman of the working group on Risk and Safety of the Generation IV International Forum. He is also the EC representative on the OECD/NEA Working Group for the Analysis and Management of Accidents (WGAMA) and the Working Group for the Safety of Advanced Reactors (WGSAR). Prior to joining the European Commission in 2007, Luca worked at Tractebel Engineering (now Tractebel Engie) in Belgium in the Thermal-hydraulics and Severe Accident Section, where he was engaged, among other projects, in the development of innovative methodologies in support of the safety assessment of the Belgian Nuclear Power Plants.
Luca received his doctorate from the Imperial College London in 2003 and his master's degree in nuclear engineering from the University of Pisa, Italy in 1999.

Risk and Safety Working Group :

The primary objective of GIF Risk and Safety Working Group (RSWG) is "Promote a consistent approach on safety, risk, and regulatory issues between Generation IV systems".

For this purpose, RSWG developed and have promoted a technology-neutral Integrated Safety Assessment Methodology (ISAM).

System	Neutron Spectrum	Coolant	Pressure (MPa)	Temperature (°C)	Fuel Cycle	Size (MW)
GFR	Fast	Helium	~9	850	Closed	1200
LFR	Fast	Lead	0.1+ (atm.	480–800	Closed	45-1500
MSR	Fast or Thermal	Fluoride or hloride salts	0.1+ (atm.	700–800	Closed	1000-1500
SFR	Fast	Sodium	0.1+ (atm.	550	Closed	50–1500
ScWR	Thermal or fast	Water	~25	510–625	Once-through or Closed	10–over 1000
VHTR	Thermal	Helium	~5.5	900–1000	Once-through	250–300

Explanation of Safety & Reliability Goals (Defence in Depth) :

GIF Safety & Reliability Goals are corresponding with the concept of Defence in Depth.

- Excel in Operational Safety and Reliability
  - ➢ DiD Level 1-2 [N.O., AOO]
- Very low likelihood & degree of reactor core damage
  - ➢ DiD Level 2-3 [Design for severe accident prevention]
- Eliminate the need for offsite emergency response
  - ➢ DiD Level 4 [Design for severe accident mitigation]

Defense-in-Depth Levels				
Level 1	Level 2	Level 3	Level 4	Level 5
Operational states		Accident conditions		EP&R
Normal Operation	Anticipated Operational Occurrences	Design Basis Accidents	Design Extension Conditions	Residual risk and practically eliminated accidents
			Severe accidents	

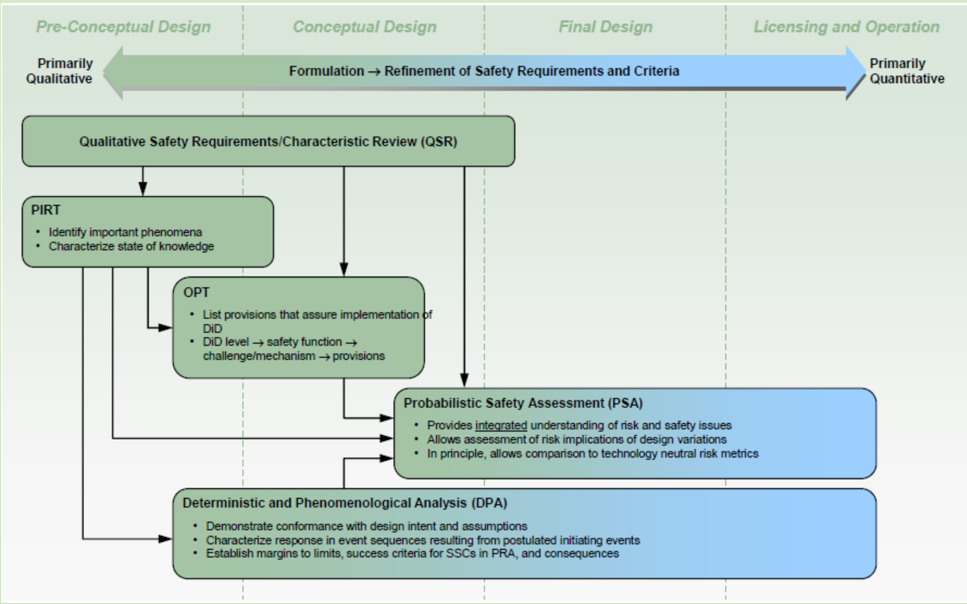
Plant states considered in design
(safety analyses)

Out of the design
(addressed in level-5 of DiD)

Integrated Safety Assessment Methodology (ISAM):

The ISAM consists of five distinct analytical tools.

- Qualitative Safety-characteristics Review (QSR)
- Phenomena Identification and Ranking Table (PIRT)
- Objective Provision Tree (OPT)
- Deterministic and Phenomenological Analyses (DPA)
- Probabilistic Safety Analysis (PSA)



Qualitative Safety-characteristics Review (QSR):

QSR is "check-list" as systematic and qualitative means of ensuring that the design incorporates desired safety attributes (preparatory step).

Phenomena Identification and Ranking Table (PIRT):

PIRT is generated for the purpose of identifying system and component vulnerabilities, and relative contributions to safety and risk.

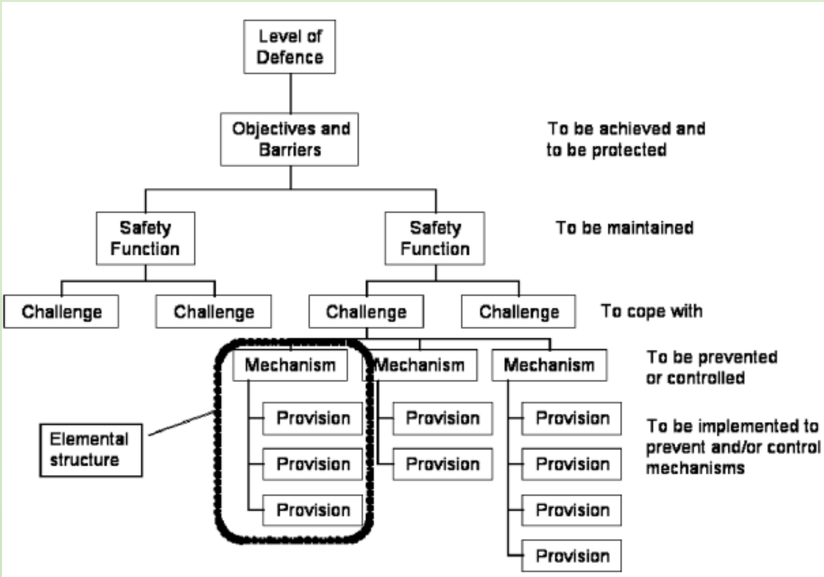
System	Component	Phenomena/Characteristics/State variables	IR		KL_1		KL_2	
			A	B	A	B	A	B
BRSS	SASS	SASS actuation temperature	H	H	1	2	3	4
Reactor	Upper core region around SASS	Coolant transport delay time from core outlet to around SASS	H	H	3	2	3	3
		Time constant of temperature response delay from coolant around SASS to SASS device	M	M	1	2	3	3
	Reactor core	Core outlet temperature of the coolant that flows to around SASS	H	H	3	3	3	3
		Doppler reactivity	M	M	4	4	4	4
		Fuel temperature reactivity	L	M	4	3	4	3
		Fuel cladding temperature reactivity	M	M	4	4	4	4
		Coolant temperature reactivity	H	H	4	4	4	4
		Coolant flow rate halving time	H	H	4	4	4	4
		Power distribution	M	M	4	4	4	4
		Flow rate distribution among core assemblies	M	M	4	4	4	4
		Coolant temperature at the core inlet and outlet	L	L	4	4	4	4
		Fuel pin gap heat transfer coefficient	M	M	4	3	4	3
		Fuel pellet thermal conductivity	I	I	4	4	4	4
		Thermal material property of fuel cladding and coolant	I	I	4	4	4	4
RPCS	Temperature I&C	Coolant temperature to be used reactor power control	M	L	4	4	4	4
PHTS	Pump	Pump rotating inertia	M	M	4	4	4	4
	-	Pressure loss in the reactor and PHTS	M	M	4	4	4	4

PIRT

Knowledge Base Gap Determination				
Adequacy of knowledge	Rank of Phenomenon			
	H	M	L	I
(4) Fully known; small uncertainty				
(3) Known; moderate uncertainty				
(2) Partially known; large uncertainty	GAP	GAP		
(1) Very limited knowledge; uncertainty cannot be characterized	GAP	GAP	GAP	

Objective Provision Tree (OPT):

OPT is a tool for identifying the provisions for prevention, or control and mitigation, of accidents that could potentially damage the reactor.

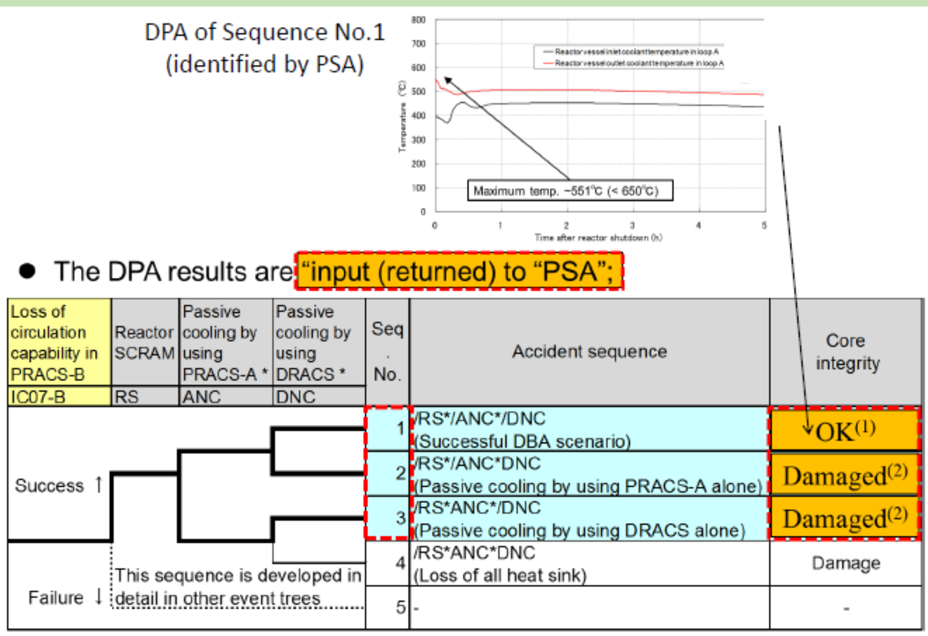


Deterministic and Phenomenological Analyses (DPA):

DPA is traditional safety analyses to assess the system's response to known challenges and guide concept/design development. Based on conventional safety analysis codes, DPA provides input to PSA.

Probabilistic Safety Analysis (PSA) :

PSA is performed in order to assure a broader coverage of the accident space. PSA is iterated from the late pre-conceptual design phase to the final design stages.



2-2. SFR Safety Design Criteria (SDC) and Safety Design Guidelines (SDGs)

Summary / Objectives:

This webinar provides the outlines of the safety design criteria (SDC) and safety design guidelines (SDG) established to achieve high development goals of Gen IV reactors including safety and reliability. Reflecting the lessons learned from the Fukushima Daiichi nuclear power plant accident, the SDC describes requirements that must be met by Gen IV Sodium-cooled Fast Reactors (SFRs), and the SDG provides guidelines on how to apply the SDC to the actual design. The Gen IV SFRs are required to adopt advanced devices and systems as a built-in safety feature, combinations of active safety systems with passive mechanisms or inherent features to prevent and mitigate core damage. Taking the characteristics of the SFR as liquid metal cooling fast reactor system into account, the SDG recommends specific design measures such as inherent / passive reactor shutdown, natural circulation decay heat removal and in-vessel retention of degraded core.

Meet the Presenter:

Mr. Shigenobu Kubo has been engaged in sodium-cooled fast reactor development since 1989. His specialties are SFR system design, safety design and related R&Ds. He is involved in the development of safety design criteria (SDC) for SFR in GIF as Chair of the GIF SDC task force, and he joined this task force since its inception in 2011. He currently occupies the position of Deputy Director, Reactor Systems Design Department, Sector of Fast Reactor and Advanced Reactor Research and Development, at JAEA. He participated in the Feasibility Study on commercialized fast reactor cycle systems (1999-2006) and the Fast Reactor Cycle Technology Development project (2006-2011). He was also involved in the France-Japan ASTRID collaboration as Design task leader and Severe accident task leader. One of his most impressive work is the EAGLE project (SFR severe accident experiments using IGR and out-of-pile experimental facility in Kazakhstan).

He earned his Master degree in nuclear engineering from the Nagoya University, Japan, in 1989.

GIF's Safety Goals& Basis for Safety Approach :

GIF's Safety & Reliability Goals

SR-1: Excel in operational safety and reliability

SR-2: Very low likelihood & degree of reactor core damage

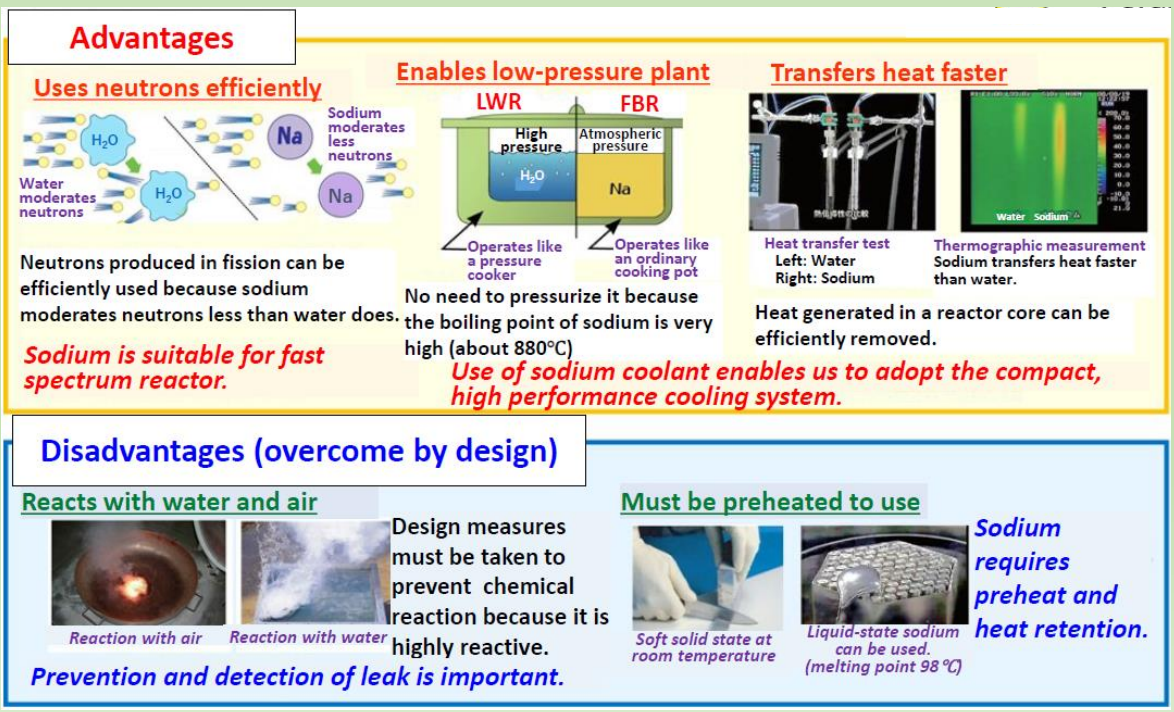
SR-3: Eliminate the need for offsite emergency response

GIF's Basic Safety Approach

- Defence-in-depth
- A combination of deterministic and risk-informed safety approach
- Safety to be built-in to the design, not added-on
- Emphasis on utilization of inherent and passive safety features

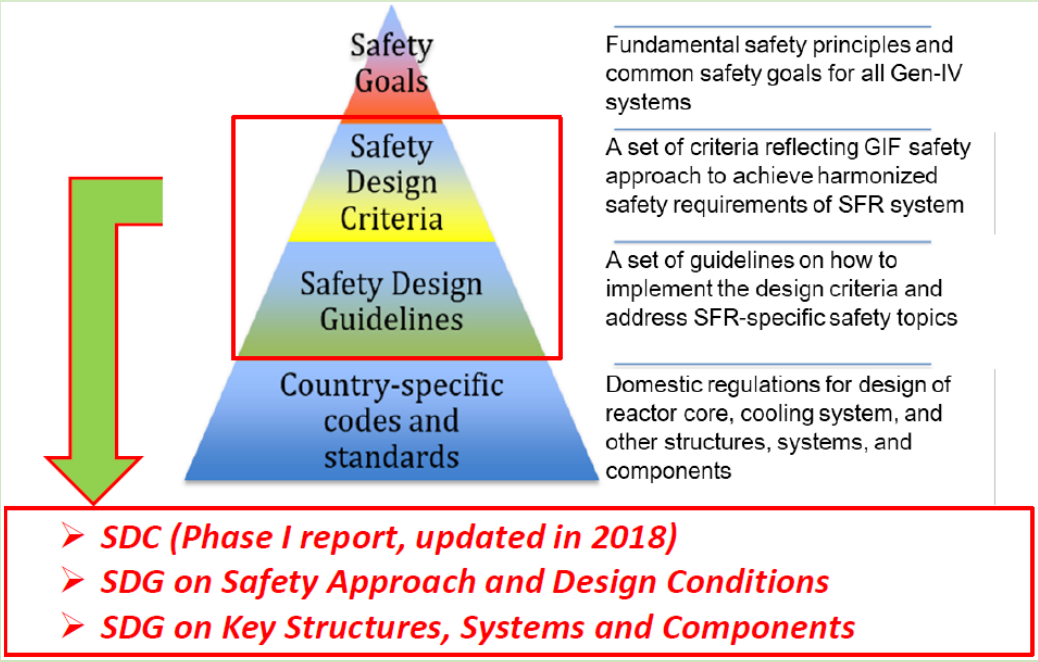
Safety Characteristics of SFR :

Though IAEA has systematically developed international safety standards with a hierarchical structure, the lower-level standards are mainly for existing LWRs. Therefore, we need to develop the global standards for Generation IV Reactors considering each characteristics of their coolant and coolant system.



Development of SDC/SDG for GEN IV SFRs :

Safety Design Criteria Task Force (SDC-TF) have developed SDC and 2 SDGs with hierarchical structure. These documents have been reviewed by external authorities such as national regulatory bodies of the countries, IAEA, and OECD/NEA WGSAR.

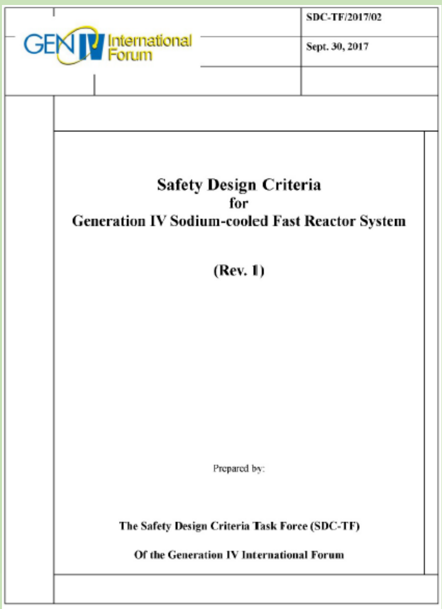


Safety Design Criteria:

The objective of the SDC is to present the reference criteria of the safety design of Structures, Systems and Components (SSCs) of the SFR system.

The criteria are clarified systematically and comprehensively to adopt the GIF's basic safety approach.

Lessons learned from Fukushima Dai-ichi NPPs accident also have been reflected into the SDC.

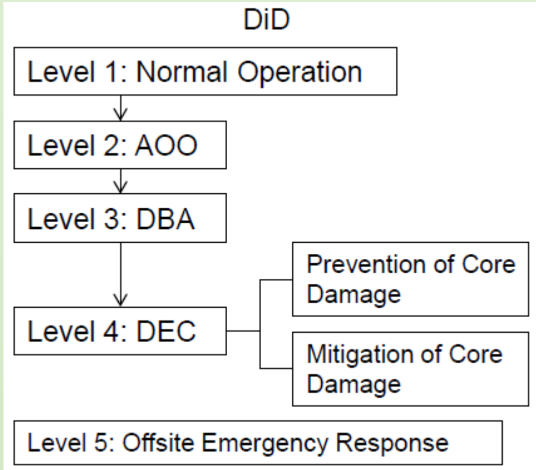


The revised SDC report (Rev.1) is available on GIF web site.
(https://www.gen-4.org/gif/jcms/c_93020/safety-design-criteria)

Safety Design Guideline on Safety Approach :

SDG on SA is intended to provide recommendations and guidance on how to comply with the SDC.

This report focuses mainly on “Design approach to Design Extension Condition (DEC)” and “Practical Elimination of Accident Situations”. These approaches are required to achieve level 4 and 5 on the Defense in Depth.



The SDG on Safety Approach report is available on GIF web site. (https://www.gen-4.org/gif/jcms/c_93020/safety-design-criteria)

Safety Design Guideline on Structures, Systems and Components :

SDG on SSCs is intended to provide detailed guidelines for SFR designers to support the practical application of the SDC in design process to ensure the highest level of safety in SFR design.

This SDG show recommendations and guidance to comply with the SDC and the Safety Approach SDG with examples, which can be applied to Gen-IV SFR systems in general. Below table shows the SFR-specific safety features and 14 focal points in this SDG.

Systems	Safety features	Focal points	SDC	SDG on Safety Approach
Reactor Core systems	Integrity maintenance of core fuels	1. Fuel design to withstand high temperature, high inner pressure, and high radiation conditions	✔	
		2. Core design to keep the core coolability	✔	✔
	Reactivity control	3. Active reactor shutdown	✔	✔
		4. Reactor shutdown using inherent reactivity feedback and passive reactivity reduction	✔	✔
		5. Prevention of significant energy release during a core damage accident, In-Vessel Retention	✔	✔
Coolant systems	Integrity maintenance of components	6. Component design to withstand high temperature and low pressure conditions	✔	
	Primary coolant system	7. Cover gas and its boundary	✔	
		8. Measures to keep the reactor level	✔	✔
	Measures against chemical reactions of sodium	9. Measures against sodium leakage	✔	
		10. Measures against sodium-water reaction	✔	
	Decay heat removal	11. Application of natural circulation of sodium	✔	✔
		12. Reliability maintenance (diversity and redundancy)	✔	✔
Containment systems	Design concept and load factors	13. Formation of containment boundary and loads on it	✔	
	Containment boundary	14. Containment function of secondary coolant system	✔	

2-3. Passive Decay Heat Removal System

Summary / Objectives:

A major design goal for Generation IV nuclear energy systems is to reduce or eliminate the likelihood and/or extent of reactor core damage incurred during an off-normal operating event, thereby eliminating the need for offsite emergency response. One approach for achieving this objective is to develop inherently safe reactor designs that can passively dissipate decay heat to the environment without relying on operator action during an event of this type. Historically, this approach has been taken for both sodium- and gas-cooled Generation IV reactor types by providing Reactor Cavity Cooling Systems (RCCS) that are designed to passively dissipate decay heat to the environment by natural convection while maintain fuel temperature below the threshold for onset of core damage. This presentation will begin by providing a high level overview of RCCS systems that have been developed for advanced reactor designs over the years. This will be followed by a summary of large scale integral effect tests that are currently underway at Argonne to provide licensing-quality data for two of these systems; i.e., air- and water-cooled RCCS concepts.

Meet the Presenter:

Dr. Mitchell Farmer is currently a Senior Nuclear Engineer and Manager for Light Water Reactor programs in the Nuclear Science and Engineering Division at Argonne National Laboratory. He has over thirty years of experience in various R&D areas related to reactor development, design, and safety. A principal early career focus was in the area has been light water reactor (LWR) severe accident analysis and experiments, followed by a rekindling of this work to address technical issues raised in the wake of the reactor accidents at Fukushima Daiichi. More recently, Dr. Farmer has been heavily involved in the analysis, design, and conduct of experiments related to operations and safety of Generation IV reactor concepts including sodium fast reactors, as well as high-temperature gas cooled reactors. He has over 200 publications in the above mentioned technical areas. Dr. Farmer also served as the Technical Area Lead for the Reactor Safety Technologies Pathway (RST) within the Light Water Reactor Sustainability (LWRS) Program at the US Department of Energy (DOE). Dr. Farmer earned his PhD in Nuclear Engineering from the University of Illinois in 1988.

MOTIVATION: The accident at the Fukushima Daiichi Nuclear Power Plant was troublesome because the system that actively cools the decay heat did not work. The study of passive cooling systems is important for advanced nuclear reactor systems.

Passive Safety Needs for GenIV

- GenIV initiative defines 8 technological goals, of which 3 are safety related:
  - "S&R 1 – System operations will excel in safety and reliability"
  - "S&R 2 – Very low likelihood and degree of reactor core damage"
  - "S&R 3 – Eliminate the need for offsite emergency response"
- The reactor cavity cooling system (RCCS) has emerged as a leading concept for meeting these goals
  - Possibility to provide inherently safe and fully passive means of decay heat removal
  - Offers a high level of performance with relative simplicity in design
  - Has been under consideration since 1950's
- Though the RCCS is our focus, our ultimate objective is to support the continued development of safe and reliable nuclear power
  - Multi-institutional effort has brought together federal, industry, national laboratories, and universities

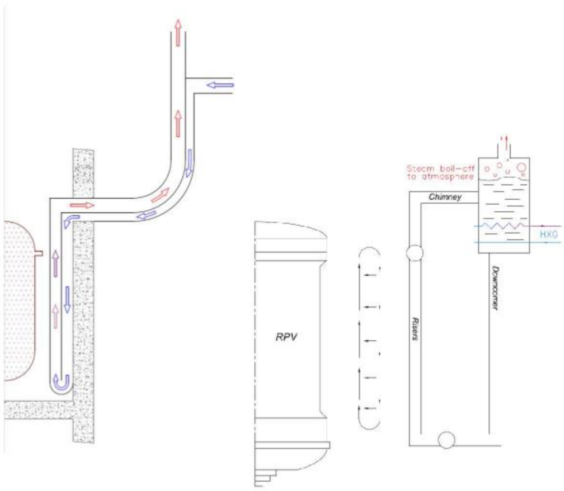
4

FOCUS: The focus is on the reactor cavity cooling system (RCCS) as a system for passive removal of decay heat. It's a simple system that utilizes the natural circulation of air and water but needs to be checked for practical applicability on a variety of scales.

RCCS Overview

- Unique to recent generation of HTGR
  - Natural circulation in laminar and turbulent flow
  - Radiative (primary) and convective heat transfer
- Air and water under consideration
- Considered for both active cooling duration normal operation, and with other designs operating solely as a passive safety system during an accident transient
- Several designs, each unique in geometry, but sharing a common concept, are under design

Reactor	RCCS Coolant	Cooling Mode	Country	Power
HTR-10	Water	Natural	China	10 MW_t
VGM	Water	Natural	Russia	20 MW_t
HTTR	Water	Forced	Japan	30 MW_t
PBMR	Water	Natural	South Africa	400 MW_t
SC-HTGR	Water	Natural	USA	625 MW_t
HTR-PM	Water / Air	Natural	China	250 MW_t
GA-MHTGR	Air	Natural	USA	450 MW_t
GT-MHR	Air	Natural	Russia	600 MW_t



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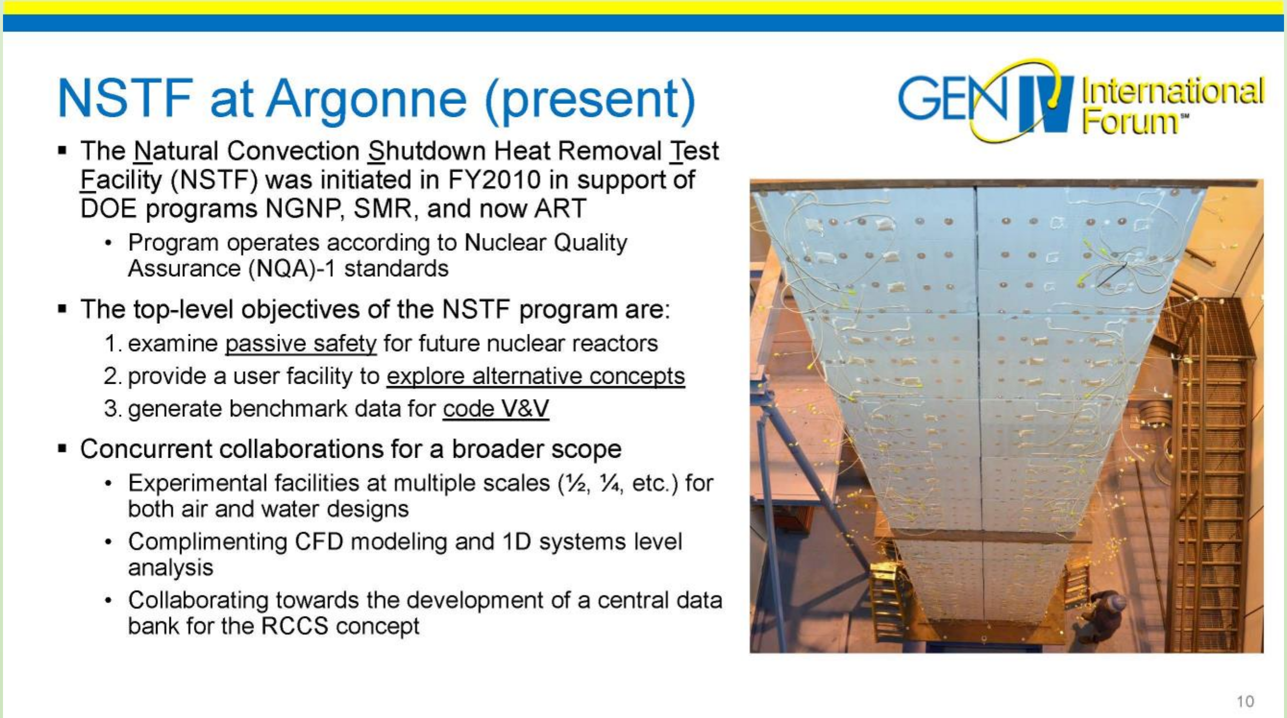
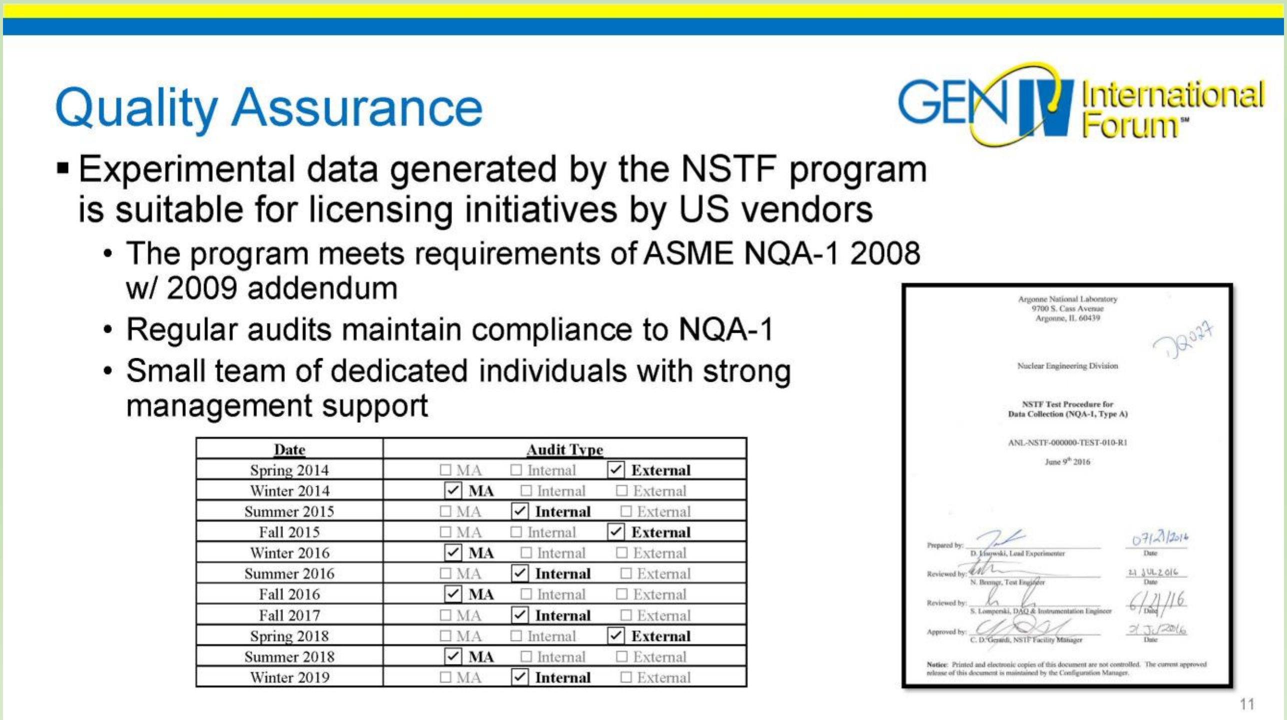
The Natural Convection Shutdown Heat Removal Test Facility: This type of experiment has been performed at ANL since the 1980s, but it has been redesigned to be applicable to advanced reactor nuclear systems.

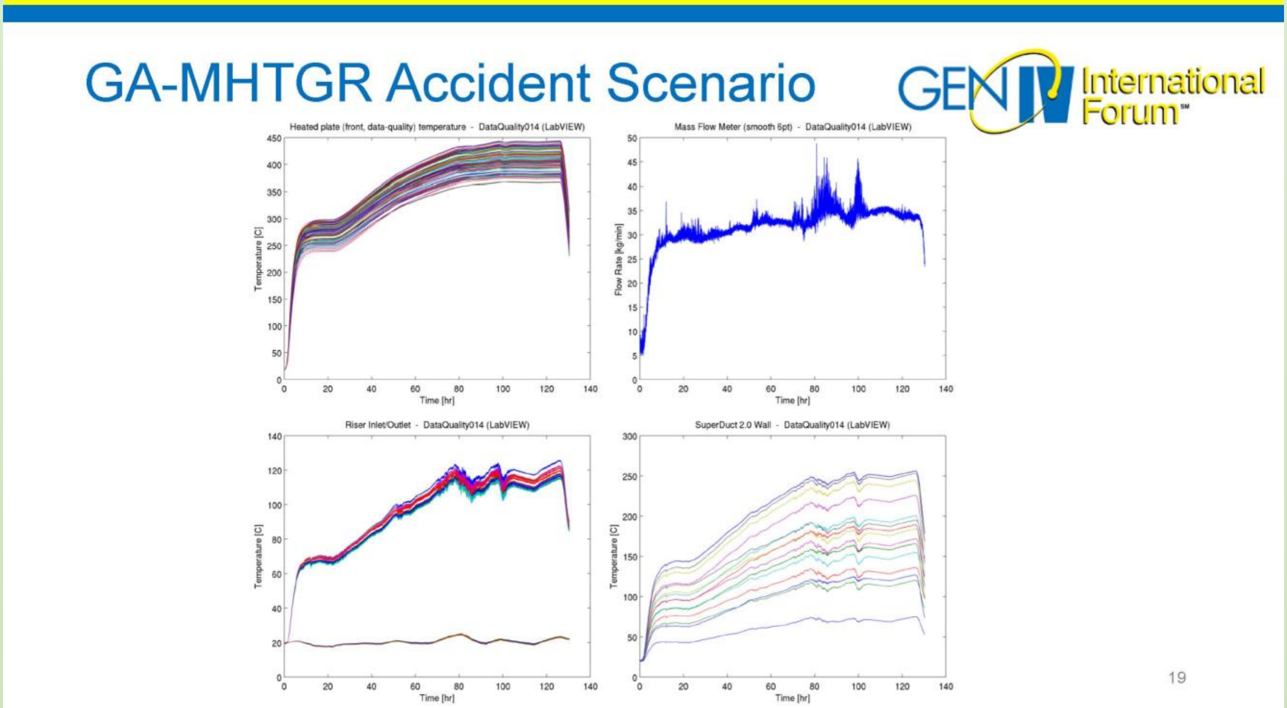


Quality: Experiments contribute to providing high quality data for code validation and to support the licensing process.



Date	Audit Type		
Spring 2014	☐ MA	☐ Internal	☑ External
Winter 2014	☑ MA	☐ Internal	☐ External
Summer 2015	☐ MA	☑ Internal	☐ External
Fall 2015	☐ MA	☐ Internal	☑ External
Winter 2016	☑ MA	☐ Internal	☐ External
Summer 2016	☐ MA	☑ Internal	☐ External
Fall 2016	☑ MA	☐ Internal	☐ External
Fall 2017	☐ MA	☑ Internal	☐ External
Spring 2018	☐ MA	☐ Internal	☑ External
Summer 2018	☑ MA	☐ Internal	☐ External
Winter 2019	☐ MA	☑ Internal	☐ External

Experimental results: An example of the experimental results of the MHTGR accident scenario is shown below.
Other performance tests have been conducted under various conditions with gas as the working fluid, and the results are presented.



Air to Water Conversion: With conclusion of air-based testing, program has shifted to a water-based operation of the existing test facility. Water-cooled NSTF based on concept design for Framatome 625 MWt SC-HTGR (formally AREVA)

Water Accomplishments

- May 2018 – Completed installation of test facility
  - Primary components: test section, water storage tank, and network piping
  - All sensors, hardware, control valves, etc.
- July 2018 – Shakedown and instrument verification
  - Signed verification sheets
- November 2018 – Single-phase demonstration test
  - Install and verify network piping sensors
  - Initial fill of test loop and system leak-test
- January 2019 – First accepted matrix test at single-phase conditions
  - Baseline ‘normal operation’; steady-state with 30°C inlet temperature
- August 2019 – Completion of single-phase parametric series

30

3-1. Closing Nuclear Fuel Cycle

Summary / Objectives:

The steps of PWR nuclear fuel cycle along with alternative fuel cycle options are described. The concepts of two methods for closing the fuel cycle, i.e., recovering the residual uranium and plutonium contained in spent fuel for reuse by wet PUREX and dry Pyroprocessing, are explained. The major issues to be considered for closing the fuel cycle are identified to provide an understanding of sustainability and nonproliferation.

Meet the Presenter:

Prof. Myung Seung Yang has been working at KAERI (Korea Atomic Energy Research Institute) for 30 years in R & D on PWR/CANDU fuel fabrication, quality control of fuel, DUPIC (direct use of spent PWR fuels in CANDU) cycle and the pyroprocessing. He gained the experience in nonproliferation through participating in GIF PRPP and INPRO activities. He served as the President of KAERI from 2007 to 2010 and is a member of the National Academy of Engineering of Korea. He is a Professor at Youngsan University since 2015. He received a decoration “Woong-Bee Order” from the Korean government in 2011, and a WNA (World Nuclear Association, London) Award in 2009 for his contribution to the peaceful use of nuclear energy.

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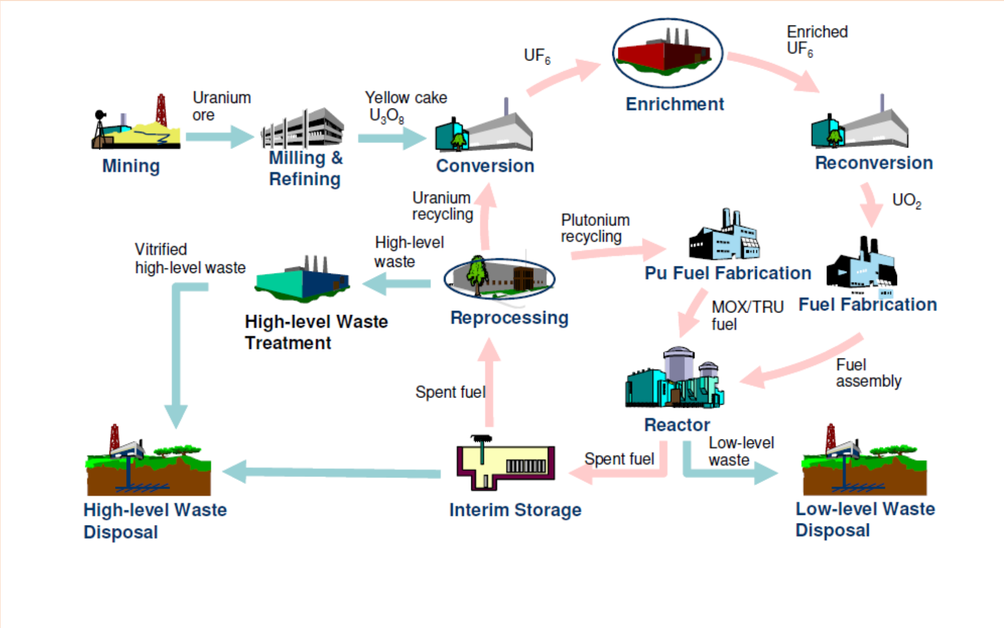
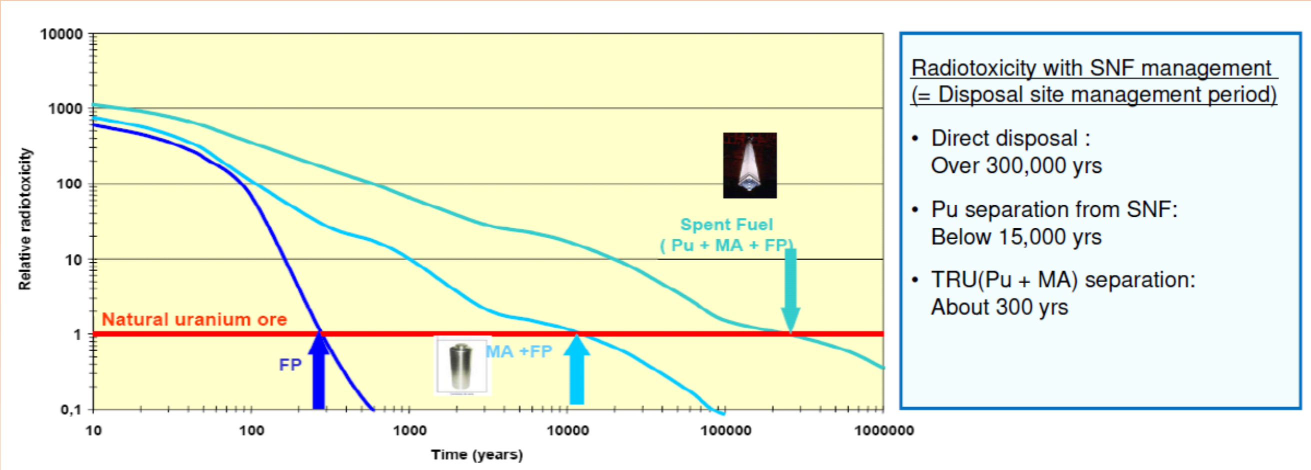
Concept of Nuclear Fuel Cycle

- Reactors are classified according to neutron energy, moderator, coolant, and nuclear fuel.
- Spent fuel (SNF) is recycled or disposed directly (once through) .



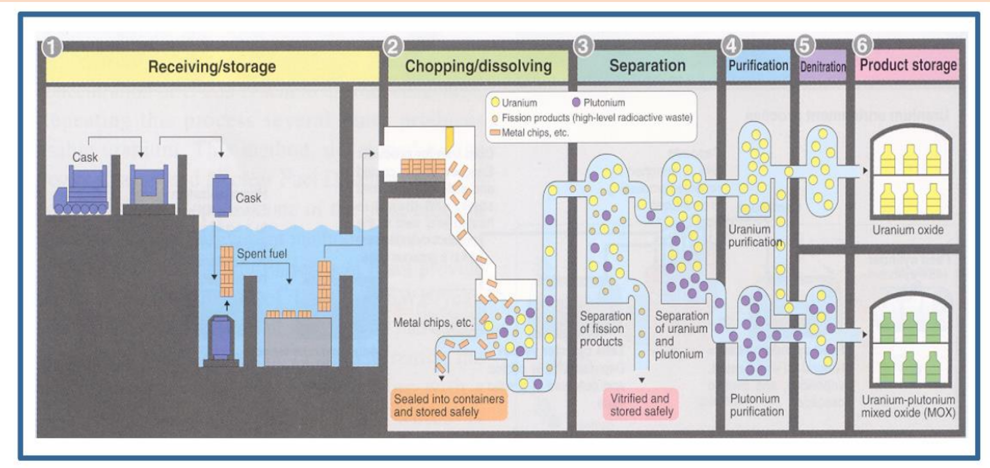
Spent Nuclear Fuel Management

- SNF contains transuranium elements(TRU), fission products(FP) and remaining uranium.
- Most of decay heat after several hundred years are caused by TRU.
- Radiotoxity decreases to natural uranium ore level after 300 years by separation of TRU.
- SNF is stored (wet or dry), packaged, and disposed in underground facility.
- Consideration on corrosion rate of canister etc, are necessary for disposal site.



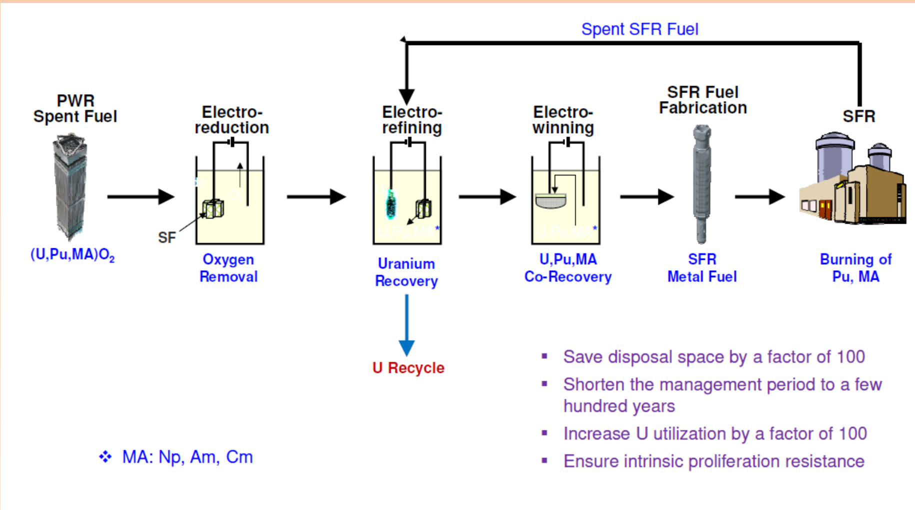
Nuclear Fuel Cycle Technology

- Proliferation resistance, sustainability, waste management, environment effect, and economics are required for innovative nuclear energy system
- PUREX is wet process, and Pyroprocess and DUPIC are dry proceees.
- PUREX process is composed of receiving/storage, chopping/dissolving/, separation, purification, de-nitration, and product storage.
- Advanced wet processes (CoDCon, ALSEP, NEXT, COEX) are under development.



Nuclear Fuel Cycle Technology

- DUPIC and Pyroprocess are appropriate for closed cycle by CANDU, PWR and Gen. IV FR (SFR).
- DUPIC process is composed of disassembling, cutting, de-cladding, oxidation/ reduction, pelletizing/sintering, welding, and assembling.
- There are several dry process technology, such as Pyro-metallurgical, Pyro-chemical, Fluoride volatility.
- Pyroprocess flow sheet is composed of de-cladding, high temperature treatment, electro-reduction, electro-refining, electro-winning, and SFR fuel fabrication.



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Nuclear Fuel Cycle Technology

- Pyro-process has merits, such as small number of components, short cooling time, low criticality hazard, and no pure Pu separation.
- Pyro-process has lower proliferation potential due to limited capability in separation Pu, etc, but has several challenges, such as less safeguard experience.
- Safeguard R&D and economic evaluation of nuclear fuel cycle have been continuing.
- Policy for SNF management on several courtiers are compared.

	Korea	USA	Japan	France	Russia	China	India
Fuel Cycle Policy	Wait & See	Direct disposal/ Wait & see (P&T)	Recycle (P&T)	Recycle (P&T)	Recycle (P&T)	Recycle (P&T)	Recycle (P&T)
Target Yr for INS	2020's	2040s	2040s	2020 ~ 2040	2020s	2020s	2020s
Recycle Method	Pyro	Wet (Advanced Aqueous) Pyro	Wet (NEXT) Pyro	Wet (COEX /GANEX)	Wet (Advanced Aqueous) Pyro	Wet (PUREX) Pyro	Wet (PUREX) Pyro
Reactor (Fuel)	SFR (Metal)	SFR (Metal, Oxide)	SFR (Oxide)	SFR (Oxide) GFR (Carbide, Nitride)	SFR (Oxide, Nitride)	SFR (Mixed oxide)	SFR (Mixed carbide, Oxide, Metal)

Summary

- Benefits of closing nuclear fuel cycle are sustainability, management of high level waste, environmental friendly, management of repository for permanent disposal, and enhanced proliferation resistance.
- Advanced wet & dry fuel cycle processes along with safeguards technology are under development.
- National policy of spent fuel management is to be decided.

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3-2. Sustainability a Powerful and Relevant Approach for Defining Future Nuclear Fuel Cycles

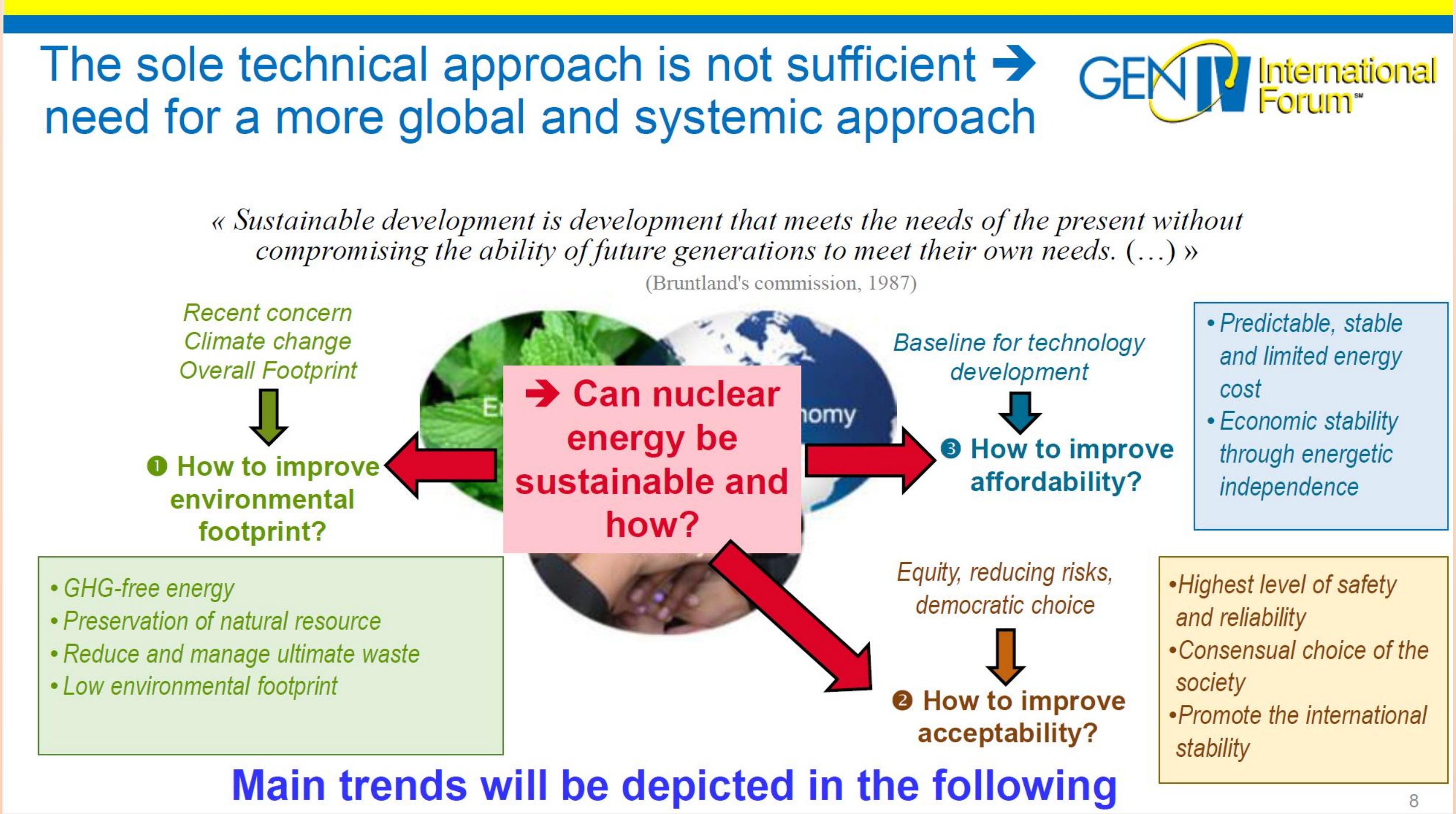
Summary / Objectives:

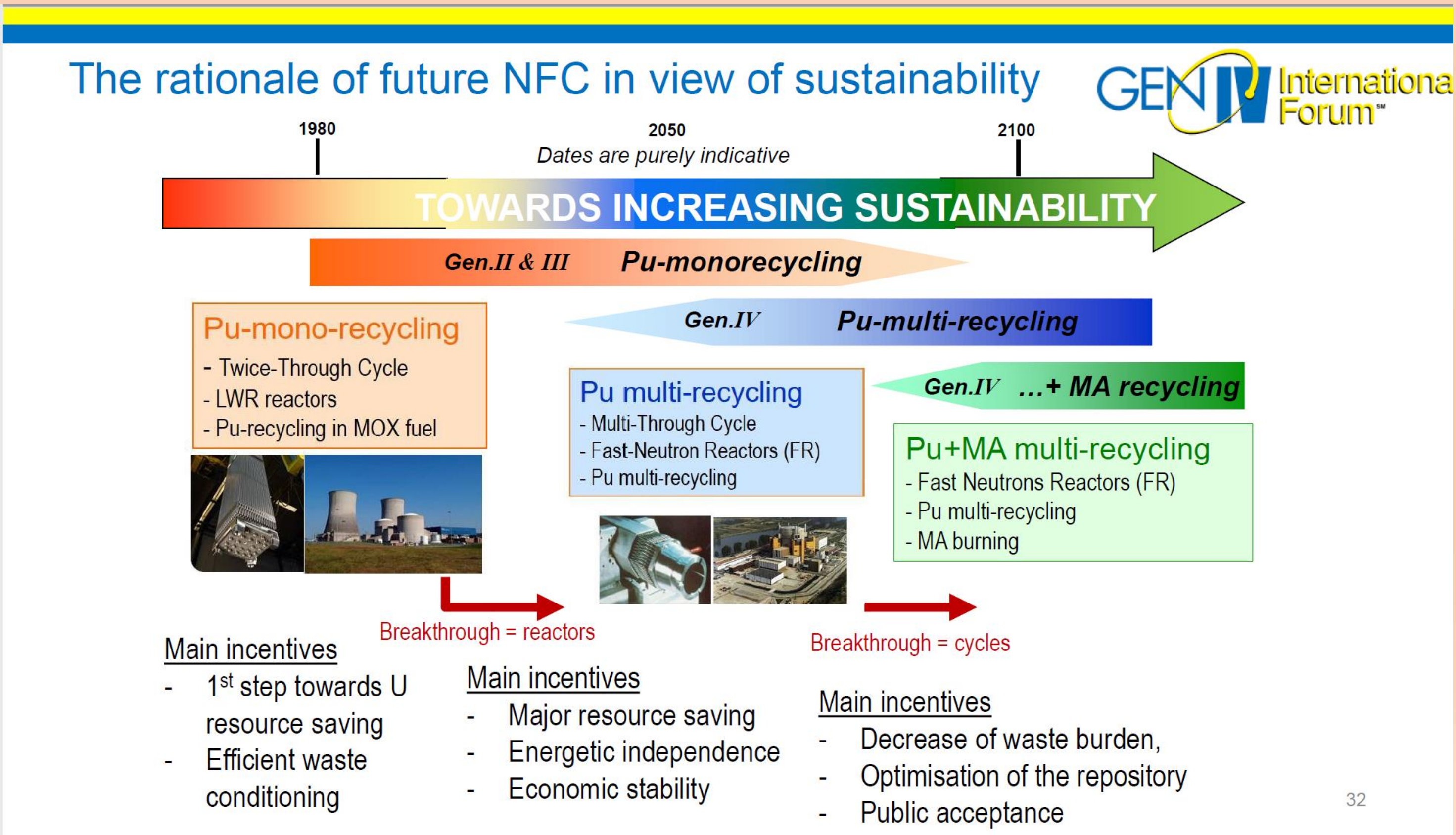
Technically, nuclear energy is anticipated to be one of the most efficient energy source to mitigate the global climate change together with the renewables, due to its low green-house-gases emissions, its reliability and its high base-load capacity. However, public opinion survey and phase-out decision regularly reminds us that political decisions are not only driven by technical criteria. **Beyond the well-known technical and economic optimization, many other criteria are of growing importance such as environmental and social concerns.** This rather recent situation requires changing our rationale technical approach to the wider sustainability approach, which also includes the overall environmental footprint and the more general social acceptability and social impact. This presentation will illustrate how sustainability can help us to identify the most promising trends for future nuclear fuel cycles in order to ensure a long-term future of nuclear energy.

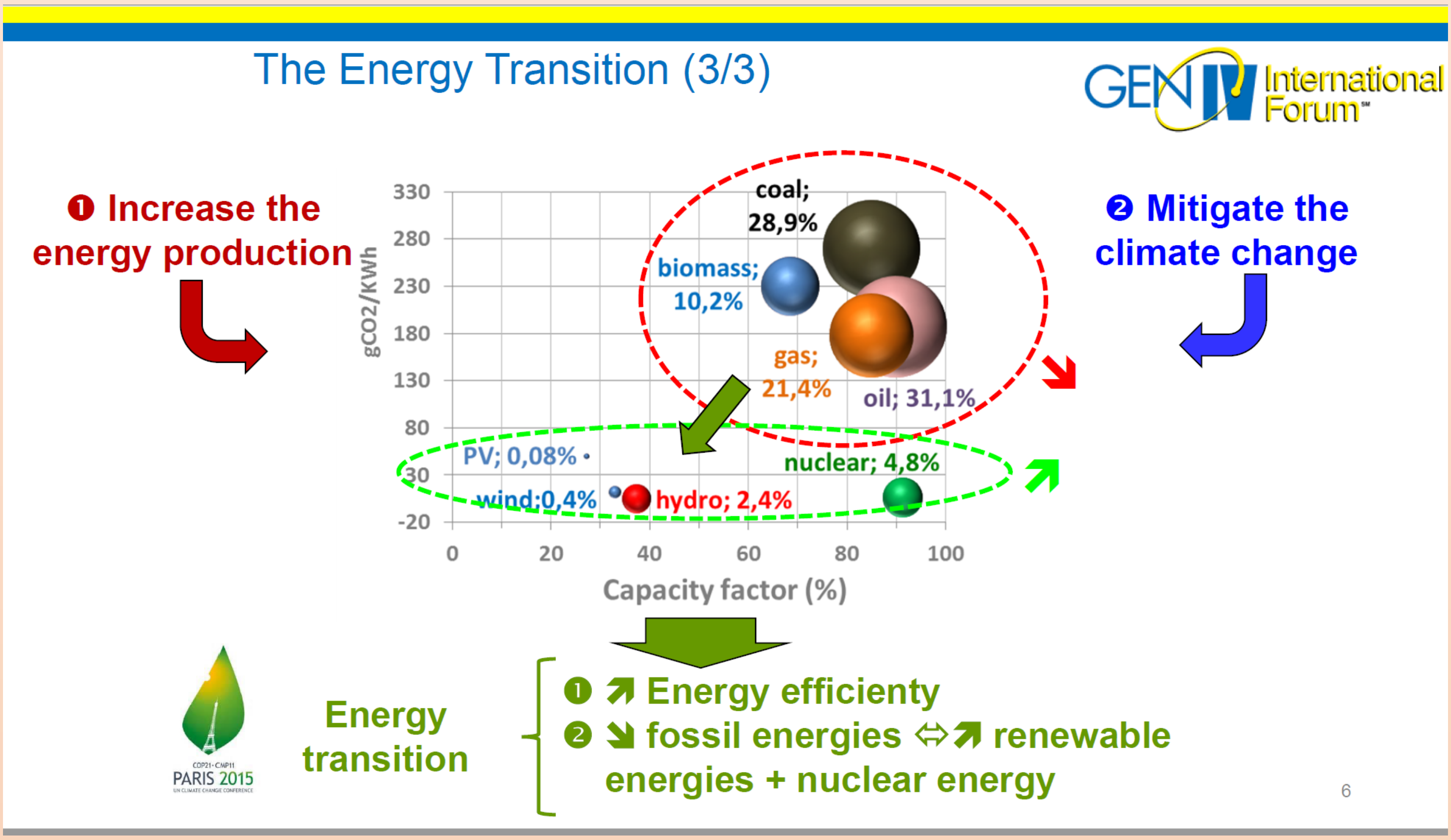
Meet the Presenter:

Christophe POINSSOT has been working at CEA (The French Alternative Energies and Atomic Energy Commission) for more than 25 years in fuel cycle R&D. He is currently heading the Research Department on Mining and Fuel Recycling Processes (DMRC), and is in charge of developing actinides recycling processes and operating the Atalante hot-lab. He is also a CEA international expert in actinides chemistry and professor in nuclear chemistry at INSTN.

He explain the energy transition to the sustainability with environmental drivers, societal drivers, and economic drivers, and show the rationale of future fuel cycles.



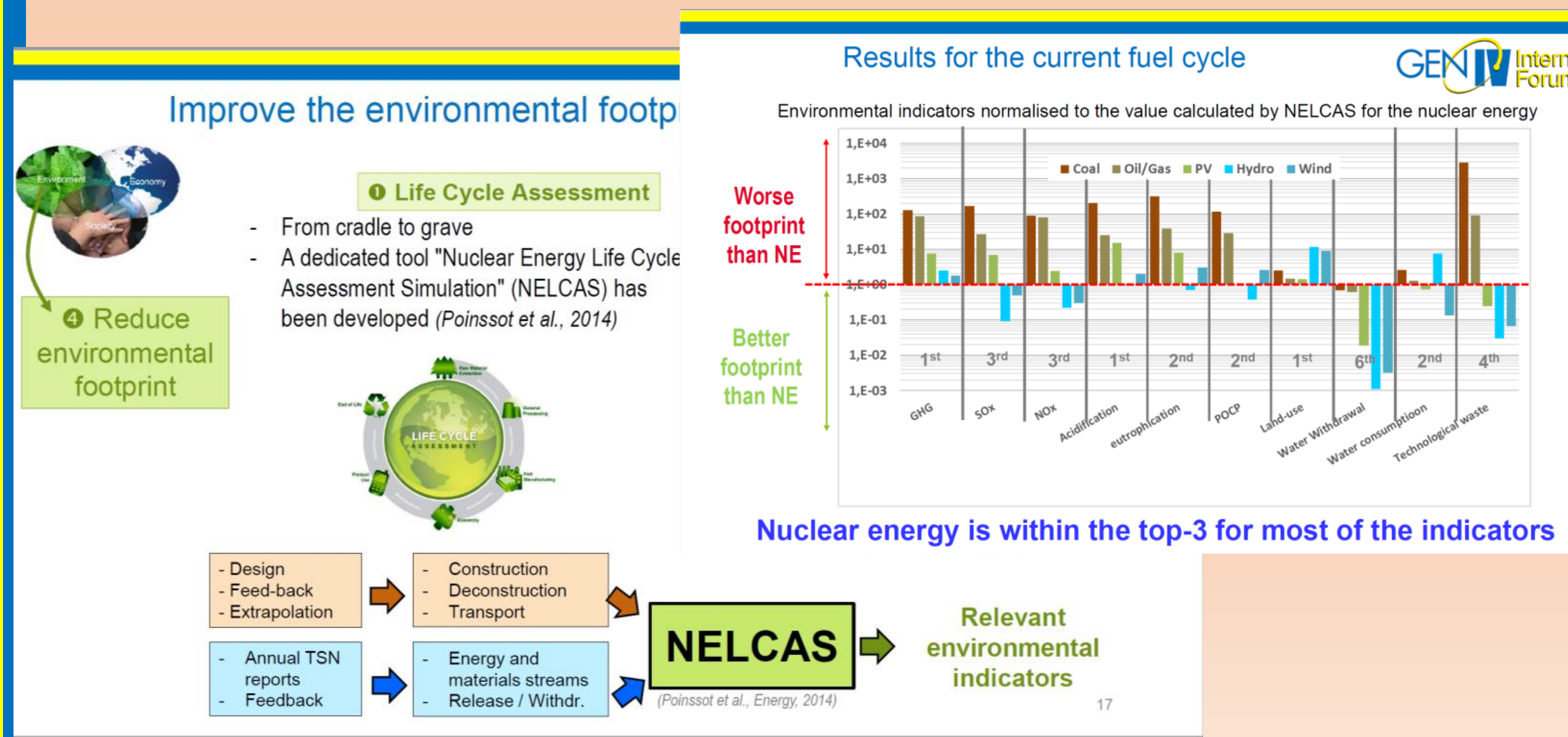




Environmental drivers

(1=Reduce GHG missions, 2=Preserve natural resource)

Life cycle assessment of environmental footprint can be performed by simulation tool. Environmental indicators for each energy source on such as GHG emissions, SOx, NOx can be shown by this simulation tool.



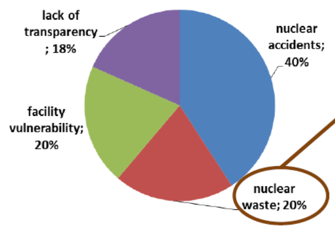
As societal drivers,

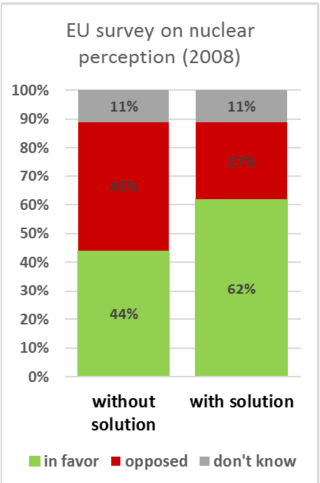
1= Improve safety, 2=Improve waste management.

As economic drivers, 1= Stable & predictable cost,

2= Ensure affordable costs, 3=Towards simpler processes

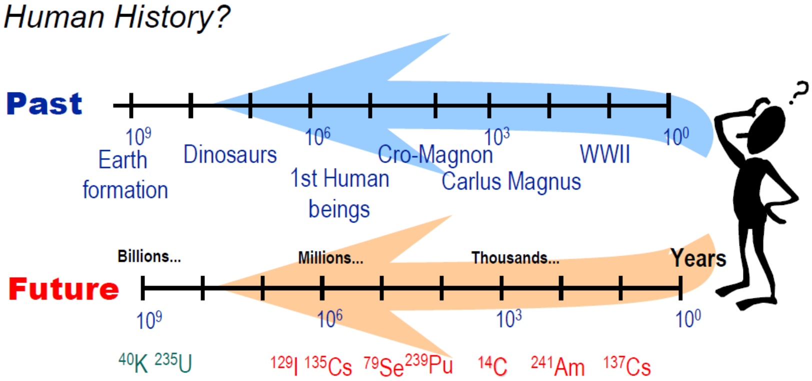
Improve waste management





❷ Improve waste management

- Waste is severely questioned by public opinion
  - Nuclear waste seen as Achille's heel of nuclear energy, mainly due to very long lifetime
  - Main concern = waste lifetime. Any reduction could help to improve acceptability. *Could we reduce waste lifetime back within Human History?*



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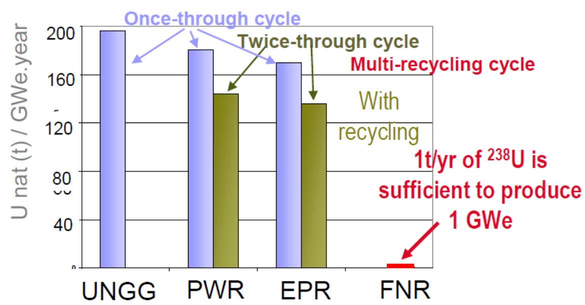
Chap.III: economic drivers

Economic optimization is already at the root of R&D for industry

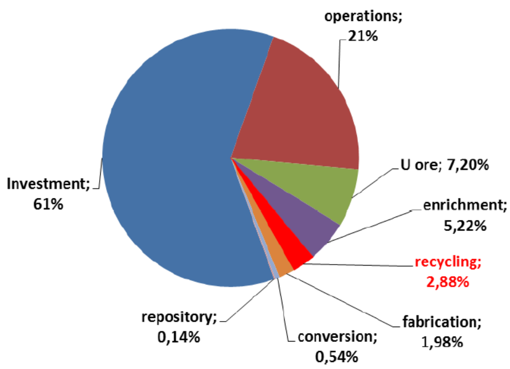
❶ Stable & predictable cost

- Recycling decreases the dependence to U market (price, availability, volatility …)
  - Possibility of using U_{rep} and U_{dep} available stockpile with FNR
  - Significant extension of U reserve



❷ Ensure affordable costs

- Back-end of the fuel cycle has a limited influence on the KWh cost



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3-3. Scientific and Technical Problems of Closed Nuclear Fuel Cycle in Two-Component Nuclear Energetics

Summary / Objectives:

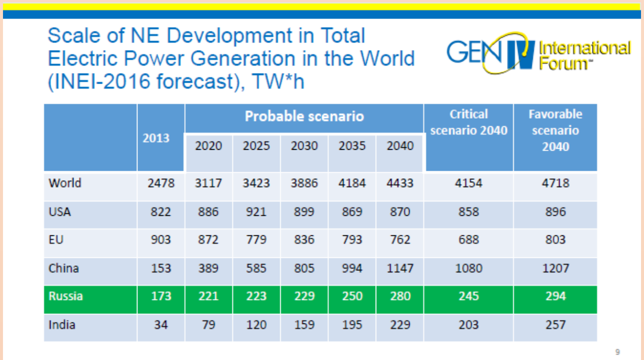
The webinar presents the overview of scientific and technical problems of closed nuclear fuel cycle in two-component nuclear energetics. The presentation will highlight the existing problems of the current technological platform of NE (thermal reactors in an open nuclear fuel cycle) and the advantages of the new technological platform (fast reactors with closed nuclear fuel cycle). Latest developments associated with the use of mixed UN fuel & spent nuclear fuel reprocessing are briefly presented as well. The remaining research challenges of the new technological platform being developed within the “Proryv” Project framework are summarized in the light of the present technology understanding.

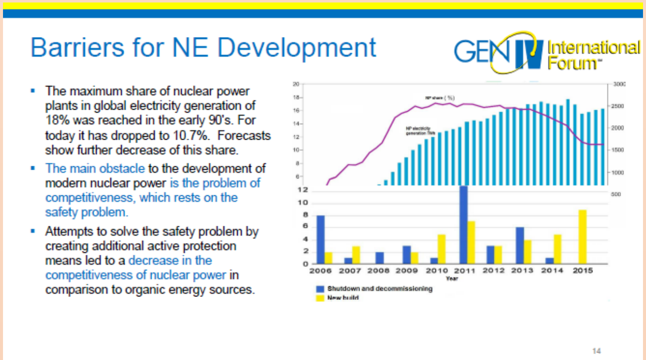
Meet the Presenter:

Mr. Alexander Orlov, Ph.D. is the advisor to the Scientific Director of R&D of the “Proryv” Project. Since 2012, he has been a member of the fast reactors with lead and sodium coolants, a new type of reactor fuel (mixed U-Pu nitride), and technologies to reprocess spent nuclear fuel in order to return it into the fuel cycle. These technologies combined are known as the “Proryv” Project.

Pessimistic forecast of future NE deployment and its obstacles:

In accordance with the analysis of world deployment scenario of nuclear power, all scenario showed pessimistic growth of nuclear deployment except China. The obstacle of nuclear deployment is lack of competitiveness by additional safety measures. The current and/or old open nuclear fuel cycle would be sufficient to mid-term fuel supply, but have limitation for use in longer-term due to low utilization efficiency of uranium, lack of environmentally acceptance, and proliferation risk.

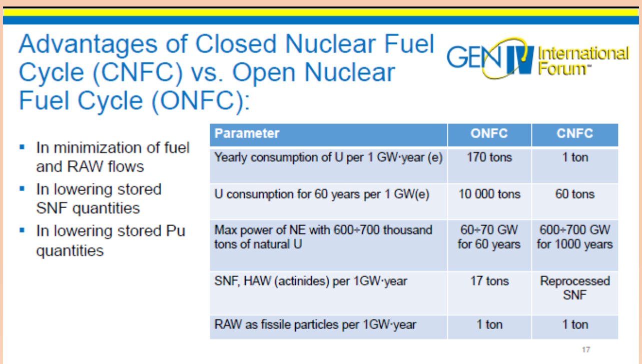


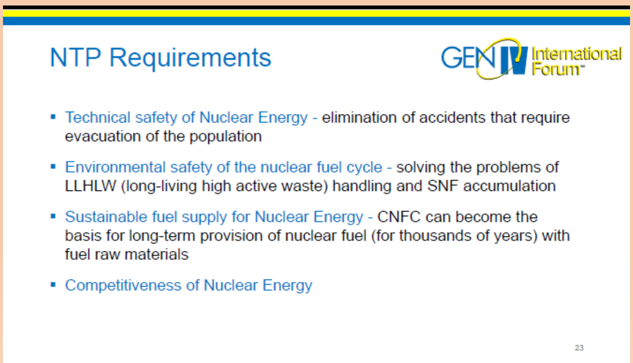


New Technology Platform (NTP) with Fast Reactor:

The closed fuel cycle with Fast Reactor have advantage in minimization of radioactive waste, lowering spent nuclear fuel (SNF) and stored plutonium. The government of Russia constructed the development strategy of NTP, Strategy-2000, and proceeded it based on the milestones by 2020.

Resolve four major challenges are required to NPT, 1) technology safety, 2) environment safety, 3) sustainable fuel supply, and 4) competitiveness.

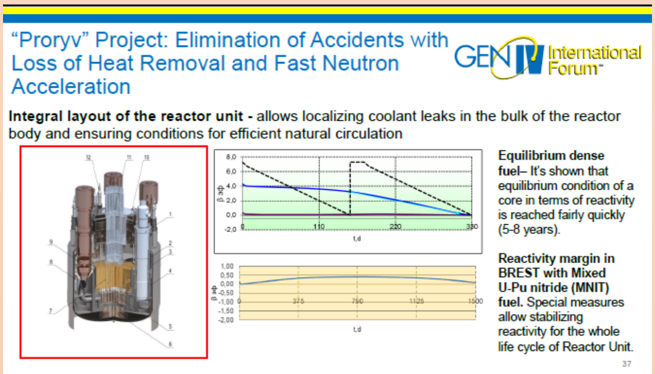


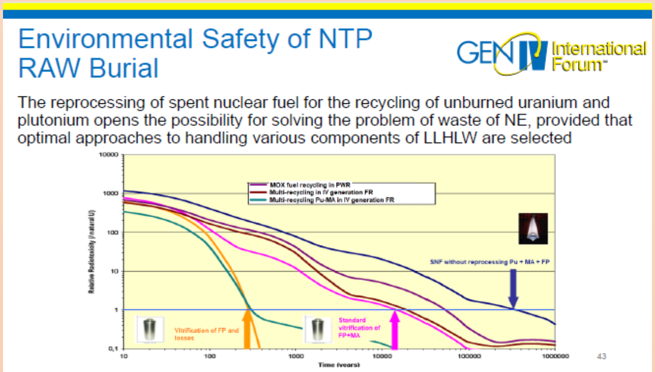


1) Technology safety and 2) Environmental safety:

The goal to achieve technological safety is elimination of accident that requires evacuation of the population at nuclear power plant and other nuclear facilities. The dense fuel in reactor core with zero reactivity margin for burnup, lead coolant, air heat exchanger for natural circulation are possible measures to eliminate reactivity accidents and accident with loss of heat removal.

For environmental safety, the goals are publicly acceptable treatment of LLHLW and avoidance of SNF accumulation. Processing SNF, MA transmutation and disposal of radioactive waste are identified as measures to prohibit RW disposal containing ecologically significant amount, reduce the amount of SNF, and isolate RW.

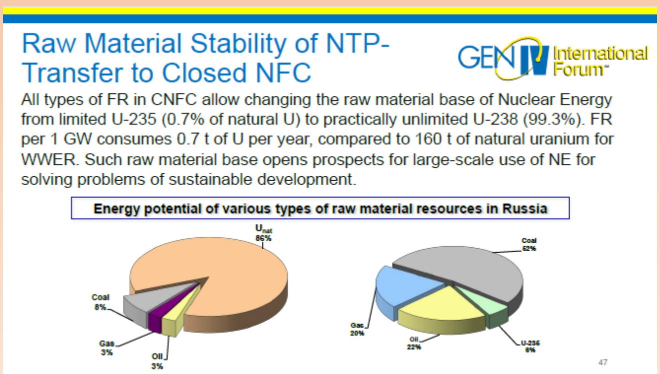


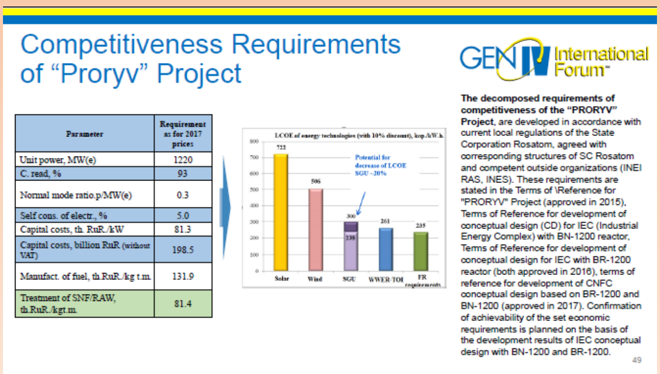


3) sustainable fuel supply, and 4) competitiveness

Having long-term provision of nuclear fuel with raw materials is the goals for sustainable fuel supply. The full reproduction of fissile nuclides in the core and transition to a closed NFC, using FR with B.R.~1, SNF reprocessing and fuel fabrication with recycled materials, are possible ways to reach the goals.

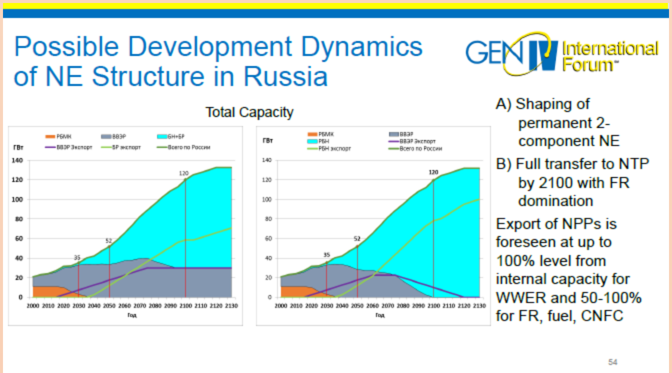
Competitiveness could be achieved by elimination and simplification of number of NPP safety systems and design of the reactor, and reduction of the fuel component, and transportation costs using on-site fuel cycle systems.

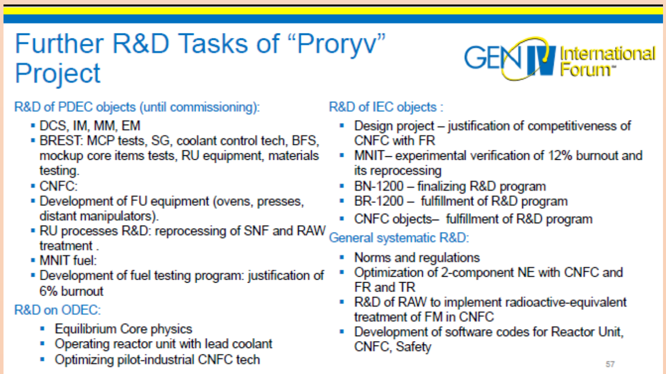




Proryv Project :

The Proryv Project have been implemented by the State Atomic Energy Corporation ROSATOM which is aimed at achieving these challenges. The seven solutions for technical safety have been studied and developed the lead coolant reactor with nitride fuel, BREST-OD-300. The multiple software evaluation and test-reactor irradiation of nitride fuel has been carried out for the development. The pyro-chemical reprocessing, no blanket design and transmutation of MA also studied for the solution of environmental safety. Preliminary results of scenario study in Russia assumed pilot energy complex, BREST-OD-300 with dense nuclear fuel and reprocessing, BN-1200 and design project of industrial energy, shows full transfer to closed fuel cycle with FR will be achieved 120 GW by the end of this Century.

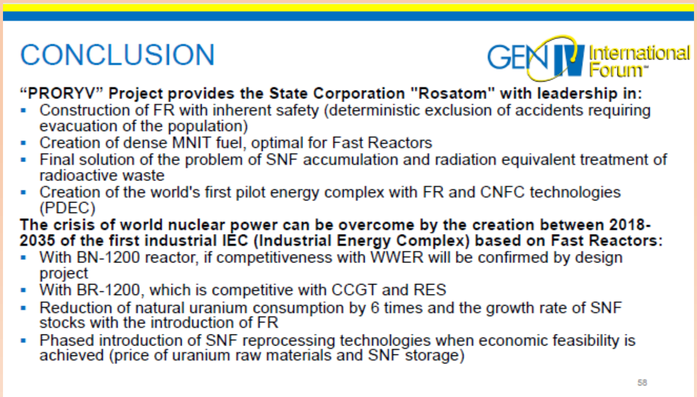




Conclusion:

“PRORYV” Project provides leadership in the studies for major challenges required to NPT.

The crisis of world nuclear power can be overcome by the creation between 2018-2035 of the first industrial Energy Complex based on Fast Reactors.



3-4. Molten Salt Actinide Recycler and Transforming System with and without Th-U support: MOSART

Summary / Objectives:

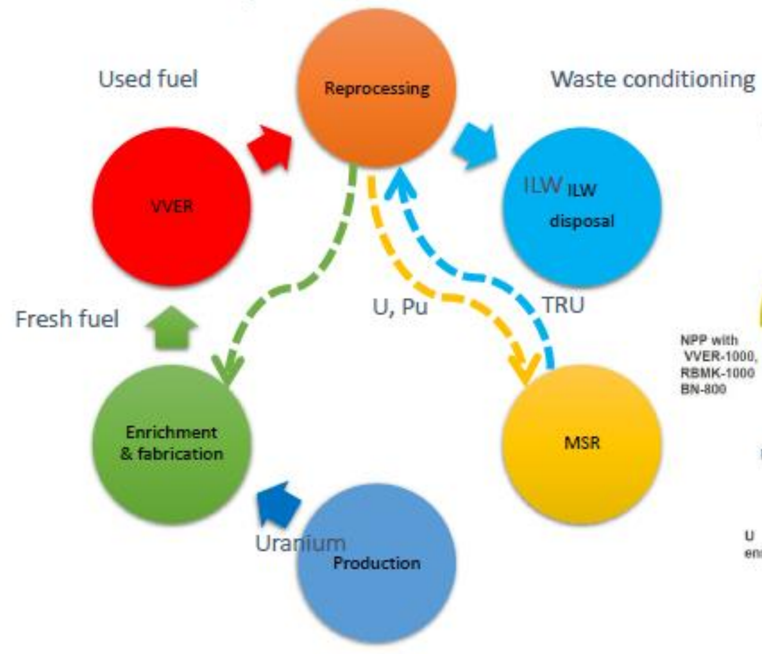
The Molten Salt Reactor designs, where fissile material is dissolved in the molten salt fluorides, under consideration in the frame work of the GIF are briefly described. The presentation mainly focuses on the MOlten Salt Actinide Recycler & Transforming (MOSART) system without and with U-Th support fueled with different compositions of transuranic elements trifluorides from spent LWR fuel. New design options with homogeneous core and fuel salt with high enough solubility for transuranic elements trifluorides are being examined at NRC "Kurchatov Institute" because of new goals. The webinar has the main objective of presenting the fuel cycle flexibility of the MOSART system while accounting technical constrains and experimental data received in this study. A description is given of the experimental results on key physical and chemical properties of fuel salt and combined materials compatibility to satisfy MOSART system requirements. In the webinar the main design choices and characteristics of MOSART concept are explained and discussed including safety, transient simulations, laboratory scale experiments and program plan for the development of the small power Demo MOSART unit.

Meet the Presenter:

Dr. Victor Ignatiev works at the NRC-"Kurchatov Institute," Moscow, Russia, both as the Head of the Molten Salt Reactor Laboratory (since 2012) and as a Professor (since 2009). He graduated from the Nuclear Power Systems Moscow Physical Engineering Institute, USSR, in 1976, and earned his Ph.D. in 1986 from the Kurchatov Institute of Atomic Energy, Moscow, USSR. His Ph.D. research focused on molten salt reactors. Since 2014, he has been the co-chair of Generation IV International Forum MSR pSSC. In 1985, he received the Kurchatov Award on the Fundamental Studies of Molten Salt Reactors; and in 2016, he received the Kurchatov Award on Engineering studies of Molten Salt Reactors. His research activities mainly focus on Molten Salt Reactor: (1) Th - U fuel cycle and TRU burners, (2) Combined materials compatibility & salt chemistry control in selected molten salt environments at parameters simulating designs operation, (3) Physical & chemical properties for fuel and coolant salt compositions, and (4) Flow sheet optimization, including reactor physics, thermal hydraulics and safety related issues.

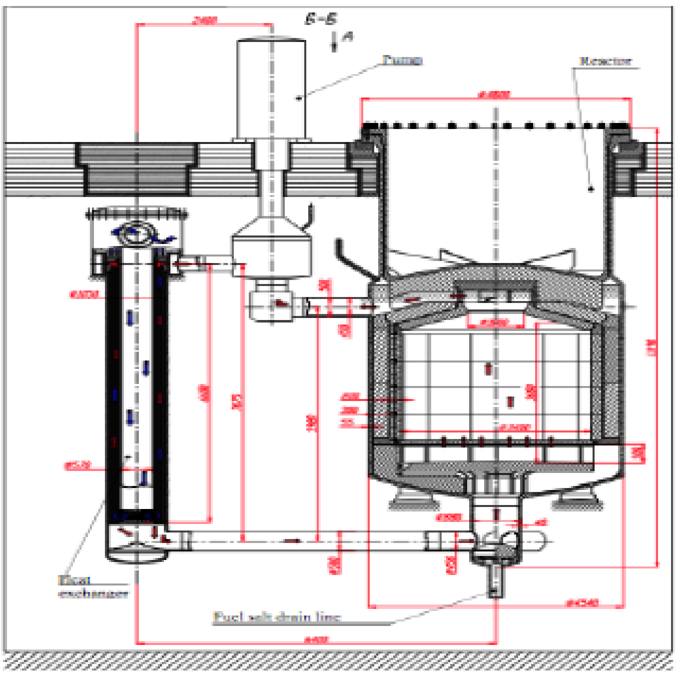
1. Introduction of MSR and MOSART:

In MSR (Molten Salt Reactor) device, solid fuel elements are replaced by liquids. Started with TRU Fluorides from LWR Spent Fuel, MOSART (Molten Salt Actinide Recycler & Transformer) can operate in different modes: Transmuter, Self-sustainable, Breeder.



2. MOSART – Transforming Reactor System

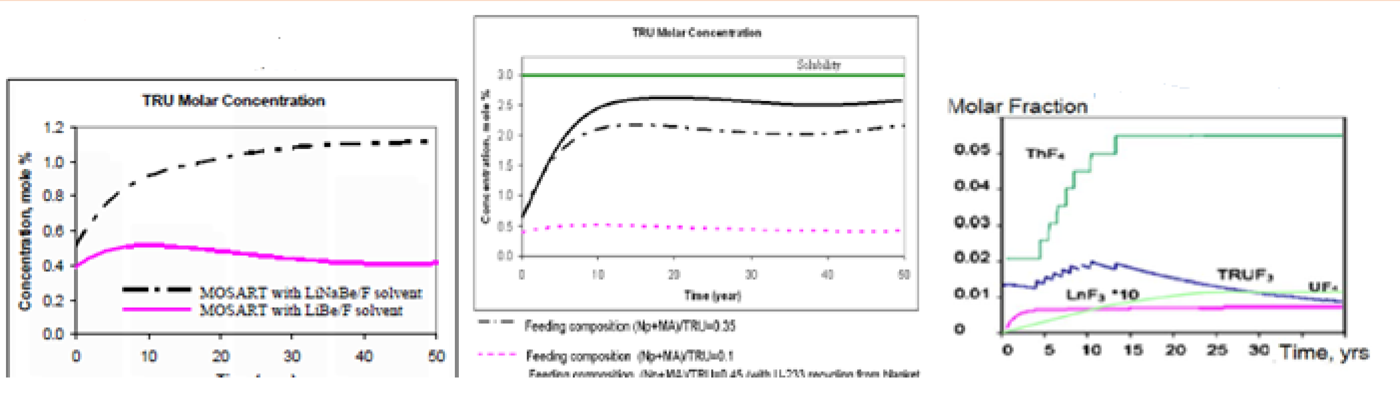
MOSART design has options with homogeneous core and fuel salt with high enough solubility for transuranic elements trifluorides.



System	burner	/ breeder
Fluid streams	1	2
Power capacity, MWt	2400	2400
Fuel salt inlet/outlet temperature, °C	600 /720	600 /720
Fuel salt composition, mole %	72LiF $27BeF_2$ $1TRUF_3$	75LiF $16.5BeF_2$ $6ThF_4$ $2.5TRUF_3$
Blanket salt composition, mole %	no	75LiF $5BeF_2$ $20ThF_4$

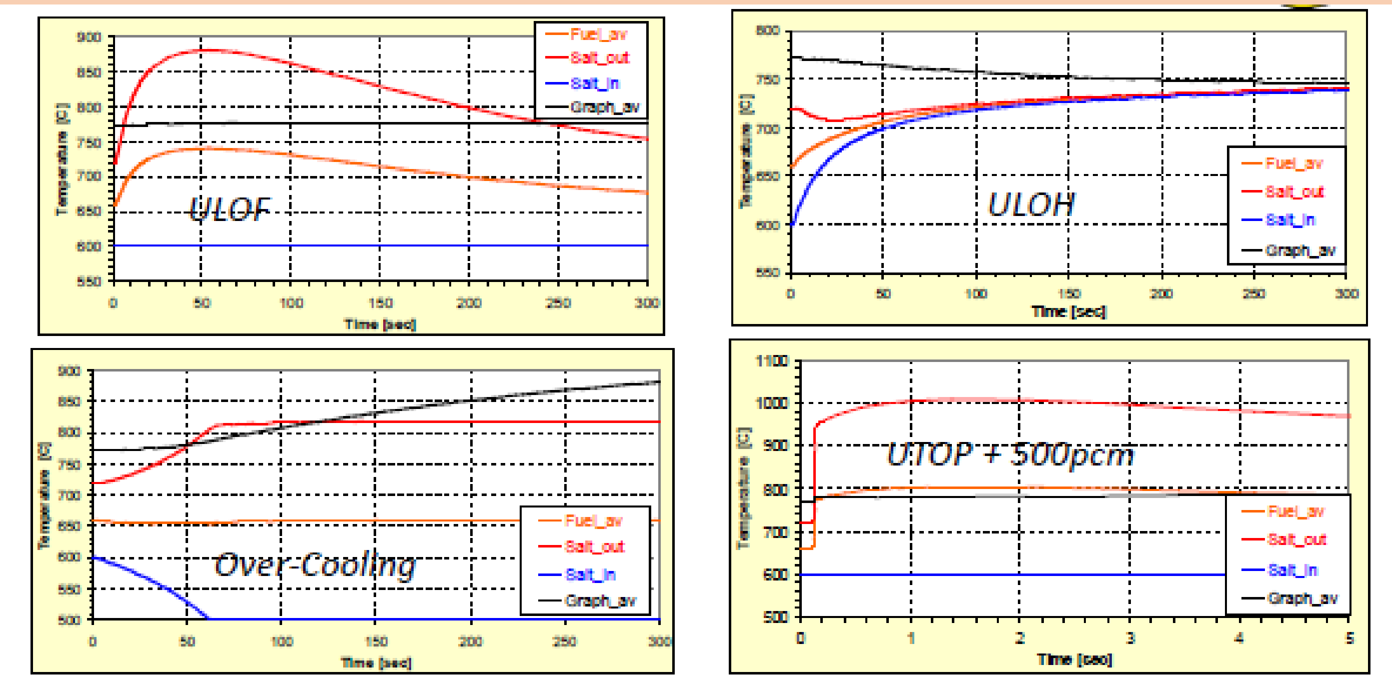
3. MOSART Fuel Cycles

- *MOSART core containing as initial loading 2 mole% of ThF_4 and 1.2 mole % of $TRUF_3$, with the rare earth removal cycle 300 epdf after 12 years can operate without any $TRUF_3$ make up basing only on Th support as a self-sustainable system.*
- *At equilibrium molar fraction of fertile material in the fuel salt is near 6 mole %.*



4. MOSART Transients Analysis :

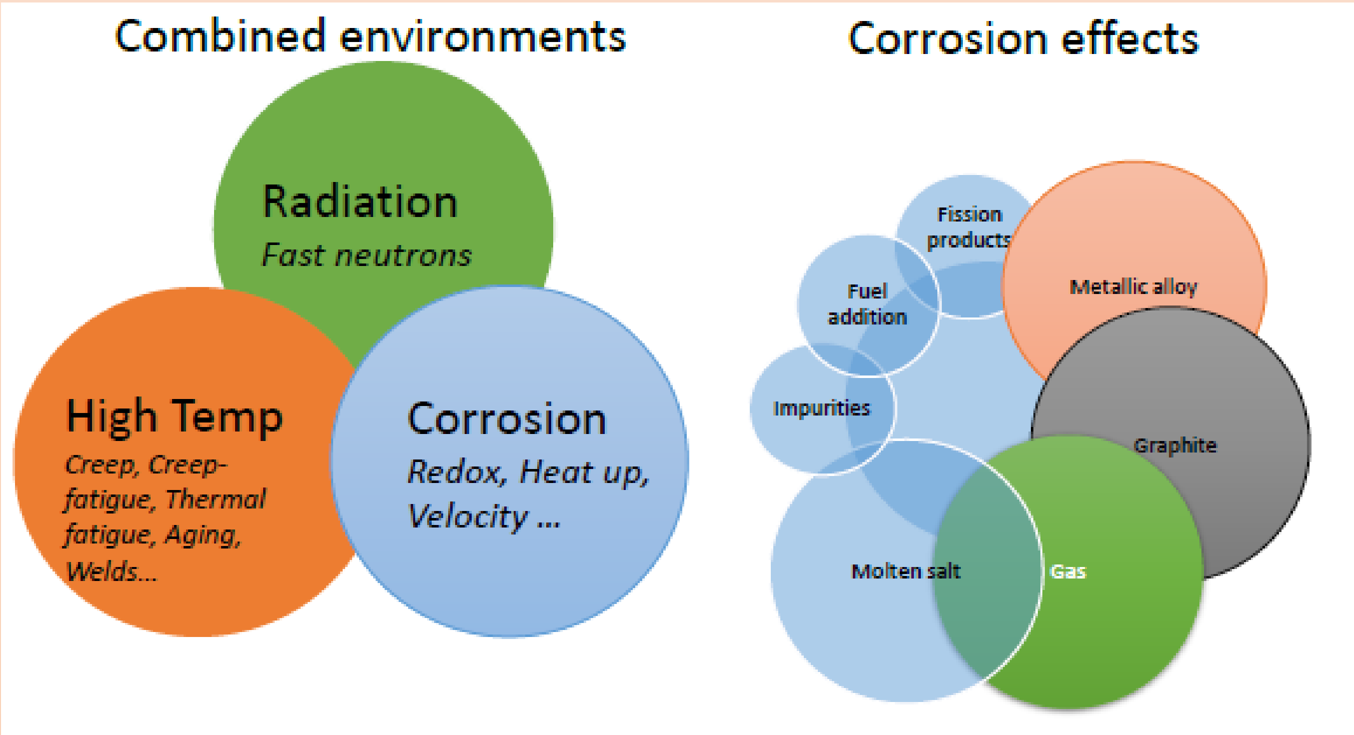
The MOSART is expected not to be seriously challenged by the major, unprotected transients such as ULOF, ULOH, overcooling, or even UTOP.



5. MSR container materials:

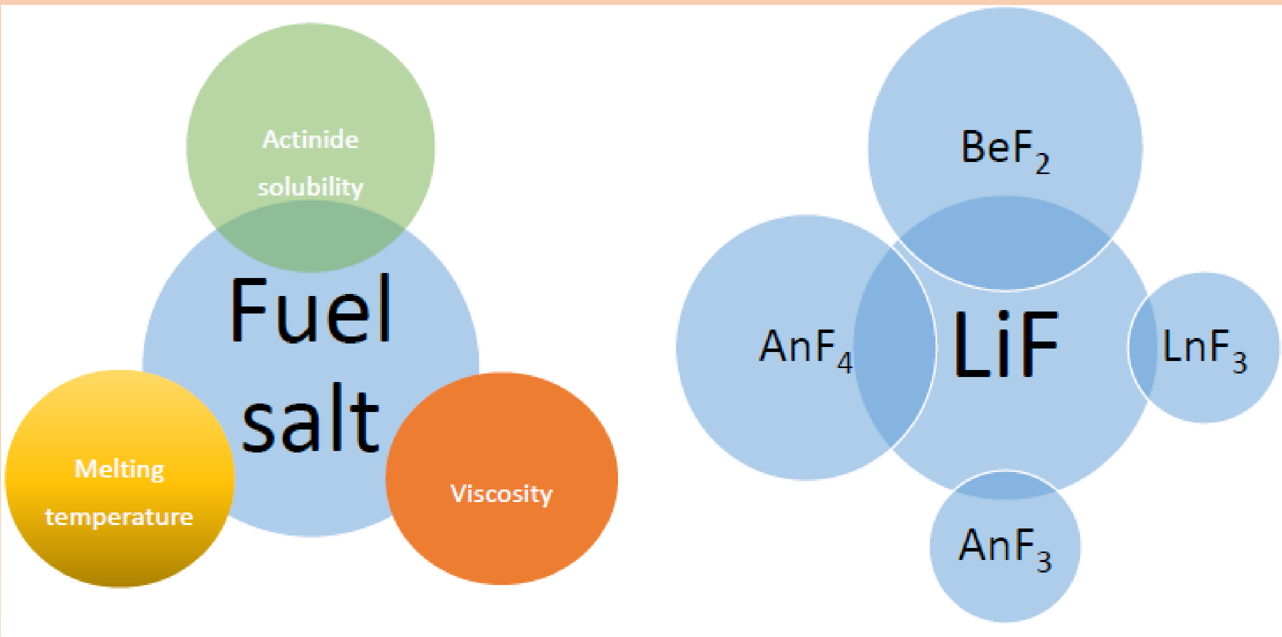
Experiments Results In polythermal loops with redox potential measurement demonstrated that operations with Li, Be/F salt, fueled by UF_4 or PuF_3, are feasible using carefully purified molten salts and loop internals.

Alloys modified by Ti, Al and V have shown the best post irradiation properties



6. Selection of Fuel / Coolant options :

In most cases the base-line fuel / coolant salt is lithium-beryllium fluoride salt as it has best properties.



4-1-1. Sodium Cooled Fast Reactors (SFR)

Summary / Objectives:

This webinar will give an overview of distinctive fast reactor characteristics and identify key performance benefits. A brief history of development and international experience with SFRs will be reviewed. Finally, the Generation-IV international collaboration on SFR technology research and development will be described.

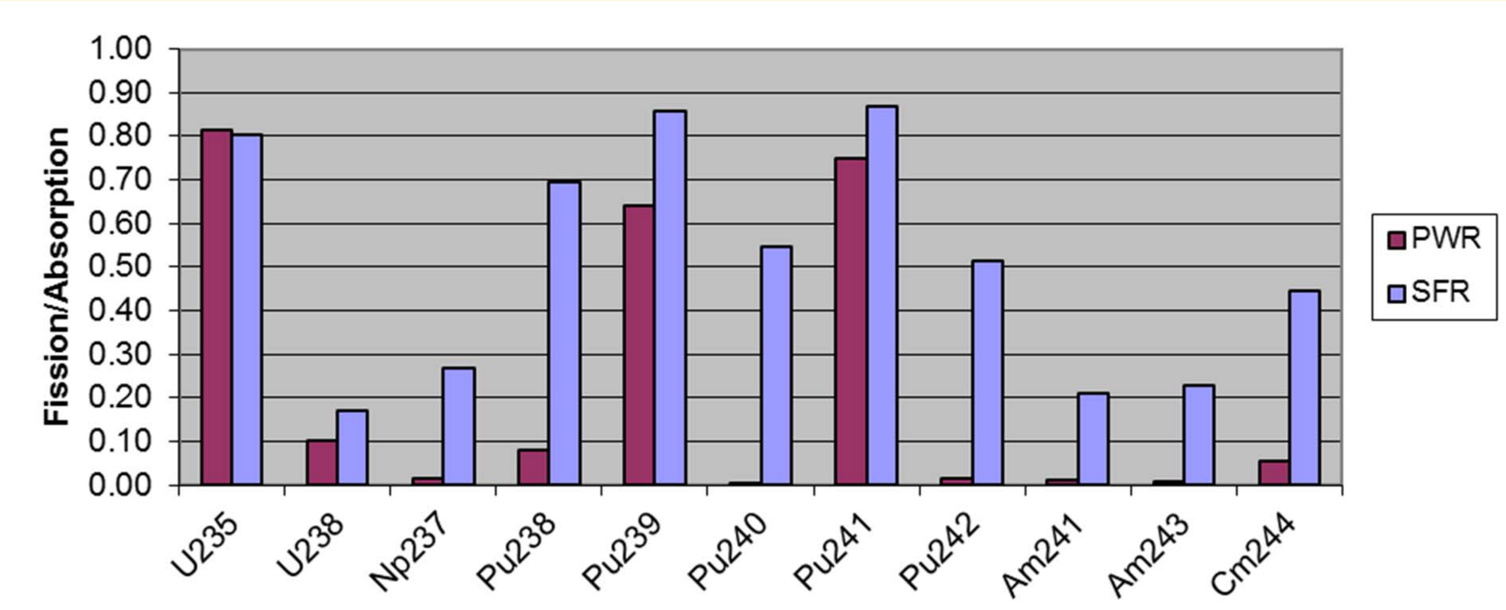
Meet the Presenter:

Dr. Robert Hill is co-National Technical Director for the DOE multi-Laboratory Advanced Reactor Technologies Program; this work includes technology innovation, safety and licensing, advanced materials, energy conversion technology, instrumentation and controls. He also serves as U.S. Member for the Generation-IV Sodium Cooled Fast Reactor and System Integration Project.

Fuel Cycle Implications of Energy Spectrum :

Fast reactors are typically intended for closed (recycle) fuel cycle with uranium conversion and resource extension

- Higher actinide generation is suppressed
- Neutron balance is favourable for recycled transuranics (Pu, Np, and Am)



Uranium Utilization :

Uranium utilization is one of the benefits of the fast reactor technologies

Through the conventional once-through systems, we have to dispose much amount of depleted uranium on the enrichment process, and total utilization of uranium is about half percent. Recycling the uranium used in fast reactor provides over 90 percent of uranium utilization.

Once-through systems

	PWR-50GWd/t	PWR-100GWd/t	VHTR	Fast Burner
Burnup, %	5	10	10.5	22.3
Enrichment, %	4.2	8.5	14.0	12.5
Utilization, %	0.6	0.6	0.4	0.8

Recycling Systems

	LWR		LWR-Fast Burner		Fast
	UOX	MOX	LWR-UOX	Fast Burner	Converter
Power sharing, %	90	10	57	43	100
Burnup, %	5	10	5	9	-
Enrichment, %	4.2	-	4.2	12.5	-
Utilization, %	0.7		1.4		~99

Sodium as a Fast Reactor Coolant :
Thermophysical and thermal-hydraulic properties of sodium are excellent and allow:
• Use of conventional stainless steels
• Smaller core with higher power density, lower enrichment, and lower heavy metal inventory
• Demonstrated natural circulation and overall passive safety performance
• Use of sodium codified in ASTM standards

Thermophysical Properties:	
Excellent Heat Transfer	✓+
Low Vapor Pressure	✓+
High Boiling Point	✓+
Low Melting Point	✓
Material Properties:	
Thermal Stability	✓+
Radiation Stability	✓+
Material Compatibility	✓+
Neutronic Properties:	
Low Neutron Absorption	✓+
Minimal Activation	✓
Negligible Moderation	✓+
Supports Passive Safety	✓+
Cost:	
Initial Inventory	✓+
Make-Up Inventory	✓+
Low Pumping Power	✓+
Hazards:	
Sodium reacts with air and water	

Worldwide Experience :
Extensive testing resulted in sodium as the primary coolant in nearly all (land-based) fast reactors constructed during the last 50 years.

Reactor	Country	MWth	Operation	Coolant
EBR 1	USA	1.4	1951-63	NaK
BR-2	Russia	2	1956-1957	Mercury
BR-10	Russia	8	1959-71, 1973-2002	Sodium
DFR	UK	60	1959-77	NaK
EBR II	USA	62.5	1963-94	Sodium
Fermi 1	USA	200	1963-72	Sodium
Rapsodie	France	40	1966-82	Sodium
BOR-60	Russia	50	1968-	Sodium
SEFOR	USA	20	1969-1972	Sodium
OK-550/BM-40A	Russia	155 (7 subs)	1969-	Lead Bismuth
BN 350*	Kazakhstan	750	1972-99	Sodium
Phenix	France	563	1973-2009	Sodium
PFR	UK	650	1974-94	Sodium
KNK 2	Germany	58	1977-91	Sodium
Joyo	Japan	140	1978-	Sodium
FFTF	USA	400	1980-93	Sodium
BN 600	Russia'	1470	1980-	Sodium
Superphenix	France	3000	1985-98	Sodium
FBTR	India	40	1985-	Sodium
Monju	Japan	714	1994-96, 2010-	Sodium
CEFR	China	65	2010-	Sodium
PFBR	India	1250	2016?	Sodium
BN-800	Russia	2000	2014-	Sodium
ASTRID	France	1500	2025?	Sodium
PGSFR	Korea	400	2028	Sodium

Generation-IV R&D Collaboration on SFR :

Several collaborative Generation-IV R&D Projects are being conducted to explore technology innovations which target to achieve the eight goals for the Generation IV nuclear energy systems

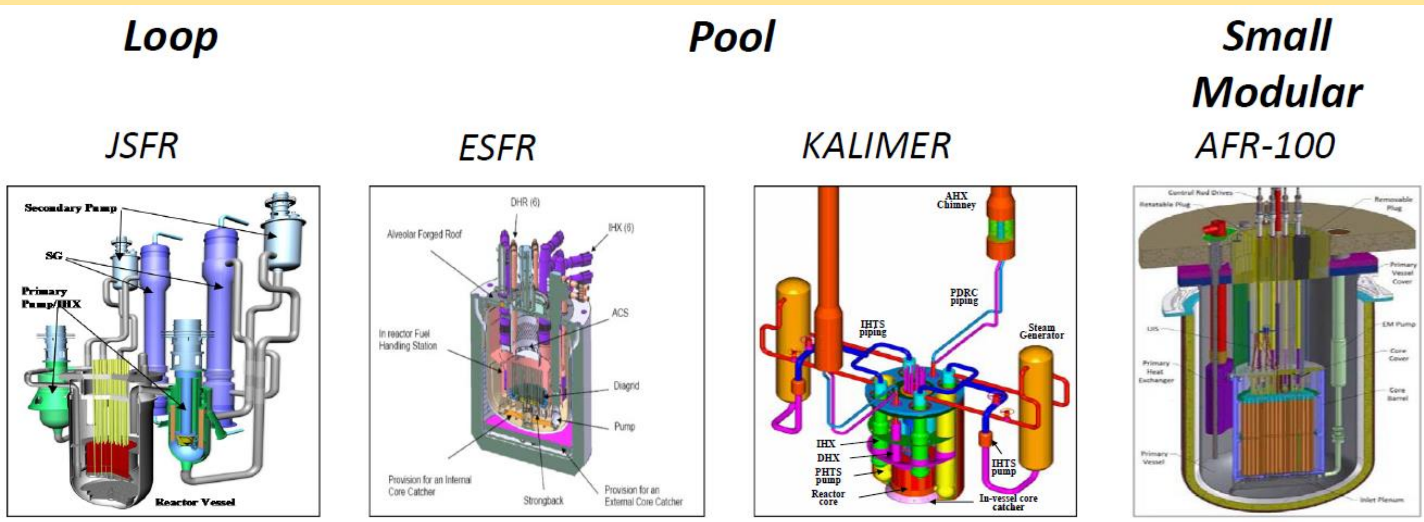
Criteria	Goal: Generation IV nuclear energy systems will….
Safety and Reliability-1	*excel in safety and reliability.*
Safety and Reliability-2	*have a very low likelihood and degree of reactor core damage.*
Safety and Reliability-3	*eliminate the need for offsite emergency response.*
Economics-1	*will have a clear life-cycle cost advantage over other energy sources.*
Economics-2	*will have a level of financial risk comparable to other energy projects.*
Sustainability-1	*will provide sustainable energy generation that meets clean air objectives and promotes long-term availability of systems and effective fuel utilization for worldwide energy production.*
Sustainability-2	*will minimize and manage their nuclear waste and notably reduce the long-term stewardship burden, thereby improving protection for the public health and the environment.*
Proliferation Resistance and Physical Protection-1	*increase the assurance that they are a very unattractive and the least desirable route for diversion or theft of weapons-usable materials, and provide increased physical protection against acts of terrorism.*

SFR System Research Plan :

System Research Plan was updated and released in July 2013.
(and further update was conducted in October 2019)

<u>Contents:</u>

<u>*Development Targets and Design Requirements*</u>
<u>*5 SFR R&D Projects*</u>
<u>*4 SFR Design Concepts*</u>



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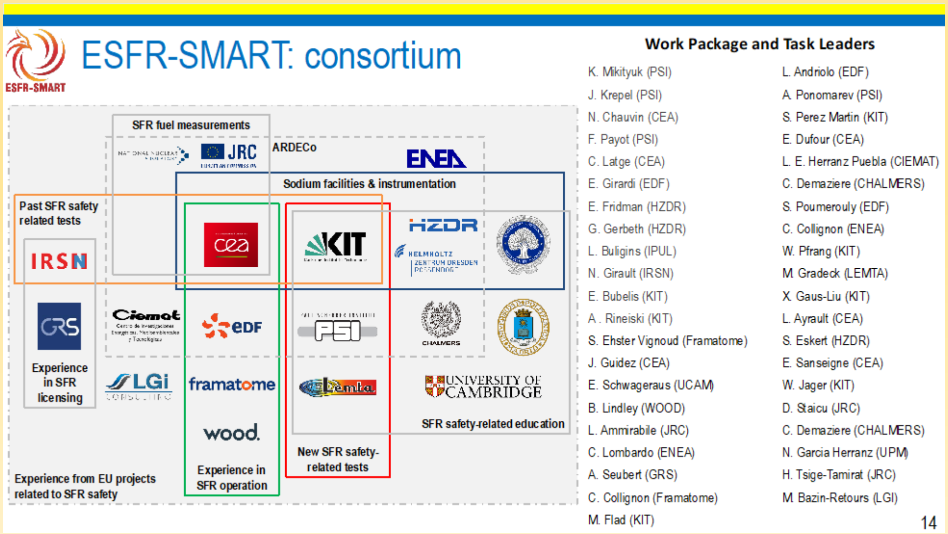
4-1-2. European Sodium Fast Reactor: An Introduction

Summary / Objectives:

This webinar presents a brief history of the conceptual development of a large-power (3600 MWth) European Sodium Fast Reactor (ESFR), discusses the status of the current R&D activities on Generation-IV ESFR safety enhancements of the Horison-2020 ESFRSMART project, and provides an overview of new safety measures proposed for improvement of the three safety functions: reactivity control, heat removal and radioactivity containment. Also, experimental programs currently on-going in Europe in support of the ESFR R&D are briefly introduced. A summary of the activities to be performed during the next phase of the project concludes the webinar.

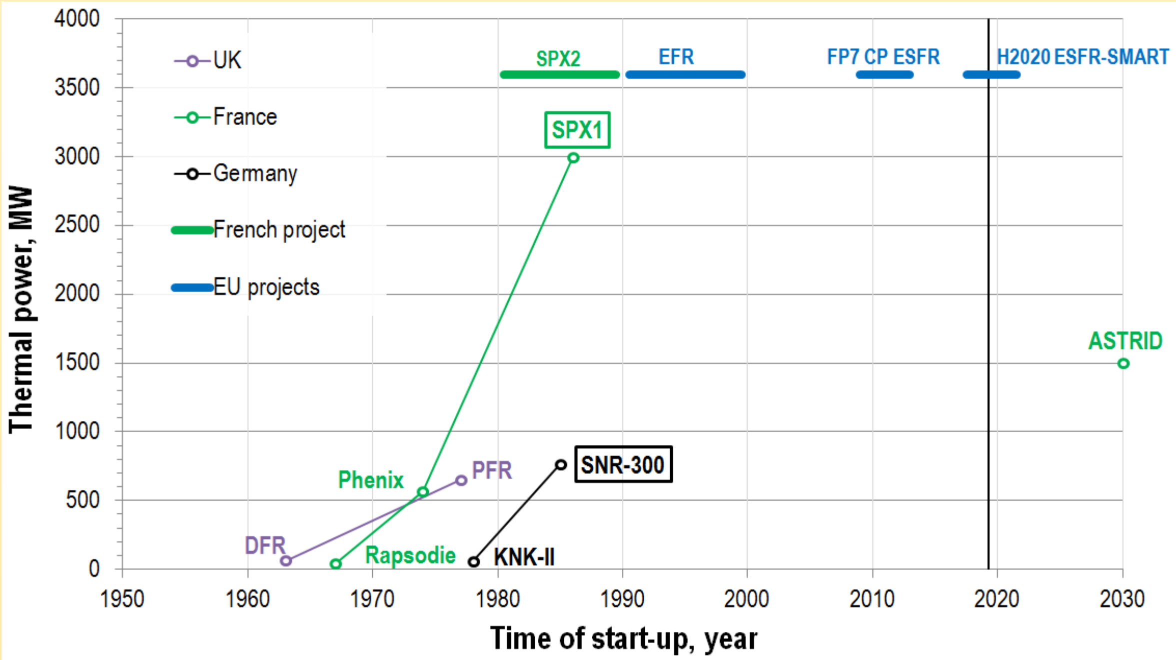
Meet the Presenter:

Dr. Konstantin Mikityuk has been involved in research of safety related aspects of various nuclear reactors with a fast neutron spectrum since he earned his doctorate from the Moscow Engineering Physics Institute in 1992: first at the Russian Research Centre "Kurchatov Institute," and then at the Paul Scherrer Institute (PSI). His current interests are safety analysis of sodium-cooled fast reactor, in particular neutronics and thermal-hydraulic aspects of sodium boiling. Dr. Mikityuk is a Group leader at PSI, Maître d'enseignement et de recherche at Ecole Polytechnique Federale de Lausanne (EPFL), Lecturer at the Eidgenössische Technische Hochschule Zürich (ETHZ). He is also the coordinator of the Horison2020 ESFR-SMART project.



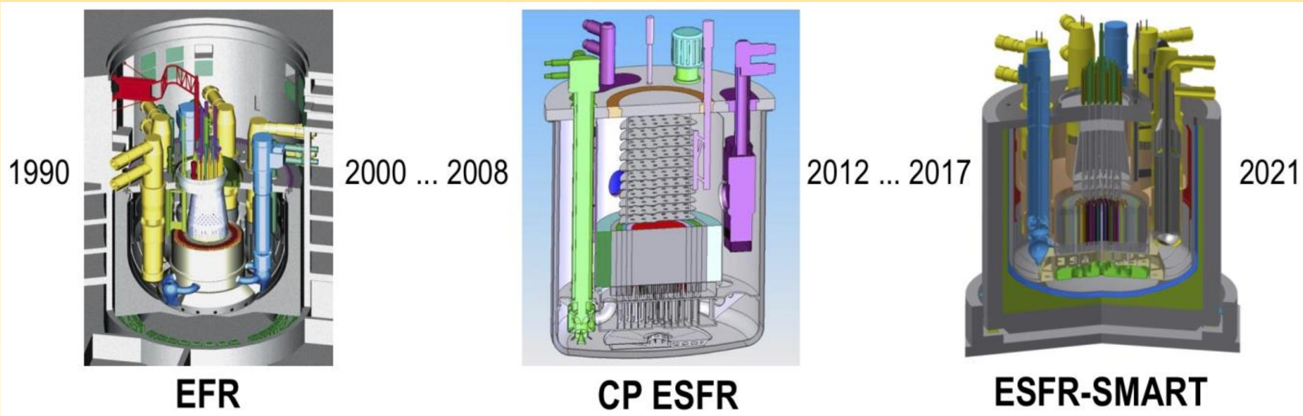
1. European Sodium Fast Reactor: brief history

The ESFR-SMART project aims at enhancing further the safety of Generation-IV SFRs and in particular of the commercial-size ESFR in accordance with the European Sustainable Nuclear Industrial Initiative (ESNII) roadmap and in close cooperation with the ASTRID program.



2. European Sodium Fast Reactor: reactor design

- Thermal / electrical power 3600 / 1500 MW
- Mass of sodium in the primary pool ~2500 t
- Primary sodium temperature 395ºC –545ºC
- 6 Heat eXchangers , 3 Primary Pumps, 36 Steam Generators



3. ESFR-SMART: project in a nutshell

Name:

- ESFR-SMART: European Sodium Fast Reactor Safety Measures Assessment and Research Tools

Goals:

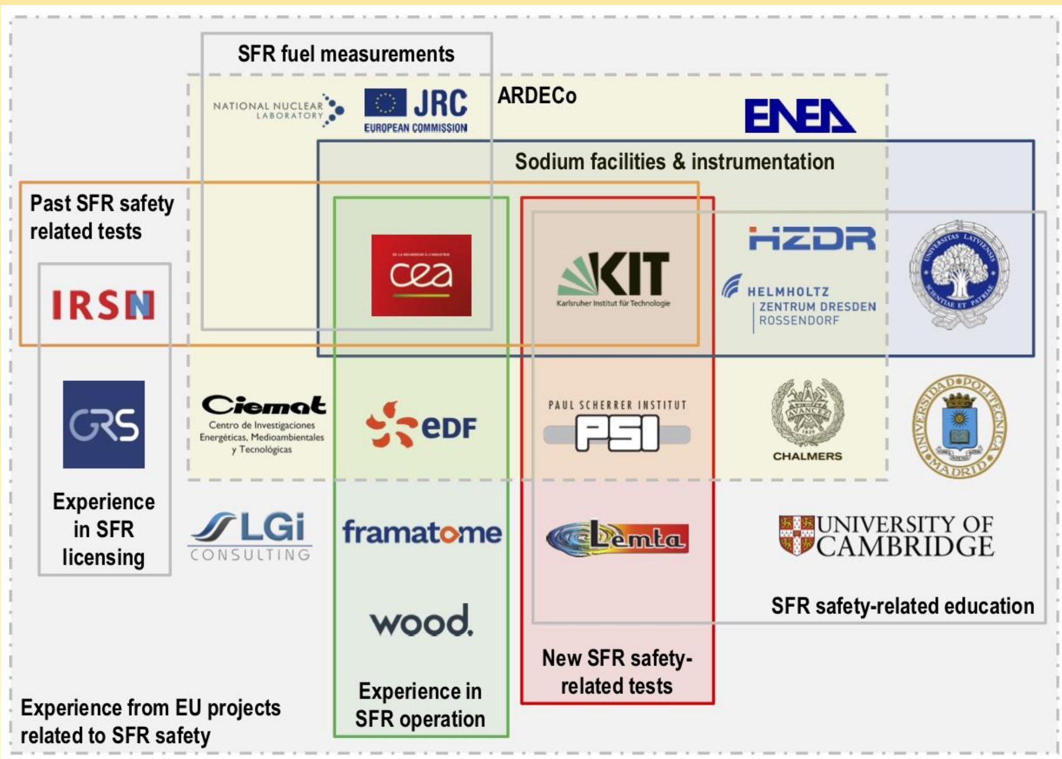
- Select and assess innovative safety measures for European SFR concept
- Develop new research tools related to SFR safety (calculational codes, experimental data and facilities)

Budget: 5 MEUR of Euratom contribution + ~5 MEUR of consortium's own contribution

Timeframe: 01.09.2017 31.08.2021

ESFR-SMART
sodium fast reactor safety

4. ESFR-SMART: consortium

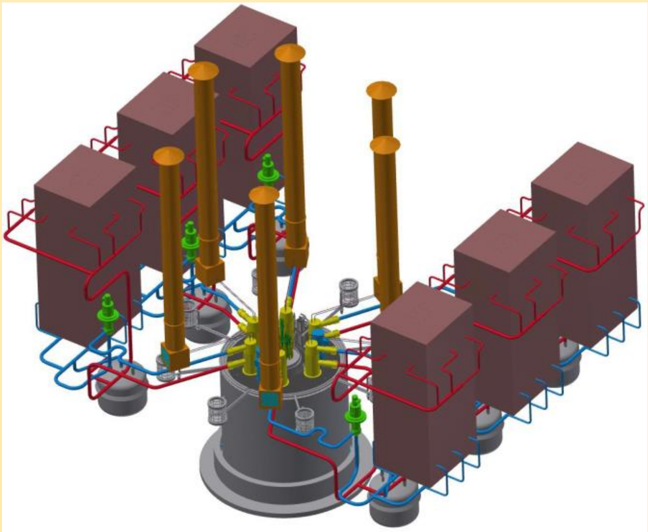


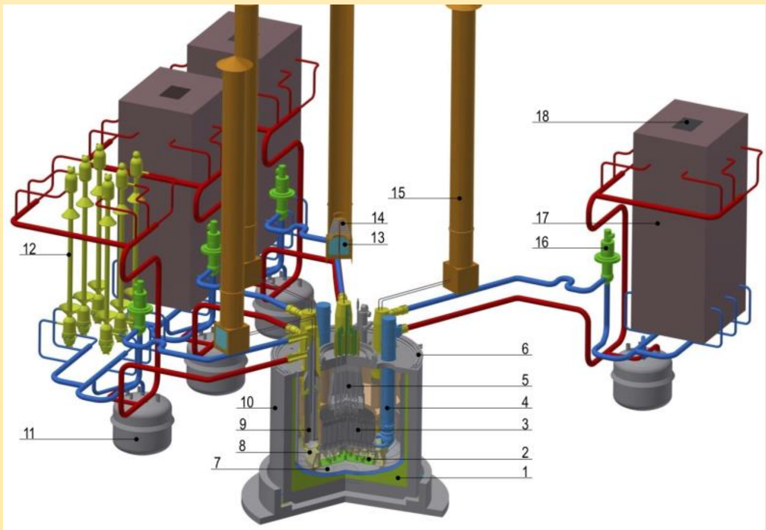
5. Overall view of new ESFR

An overview of new safety measures proposed for improvement of the three safety functions:

- Reactivity control, Heat removal and Radioactivity containment.

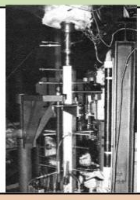
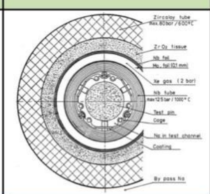
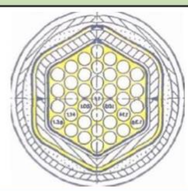
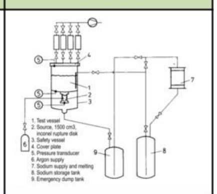
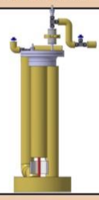
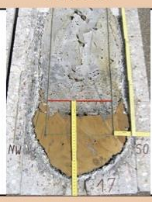
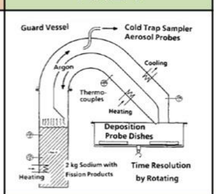
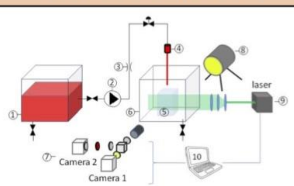
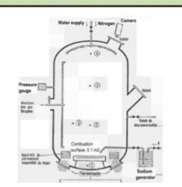
New ESFR consists of tall chimney for decay heat removal, six steam generators inside the boxes, six secondary loops and the primary sodium pool with core, 3 pumps and 6 heat exchangers.





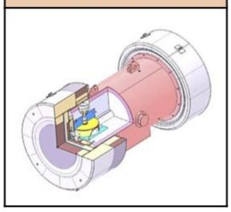
6. ESFR-SMART: past and ongoing tests

- Legacy data obtained in past tests are used for validation of computer codes.
- The new facilities for ongoing test are designed and under development.

Normal operation	Sodium boiling	Severe accident (SA) management		SA mitigation
Superphenix	KNS-37	CABRI	SCARABEE	FAUST
				
KASOLA	KARIFA	LIVE	JIMEC	NALA
				
ECFM	CHUG	HAnSOLO and JEDI		FANAL
				

Past tests
Ongoing tests

MOX fuel measurements



4-1-3. Lead Fast Reactor (LFR)

Summary / Objectives:

The Lead-cooled Fast Reactor (LFR) is characterized by a fast neutron spectrum; a liquid coolant with a very high margin to boiling and relatively inert interaction with air or water; and design features that capitalize on these attributes. As with other fast spectrum reactors, the LFR offers fuel cycle options that greatly enhance resource utilization and sustainability. LFR concepts offer great potential in terms of safety, simplification, proliferation resistance and the economic performance. The webinar presents background on fast reactor physics, the historical development and present status of LFR technology and the main characteristics of LFR concepts under current consideration.

Meet the Presenter:

Professor Craig Smith, Research Professor at the Naval Postgraduate School, Monterey, CA, USA, is a nuclear engineer with broad experience in nuclear energy technology, radiation detection and information science. His previous employment includes a career at Lawrence Livermore National Laboratory (LLNL) where he led the Fission Energy and System Safety Program. Beginning in 2004, he served as the LLNL Chair Professor at the Naval Postgraduate School (NPS) in Monterey, CA. After retiring from LLNL, he assumed his current position as Research Professor of Physics at NPS.

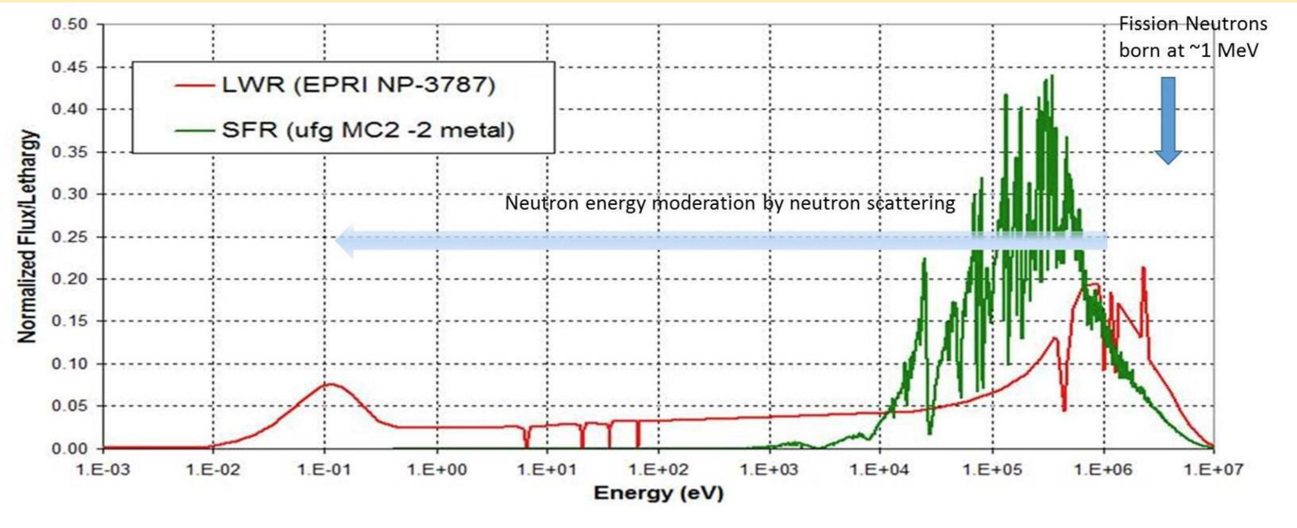
Why LFR Technology?

- As with other Fast Reactors, LFRs offer:
  - Significant advantage in sustainability/uranium utilization – better use of natural resources
  - Potential for dramatic reduction of high level waste if full recycle (closed fuel cycle) is used
- Relative to other fast reactors, LFRs have a unique combination of favorable features:
  - Very high boiling point (1737°C)
  - Benign chemistry (no rapid chemical reaction with water/air)
  - Low vapor pressure
  - Excellent neutronic properties for fast spectrum operation
- These features are inherent in the properties of the lead coolant and can be exploited through proper plant design.

9

1. **A Recap on Fast Reactor Physics: Comparison of fast (SFR) vs. Thermal (LWR) spectra**

- In thermal reactors such as LWRs, most fissions occur around the ~0.1 eV "thermal" peak.
- In fast reactors such as LFRs or SFRs , neutron energy moderation is avoided fissions occur mainly in "fast" energy range.



2. **Some Chemical and Thermal Characteristics of Liquid Metal Coolants**

- Both of lead-based coolants are practically inert in terms of chemical reactivity with water and air, and this has important and favorable implications for the design, safety, and economic potential of LFRs.

Coolant	Melting Point (°C)	Boiling Point (°C)	Chemical Reactivity (w/Air and Water)
Lead-Bismuth (Pb-Bi, LBE)	125	1670	Practically Inert
Lead (Pb)	327	1737	Practically Inert
Sodium (Na)	98	883	Highly reactive

3. Stored Potential Energy for Different Reactor Coolants

- The very low comparative amount of stored energy in lead-cooled fast reactor coolants is an indication of their enhanced safety potential based on the intrinsic properties of the coolant.

Coolant	Water	Sodium	Lead, LBE
Parameters	P = 16 MPa T = 300 °C	T = 500 °C	T = 500 °C
Maximal potential energy, GJ/m^3, *including:*	~ 21.9	~ 10	~ 1.09
Thermal energy / *including* compression potential energy	~ 0.90 / ~ 0.15	~ 0.6 / None	~ 1.09 / None
Potential chemical energy of interaction	With zirconium ~ 11.4	With water 5.1 With air 9.3	~0
Potential chemical energy of interaction of released hydrogen with air	~ 9.6	~ 4.3	None

4. Recap of Design Parameters of Gen IV Reference LFR Concepts

Within the SRP for LFR, there are reference systems adopted by the committee, and they include, the ELFR (large reactor), BREST-OD-300 (under construction), or SSTAR (transportable, small modular reactor with the supercritical CO2 gas turbine cycle as a secondary cycle).

Parameter	ELFR	BREST-OD-300	SSTAR
Core power (MW_{th})	1500	700	45
Electrical power (MW_e)	600	300	20
Primary system type	Pool	Pool/loop	Pool
Core inlet T (°C)	400	420	420
Core outlet T (°C)	480	535	567
Secondary cycle	Superheated steam	Superheated steam	S-CO_2
Net efficiency (%)	42	43.5	44

5. LFRs Have the Potential to Excel in Safety

To summarize this part of the discussion, lead-cooled fast reactors have the potential to excel in safety for reasons outlined on this slide.

LFRs Have the Potential to Excel in Safety

- The very high boiling point of lead (~1737°C):
  - Allows reactor operation at near atmospheric pressure
  - Eliminates the risk of core voiding due to coolant boiling
- No rapid chemical reactions between lead and either water or air
  - No energetic releases or hydrogen production from chemical reactions
  - Use of water as ultimate heat removal fluid is conceivable, should other heat removal systems fail
- The thermal capacity of lead combined with the large mass of coolant
  - Significant thermal inertia in the event of hypothetical accident initiators.
  - Long grace time (the need for operator's intervention is eliminated or significantly delayed)
- Lead shields gamma radiation and retains iodine and cesium up to 600°C
  - Reduced source term in case of fuel rod failure → enhanced Defense-in-Depth.
- The low neutron moderation of lead allows greater fuel spacing without excessively penalizing neutronic performance:
  - Reduced risk of flow blockage
  - Reduced core pressure drop and simple coolant flow path allow decay heat to be removed through natural circulation

13

6. There are challenges to address, and the first is corrosion potential, and this is the one that gets the most attention. Other challenges that need to be considered include the high melting or freezing point of lead, which is 327°C. Another challenge relates to seismic or structural considerations due to the high density and weight of the coolant.

However, There are Challenges to Address

- **Corrosion potential**
  - Operate at temperatures low enough to avoid corrosion (current materials can be used)
  - Use advanced materials for higher temperature operation, to enhance economics
    - Silicon or Aluminum enhanced materials (i.e., Alumina Forming Austenitic (AFA) steels and Silicon enhanced steels)
    - Surface coating with corrosion-protective materials for higher temperature operation (cladding + steam generator)
    - Functionally graded composite materials
  - In any case, methods must be implemented to monitor/control oxygen content to maintain protective oxide coatings and avoid the formation of PbO
- **High melting point (327°C)**
  - Proper engineering to avoid lead freezing
- **Seismic/structural considerations due to heavy coolant**
  - Compact size mitigates this challenge
  - Seismic isolation
- **Opaque, high-temperature coolant**
  - Similar in service inspection issues and solutions as for SFR
  - Accessibility/replaceability of components
  - Newer acoustic methods

These challenges are generally technical in nature and can be overcome through proper engineering and R&D work

10

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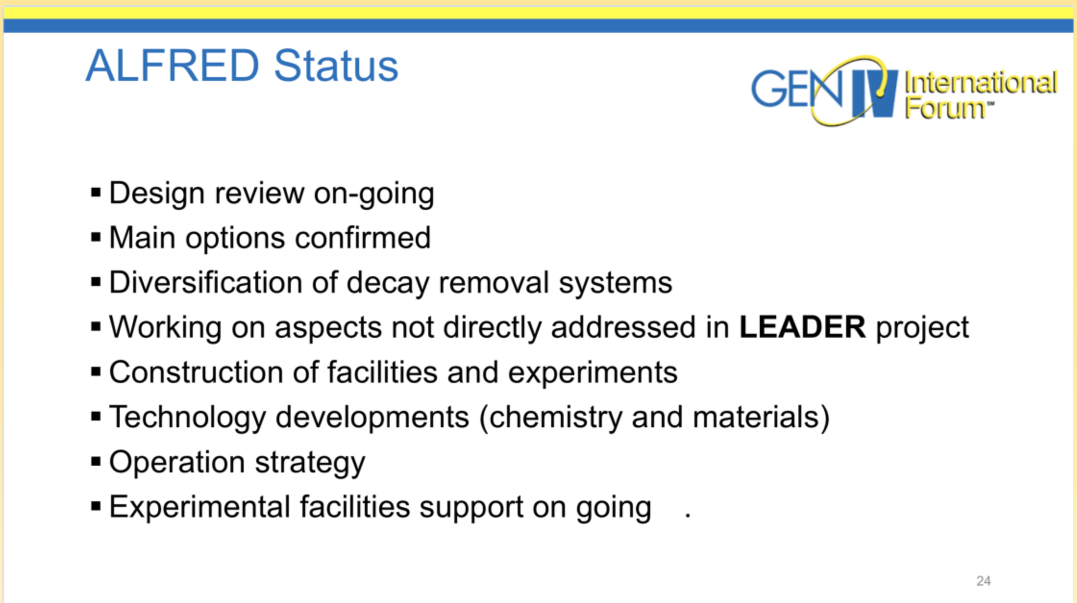
4-1-4. Advanced Lead Fast Reactor European Demonstrator - ALFRED Project

Summary / Objectives:

The webinar presents the main design features of the ALFRED nuclear reactor demonstrator as developed in the frame of the Collaborative projects funded by the European Community Framework Program. The presentation will provide an overview of specific design solutions, safety approach and safety characteristics of ALFRED, touching the most important aspects of the demonstrator. Latest developments are briefly presented as well. The remaining research challenges are then explained at the light of the present technology understanding to highlight the present status of knowledge and further steps to be pursued.

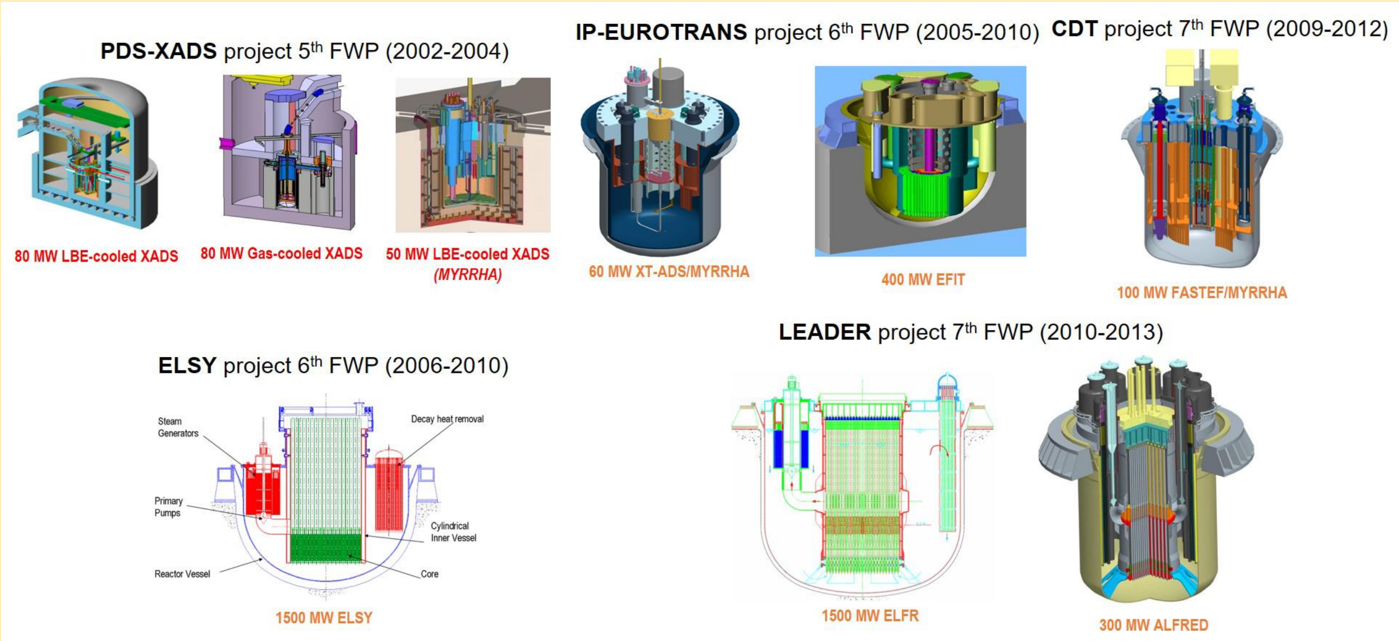
Meet the Presenter:

Dr. Alessandro Alemberti is the Nuclear Science Development Manager of Ansaldo Nucleare (Italy) and in this position takes care of the Research & Development activities of the company. He coordinated the ELSY and LEADER projects in the frame of the 6th and 7th Framework Programs of the European Community, projects devoted to Lead cooled Fast Reactors development and participated as well to the main EU projects related to Lead and Lead Bismuth Eutectic (LBE) coolant technologies in recent years. Since 2012, he has has served as the chairman of the Generation IV International Forum (GIF) Lead Fast Reactor provisional System Steering Committee representing EURATOM.



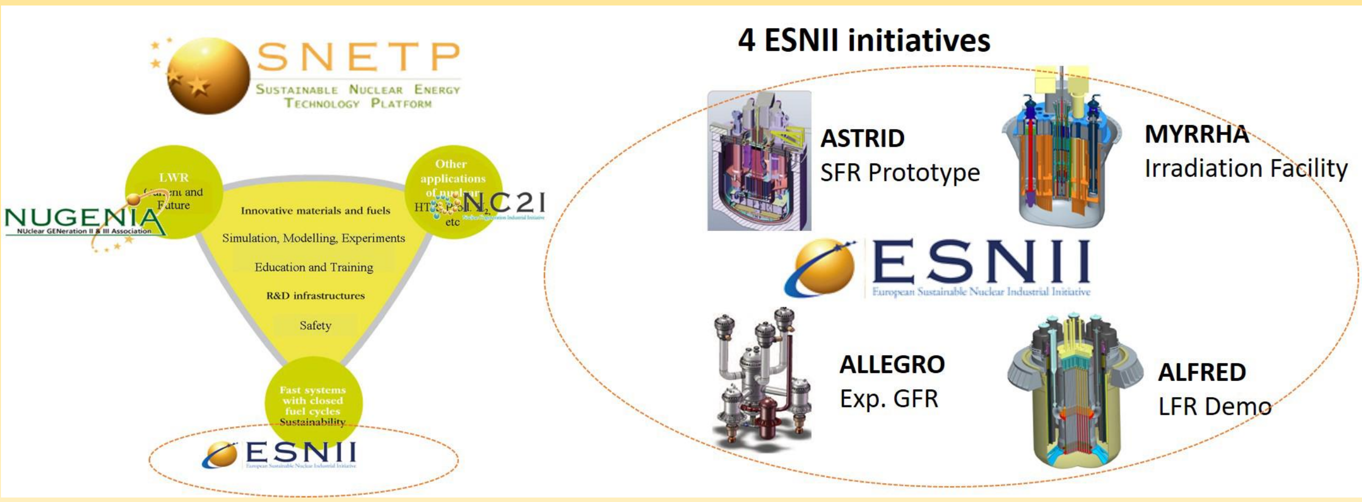
1. Heavy Liquid Metal Technology Development in Europe

Works on the development of lead-cooled fast reactors are actively carried out in European Union countries (concept projects ELFR, ELSY, LEADER, ALFRED) have been proposed.



2. The European Context: Sustainable Nuclear Energy Technology Platform

- LFR technology can offer a safe, sustainable and competitive alternative to address market opportunities
- More than **200 M€** invested in LFR technology in the last **10 years**

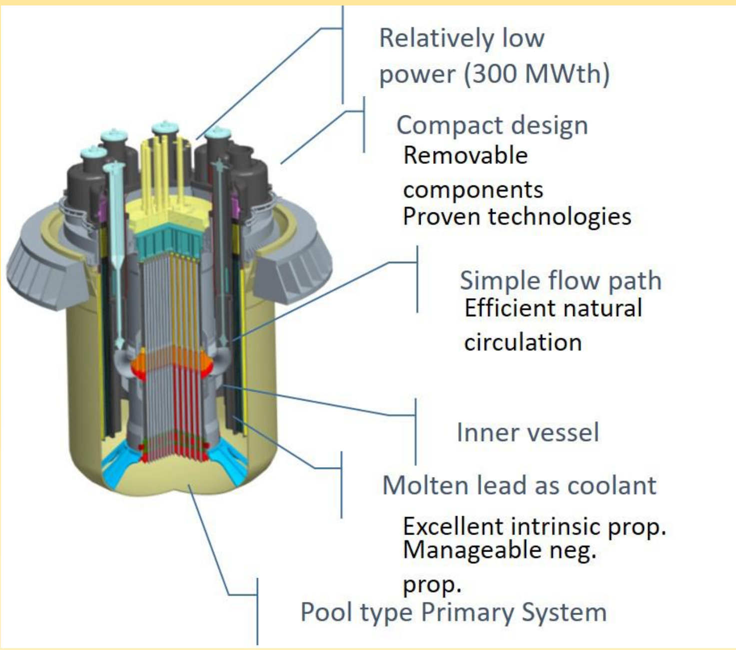


3. ALFRED Support: The FALCON Consortium (**FALCON** – **F**ostering **A**lfred **CON**struction)

- FALCON Consortium Agreement was established in 2013 to bring LFR technology to industrial maturity
- FALCON recently evolved to better cope with European context.
- Main objectives are:
    - Firm commitment to ALFRED as a Major Project in Romania
    - Finalization of ALFRED feasibility study
    - Initiation of construction of supporting R&D facilities

4. ALFRED – Design Guidelines

ALFRED design should be based on available technology as much as possible, in order to speed up the construction time.



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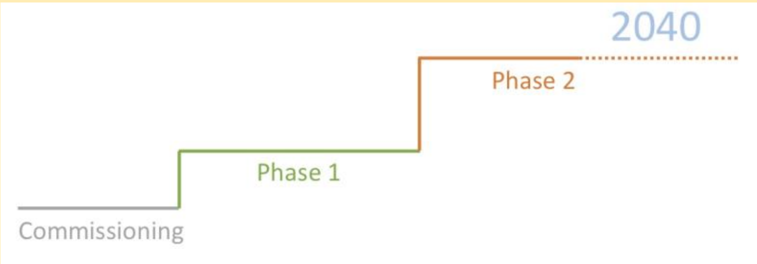
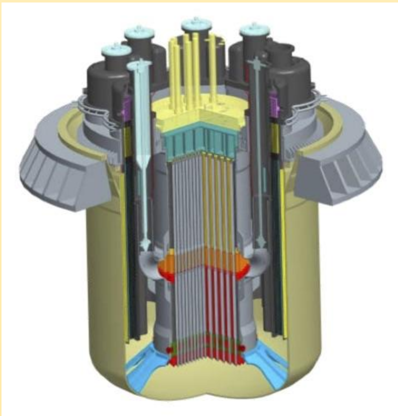
5. ALFRED DEMONSTRATOR: a way to achieve technology maturity

The operation of ALFRED will be based on a stepwise approach:

- Phase 1: operation at low power in low-temperature range
    - Presently existing proven materials working without corrosion protection
- Phase 2: operation at full power in high-temperature range
    - Coated materials fully qualified during phase 1

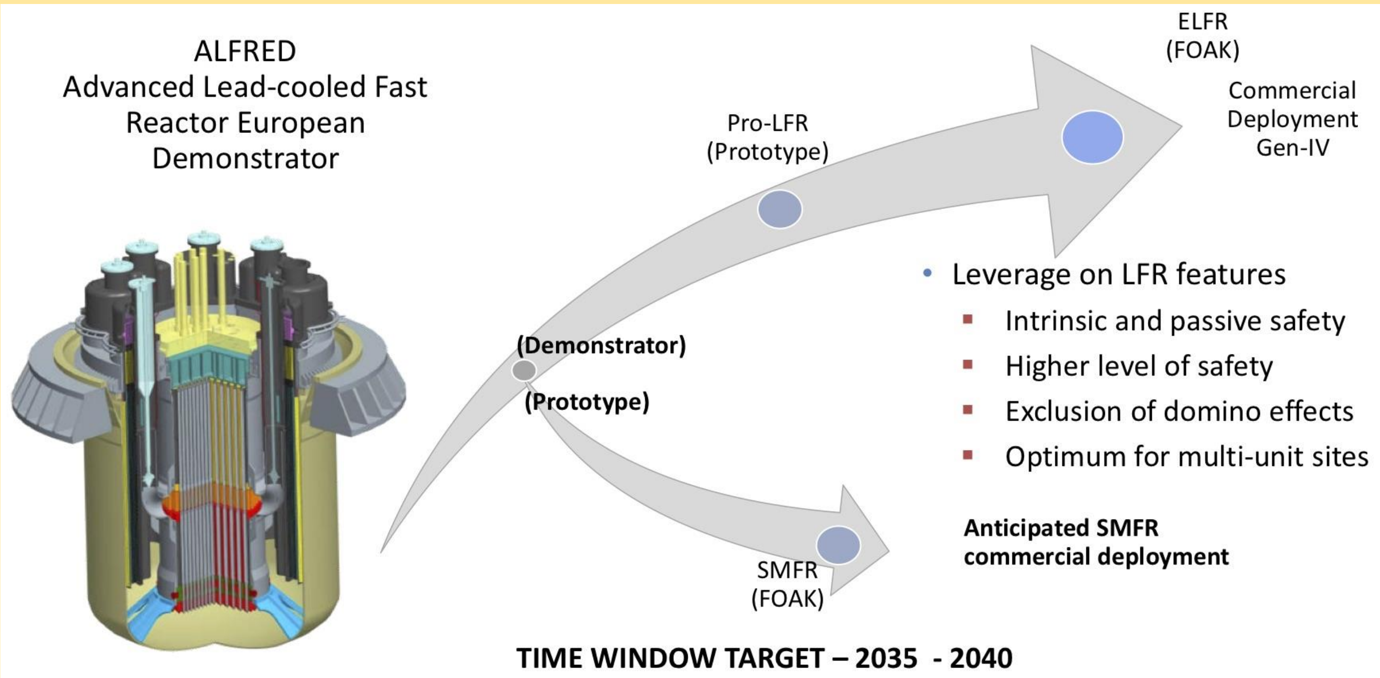




6. ALFRED: a LFR Demo with SMR-oriented features

Example of fast neutron reactor cooled by molten lead having SMR-oriented features are:

- SMR derived from the ALFRED concept, FALCON consortium, Europe.



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4-1-5. MYRRHA an Accelerator Driven System Based on LFR Technology

Summary / Objectives:

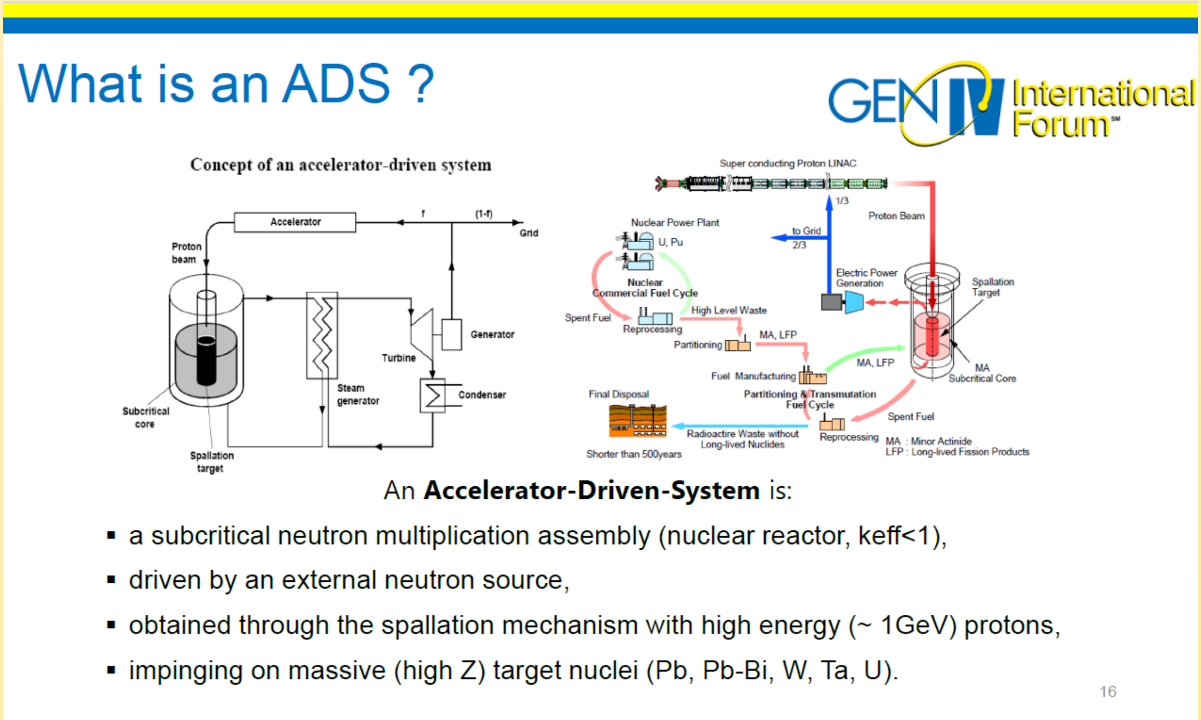
SCK•CEN is actively working on designing and building a new multifunctional research installation: MYRRHA as in Multi-purpose hYbrid Research Reactor for High-tech Applications. This webinar will present the MYRRHA project, an accelerator driven system coupling a sub-critical Pb-Bi cooled reactor and a high power proton accelerator through a spallation target which is the very first prototype of a nuclear reactor driven by a particle accelerator in the world. As an external source of neutrons, this particle accelerator maintains the nuclear fission chain reaction. It is referred to as a subcritical reactor: the core does not contain enough fissile material to spontaneously maintain the chain reaction. This innovative nuclear technology is safe and easy to control. When the particle accelerator is stopped, the chain reaction also stops automatically within a fraction of a second.

Meet the Presenter:

Dr. Hamid Aït Abderrahim is both the Deputy Director General of SCK•CEN, the Belgian nuclear research center, and a professor of reactor physics and nuclear engineering at the "Université Catholique de Louvain" at the Mechanical Engineering Department of the "Ecole Polytechnique de Louvain". Since 1998, he has been the director of the MYRRHA project. He is a partner and/or coordinator of various projects of the European Commission framework programme related to advanced nuclear systems or to partitioning and transmutation of HLW management. From September 2007 to December 2011, he chaired the Strategic Research Agenda working group of the SNETP and has been the chairman of the Governing Board of SNETP since 2015. He represents Belgium in the Governing Board of the project JHR.

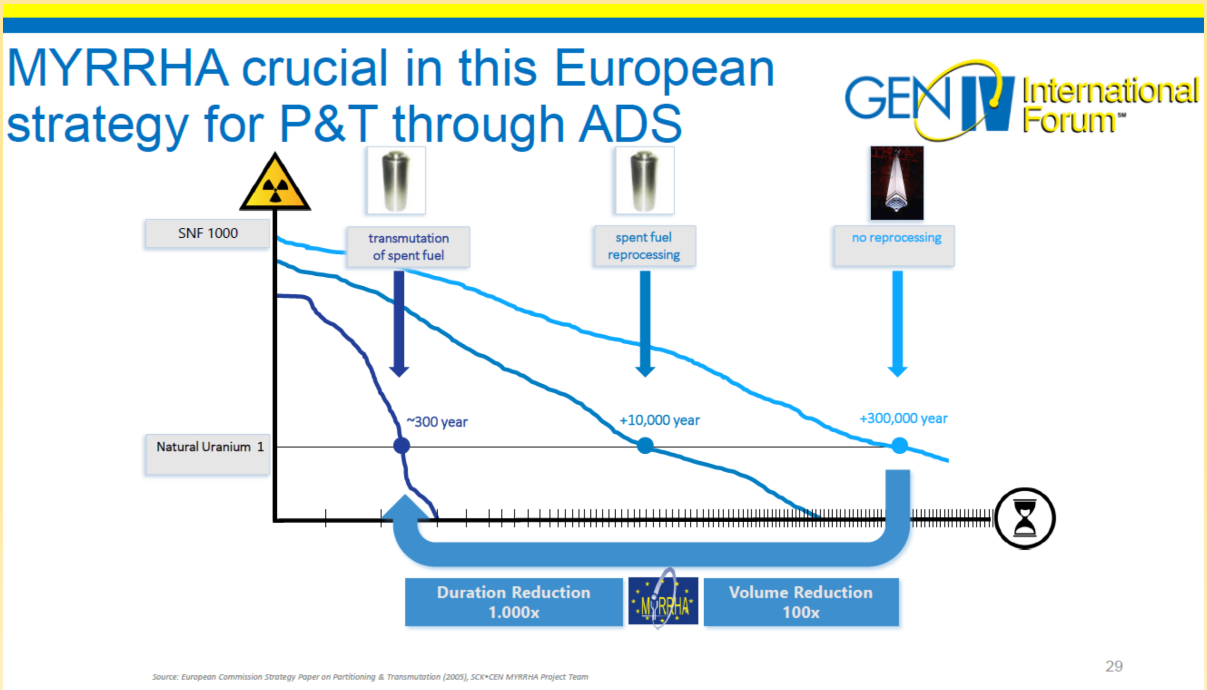
Introduction of an Accelerator driven system (ADS):

The ADS is simply reactor. This system is need an external source of neutrons that source is produce to a linear accelerator into the center of core on heavy metals. These heavy metals are led, bismuth, tungsten, tantalum etc.



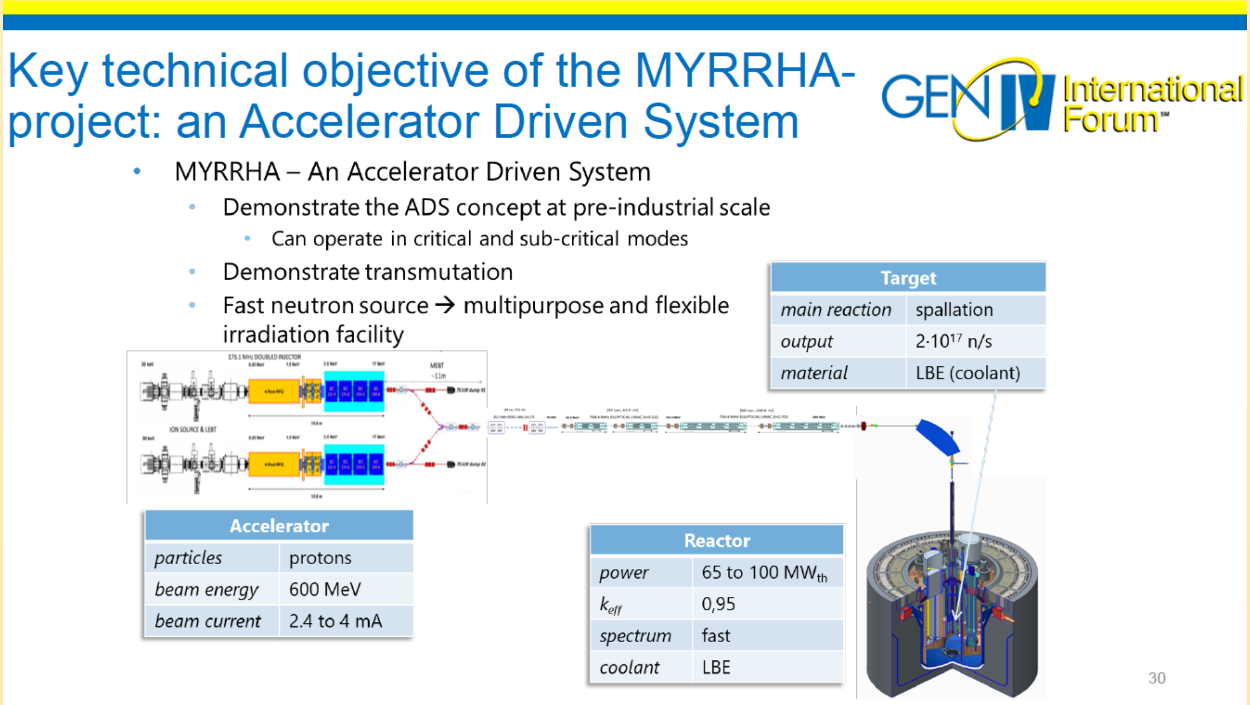
Transmutation impact:

The time scale needed for the radiotoxicity of the waste to drop to the level of natural uranium will be reduced from a 'geological' value (300,000 years) to a value that is comparable to that of human activities (several hundreds of years).



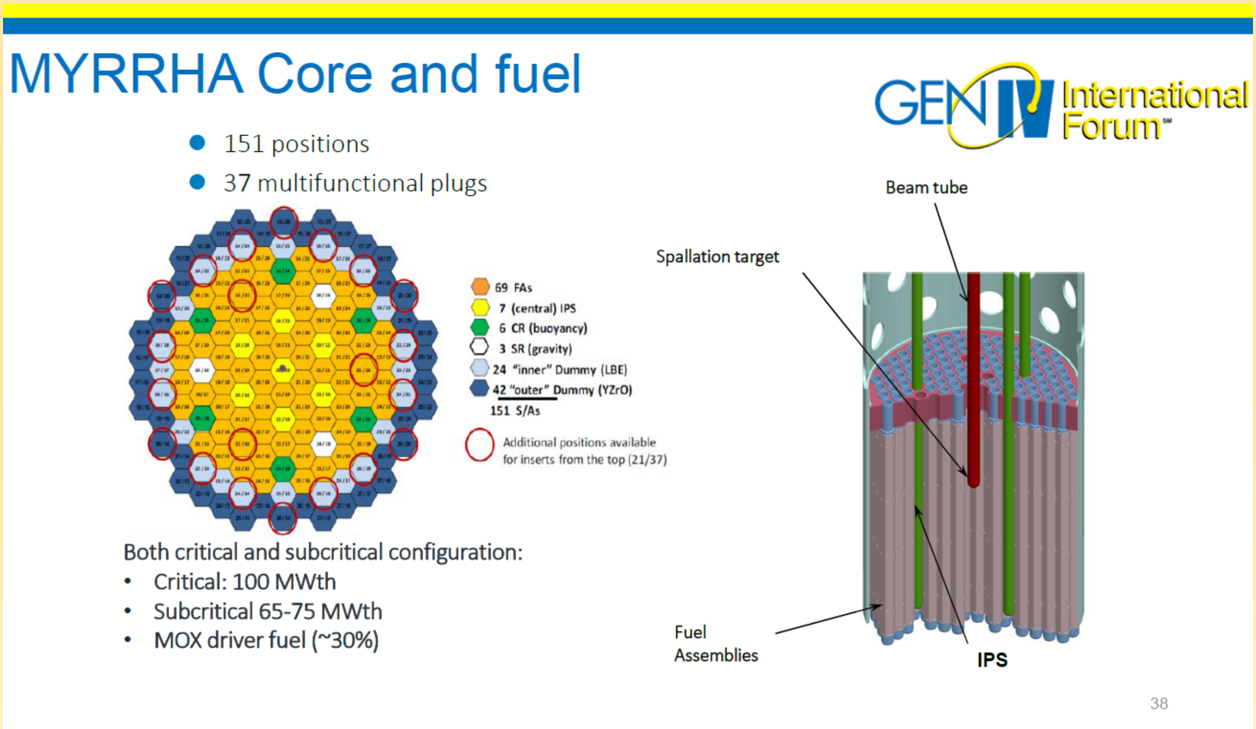
Introduction of the MYRRHA project:

The MYRRHA is an ADS, but the operate mode has critical and sub-critical mode. The neutron source in sub-critical mode is created by shooting a proton beam of 600 MeV at maximum on a led-bismuth target in the center core.



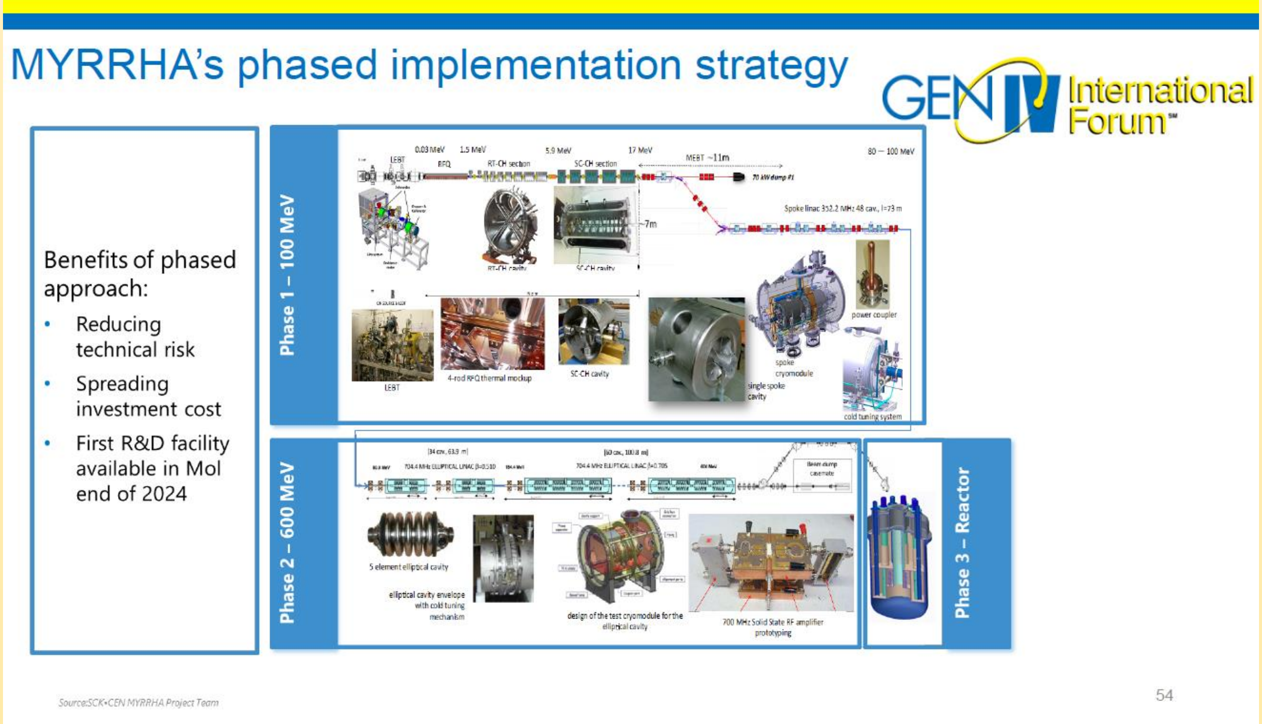
MYRRHA Core and Fuel Overview:

The MYRRHA core has the hexagonal fuel assemblies with MOX fuel, the control rods etc. The central place in the core is the beam tube with spallation target.



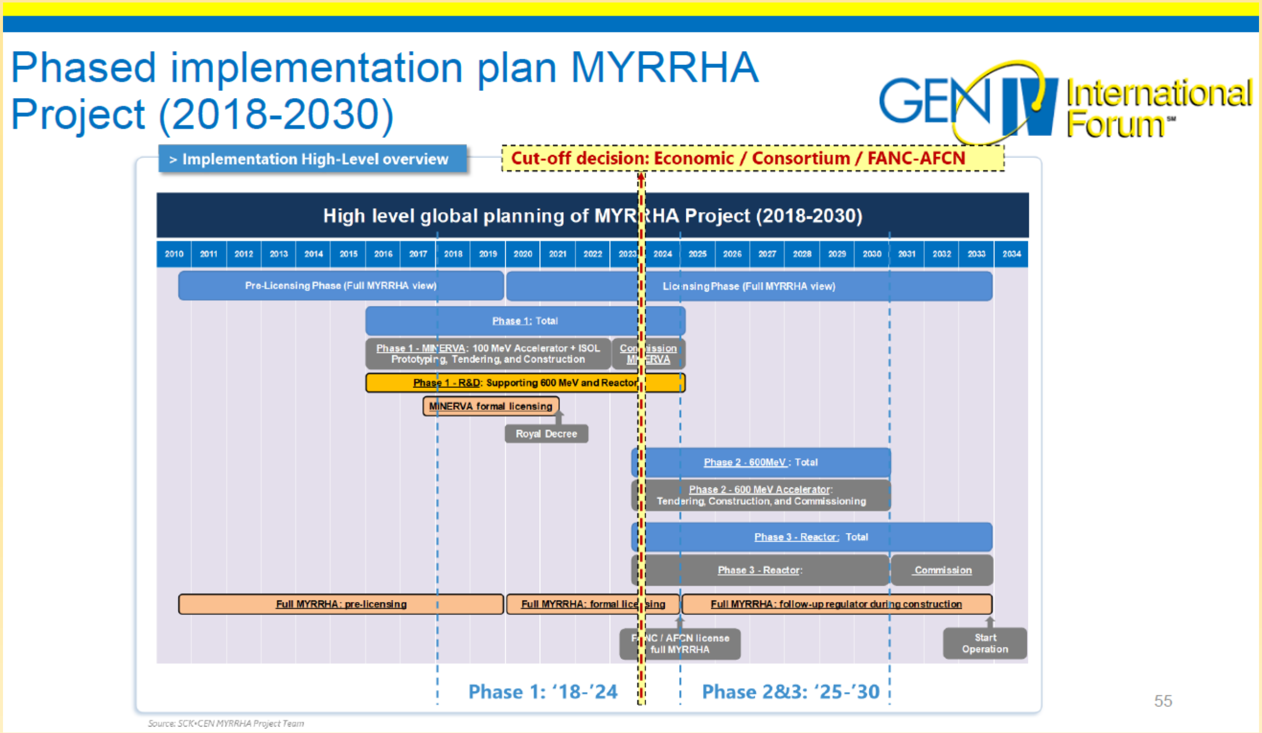
MYRRHA Project strategy:

The phase 1 is the accelerator with two injectors until 100MeV. The phase 2 is upgrade the accelerator to 600MeV, and the phase 3 is construct the reactor.



MYRRHA Project Plan:

The accelerators and targets for regular isotope for phase 1 will be built by 2022. And we take the decision by 2024 to upgrading the accelerator to 600MeV of phase 2 and constructing the reactor of phase 3.



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4-1-6. Gas Cooled Fast Reactor (GFR)

Summary / Objectives:

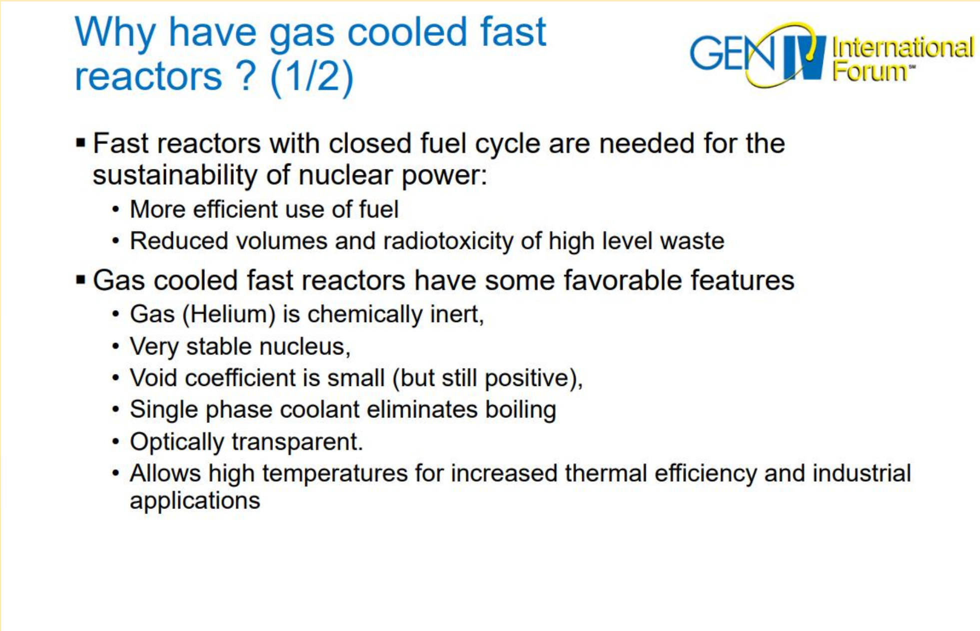
The Gas Cooled Fast Reactor (GFR) is one of the six promising technologies selected by the GIF. The presentation summarizes the main advantages and drawbacks of GFRs and the key design and safety issues as well as the related research and development programs.

Meet the Presenter:

Dr. Alfredo Vasile earned a Master of Physics Degree at the Balseiro Institut (CNEA, Argentine) and his Doctorate in Nuclear Engineering at the Grenoble University (France) in 1977. He joined CEA in 1981 working at the RAPSODIE sodium cooled experimental fast reactor at Cadarache. He has held laboratory head positions on core physics and safety studies both for light water reactors and fast reactors. Dr. Vasile participated at the Gen IV Roadmap definition process as a member of the Light Water Reactors Technical Group and was the French representative of the INPRO Steering Technical Committee for the Joint Study on Closed Nuclear Fuel Cycle with Fast Reactors. He is presently project manager of the ESNII Plus European Project on fast reactors, the French representative at the IAEA Technical Working Group on Fast Reactors, GIF GFR Steering Committee, GIF GFR Conceptual Design and Safety and GIF SFR Safety and Operation Project Management Boards. Dr. Vasile also serves as the CEA representative for the ALLEGRO GFR experimental reactor project.

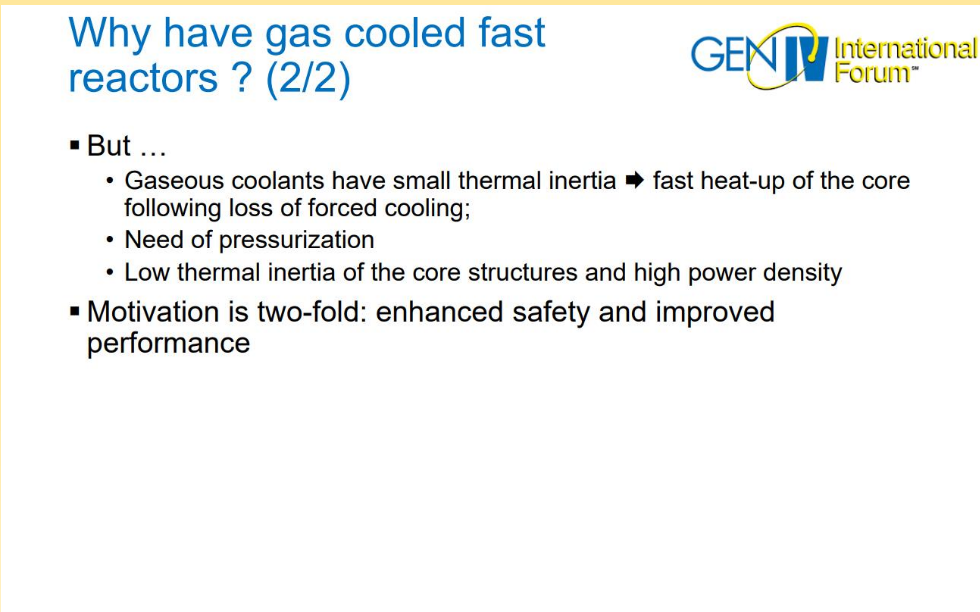
1. Motivations of fast reactor and GFR:

Fast reactor with closed fuel cycle can use nuclear fuels more efficiently, and reduce volumes and radiotoxicity of high level waste. GFR has some favorable features compared to fast reactors using liquid coolant.



2. Drawbacks of GFR:

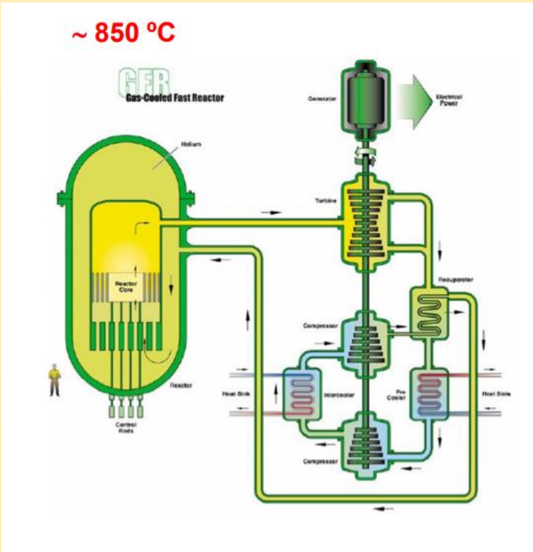
Typically gaseous coolant has a low thermal inertia, which leads fast heat-up of the core following loss of forced cooling. We need to have pressurized systems even in a normal operation roughly in range of 7 MPa. Low thermal inertia of the core makes the decay heat removal difficult.



3. The Gen IV GFR system:

The Gen IV GFR uses uranium-plutonium carbide with SiC cladding.

The core outlet temperature is 850 degree Celsius, which is very interesting characteristic for high efficiency and other applications of heat. The average power density is 100 MWth/cm^3, which is about 10 times higher than typical HTR, but lower than that of sodium cooled fast reactor.



Reactor Parameters	Reference Value
Reactor power	600 MWth
Net plant efficiency (direct cycle helium)	48%
Coolant inlet/outlet temperature and pressure	490°C/850°C at 90 bar
Average power density	100 MWth/m3
Reference fuel compound	UPuC/SiC (70/30%) with about 20% Pu content
Volume fraction, Fuel/Gas/SiC	50/40/10%
Conversion ratio	Self-sufficient
Burnup, Damage	5% FIMA; 60 dpa

4. Present project ALLEGRO:

ALLEGRO is an experimental reactor that has been developed in the framework of the V4G4 consortium.

ALLEGRO has three decay heat removal systems, two main primary loops with an additional loop to test high temperature components.

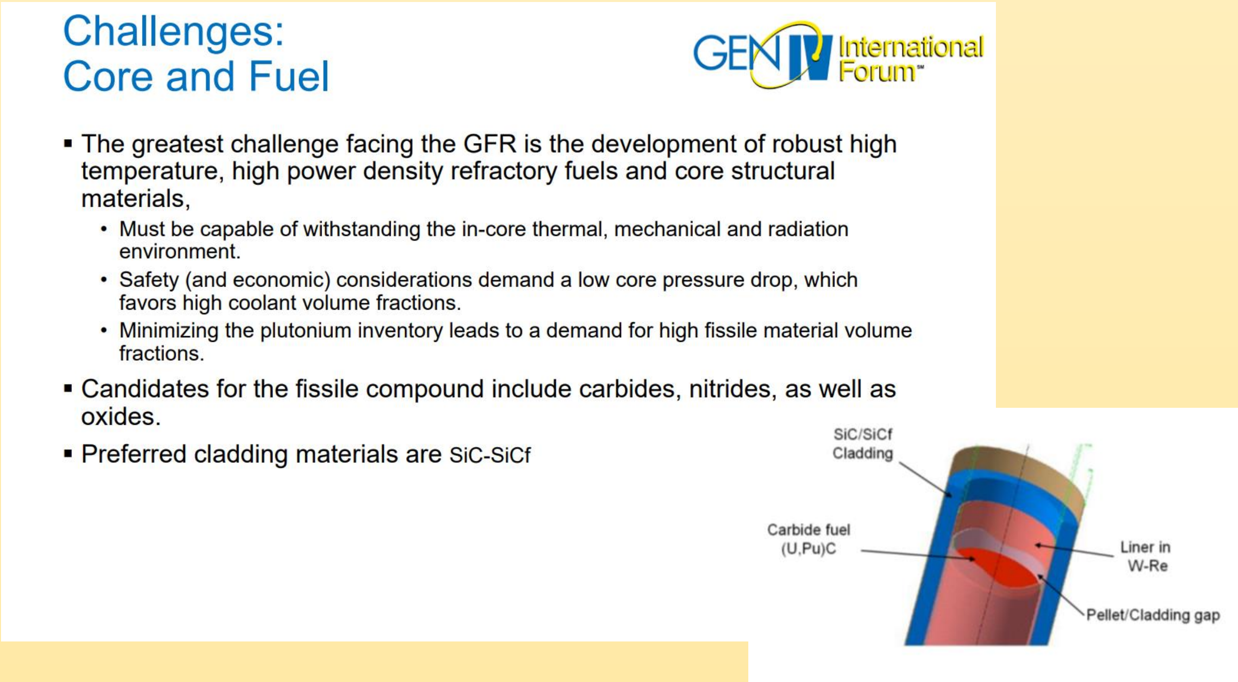
The objective of ALLEGRO is to demonstrate the key GFR technologies.

Objectives of ALLEGRO

- Demonstration of key GFR technologies:
  - Core behavior and control.
  - Development of ceramic fuels
  - Helium circuits and components
  - Decay heat removal
- Fast neutron irradiation capacity
- Potential for coupling with high temperature components or direct use of heat
- Development of safety standards for GFRs

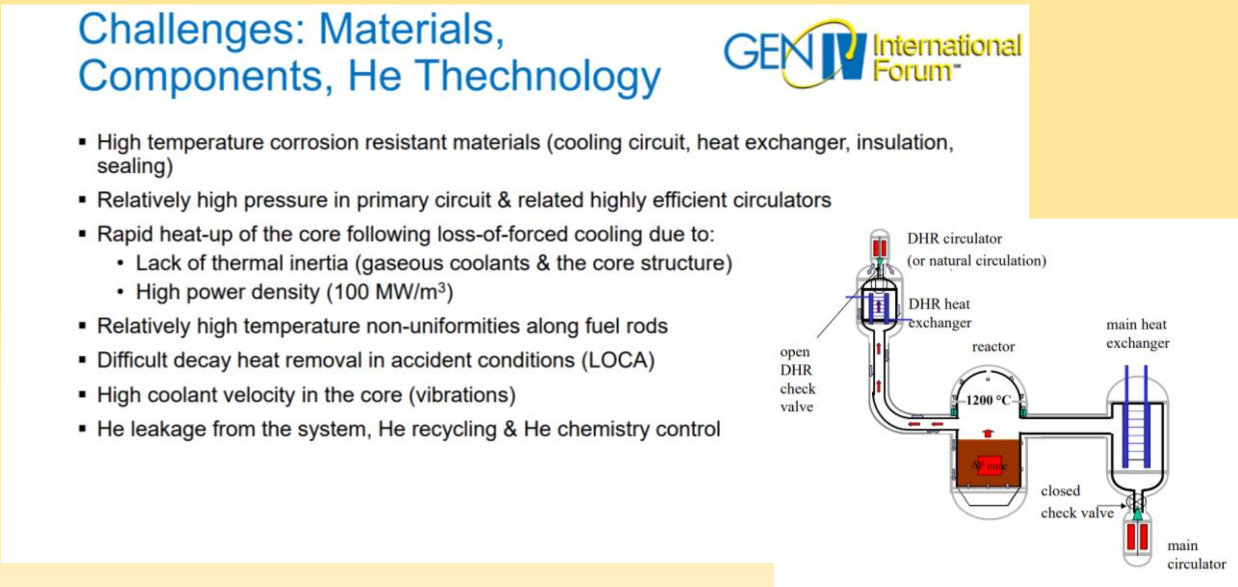
5. Challenges and R&D for the fuel material:

The greatest challenge is the development of a robust high temperature and power density refractory fuels and core structural materials. Some R&D is under way such as the design of carbide fuel with SiC cladding.



6. Challenges and R&D for the decay heat removal system

Challenges of materials, components and He technology must be addressed. Difficulties related to the decay heat removal in LOCA are also concern. Some R&D for the challenges are under way. For example, the decay heat removal system design that can change flow path when forced convection change to natural convection in accidental condition.



4-1-7. The ALLEGRO Experimental Gas-Cooled Fast Reactor Project

Summary / Objectives:

The webinar presents the main design features of the ALLEGRO nuclear reactor demonstrator as developed in the frame of the European V4G4 Consortium "V4G4 Centre of Excellence" associating nuclear research organizations from the Czech Republic, Hungary, Poland, Slovakia and France. The presentation provides an overview of the existing concepts of ALLEGRO, goals of the development, specific design solutions, and the safety approach and safety characteristics of ALLEGRO, touching the most important aspects of the demonstrator. Latest developments associated with both the use of UOX fuel and the new safety features are briefly presented as well. The remaining research challenges are summarized in the light of the present technology understanding to highlight the present status of knowledge and further steps to be pursued.

Meet the Presenter:

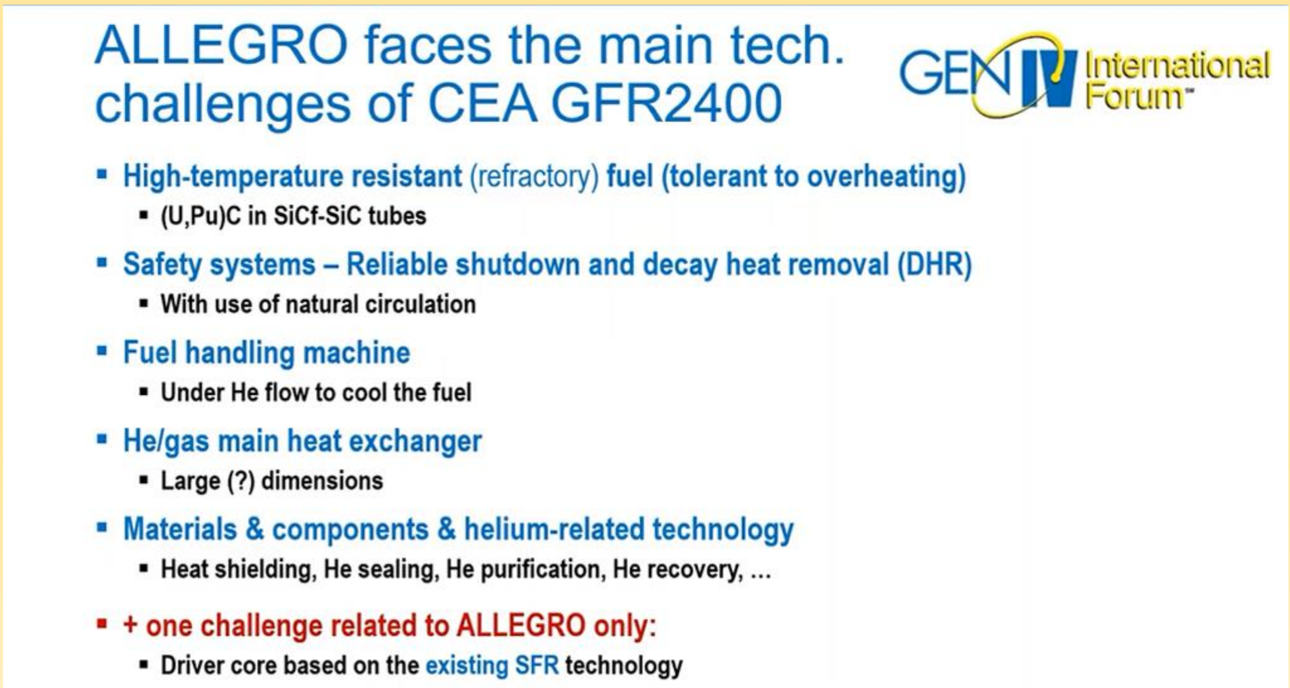
Dr. Ladislav Bělovský works at the ÚJV Řež, a. s., Husinec-Řež close to Prague, Czech Republic as a senior engineer and has over 30 years of experience in nuclear energy research. At ÚJV Řež, Dr. Bělovský participates in the development of the helium-cooled demonstration Fast Reactor ALLEGRO in the frame of the international association "V4G4 Centre of Excellence" in the following areas: 1) Design & Safety of the reactor, 2) Related R&D focused on safety, helium technology and material research. His background in the Czech republic and France in the period from 1988 to 2011 is mainly characterized by activities in the development & application of computer codes for modelling of LWR fuel behavior in design basis & severe accident conditions.

1. A first ever GFR demonstrator ALLEGRO

The purpose of a first ever GFR demonstrator ALLEGRO is verification and validation of the fuel, proving that it works safely and getting the experience of gas cooled fast reactor.

2. The main technological challenges of ALLEGRO:

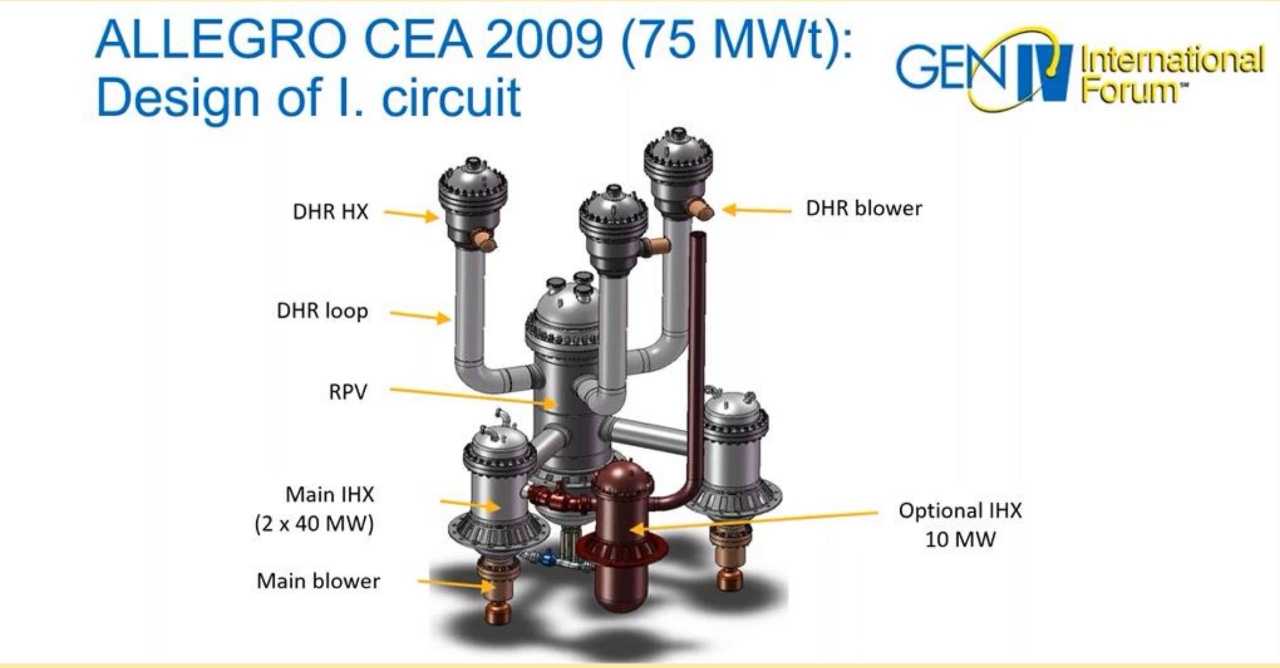
ALLEGRO will touch the challenges concerning the high temperature resistant, safety, fuel handling and so on.



3. Pre-conceptual design of ALLEGRO:

Characteristic of Pre-conceptual design of ALLEGRO are:

- Two main circuits and loops, which would be a safer solution
- Three decay heat removal heat exchanger using the Chimney effect
- Optional gas heat exchanger



4. Advantages and disadvantages of the latest version of ALLEGRO:

The latest version of ALLEGRO has advantages such as core cooling without any active system (except some cases), no more LOFA transients, etc. The disadvantages are complex management for start-up and shutdown, etc.

ALLEGRO CEA 2010:
Innovative option 3

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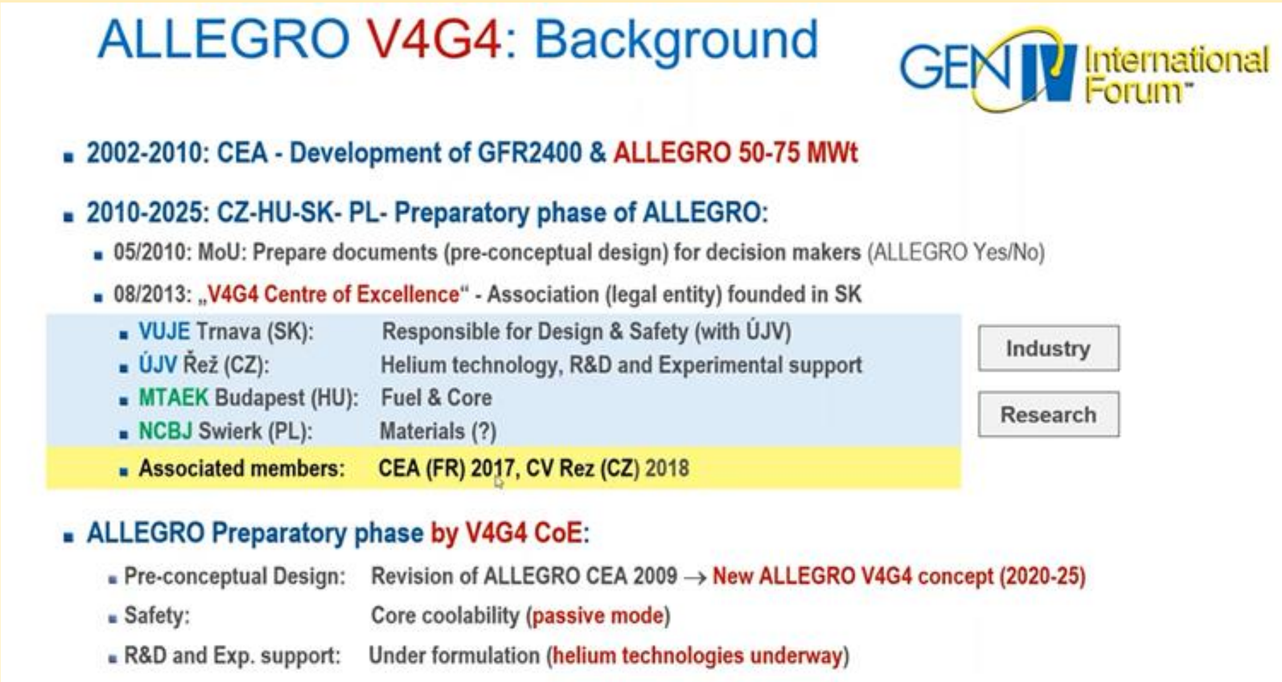
- **ADVANTAGES** (MOX ALLEGRO 530 °C):
  1) Increase of inertia: Core cooling (few hours) without any active system except the SCRAM actuation and the depressurisation of the secondary circuit (could be passive, and even without depressurization the "grace delay" would be significantly longer than few minutes).
  2) No more LOFA transients: This initiating event is no more possible because the primary blowers are driven by the secondary circuits turbomachinery.
  3) Limitation of water ingress risk: Because of gas in the II. circuit
- **DISADVANTIGES:**
  1) Operation: Complex management of the single shaft for start-up and shutdown
  2) Technology: Very complicated to make it feasible (rotating seal in GV)
  3) Once the TM stops in passive operation it cannot restart

5. ALLEGRO V4G4 Centre of Excellence:

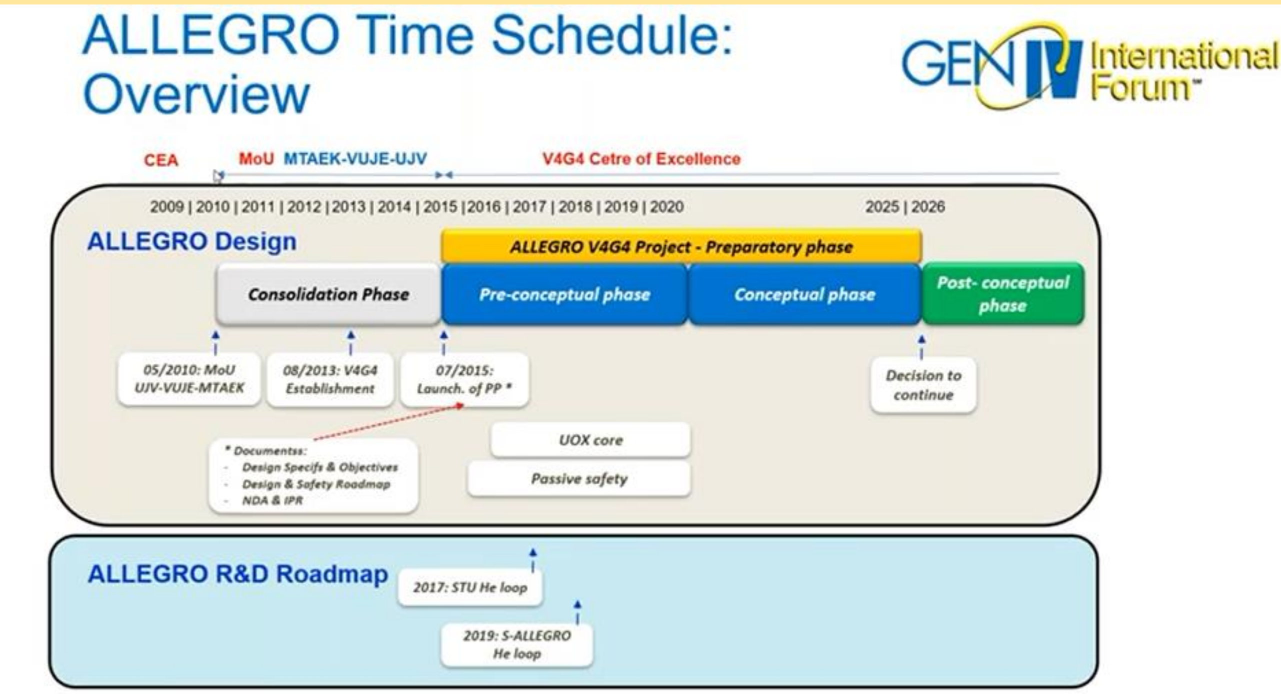
V4G4 Centre of Excellence is an association system for ALLEGRO preparatory phase between SK, CZ, HU, PL and FR. Each of them is in charge of an assigned development topic.



6. Time schedule overview:

ALLEGRO project is planed to proceed with the time schedule below:

- 2020 : Providing pre-conceptual design
- 2025 : Providing conceptual design
- 2026 - : Decision to continue and post-conceptual phase



4-2-1. Very High Temperature Reactors (VHTR)

Summary / Objectives:

Among the six Generation IV concepts eventually selected for international cooperative development, the Very High Temperature Reactor (VHTR) was seen as an early favorite among many of the members. Indeed, among the seven original members of the VHTR System Arrangement (SA), three had already operated or tested high temperature gas-cooled reactors. The accession of the People's Republic of China to the VHTR SA in 2008 brought that number to five. This presentation will describe how the continued cooperative development of the VHTR concept as a Generation IV system will deliver on nuclear energy's promises of sustainable, economic, safe, reliable and proliferation resistant power and energy supply.

Meet the Presenter:

Carl Sink has been working for the U.S. Department of Energy (DOE) for 24 years in various roles. Currently a Program Manager for Advanced Reactor Deployment within the Office of Nuclear Energy, he is responsible for coordinating cooperative research, development and demonstration projects conducted by DOE national laboratories and U.S. nuclear industry partners. Since 2004 he has been closely associated with the Next Generation Nuclear Plant Project, the DOE initiative to develop and demonstrate a high temperature gas-cooled reactor (HTGR). From 2006 through 2009 he was the program manager for the Nuclear Hydrogen Initiative, coordinating DOE efforts to develop high temperature water-splitting technologies to take advantage of HTGR outlet temperatures. Within GIF, Mr. Sink has served on the VHTR System Steering Committee since 2008, and currently chairs that group. He holds a Masters Degree in Engineering Management from the Catholic University of America, and is a graduate of the United States Naval Academy. Before joining the DOE, Mr. Sink spent nine years as a qualified Nuclear Engineering Officer in the United States Navy, with reactor operations assignments in a nuclear powered cruiser and a nuclear powered aircraft carrier.

1. Why HTGRs ?:

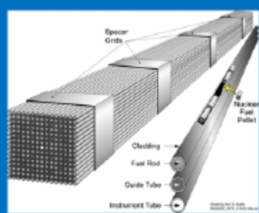
HTGR is one of the six generation IV concepts, and it has good inherent safety characteristics, diverse industrial applications in addition to electricity, proliferation resistant, and high burnup fuel cycle with growth potential for advanced fuels and cycles.

- **Inherent safety characteristics**
  - Ceramic fuel particles – won't melt
  - Graphite core – stable moderator and thermal buffer
  - Helium coolant – inert gas does not interact with fuel, graphite or structural metals
- **Diverse industrial applications in addition to electricity**
  - High efficiency power conversion capability: modern Rankine cycle (Eff ~40%) to advanced closed cycle Brayton (efficiency up to ~47%)
  - High temperature process steam and process heat capability offer cogeneration opportunities now; very high temperatures in future
- **Proliferation resistant, high burnup fuel cycle with growth potential for advanced fuels and cycles (e.g. Plutonium, Thorium), including deep burn cycles with LWR spent fuel**

2. HTGR / LWR Comparison:

Briefly to compare for those of you who are familiar with Light Water Reactor (LWR) how HTGR is significantly different:

Item	HTGR	LWR
Moderator	Graphite	Water
Coolant	Helium	Water
Avg coolant exit temp.	700-950°C	310°C
Structural material	Graphite	Steel
Fuel clad	SiC & PyC	Zircaloy
Fuel	UO_2,UCO	UO_2
Fuel damage temperature	1600-1800C (design dependent)	1260°C (due to Zircaloy clad properties)
Power density, W/cm³	4 to 6.5	58 - 105
Linear heat rate, kW/ft	1.6	19
Neutron migration length	57 cm	6 cm

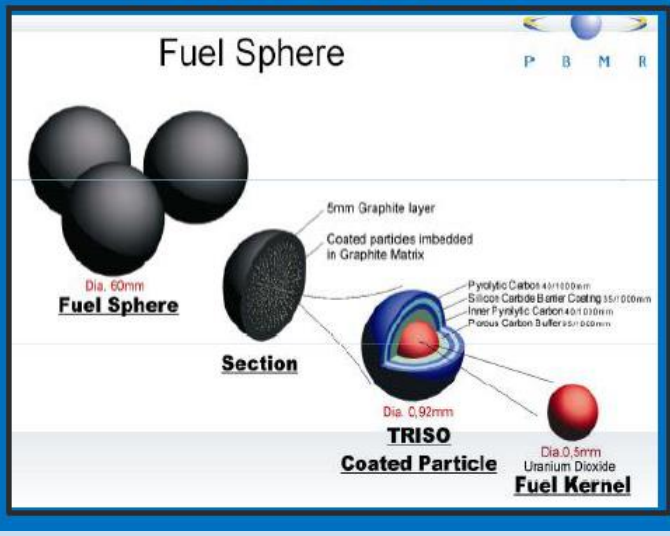


TRISO Coated-particle Fuel:

TRISO coated-particle fuel as the basic element is used for both prismatic and pebble bed type HTGRs. In the prismatic type HTGR, TRISO coated fuel particles are formed into fuel rods and inserted into graphite fuel elements, and in the pebble bed type HTGR, TRISO coated fuel particles are formed into fuel spheres.

Prismatic Fuel

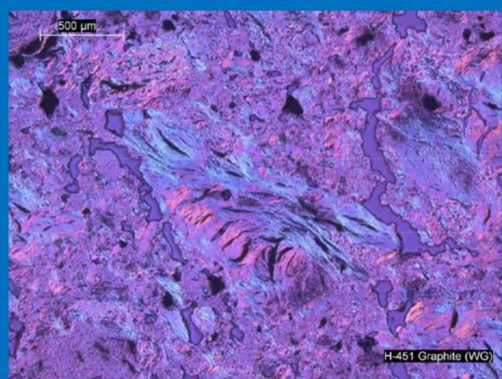
Pebble Bed Fuel



Role of Graphite in HTGRs:

Graphite plays a key role in the core of HTGR as shown in the figure. The other roles are as follows: in prismatic cores, graphite fuel element blocks retain the nuclear fuel compacts, and in a pebble bed reactor, a graphite reflector structure retains the fuel pebbles; the graphite reflector structure contains vertical penetrations for reactivity control; reactivity control channels are also contained in prismatic graphite fuel elements.

- Neutron moderator (carbon & graphite)
  - Thermalize fast neutrons to sufficiently low energies that they can efficiently fission U-235
- Neutron reflector – returns neutrons to the active core
- Graphite (nuclear grade) has a low neutron capture cross section
- High temperature tolerant material



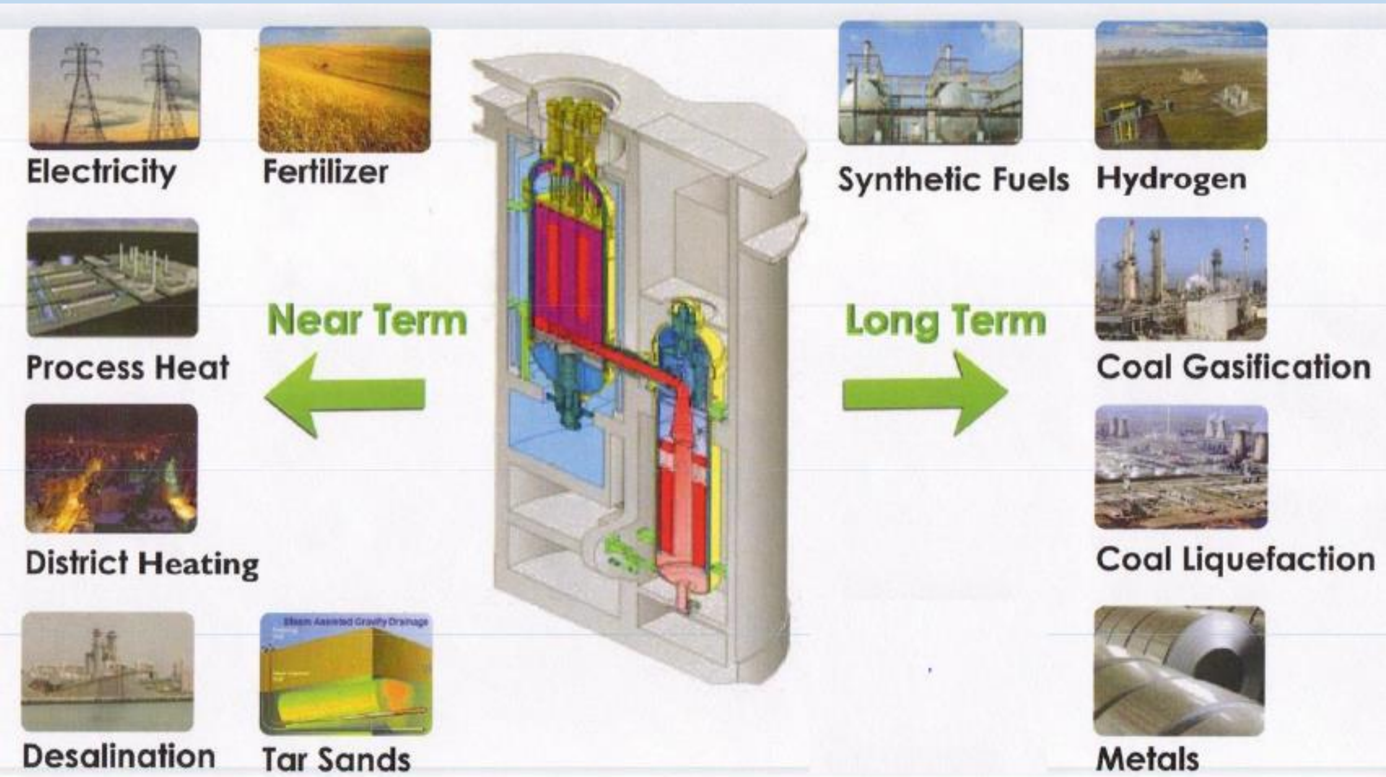
Important HTGR Safety Paradigm Shifts:

HTGR has some safety paradigm shifts from LWR, and it's just a different way of thinking about reactor safety and this has been an issue which has caused us to have to rethink how we regulate HTGR and how we think about accident scenarios for HTGR.

- The fuel, helium coolant, and graphite moderator are chemically compatible under all conditions
- The fuel has very large temperature margins in normal operation and during accident conditions
- Safety is not dependent on the presence of the helium coolant
- Response times of the reactor are very long (days as opposed to seconds or minutes)
- Loss of forced cooling tests have demonstrated the potential for walk-away safety
- There is no inherent mechanism for runaway reactivity excursions or power excursions
- The HTGR has multiple, nested, and independent radionuclide barriers
- An LWR-type containment is neither advantageous nor necessarily conservative.

HTGRS for Production of a Wide Variety of Energy and Commercial Products:

HTGR can supply a wide range of heat from law temperature to high temperature, and the various applications such as hydrogen production are proposed to be used in commercial form.



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4-2-2. Design, Safety Features and Progress of the HTR-PM

Summary / Objectives:

The high-temperature gas-cooled reactor pebble-bed module (HTR-PM) is aimed to extend nuclear energy application beyond the grid, including cogeneration, high-temperature heat utilization, and hydrogen production. The first concrete of HTR-PM demonstration power plant, which has been approved as part of the National Science and Technology Major Projects, was poured five years ago, in Rongcheng, Shandong Province, China. The thermal power of a single HTR-PM reactor module is 250 MWth, the helium temperatures at the reactor core inlet/outlet are 250/750 ° C, and a steam of 13.25 MPa/567 ° C is produced at the steam generator outlet. Two HTR-PM reactor modules are connected to a steam turbine to form a 210 MWe nuclear power plant. The progress of HTR-PM project in China has drawn considerable attention worldwide. In this webinar, the design basis, design principles, general design features and safety characteristics of HTR-PM will be presented. Main engineering verification experiments of components and systems for the HTR-PM, such as helium blower, steam generator, will be introduced. Progress of the HTR-PM demonstration power plant, including civil engineering, first-of-a-kind equipment manufacturing, licensing, installation of the main equipment, will be described. In addition, the irradiation test results of pebble fuel samples and the status of commercial fuel production plant will be explained.

Meet the Presenter:

Dr. Yujie Dong is a Professor in Nuclear Engineering at Tsinghua University, Beijing, China, where he earned his PhD degree in Nuclear Reactor Engineering and Safety. From 1997 he worked to develop advanced nuclear reactors at the Institute of Nuclear and New Energy Technology, INET, Tsinghua University. He was Head of the Division of Reactor Thermal Hydraulic Calculation, Head of the Division of Reactor Physics, Thermal hydraulics and system simulation. From 2006 he was responsible for the Division of General Design of High Temperature Gas-cooled Reactor (HTGR). Currently, he is the Deputy Director and Deputy Chief Engineer of INET in charge of HTGR projects. Also, he has been appointed by the National Energy Administration as Deputy Technical Director of the HGTR Nuclear Power Plant Project, which is one of the National Science and Technology Major Projects. He was actively involved in planning the System Arrangement of VHTR as a member of System Steering Committee in the frame of GIF.

Technical Goals of HTR-PM:

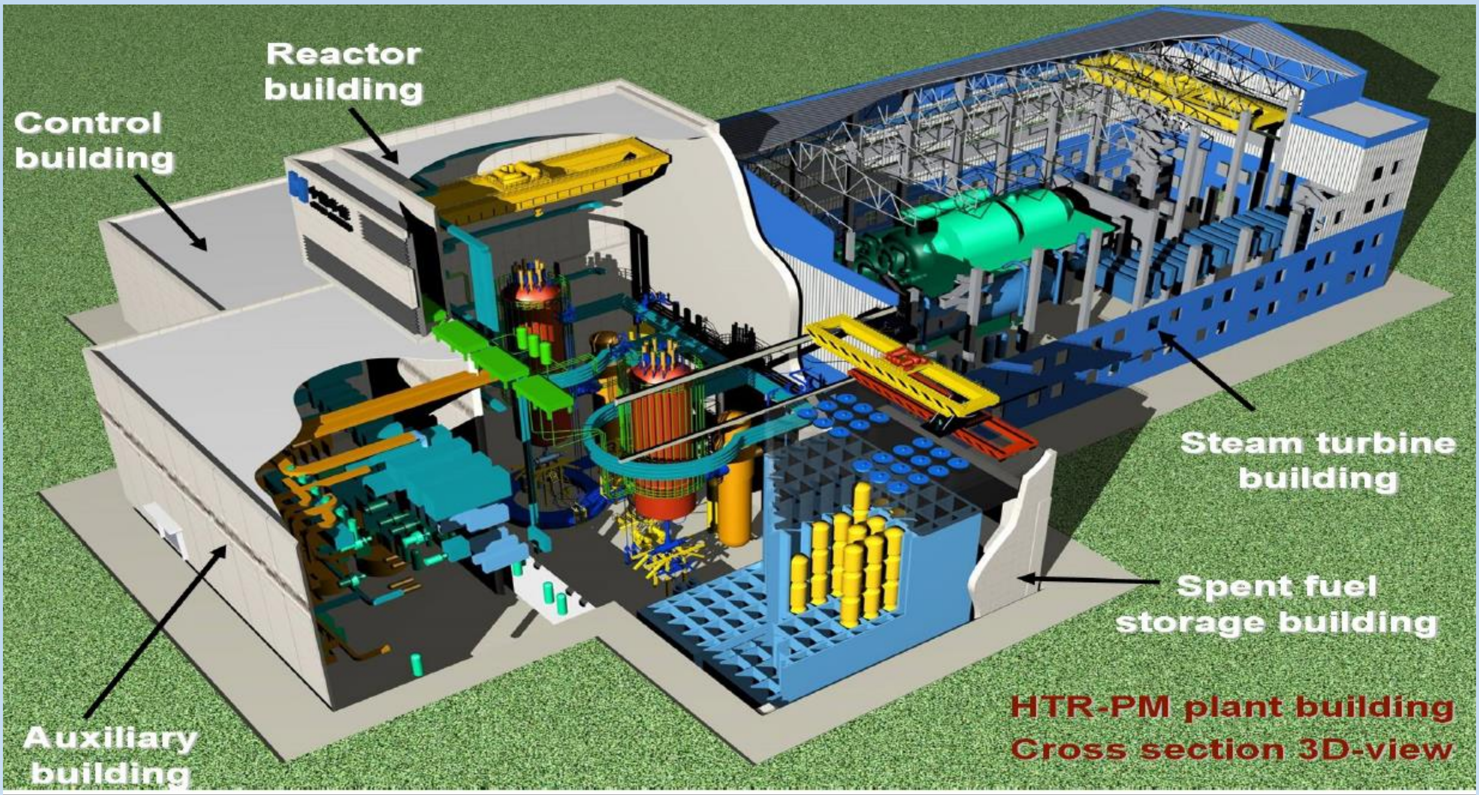
Technical goals of HTR-PM which is a HTGR demonstration power plant comprises four points:

- Keep inherent safety
- Achieve economic competitiveness
- Realize standardized design
- Use proven technology as much as possible
  - HTR-10 proven technology
  - Global experience
  - Steam turbine
  - Global purchase of some key components

HTR-PM: High Temperature Reactor- Pebble-bed-Module

HTR-PM Plant Building Cross Section 3D-view:

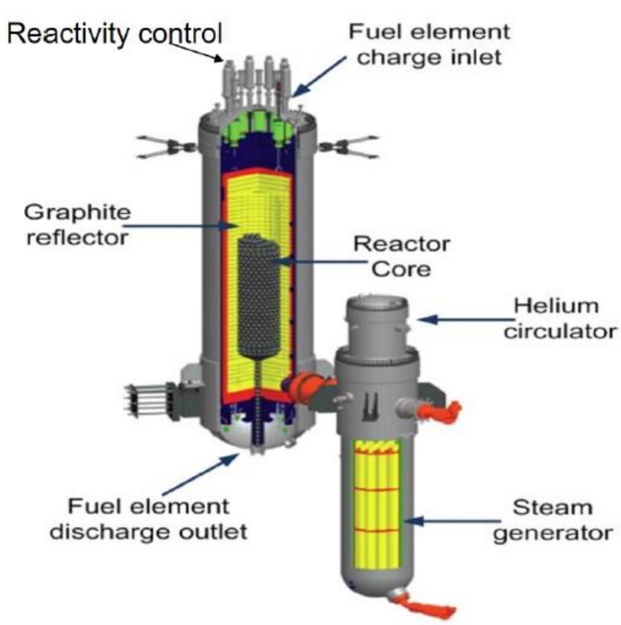
HTR-PM consists of a reactor building, a control building, an auxiliary building, a spent fuel storage building, and steam turbine building. There are two modules in the reactor building and they are connected to one steam turbine generator.



Overview of Design:

The left-side figure illustrates the one module of HTR-PM, and there are one reactor, one steam generator and one helium circulator. The reactor and the steam generator are connected by side-by-side arrangement.

The right-side table shows the main parameter of HTR-PM.



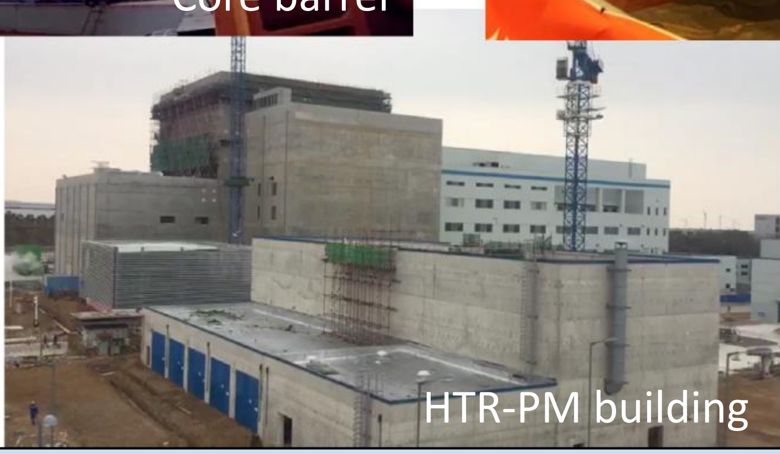
Plant electrical power, MWe	211
Core thermal power, MW	250
Number of NSSS Modules	2
Core diameter, m	3
Core height, m	11
Primary helium pressure, MPa	7
Core outlet temperature, ℃	750
Core inlet temperature, ℃	250
Fuel enrichment, %	8.5
Steam pressure, MPa	13.24
Steam temperature, ℃	567

Situation of Construction:

Most components delivered on schedule, and the HTR-PM construction is smoothly going.


Core barrel

Head of RPV


HTR-PM building

Fuel Fabrication:

The fuel production plant for HTR-PM put into operation successfully.

- Commercial fuel plant, 300,000/a, Baotou, CNNC fuel plant
  - 2013/03/ started construction
  - 2016/03/ finished plant installation and commission
  - 2016/08/ started production
  - 2017/12/ 300,000 fuel pebbles produced

38

HTR-PM600:

The next step of HTR-PM, 6-module commercial 600 MWe unit (HTR-PM600), can be deployed, as supplement to PWRs, such as replacing coal-fired power plant, co-generation of steam and electricity.

- 6 reactor modules connected to one steam turbine, **650 MWe**
  - the same safety features,
  - the same major components,
  - the same parameters,
  - comparing with HTR-PM demonstration plant;
- the same site footprint and the same reactor plant volume comparing with the same size PWRs.
- Plant Owner: China HUANENG Cor.，China Nuclear Engineering Cor.(CNEC) ，China General Nuclear Power Cor.(CGNPC)
- Feasibility study of sites:
  - ***Sanmen,Zhejiang***; *Ruijin, Jiangxi; Xiapu, Fujian; Wan'an Fujian; Bai'an, Guangdong*

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4-2-3. GIF VHTR Hydrogen Production Project Management Board

Summary / Objectives:

The objective of the GIF VHTR Hydrogen Production Project Management Board is to provide a collaborative environment among the signatories for the development, optimization and demonstration of economical large-scale hydrogen production processes that do not emit greenhouse gases through the use of nuclear energy. The main processes considered by the signatories include Sulphur-Iodine (S-I), High Temperature Steam Electrolysis (HTSE), Copper-Chlorine (Cu-Cl) and Hybrid Sulphur (HyS). The signatories include Canada, EU, France, Japan, Korea and the USA. China has been an observer, waiting to join the group formally, but contributing strongly to the developments. The S-I process has been demonstrated for short term operation by China, Korea and Japan. EU, France and the USA have been very active in HTSE. Canada has been focusing on the Cu-Cl Cycle with plans for demonstration of an integrated lab-scale system in 2021. This webinar will provide an overview of these activities and their relevance to mitigating global warming.

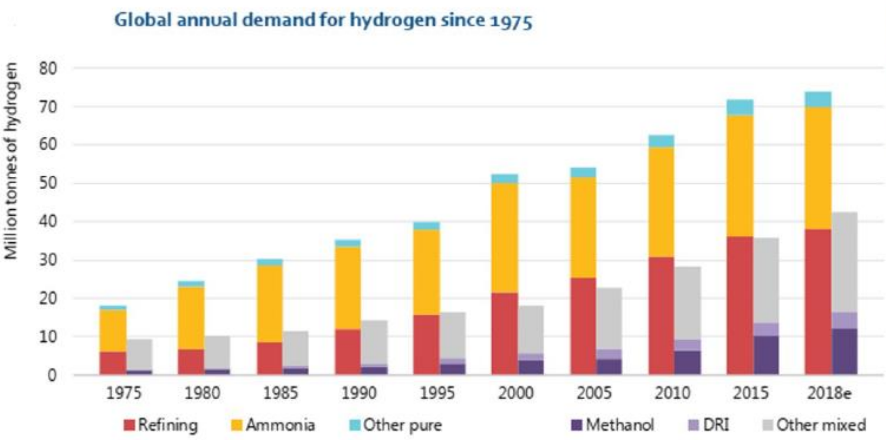
Meet the Presenter:

Dr. Sam Suppiah is currently the manager of the Chemical Engineering Branch and the Facility Authority for Tritium Facility Operations at the Canadian Nuclear Laboratories (CNL), Chalk River, Ontario. He earned his chemical engineering degree and PhD from the University of Birmingham, UK, and worked for a contracting company and British Gas Corporation in the UK before joining AECL (now CNL). He is a Professional Engineer in Ontario, and a certified Project Management Professional (PMP).

He has more than 35 years of expertise in the areas of Heavy Water and Tritium, Catalysis, Electrolysis Technologies, Fuel Cell Technologies, Nuclear and non-Nuclear Battery Technologies, Hydrogen Production from High and Medium Temperature Thermochemical Processes, Steam Electrolysis and Energy Storage. His current focus at CNL in the area of hydrogen production is in the development of the hybrid copper-chlorine cycle. This development is approaching lab-scale continuous operation demonstration in 2021. Dr. Suppiah has been leading collaborations in many of the above areas with industry, institutes and universities. He is the Canadian delegate for and the current Chair of the GEN IV VHTR Hydrogen Production Project Management Board. He is also a board member of the Canadian Hydrogen and Fuel Cell Association (CHFCA). He has been a regular presenter at IAEA's technical meetings and other national and international meetings on hydrogen production.

Current & Future Demand & Use of Hydrogen:

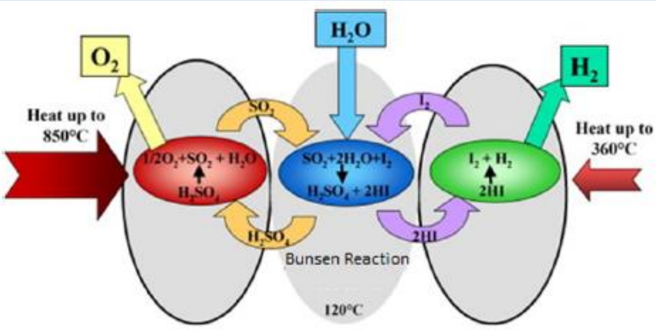
The demand of hydrogen over the years has been growing with the expanding population of the world because it is a raw material used to produce fertilizers and various other materials. It's only expected to grow faster with increasing living standards, the demand for hydrogen is forecast to grow very rapidly. In the future, to minimize the greenhouse gas emissions from heavy duty vehicles, a shift will have to be made to hydrogen fuel all.



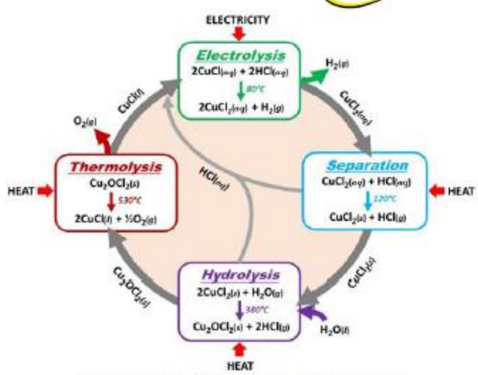
Transportation:
- Heavy vehicles
- Trains
- Ships
- Aviation

Hydrogen from GEN IV Nuclear Technologies:

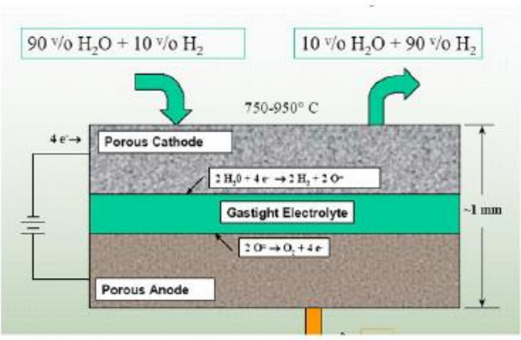
These four hydrogen production processes have been receiving the most attention over the last decade or two, and the hydrogen production PMB member countries (Canada, EU, France, Japan, Korea, USA, China (observer)) are mainly focused on these processes.



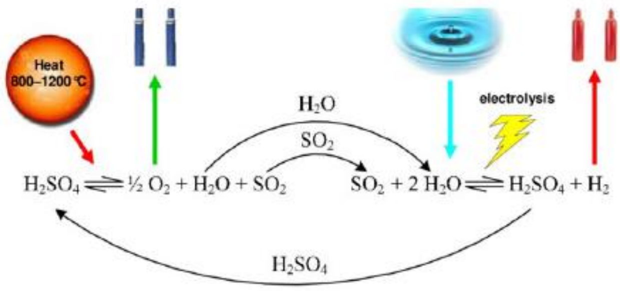
Sulphur-Iodine Process



Copper-Chlorine Process



High Temperature Steam Electrolysis



Hybrid-Sulfur Process

H2 Production PMB Goals and Objectives 1:

The development of the Sulfur-Iodine cycle has been carried out by JAEA of Japan, INET of China and KAERI of Korea. The operation of the integrated Sulfur-Iodine process has been demonstrated. However, materials related issues require resolution for industrial demonstration.

Development of the Sulfur-Iodine Cycle:

- Process evaluation including flowsheet optimization, selection of construction materials with suitable corrosion and mechanical properties and selection of catalysts for SO_3 and HI decomposition.
- Bench-scale experiments to optimize process conditions.
- Pilot-scale plant construction and performance testing to confirm scaling parameters and materials performance.
- Long-term testing for validating catalyst performance and suitability of construction materials.

H2 Production PMB Goals and Objectives 2:

The development of the high temperature steam electrolysis has been carried out by INET of China, KAERI of Korea, CEA of France, INL of USA and EU. The high temperature steam electrolysis technology has reached mature state. The degradation of cell components requires continuing advances.

Development of High Temperature Steam Electrolysis:

- Process evaluation including flow sheet optimization and development of methods for separation of hydrogen from the residual steam.
- Development of advanced materials for electrodes, electrolytes and interconnections, particularly for achievement of low cell and stack resistance and for decreased degradation rates.
- Development of advanced cell and stack designs.
- Experimental testing of promising cell configurations and materials at scales ranging from watts to multi-kW, and in pressurized stack experiments.
- Pilot-scale plant (200 kW) construction and demonstration.
- Theoretical and experimental feasibility studies of high-temperature co-electrolysis of steam and CO_2 while integrating different primary energy sources

H2 Production PMB Goals and Objectives 3:

The development of the Copper-Chlorine (Cu-Cl) cycle has been carried out by CNL of Canada. The Cu-Cl cycle development is approaching lab-scale demonstration. The assessment of the other alternative cycles such as Hybrid-Sulfur process and the economic evaluation has been also carried out by the hydrogen production PMB members.

Development of Copper-Chlorine (Cu-Cl) Cycle and Assessment of other alternative cycles and economic evaluation

- Cu-Cl Cycle evaluation including determination of process options, flow-sheet optimization and selection of materials.
- Cu-Cl Cycle component and bench-scale experiments to define and evaluate key parameters such as thermodynamic properties, rate constants, and equipment selection.
- Integrated testing of lab-scale system for 100 L/h hydrogen production.
- Development of HyS process: SO_2 Depolarization Electrolyser (SDE) development, and laboratory-scale tests and optimization.
- *Technical evaluation of potential alternative cycles with reference to S/I and HTSE regarding methodology, feasibility and process efficiency and economics.*
- *Basic R&D as proof of principle for process development.*
- *Economic evaluation for all hydrogen production processes coupled to nuclear reactors.*

H2 Production PMB Goals and Objectives 4:

The hydrogen production and nuclear reactor coupling has been investigated by the hydrogen production PMB members.

Hydrogen Production and Nuclear Reactor Coupling

- System evaluation and optimization of coupling circuits.
- Develop standards on the separation of nuclear reactor and hydrogen production process.
- Develop methodology and requirements for all safety aspects.
- Develop methodology for system integration.

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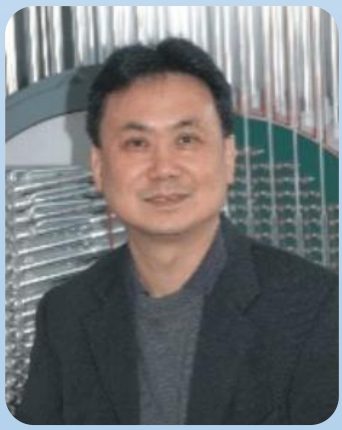
4-2-4. Supercritical Water Cooled Reactors (SCWR)

Summary / Objectives:

Supercritical Water-Cooled Reactors (SCWRs) are a class of high temperature, high pressure water-cooled reactors that operate above the thermodynamic critical point of water (374° C, 22.1 MPa). These concepts combine the design and operation experience gained from hundreds of water-cooled reactors with the experience from hundreds of fossil-fired power plants operated with supercritical water. The main goals of using supercritical water in nuclear reactors are to increase the efficiency of modern nuclear power plants, decrease capital and operational costs, and finally decrease electrical energy costs. This presentation describes SCWR concepts being pursued in the international community and highlights the technical advancements and challenges in the development.

Meet the Presenter:



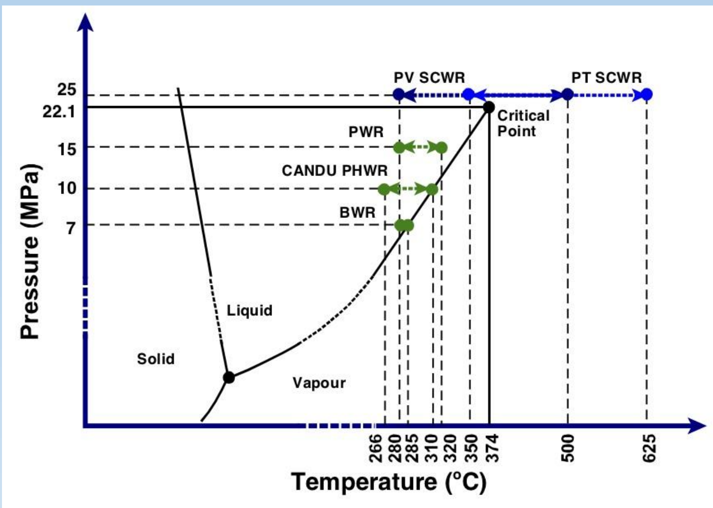
Laurence Leung has been working at Canadian Nuclear Laboratories (formerly Chalk River Laboratories of Atomic Energy of Canada Limited) since 1987 in the field of thermal-hydraulics. He completed his Ph.D. degree at University of Ottawa, Canada, in 1994. Laurence is currently Manager of R&D Facilities Operations and is also responsible for the development of the Canadian Super-Critical Water-cooled Reactor (SCWR) concept. He received 13 awards from AECL (CNL) and external organizations, and delivered short courses on thermal-hydraulics and SCWRs. Laurence is one of Canada's representatives to the GIF SCWR System, and is the Co-Chair of the System Steering Committee and the Thermal-hydraulics and Safety Project Management Board.

1. Why SCWR?

- Merging proven advanced technologies of nuclear and fossil-fuel power plants
- Many utilities operate both nuclear and supercritical fossil plants
- Many years of design and operating experiences

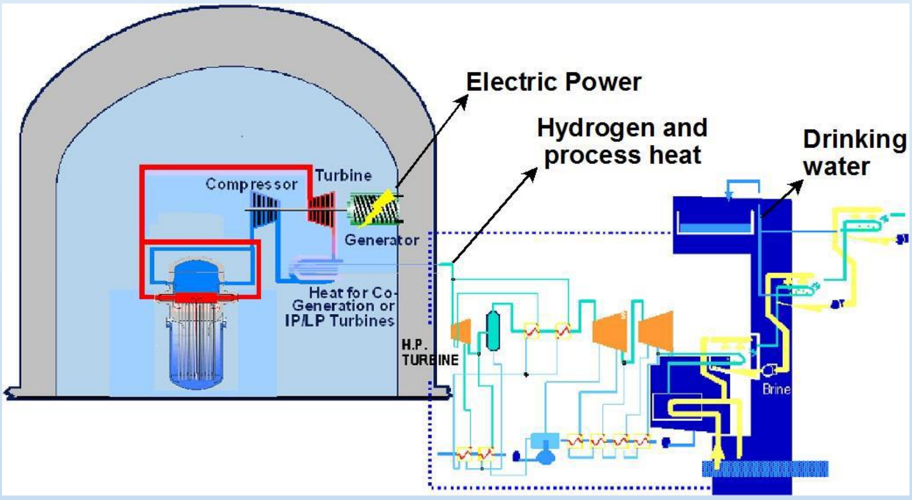
2. SCWR Main Features

- High efficiency with supercritical pressures and temperatures at core outlet
  - Increasing the power output for the same fuel input (specific fuel utilization)
  - Reducing waste heat from turbines and condensers (environmental discharges)
  - Building fewer plants for meeting demand (capital and operating cost savings)
- Simplification of plant components and layout
  - Direct cycle eliminating heat exchangers, steam generators, steam dryers, and moisture separator reheaters
  - Reduction in capital and operational costs
- Design flexibility
  - Thermal or fast spectrum
  - Advanced fuel cycles and fuel design optimization
  - Reduction in electrical energy costs
  - Opportunities for co-generation



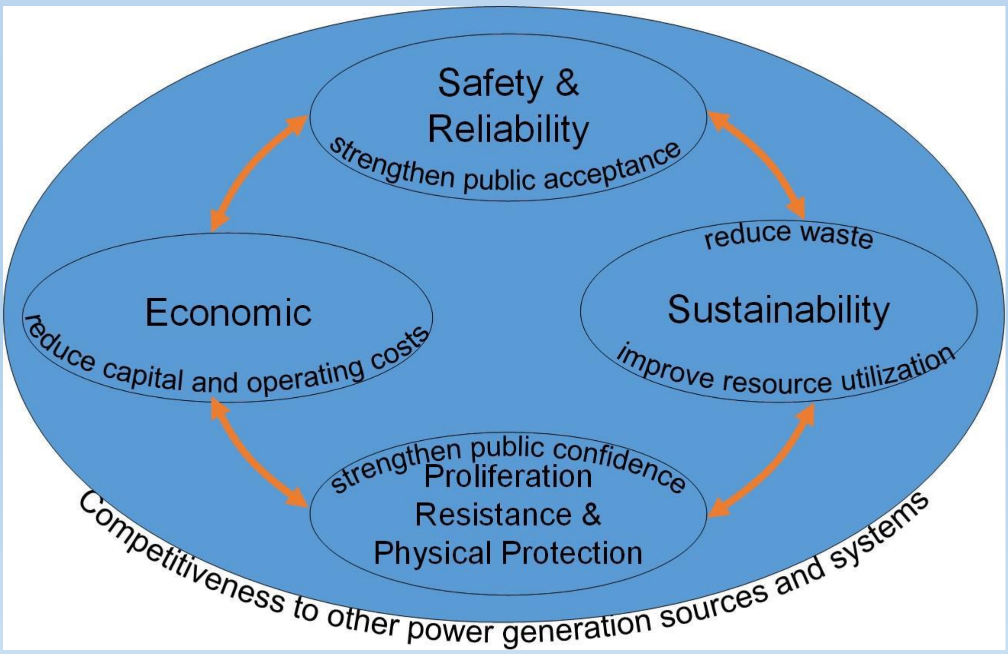
3. SCWR Applications

- Primarily for electric power generation
- Heat can be extracted for co-generation
  - Hydrogen production
  - Oil extraction (Steam-Assisted Gravity Drainage process)
  - Desalination
  - Process heat



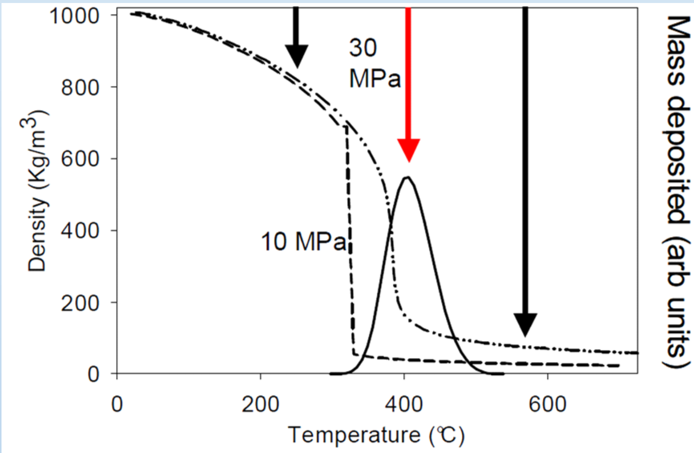
4. GIF Technology Goals

A pressure-tube-type SCWR concept can potentially meet key technology goals of the GIF (i.e., improving economics and sustainability, as well as enhancing safety and proliferation resistance).



5. SCWR Design Challenges: Chemistry

- Changes in chemical properties due to marked change in SCW density through the critical point
- SCWR In-core radiolysis is markedly different from those of conventional water-cooled reactors
  - Extrapolation of the behavior is inappropriate
  - Strong impact on corrosion and stress corrosion cracking
- Identification of an appropriate water chemistry to minimize
  - Corrosion rates
  - Stress corrosion cracking
  - Deposition of deposits on fuel cladding and turbine blades
- Establish a chemistry-control strategy



6. Collaborations

- Leverage resources and expertise to expedite the development
  - Generation-IV International Forum (GIF)
  - International Atomic Energy Agency (IAEA)
  - Bilateral agreements
- Exchange of technical information
  - International Symposium on SCWRs
  - Information Exchange Meetings
  - IAEA Coordinated Research Projects and Technical Meetings

4-2-5. Overview of FHR Technology

Summary / Objectives:

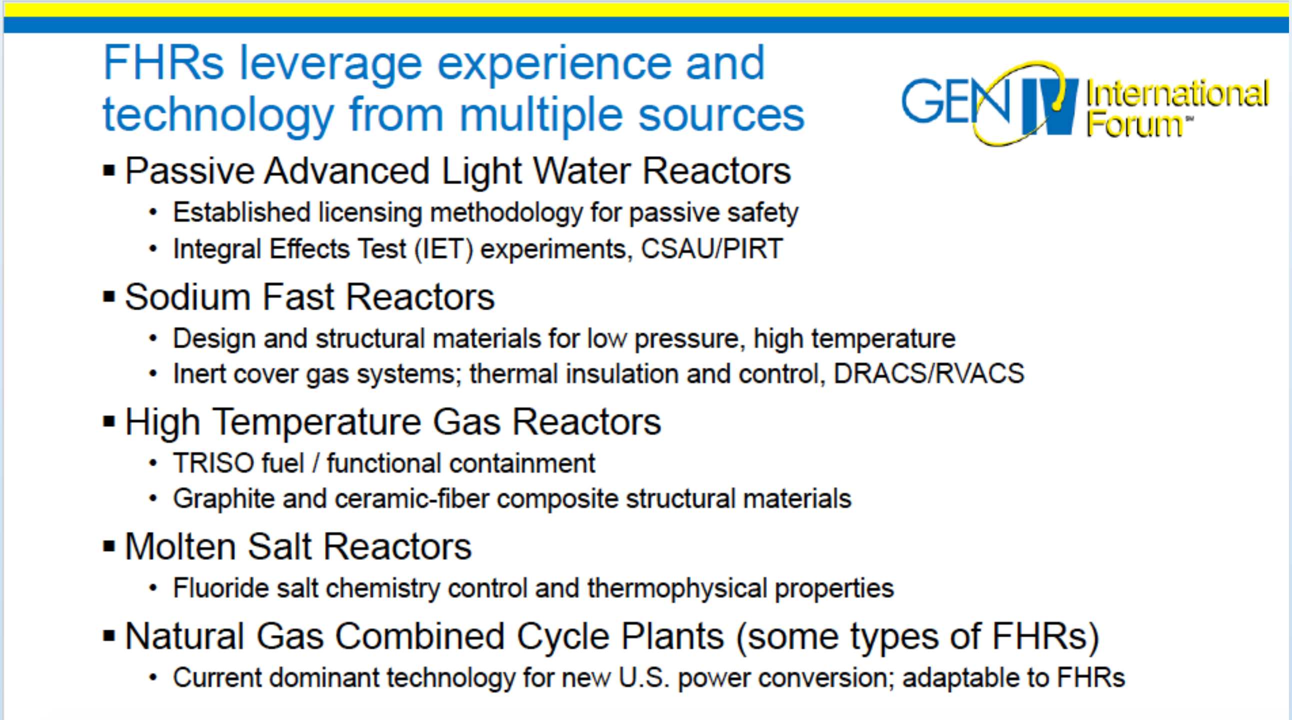
Fluoride Salt Cooled High Temperature Reactors (FHRs) use solid, ceramic fuel with a molten salt coolant, and deliver heat in the temperature range from 600° C to 700° C. This presentation will review key design features of FHRs and recent work to develop the technical basis for safety analysis and licensing.

Meet the Presenter:

Per F. Peterson holds the William and Jean McCallum Floyd Chair in the Department of Nuclear Engineering at the University of California, Berkeley. He performs research related to high-temperature fission energy systems, as well as studying topics related to the safety and security of nuclear materials and waste management. He participated in the development of the Generation IV Roadmap in 2002 as a member of the Evaluation Methodology Group, and cochaired its Proliferation Resistance and Physical Protection Working Group. His research in the 1990's contributed to the development of the passive safety systems used in the GE ESBWR and Westinghouse AP-1000 reactor designs. Currently his research group focuses primarily on heat transfer, fluid mechanics, and regulation and licensing for advanced reactors.

1. FHRs leverage experience and technology from multiple sources

FHR design concept is based on technologies and experiences from multiple fields such as LWR passive safety, SFR, HTGR, MSR, and gas combined cycle.



2. R&D has developed an improved foundation for understanding FHRs

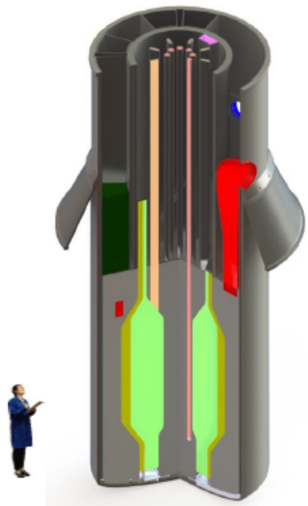
The base technology related to FHR concept has been improved and documented through design studies and various experiments.

3. Nominal Mk1 PB-FHR Design parameters

Main plant parameters, core structure, power output, and mitigation measures for Tritium are shown.

Nominal Mk1 PB-FHR Design Parameters

- Annular pebble bed core with center reflector
  - Core inlet/outlet temperatures 600° C/700° C
  - Control elements in channels in center reflector
  - Shutdown elements cruciform blades insert into pebble bed
- Reactor vessel 3.5-m OD, 12.0-m high
  - Vessel power density 3 x higher than S-PRISM & PBMR
- Power level: 236 MWth, 100 MWe (base load), 242 MWe (peak w/ gas co-fire)
- Power conversion: GE 7FB gas turbine w/ 3-pressure HRSG
- Air heaters: Two 3.5-m OD, 10.0-m high CTAHs, direct heating
- Tritium control and recovery
  - Recovery: Absorption in fuel and blanket pebbles
  - Control: Kanthal coating on air side of CTAHs

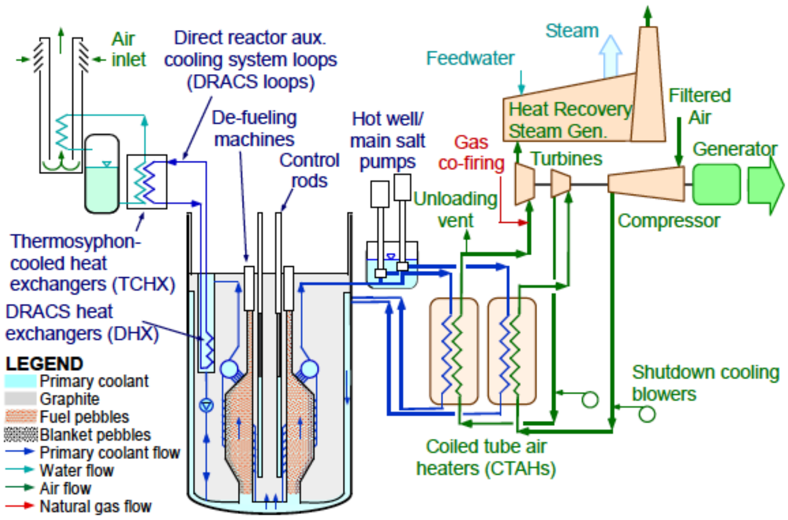


PB-FHR cross section

4. Mk1 PB-FHR flow schematic

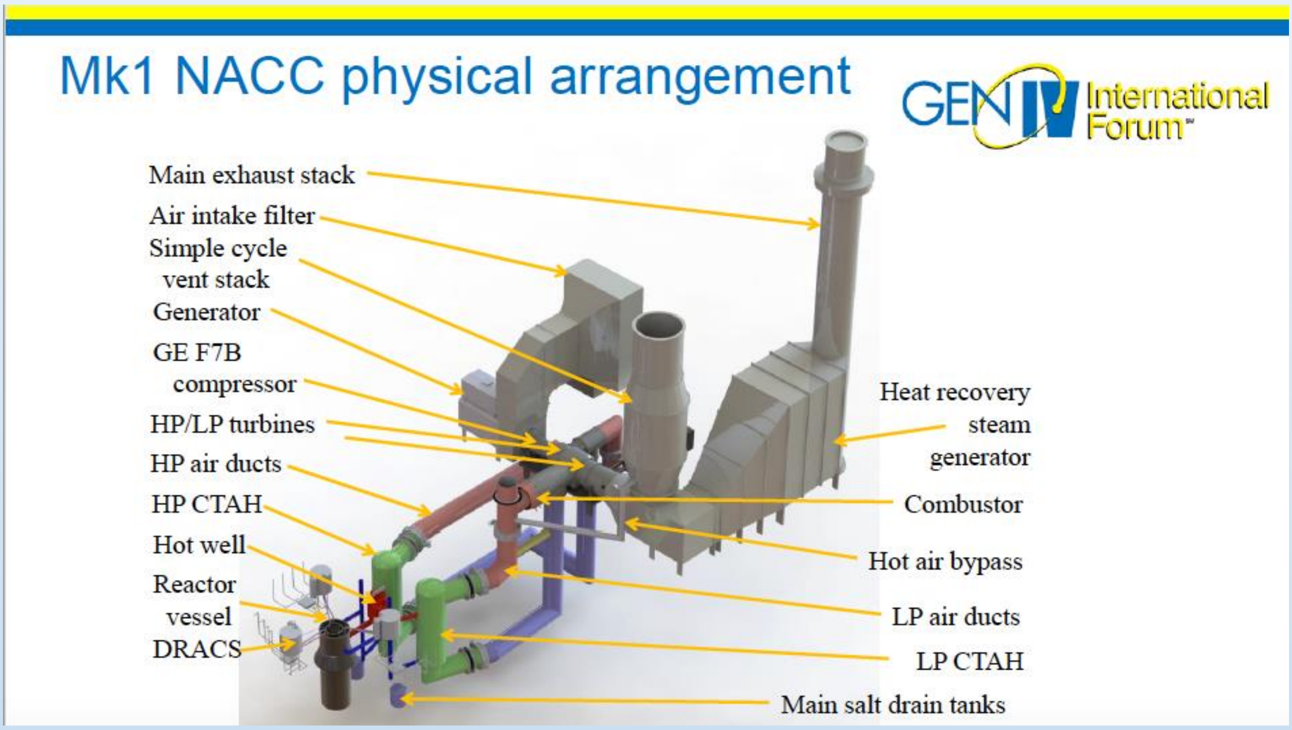
The main heat transport system transfer the core heat to the power conversion system (PCS) through coiled tube air heaters.

Mk1 PB-FHR flow schematic



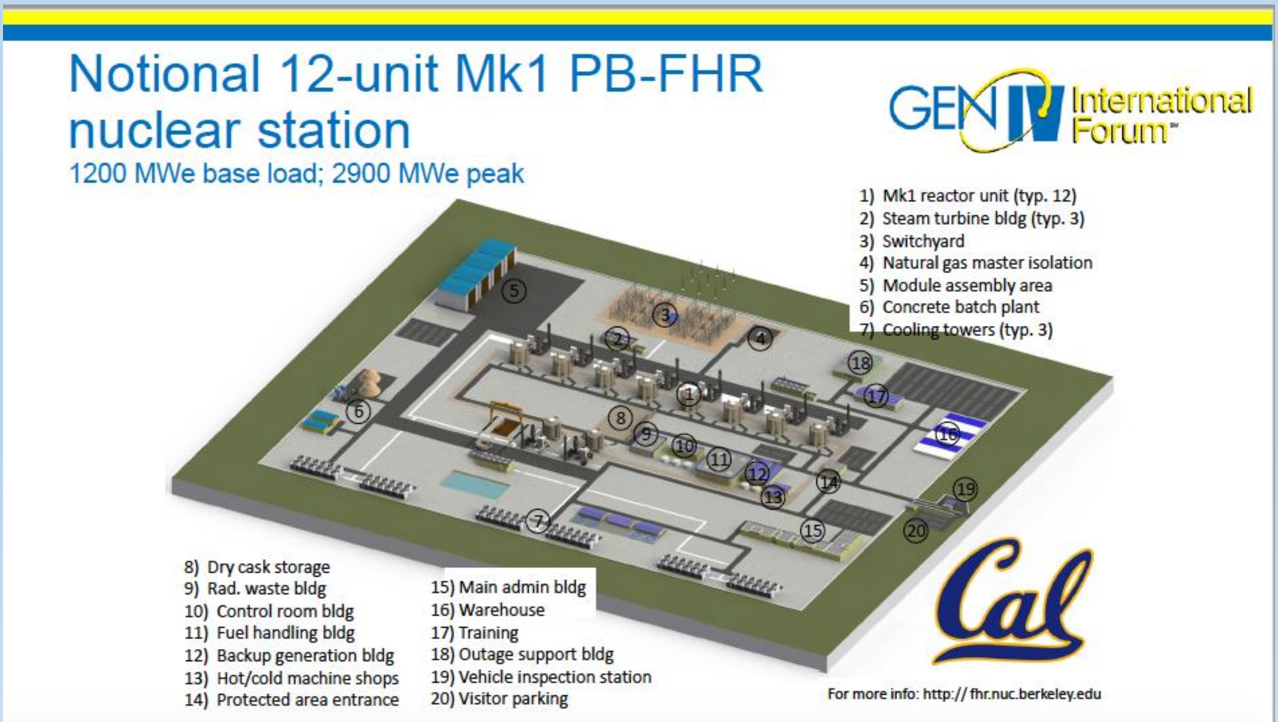
5. Mk1 NACC physical arrangement

Each FHR unit has one PCS (NACC: nuclear air-brayton combined-cycle) .



6. Notional 12-unit Mk1 PB-FHR nuclear station

The total of 12 units can produce 1200 MWe base load and 2900 MWe for peak load with natural-gas co-firing boost function.



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4-2-6. Concept of European Molten Salt Fast Reactor (MSFR)

Summary / Objectives:

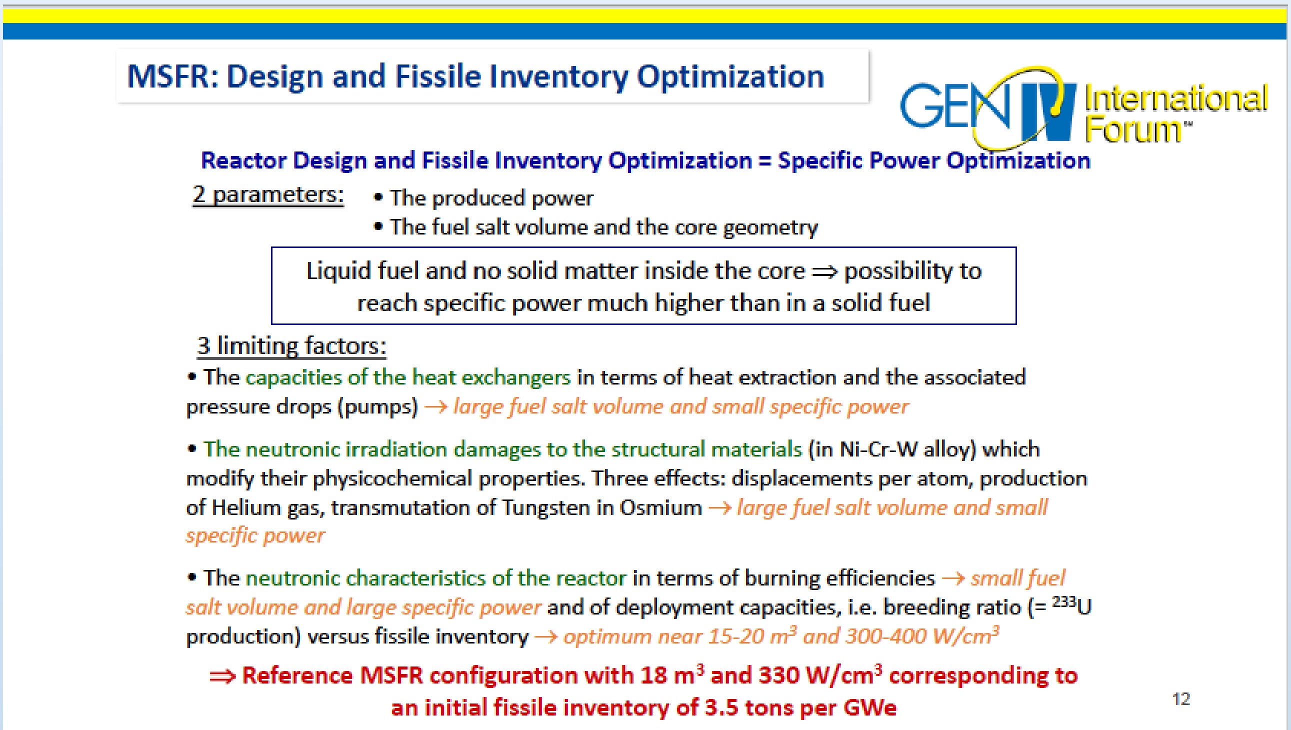
Liquid-fueled reactors exhibit unusual and interesting properties in terms of operation and safety compared to solid-fueled reactors, requesting a revision of some well-known conception and safety rules. In this webinar, such characteristics of the Molten Salt Reactors (MSRs) will be presented, together with the past and current R&D activities. The concepts studied in the frame of the Generation-IV international collaboration will be briefly described, and the presentation will then focus on the concept of Molten Salt Fast Reactor (MSFR), reactor based on a fast spectrum and studied since almost a decade mainly by calculations and determination of basic physical and chemical properties, initially at CNRS in France and now more largely in the European Union. The main design choices and characteristics of this MSFR concept will be explained and discussed including transient simulations, chemistry and material issues, safety analysis, research roadmap and laboratory scale experiments.

Meet the Presenter:

Prof. Elsa Merle is the director of the Master's Program in Reactor Physics and Nuclear Engineering at the PHELMA engineering school of Grenoble Institute of Technology, France. She is also working, as a research staff member, at the Laboratory for Subatomic Physics and Cosmology of Grenoble. Since 2000, she has been actively involved with the French National Center for Scientific Research (CNRS) programs dedicated to the conceptual design of innovative Generation IV reactors. As such, she is contributing to various studies and validations of the concept of Molten Salt Reactors and more specifically since 2008 on the definition and optimization of the concept of Molten Salt Fast Reactor (MSFR). Dr. Merle is in charge of the work-package 1 “Integral safety approach and system integration” of the Euratom project SAMOFAR of Horizon2020, and she represents CNRS at the GIF steering committee on Molten Salt Reactors.

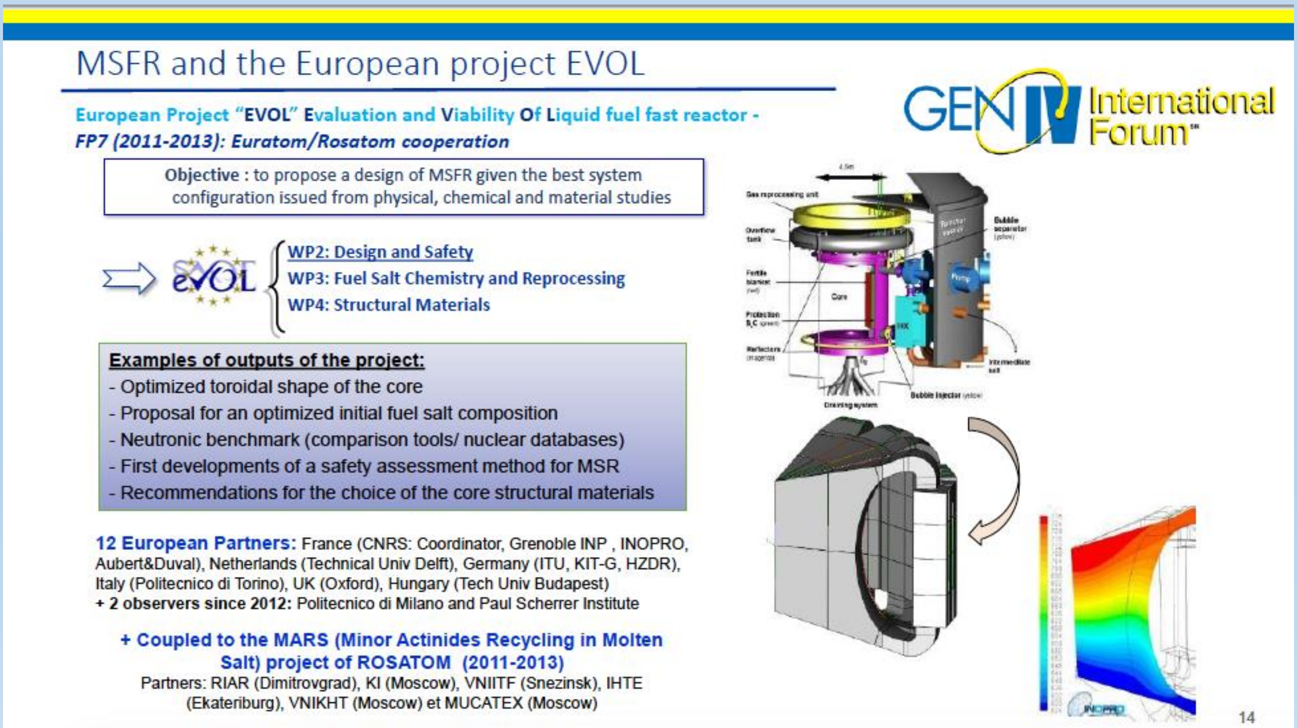
1. MSFR: Design and Fissile Inventory Optimization

The reference design parameters of power, fuel salt volume and core geometry have been decided considering some limiting factors.



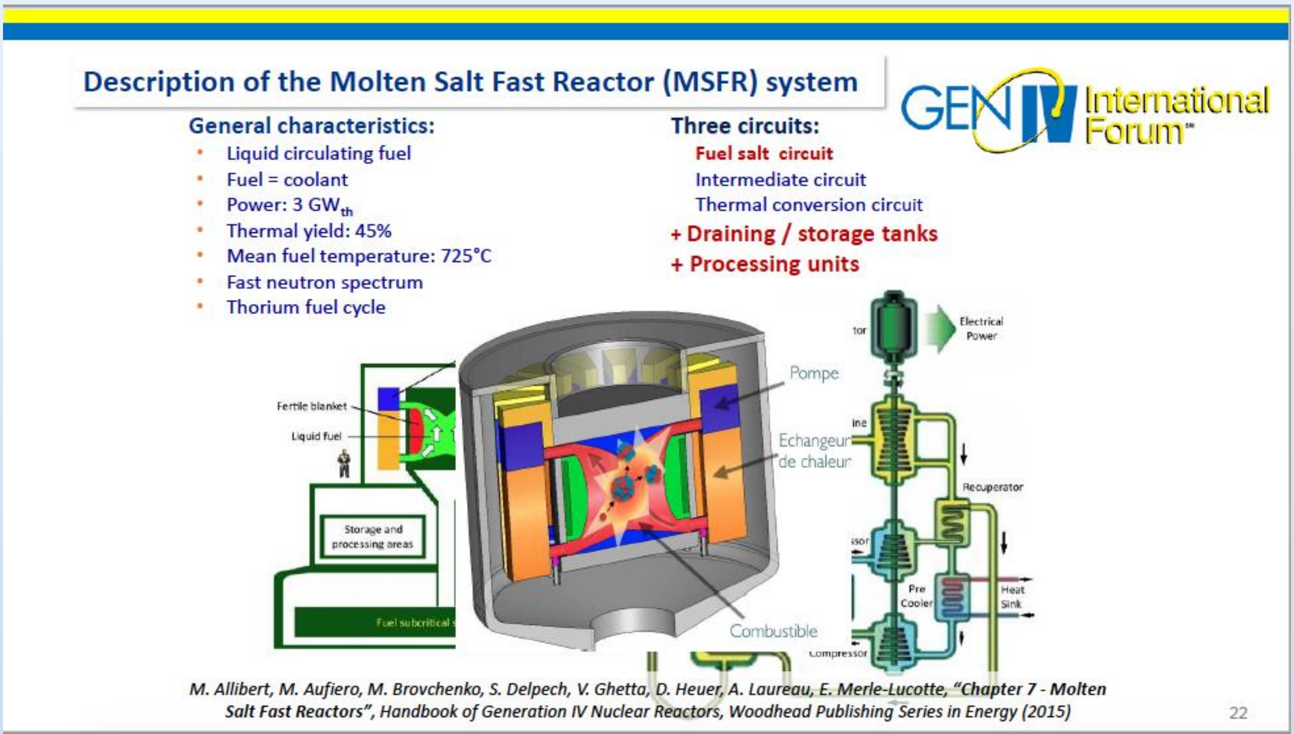
2. MSFR and the European project EVOL

EVOL project has been implemented during 2011-2013, in order to propose best MSFR system based on physical and material studies



3. Description of the Molten Salt Fast Reactor (MSFR) system

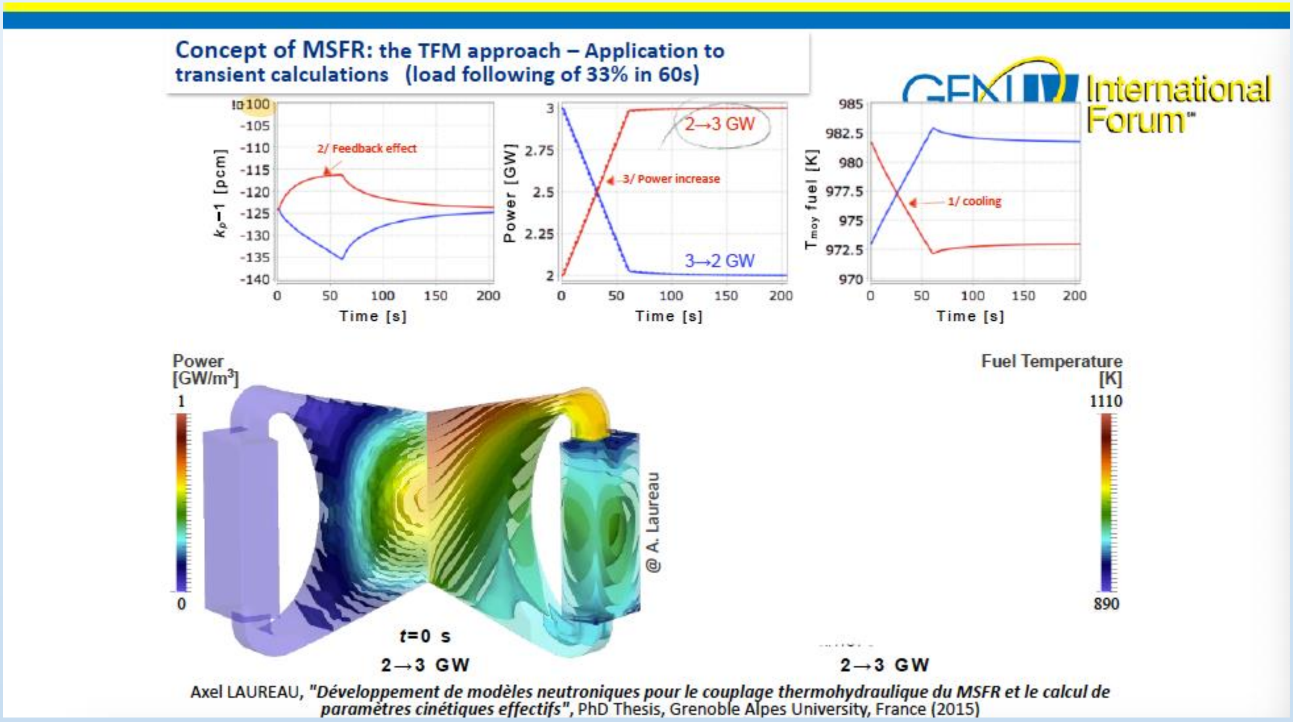
The main plant parameters, the heat transport configuration are shown.



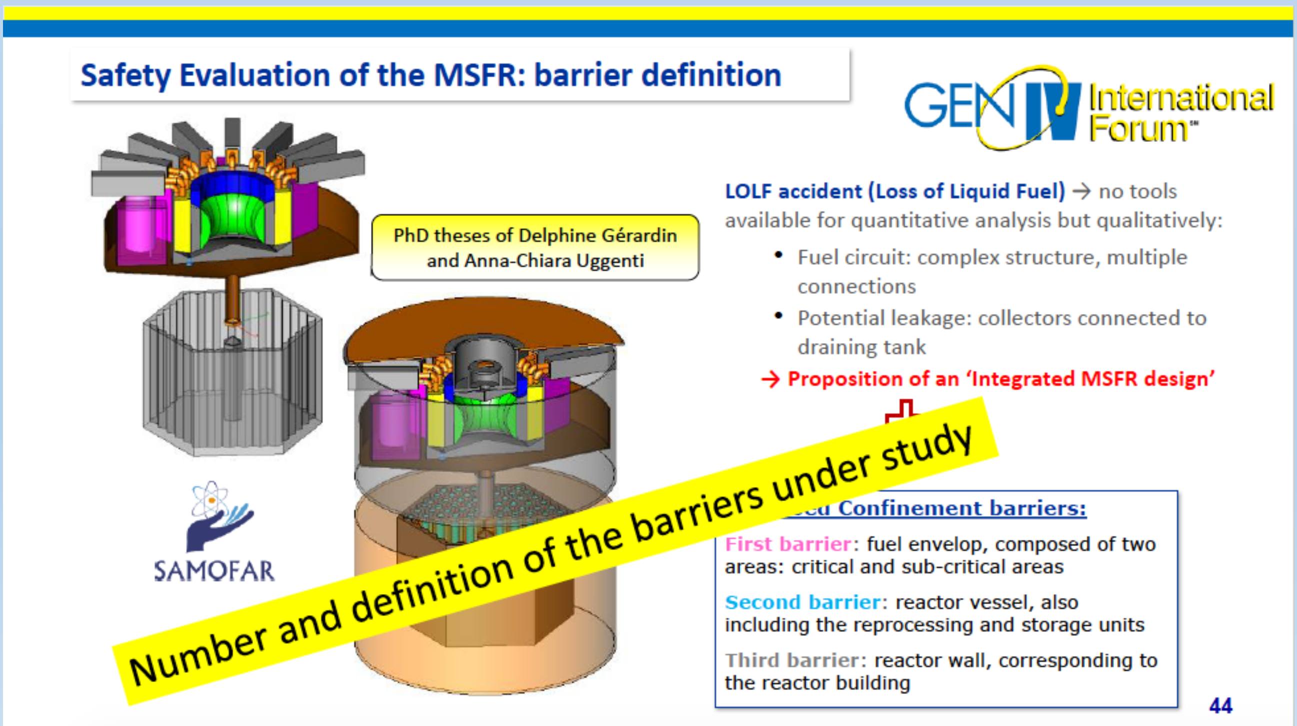
4. SAMOFAR (Safety Assessment of a MOlten salt FAst Reactor) project

This European project has been performed during 2015-2019. They have discussed the safety approach considering the MSFR specific safety features.

5. An example of transient calculations (load following of 30% in 60s)
The Load following is driven by only the extracted power (no control rods needed). The excellent load following capacities of MSFR has been confirmed.



6. Safety Evaluation of the MSFR: barrier definition
How to assign the multiple confinement barrier function to the MSFR SSC (Structure, System, Components) is studied.



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4-2-7. Czech Experimental Program on MSR Technology Development

Summary / Objectives:

The webinar will give an overview of the existing experimental development of Molten Salt Reactor (MSR) technology in the Czech Republic. A technology of nuclear reactor systems with liquid molten salt fuel has been investigated in the Czech Republic since 1999. After 2005, the studies cover also thorium - uranium fuel cycle technology, material research and development of selected components of the MSR technology. Today a new, four-year (2017 – 2020) project of MSR technology development is the key component of the Czech MSR R&D program on fluoride salt-cooled nuclear reactor systems. The aim of the project is to contribute to the development of MSR and FHR reactor technology in the area of reactor physics, nuclear – chemical engineering and material research.

Meet the Presenter:

Dr. Jan Uhlíř works for the Research Centre Řež, Czech Republic as a Senior Researcher of the Nuclear Fuel Cycle Program. Prior to that, he worked for more than 30 years for the ÚJV Řež - Nuclear Research Institute, which is the mother company of the Research Centre Řež. From 1990 to 2012 his positions were Head of Fluorine Chemistry Department and Deputy Director of Fuel Cycle Division. His long-term expertise is mainly in the development of Fluoride volatility reprocessing method and other fluoride pyrochemical partitioning technologies, recently of those devoted to MSR fuel cycle. Jan Uhlíř has been a leader of several national projects devoted to the nuclear fuel cycle, pyrochemistry and molten salt technology granted mainly by the Ministry of Industry and Trade of the Czech Republic. He was also responsible for the chemical part of the national project SPHINX devoted to the experimental development of MSR technology. He participated in several European projects devoted mainly to pyrochemical partitioning and MSR technology. Dr. Uhlíř is a representative of the Czech Republic in the Working Party on Scientific Issues of the Fuel Cycle of the OECD-Nuclear Energy Agency, a member of the MSR Provisional System Steering Committee of the Gen IV International Forum as a representative of EURATOM and a member of the High Scientific Council of the European Nuclear Society. He earned his M.S. in Chemical Engineering and PhD. in Nuclear Fuel Technology from the University of Chemistry and Technology in Prague.

1. Main aims of the Czech Program on MSR Technology Development

The R&D program in Czech covers MSR technologies such as reactor physics, structural material, and Th-U fuel cycle, with experimental verifications.

Main aims of the Czech Program on MSR Technology Development

- To appropriately contribute to the knowledge of MSR reactor physics, core design and safety, structural material development and to the technology of Th – U fuel cycle.
- To focus on R&D of technologies applicable within the MSR on-line reprocessing of liquid fuel.
- To verify experimentally selected important areas of MSR technology and to solve existing bottlenecks.
- Three main domestic projects solved or launched during the first decade of the century contributed to the development of MSR technology:
  - "Transmuter LA-10"
  - "System SPHINX with liquid fluoride fuel"
  - "Fluoride reprocessing of spent fuel from GEN-IV reactors"
- Moreover Czech scientists and researchers also actively participated in several MSR projects of EC-EURATOM, IAEA and contributed to the work of Gen-IV as representatives of EURATOM.

7

2. Structural material development

A new nickel-alloy called MONICR has been developed and further technological activities on the production, corrosion, high temperature integrity, and irradiation damage are ongoing.

Main experimental activities

Structural material development

- Development of structural material for MSR technology, which started in ŠKODA JS - Nuclear Machinery and continued in COMTES FHT company, was crowned in 2011 by experimental production of tubes and sheets from new nickel-alloy called MONICR (Ni-Mo-Cr type super-alloy)

Present development of MONICR alloys is under way in COMTES FHT in the collaboration with other companies including the Research Centre Řež.

The composition of original MONICR alloy is:

Ni	Mo	Cr	Fe	W	Al	Ti	C	Co, Nb, Zr
bulk	13.2 %	6.85 %	2.27 %	< 0.1 %	< 0.1 %	< 0.1 %	< 0.1 %	< 0.1 %

COMTES FHT company reached the experimental pilot production of MONICR alloy (ingots, sheets, wires, tubes).

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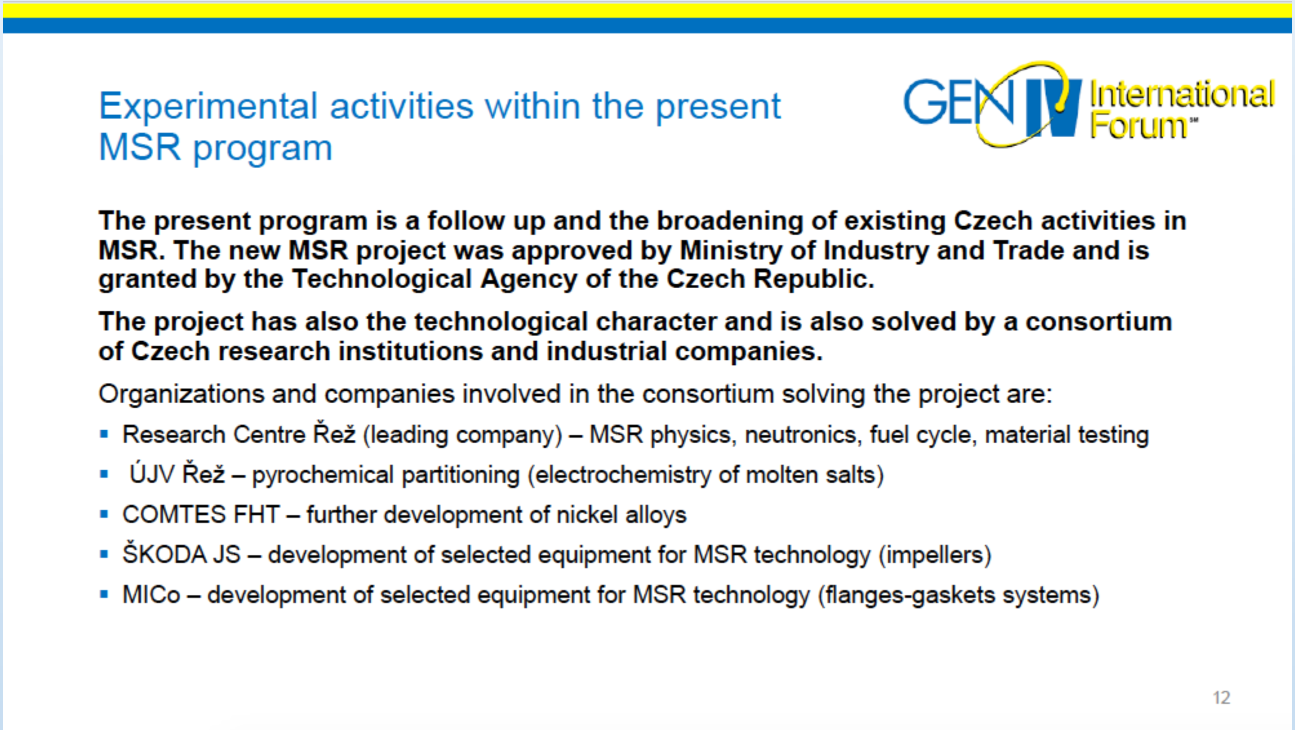
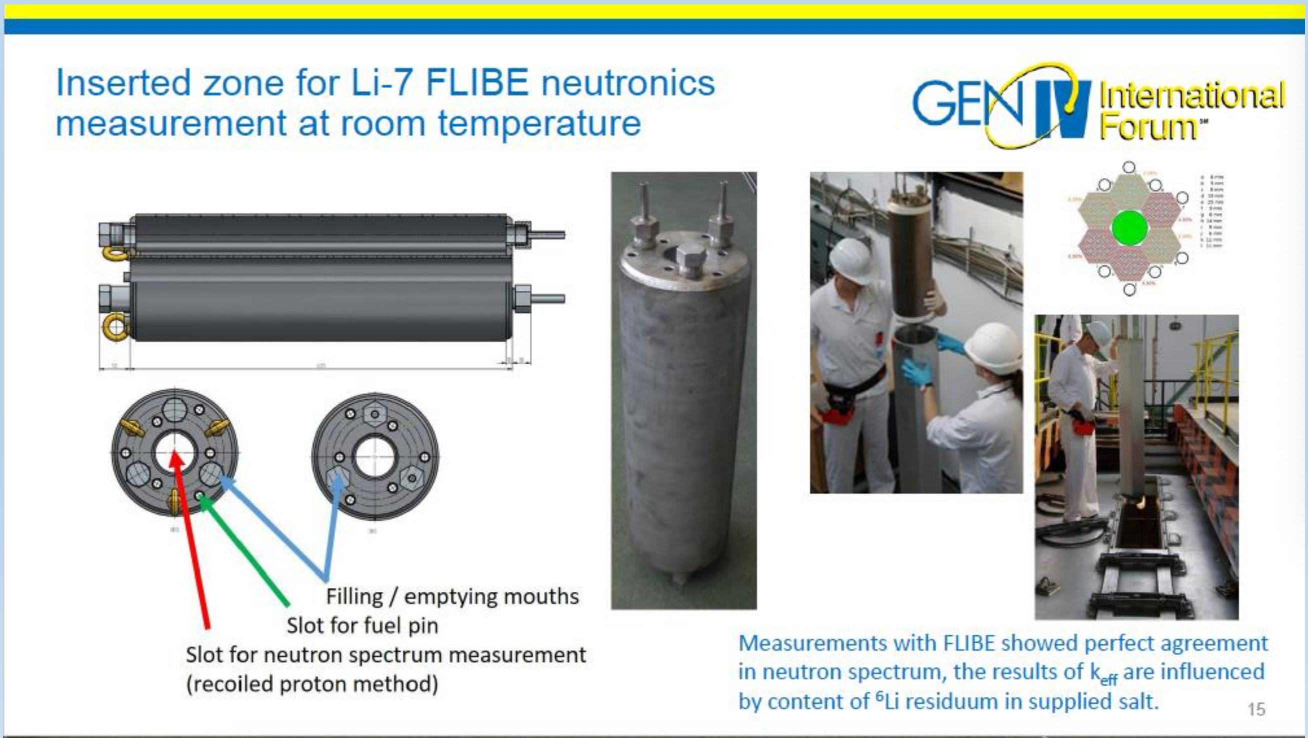
3. Experimental activities within the present MSR program

The new MSR project broadening the existing project was approved by Ministry of Industry and Trade. The collaboration with US-DOE is included in this program.

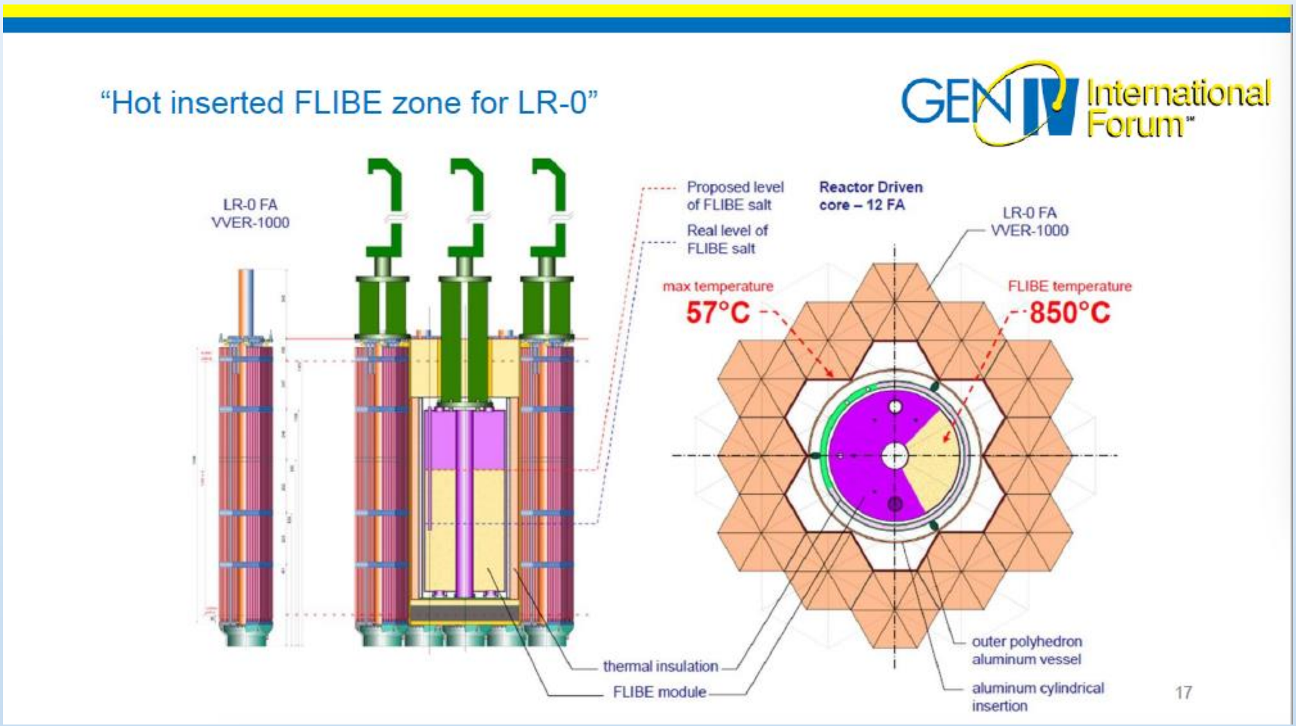


4. Results achieved in MSR physics and salt neutronics with in-pile experiments

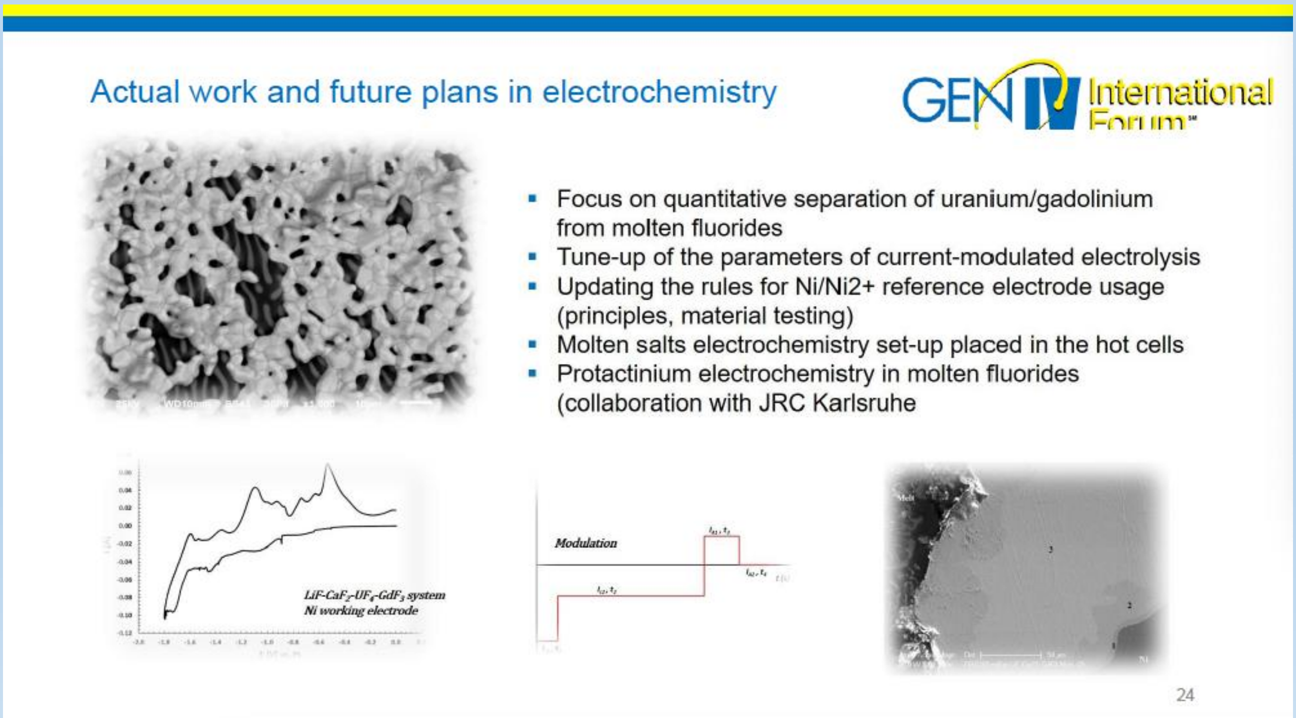
Measurements at room temperature with FLIBE showed perfect agreement in neutron spectrum, the results of k eff are influenced by content of 6 Li in the salt.



5. In-pile test of FLIBE under high temperature condition using LR-0 reactor.
The new heated inserted FLIBE zone (for the measurement at the temperature range 500 -750 ° C) is under development.



6. Studies on MSR fuel cycle technologies
Electrochemical behavior in molten salts and the electrochemical extraction of U, Th and several Lns are investigated



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4-2-8. Micro-Reactors: A Technology Option for Accelerated Innovation

Summary / Objectives:

Micro-reactors are very small nuclear reactors capable of operating independently from the electric grid to supply highly resilient power, and are well suited to serve the power needs for remote communities that currently do not have access to reliable, resilient and affordable energy. A typical commercial micro-reactor is envisioned to be a mobile nuclear power plant in a 2-20 MWe range that is fully factory built, fueled and assembled. It is transportable to the remote site via ground, sea or air with black start, renewable integration and island mode operation capability. They are designed to be self-regulating and walk-away safe with minimal operator intervention. NEI estimates that Micro-reactors could deliver electricity at rates between \$0.09/kWh and \$0.33/kWh. This presentation will describe 'genericized' micro-reactor designs being pursued by various vendors, technology gaps and the role of DOE's Micro-reactor R&D.

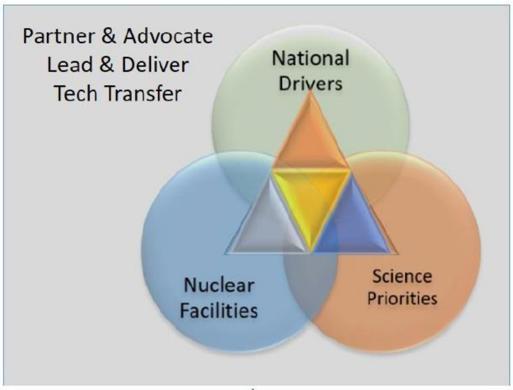
Meet the Presenter:

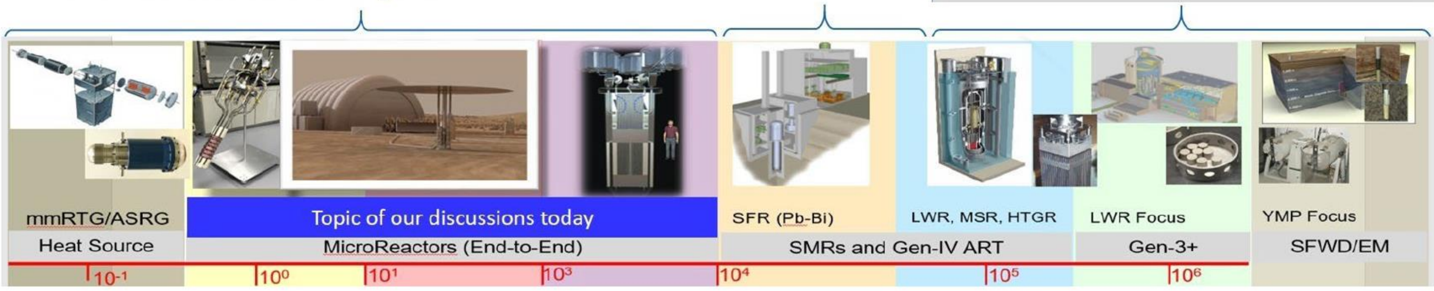
Dr. Dasari V. Rao is a nuclear and mechanical engineer with 25 years of experience in safety and safeguards of nuclear and high hazard facilities. His technical areas of expertise include computational fluid dynamics, neutron and radiation transport, and risk assessment of nuclear energy systems. Dr. Rao is presently Director of the Office of Civilian Nuclear Programs at the Los Alamos National Laboratory. He is also Technical Advisor to Dr. Jess Gehin, National Technical Director for DOE Microreactor Program, and Principle Investigator for the NASA's Fission Surface Power project.

Microreactor R&D at a Glance

GEN IV International Forum

- ❖ National Drivers
  - Innovative, Affordable and Rapid
  - DoD and Civilian Microgrids
- ❖ Nuclear Facilities and Technologies
  - Fuels (HALEU)
  - High Temperature Moderators
  - Nuclear Data
- ❖ Prototypes
  - Advanced Manufacturing
  - Sensors and Structures
  - Sub-scale simulation test objects
- ❖ Integration
  - Multi-scale, nuclear validated codes
  - Test Beds: EDU and NDU
  - NRIC





Common strategy between multi-mission Radioisotope Thermoelectric Generator (2kWt) developed for NASA's mars mission, Micro-reactors (2-20 MWe), SMRs, Gen III+, IV (up to 1500MWe).
That is **diagram by National drivers, Nuclear Facilities, and Science priorities**.
By applying this strategy for Micro-reactors, **Micro-reactors become Factory fabricated, Transportable and Self regulating**.

Reimagine Nuclear Generation…

GEN IV International Forum

Factory fabricated

The majority of components of a microreactor are anticipated be fully assembled in a factory and shipped out to its location. This can eliminate difficulties associated with large-scale construction, reduce capital costs, and help get the reactor up and running quickly.

Transportable

Smaller unit designs can enable microreactors to be very transportable. This can make it easier for vendors to ship the entire reactor by truck, shipping vessel, airplane, or railcar.

Self-regulating

Simple and responsive design concepts can enable remote and semi-autonomous microreactor operations that may significantly reduce the number of specialized operators required on-site. In addition, microreactors plan to use utilize passive safety systems that can prevent the potential for overheating or reactor meltdown.

DOE Microreactor Program is undertaking some of the most important and challenging research and development efforts to accelerate microreactor deployments by mid-2020s

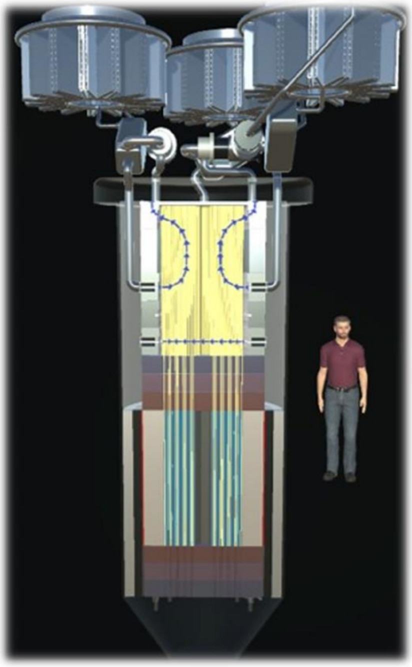
National drivers for Micro-reactors are,
+ Innovative, Affordable and Rapid
+ Military and Civilian Microgrids

Key technology are
+Factory built with advanced manufacturing, instrumentation/sensors, and advanced heat removal systems.
+Easy to operate and licensed by power controllability which brings easy load following.

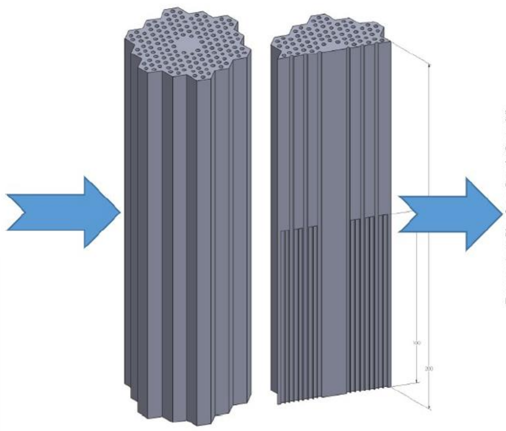
Technology neutral with the common strategies
= Accept various types of fuel including nationally supplied HALEU fuels.

Key Technology Enablers

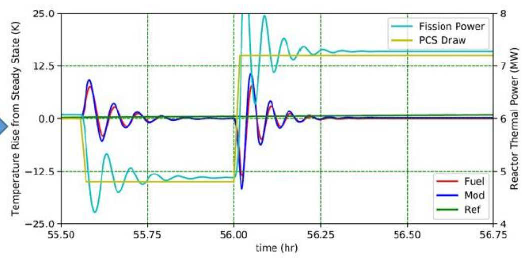
Factory Built ◆ Easy to operate ◆ Easy to license



Designs may vary, but challenges are similar…..
…. So, R&D focus is concept and technology neutral



Understanding manufacturability and licenseability



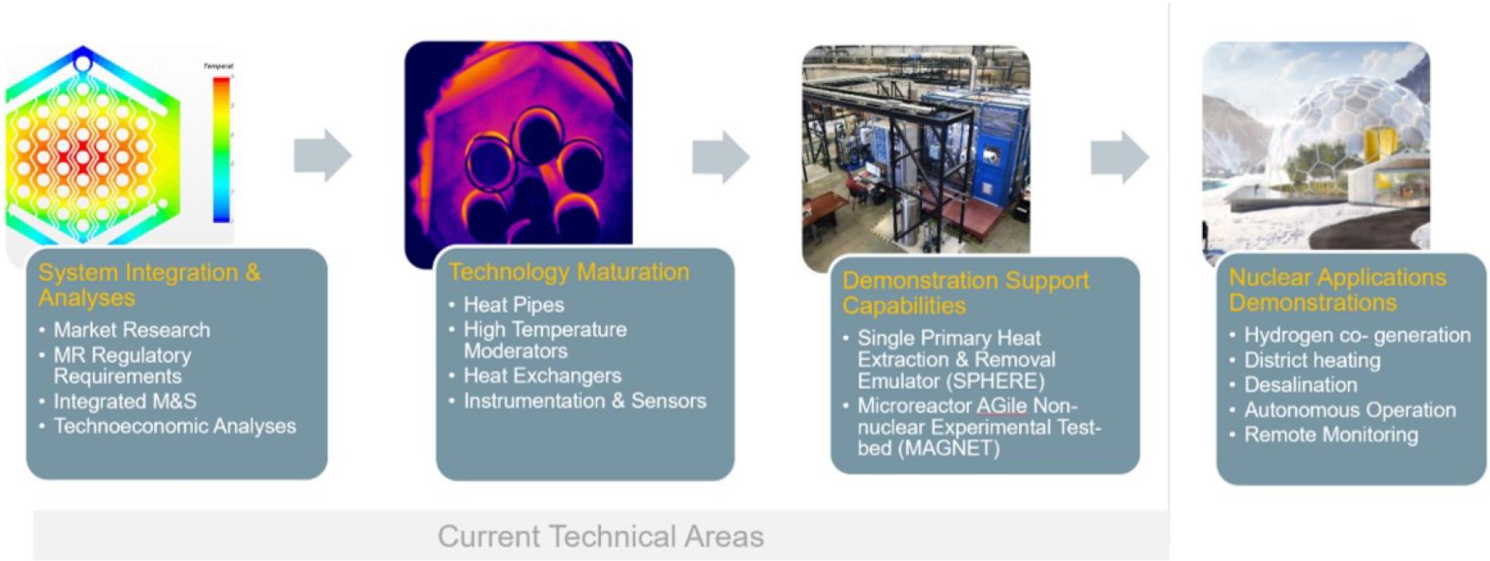
Demonstrating safety, stability and ease of operability

6

Typical Microreactor Design

- Reactor designs include following options:
  - HALEU Metallic, Ceramic or TRISO Fuels
  - Fast, intermediate or thermal neutron spectrum enabled by a mixture of high temperature hydrides, beryllium and graphite
  - A large reflector that also performs as a thermal sink and houses control drums
  - Heat pipe-, gas-, molten salt- cooled
  - Brayton power conversion (with or without intermediate HX)
- Structural material options include
  - Metals
    - High temperature creep-resistant steel
    - Molybdenum
  - Ceramics
  - Graphite

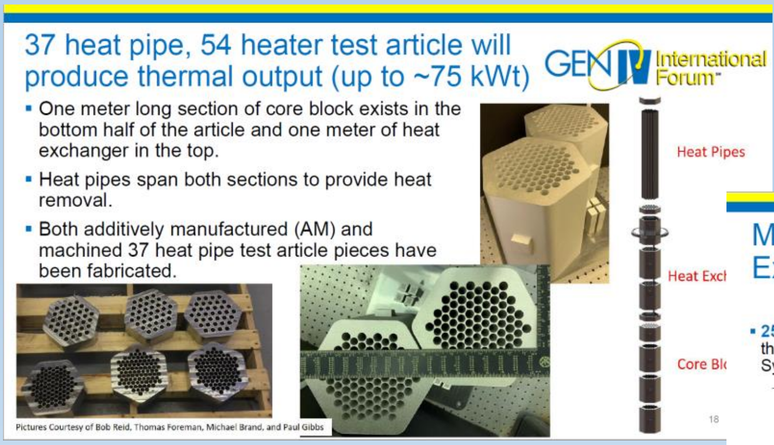
DOE Microreactor Program R&D Focus



Dr. Holly Trellue is a team leader at Los Alamos National Laboratory, the Technical Area Lead for Technology Maturation for the DOE-NE Microreactor Program.,

She introduced Technology Maturations.

+ Possible fuel materials
+ Advanced moderators including metal hydrides
+ Advanced heat removal mechanisms
+ Instrumentation / Sensor developments

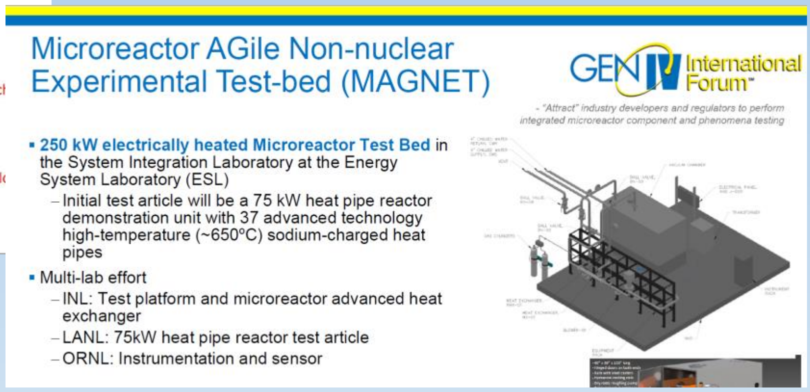


Mr. Yasir Arafat is currently serving as the Technical Advisor to the DOE Microreactor Program from Idaho National Laboratory. He was the founder and Technical Lead of the Westinghouse eVinci™ Micro Reactor Program.

He introduced Two demonstration test programs.

SPHERE: Single Primary Heat Extraction & Removal Emulator

MAGNET: Microreactor Agile Non-nuclear Experimental Test-bed



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5-1. Metallic Fuels for Fast Reactors

Summary / Objectives:

This webinar will provide an overview of metallic fuels used in sodium-cooled fast reactors. Topics to be briefly surveyed will include: a history of metallic fuel development and use; benefits of metallic fuel technology for fuel reliability and safety; and current development directions in the areas of actinide transmutation and ultra-high burnup.

Meet the Presenter:

Dr. Steven Hayes is a Fellow of the Nuclear Science & Technology engaged in the development, testing and modeling of a variety of nuclear fuels, including metallic, oxide, and nitride fuels for liquid metal reactors and high-density dispersion fuels for research reactors. He led numerous fuels and materials irradiation experiments in the Experimental Breeder Reactor II prior to its shutdown, and today he maintains an active fuel testing program in the Advanced Test Reactor. Dr. Hayes is a national leader in the development and testing of metallic fuels for the US-DOE's Advanced Fuels Campaign and in the development of multiscale, multiphysics fuel performance codes for the US-DOE's Nuclear Energy Advanced Modeling and Simulation program.

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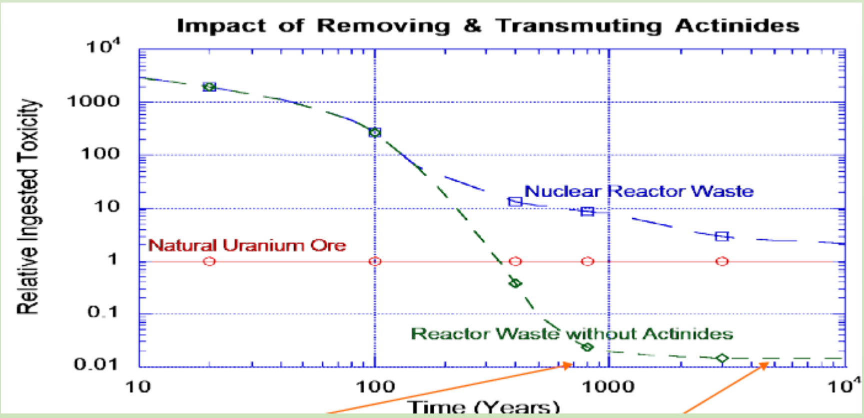
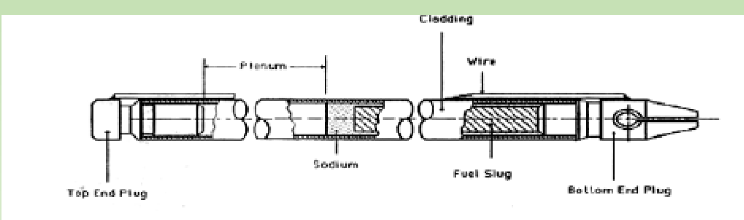
Background: Motivation for Actinide Transmutation

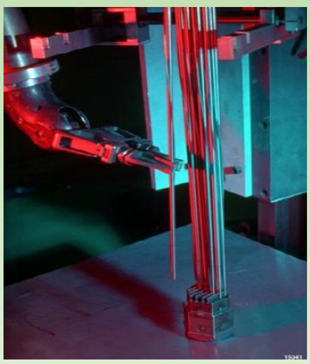
- Plutonium and minor actinides are responsible for most of repository hazard beyond ～400 years.
- Fast reactors are appropriate for actinides transmutation mission, because of large number of excess neutrons, neutrons of high energy, and variety of actinide management strategy.
- SFR Transmutation fuels contain minor actinides and rare earth fission product in significant quantities. So, remote fuel fabrication, new fabrication methods, and determination of effects on fuel performance are necessary.



Metallic Fuels: History & Benefits

- Metallic fuels are used in EBR-1, UK Dounreay Fast Reactor, Enrico Fermi FBR, EBR-II, and FFTF.
- Metal fuels have historical benefit, including reliability to high burnup, compatibility with proliferation-resistant electrochemical recycle, simple and compact fabrication process, and synergistic with passive approach to reactor safety.
- Fabrication of metallic fuels on large scale and remote environments are easy historically. Metallic fuels has demonstrated high-burnup reliability; lower-density alloys for transmutation offer even higher burnup potential.





Casting Process Development

- Traditional casting (Injection casting (counter-gravity)) is employed for remote fabrication of 39,000 metallic fuel pins for EBR-II over a 3-year period in 1960's.
- Application of the traditional casting to metallic transmutation fuels has issues on fuel losses, high level waste, and crucible cleaning and coating.
- New casting process (Bottom casting) was to developed to greatly improve melt utilization, and near-zero Am loss during fabrication.
- Issue of Am volatility during casting has been resolved at bench-scale using surrogate system; validation testing with Am is underway.

Performance of Metallic Fuels with MAs

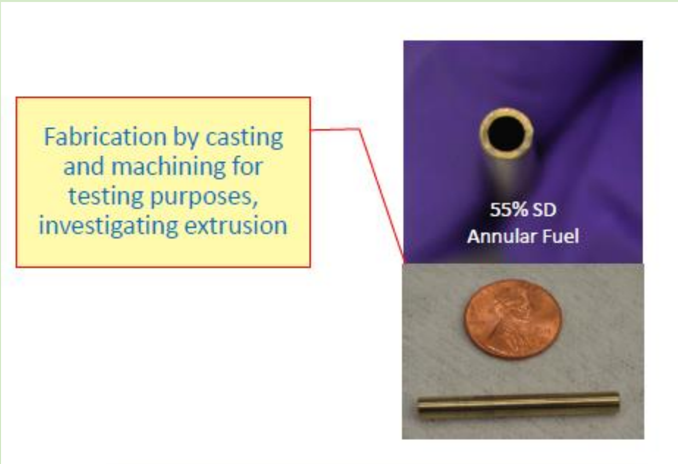
- Wide spectrum of U-Pu-Am-Zr fuel alloys have been conducting in the ATR (AFC-1～4, IRT).
- With double encapsulated testing approach, the tests could be conducted 500W/cm in linear power and 600℃ in cladding temperature.
- Cd-shroud removed thermal neutrons from neutrons of ATR.
- Irradiation performance tested fuels has been shown to be typical of historic understanding for wide variation of U, Pu, Zr, & MA contents.
- Comparison Report (FY17) will validate ATR Cd-shrouded test results vs. data from EBR-II, FFTF, and Phenix.

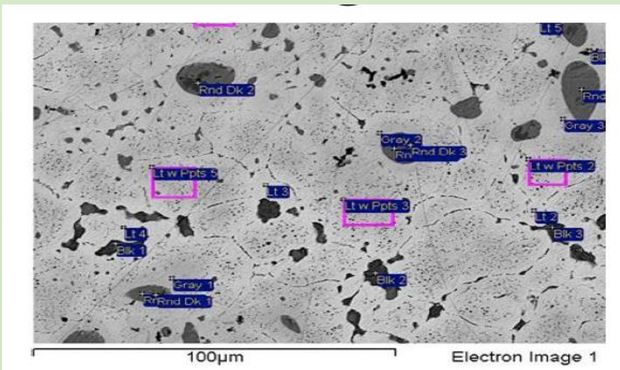
		AFC-1	AFC-2	AFC-3/4	IRT
Test Strategy		Scoping – Many compositions	Scoping – Focused compositions	Focused compositions	Focused compositions
		Nominal conditions	Nominal conditions	Nominal+ conditions	Nominal+ conditions
Capsule Type		Drop-in	Drop-in	Drop-in	Drop-in
Fuel Types		Metallic Nitrides	Metallic Oxides	Advanced Metallic Concepts	Metallic
Key Features		Baseline + MA	Baseline + MA + RE	FP control, annular fuel, FCCI barriers, ultra-high burnup	Recycle feed Remote fabrication
Time Frame		FY 2003 – FY 2008	FY 2008 – FY 2012	FY 2011 – FY 2017 +	FY 2018 – 2020

Past test series   Test series in progress   Future test series

Future Directions: Innovative "Advanced Metallic Fuel Concept"

- Development of the "Advanced Metallic Fuel Concept"
- Additives for Ln FP stabilization and immobilization
- Cladding coating/liners
- Low SD annular fuel, fabrication by extrusion
- Demonstration reliable performance to ultra-high burnups (30-40%)





Back scattered electron image of
U-15Zr-3.86Pd-4.3Ln
(Ln = 53Nd-25Ce-16Pr-6La, in wt%)

5-2. TRISO Fuels

Summary / Objectives:

TRISO (TRi-structural ISOtropic) particle fuel has been developed for use in modular high temperature gas reactors (HTGR) designed to passively maintain core temperatures below fission product release thresholds under all licensing basis events and accident scenarios. This webinar will give an overview of the US DOE Advanced Gas Reactor (AGR) TRISO Fuel Qualification and Development Program's activities focused on enhancing TRISO fuel performance by using uranium oxicarbide (UCO) fuel kernels and improving coated particle and compact fabrication methods for deployment in advanced HTGRs. Topics include fuel characterization and qualification methods, TRISO production scale fabrication process improvements, AGR TRISO irradiation experiments, post-irradiation examination and safety heating test results, and fuel performance modeling efforts. Current US TRISO fuel reactor vendor efforts, and the first TRISO topical report submitted to the NRC will be presented.

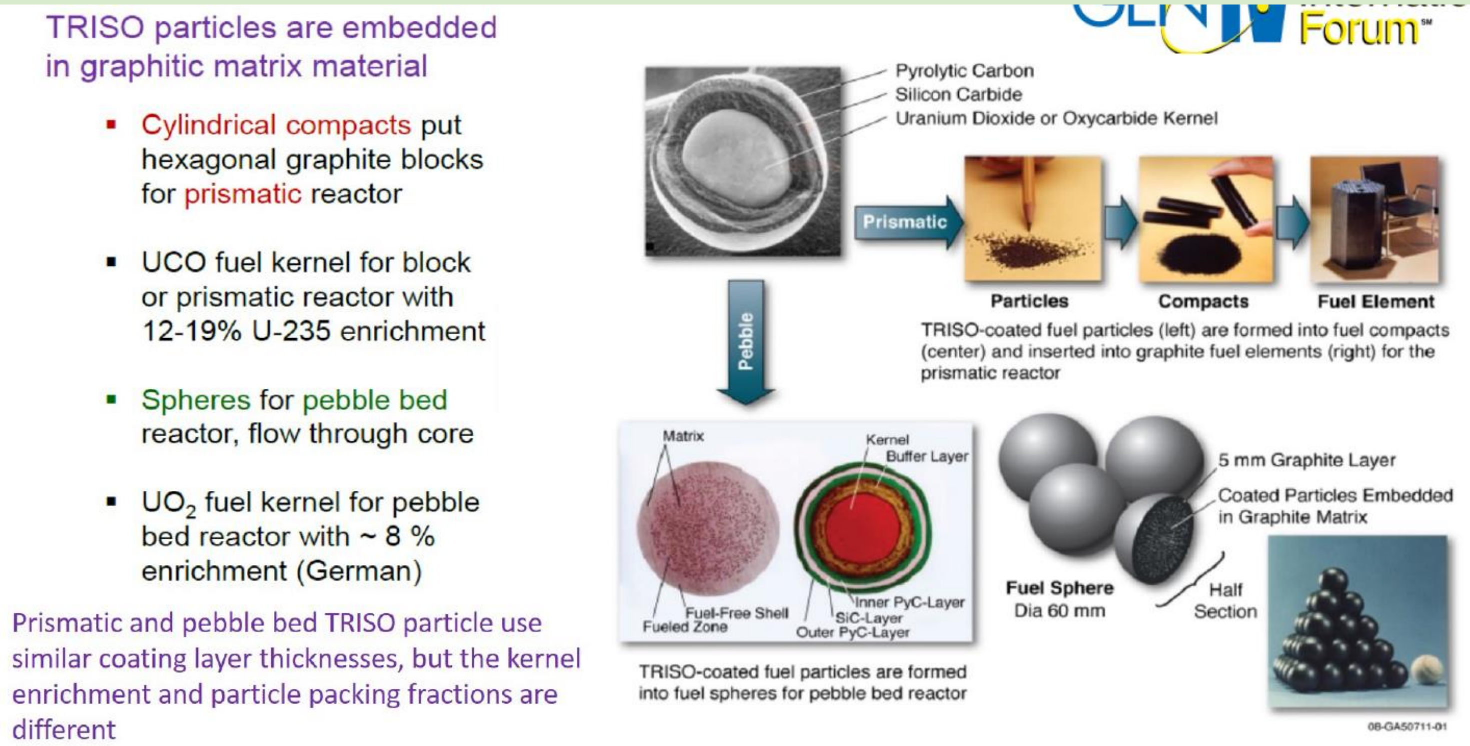
Meet the Presenter:

Dr. Madeline Feltus has led the DOE Office of Nuclear Energy's Advanced Gas Reactor TRISO Fuels Qualification and Development Program since 2003. She provides technical support for DOE's advanced nuclear fuel research and development (R&D), light water reactor accident tolerant fuel R&D, and reactor development projects where she focuses on improving reactor fuels and materials irradiation performance for current and advanced fuel designs to have safe, accident-tolerant, robust, and reliable reactor fuel that can be used in existing and future advanced light water, gas-cooled, and sodium cooled reactors.

She has been involved in writing and providing input for OECD NEA Experts Committee reports, IAEA technical documents, and reviewing manuscripts for technical journals. She is responsible for managing various university grant projects, vendor/industrial projects and small business R&D efforts. Prior to joining DOE in 1999, Dr. Feltus was an assistant professor of nuclear engineering at the Pennsylvania State University (1991-1999). Madeline received her B.S. in Nuclear Engineering from Columbia University in 1977. While working full-time as a nuclear engineer at Burns and Roe, Public Service Electric and Gas (N.J.) and the New York Power Authority, she continued her graduate studies at Columbia and earned her M.S. in Nuclear Engineering (Reactor Physics, 1980), her M. Phil. in Mechanical Engineering (Thermal-Hydraulics, 1989) and her Ph.D. in Nuclear Engineering (1990) with her thesis on 3D time-dependent coupled kinetics-neutronics and thermal-hydraulics analyses.

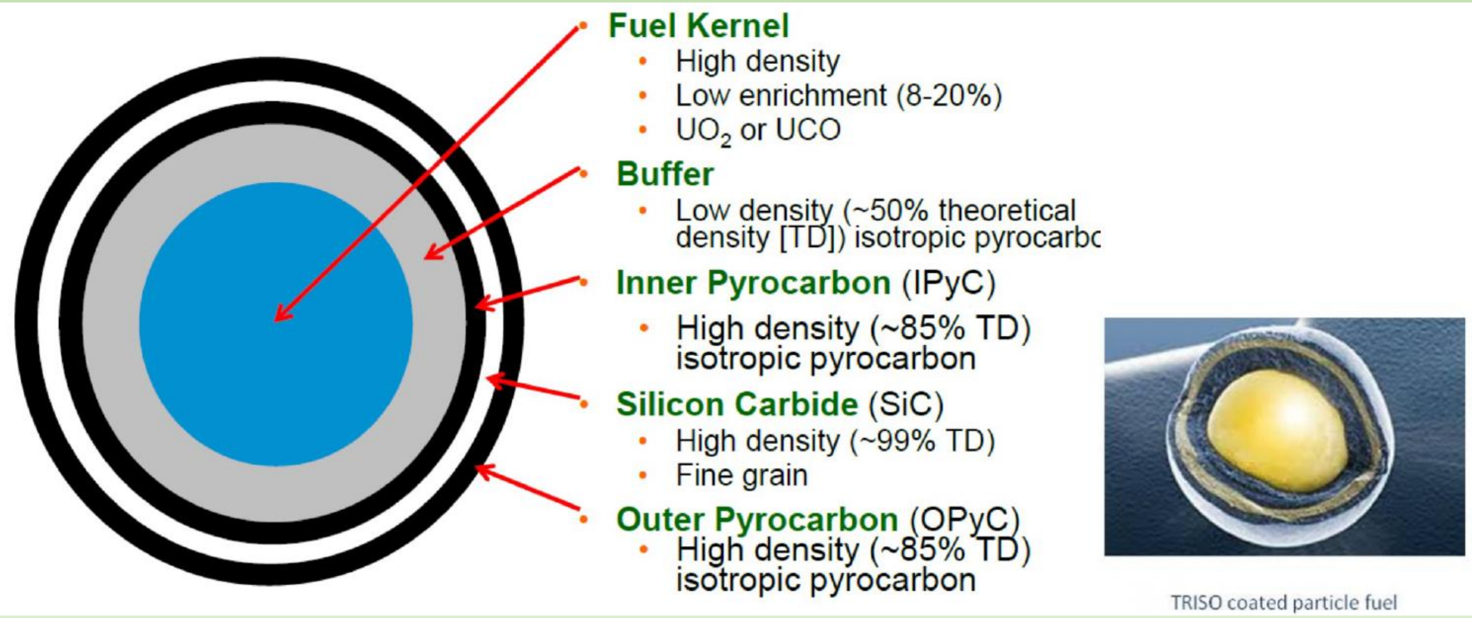
TRISO Particle Fuel:

TRI-Structural ISOtropic (TRISO) particles are embedded in graphite matrix material.



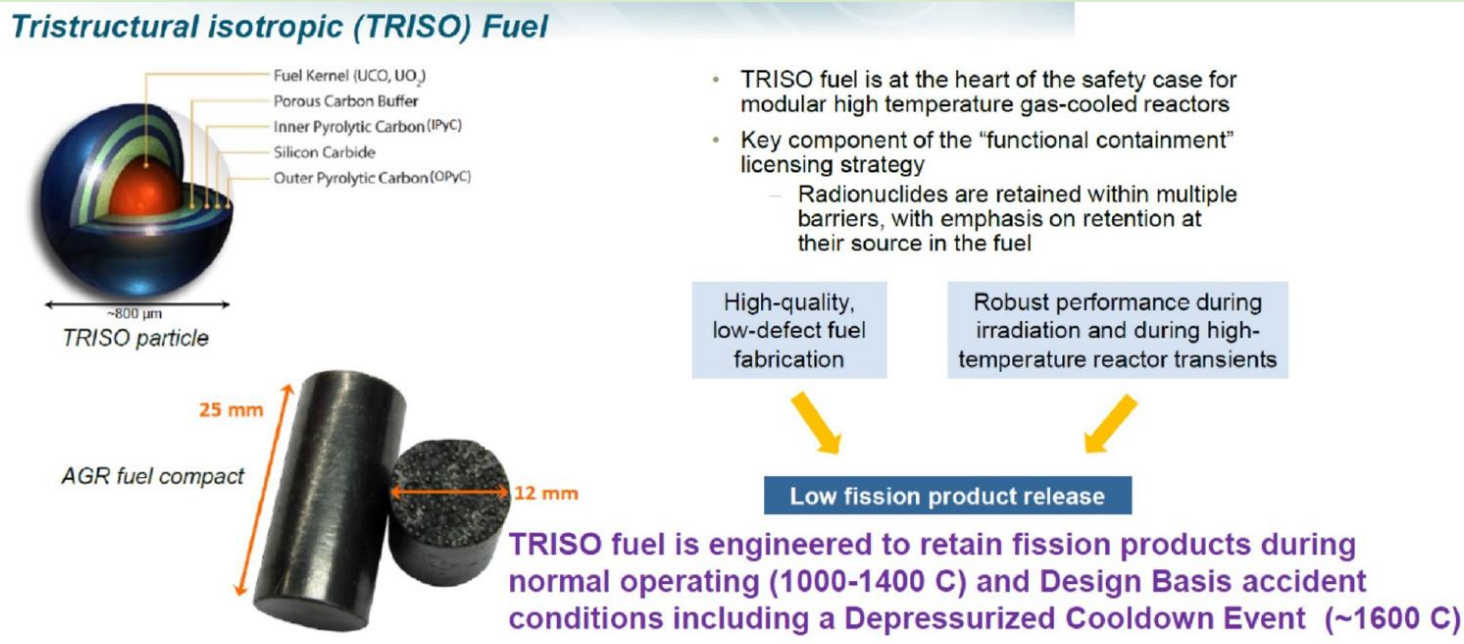
TRISO Particle Fuel Design:

TRISO particle fuel consists of fuel kernel, buffer, inner Pyrocarbon, Silicon Carbide, and outer Pyrocarbon.



TRISO Particle Coatings Retain Fission Products:

TRSO fuel is engineered to retain fission products during normal operating (1000ºC-1400ºC) and design basis accident conditions including a depressurized coolant event (-1600ºC).

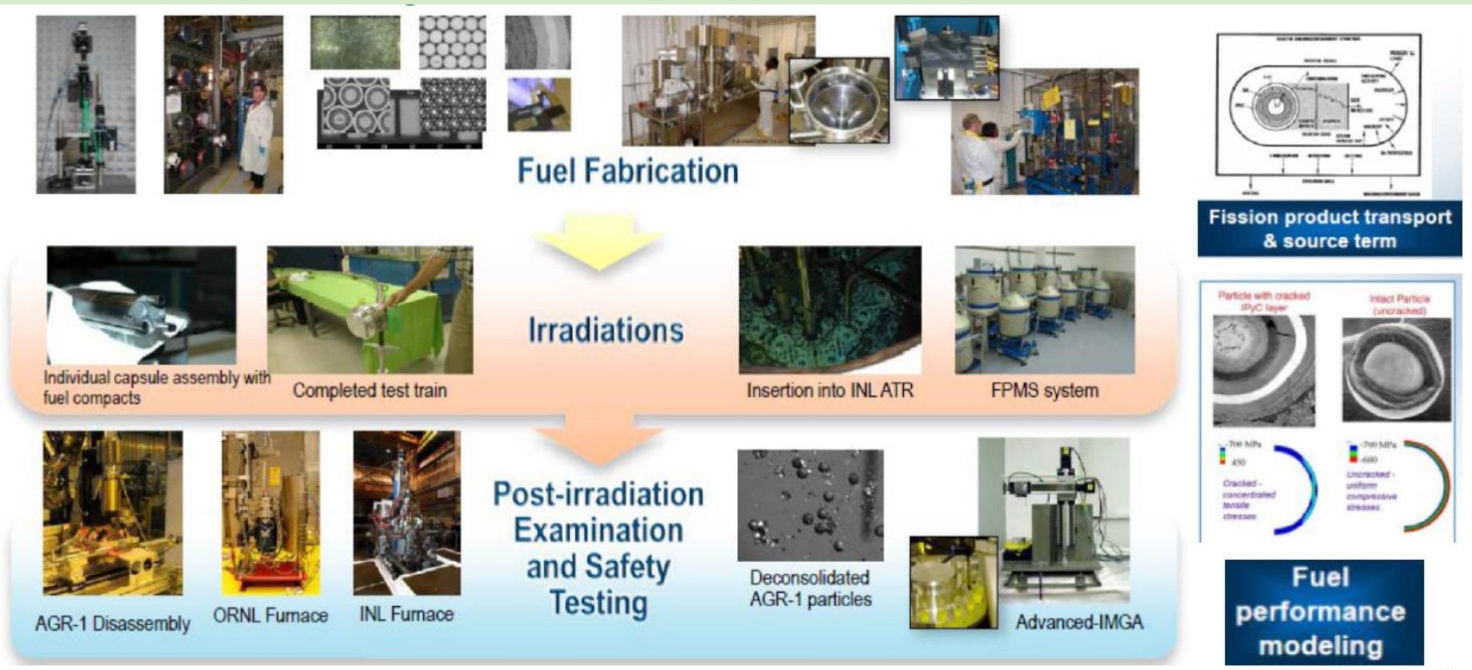


TRISO Particles act as individual fission product "Containments" for Gas-Cooled Reactors:

TRISO coated particle fuel performance and fission product retention is key factor for making the HTGR/VHTR/NGNP safety case.

Advanced Gas Reactor TRISO Fuel Qualification Program:

The objectives and motivation of the advanced gas reactor TRISO fuel qualification program in USA is to provide data for fuel qualification in support of reactor licensing and to establish a domestic commercial vendor for TRISO fuel.



Beyond the AGR TRISO Program:

TRISO fuel can be used in other reactor designs.

- Molten Salt-cooled (e.g., FLiBe, FLiNaK,) reactor concepts use graphite matrix TRISO fuel directly, e.g. Kairos Power based on University of California – Berkeley pebble bed design
- Fast Gas Reactors, using SiC or other non-graphitic matrix compacts
  - French helium fast gas design ZrO_2 coating
  - UC fuel kernels in metallic cladding
  - GA's EM^2 alternate design
- Encapsulated fuel for LWR Accident Tolerant Fuel
  - TRISO in SiC matrix with SiC tubes or Zircalloy cladding (ORNL)
- Fast sodium/metal cooled reactors
  - Dispersion fuels, TRISO-like fuel in metallic matrix, metallic clad
  - TRISO in SiC Mixed Oxide fuel pellets (FFTF or MOX cores)
- Extreme high temperature reactors using refractory metals, UC or UN fuels
  - Space reactors, or niobium (Nb), tantalum (Ta), molybdenum (Mo), rhenium (Re), vanadium (V) and tungsten (W) alloys.

5-3. On Thorium As Nuclear Fuel

Summary / Objectives:

This webinar will present an overview of the basic concepts behind the historical interest on the use of thorium as a nuclear fuel. It will aim at reviewing thorium's real potential and the many challenges it is facing before it can be part of the solution to the world's energy problems. It is aimed at giving some of the scientific elements to a general audience in order to "demystify the thorium question" which has regained some prominence in recent years when talking about future nuclear concepts.

Meet the Presenter:

Dr. Franco Michel-Sendis is responsible for the co-ordination of Nuclear Data Services and Criticality Safety Activities at the OECD Nuclear Energy Agency (NEA) under the Data Bank and the Nuclear Science Division, since 2010. From 2011 to 2016 he also served as NEA scientific secretary to the Generation IV Molten Salt Reactor System Steering Committee and coordinated the NEA report "Introduction of Thorium in the Nuclear Fuel Cycle". Dr. Michel-Sendis holds a B.Sc and M.Sc in physics from the University of Paris (UPMC) and a Ph.D. in nuclear reactor physics from the University of Paris-Sud Orsay.

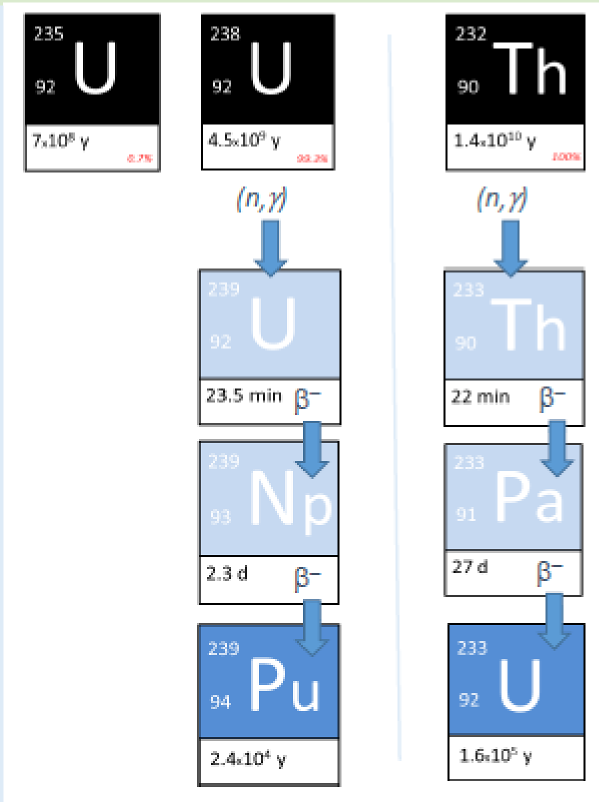
1. U or Th? Not that much of a choice in fact :

- **Only three actinides are naturally present on Earth**
- **Thorium is likely abundant**
- **But Thorium lacks a fissile isotope; only ^{235}U is fissile**

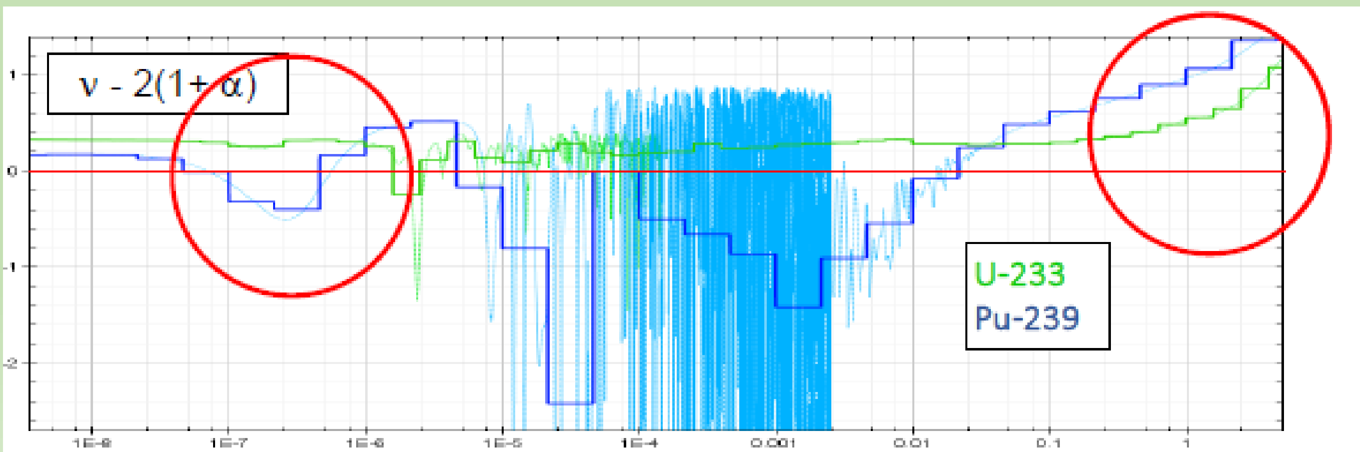
Under neutron irradiation :

- 238U will produce 239Pu
- 232Th will produce 233U

- 232Th excellent fertile
- 233U excellent fissile (in harder neutron spectra)



2. Neutron Economy:
Breeding with Th-U233: possible in thermal spectrum
U/Pu cycle : best neutron economy in fast spectrum



3. Thoria(ThO_2)-based fuels (in current technologyies) :

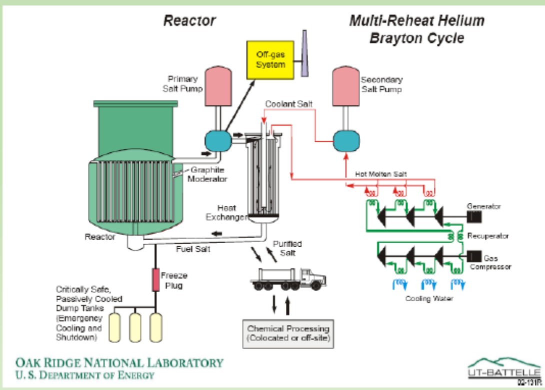
- Thoria-based fuels for LWRs and PHWRs exhibit improved defect performance and are a highly prospective technology for consuming or transmuting transuranic (Pu + MA) nuclides
- Thoria-based fuels must first be qualified to assure their safe performance in the usual suite of normal/accident scenarios; Processes will require significant further development and test programmes to manufacture and qualify optimal industrial thorium-based fuels.

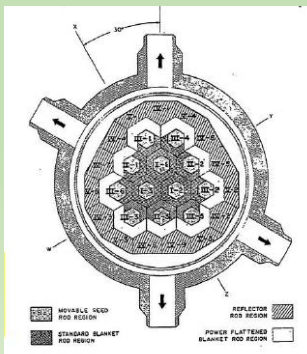
ThorEnergy@ IFE, Norway, (Th, Ce)O_2
Irradiation tests at OECD Halden Reactor

4. Past Experience of Thorium development:

In 1960/70's, some reactors have used Thorium based fuels.

YEAR	Country	Reactor	Type	P (MWe)	Fuel Type	
1962	USA	IndianPoint1	PWR	275	Th/HEU-235	Mixed Oxide
1964-1969	**USA**	**MSRE**	**MSR**	**2-3**	**U-233 FLiBe**	**Molten salt**
1967-1974	USA	Peach Bottom	HTR	40	Th/HEU carbide	Microspheres
1976-1989	USA	Fort St Vrain	HTR	330	Th/HEU carbide	Microspheres
1977-1982	**USA**	**Shippingport**	**PWR**	**70**	**Th/U-233 ox**	**Seed/Blanket**
1983-1989	Germany	THTR	HTR	300	Th/HEU-235	Pebble – 90% U-235

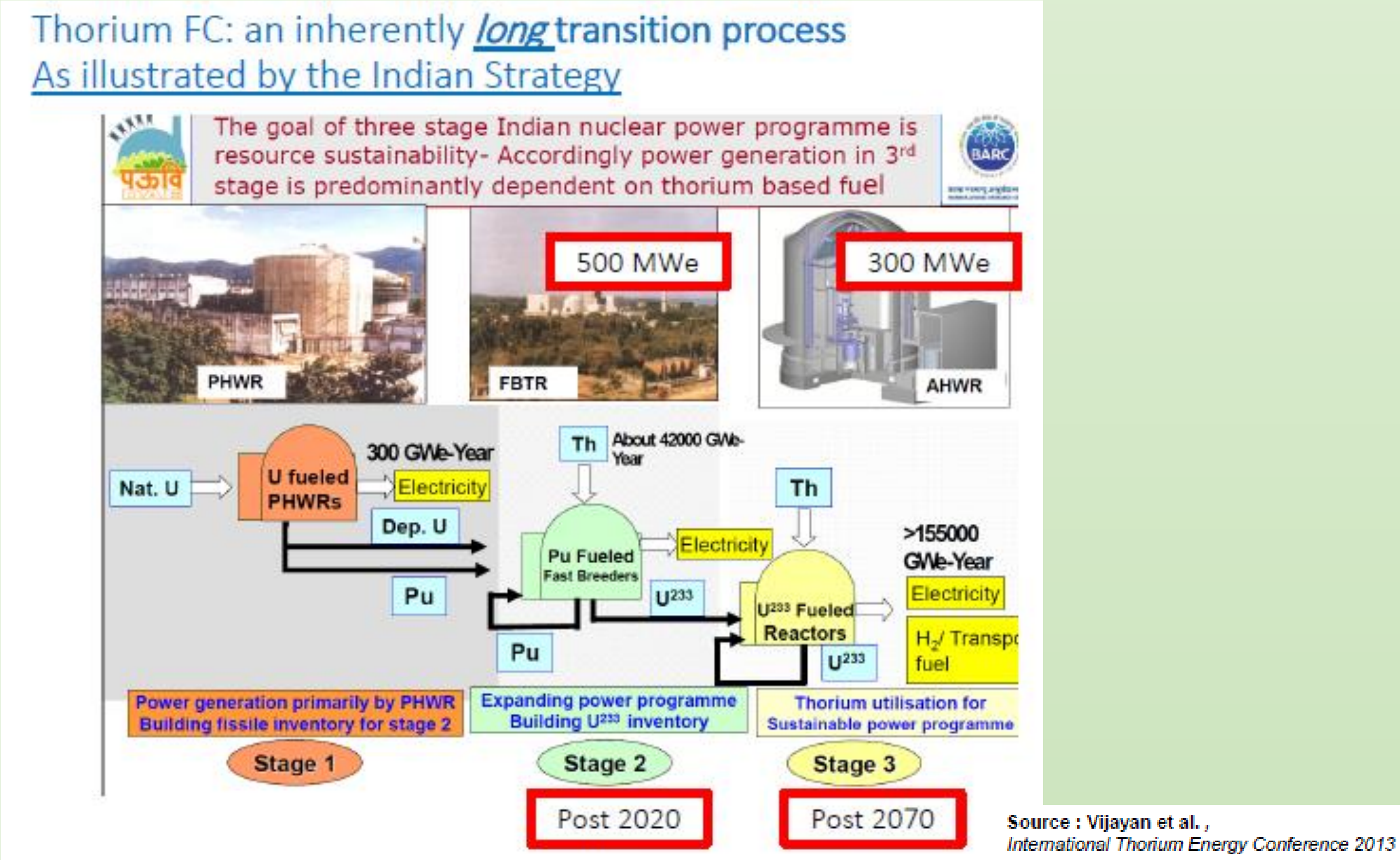




Shippingport's LWBR core

5. Thorium Fuel Cycle

An inherently long transition process, as illustrated by the Indian Strategy

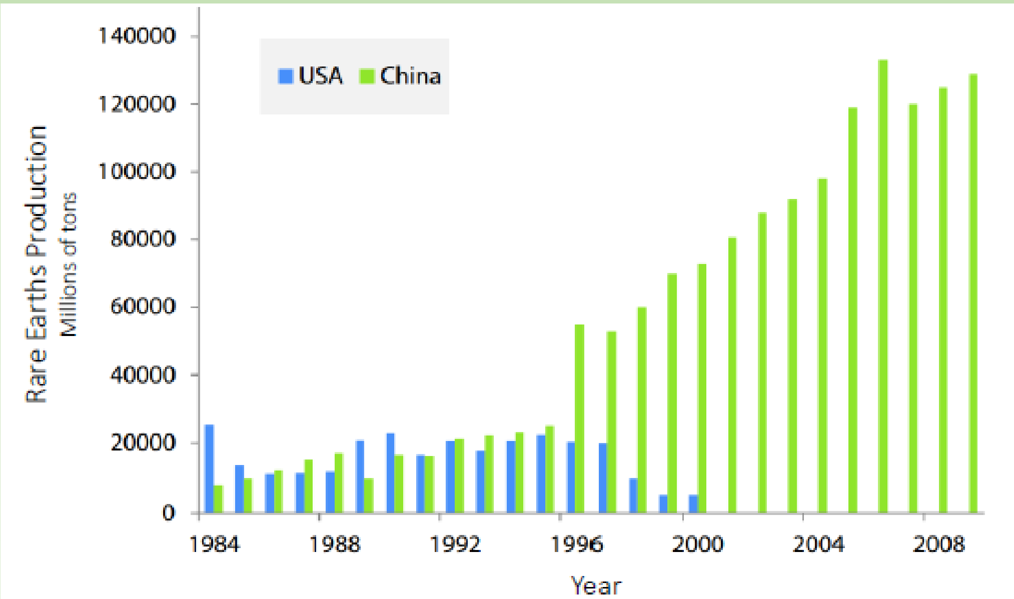


Source : Vijayan et al. , *International Thorium Energy Conference 2013*

6. Resource availability of Thorium

By-product Production of thorium from other industrial mining activities can provide more than ample quantities of thorium for potential use in the nuclear industry for this century and beyond:

- Rare Earth ore mining
- Ilmenite (titanium ore) mining
- Iron ore mining



5-4. Lead Containing Pb-208: New Reflector for Improving Safety of Fast Neutron Reactors

Summary / Objectives:

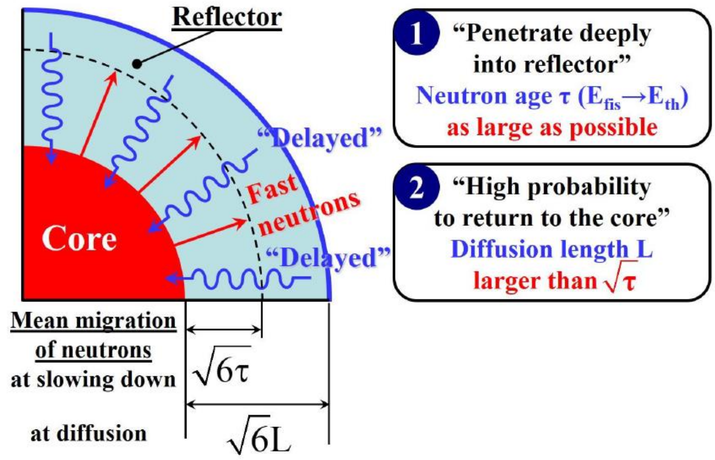
This webinar considers improvement of fast reactor safety through slowing-down power runaway. The idea is surrounding the core by the neutron reflector made of lead-208, a material of heavy atomic weight and extremely low neutron absorption. The power runaways can be slowed down because of a long way for leakage neutrons to come back from distant layers of neutron reflector to the core. It is demonstrated that mean prompt neutron lifetime can be elongated roughly by three orders of magnitude with appropriate slowing-down the reactor power runaway.

Meet the Presenter:

Dr. Evgeny Kulikov earned his PhD at the National Research Nuclear University MEPhI in Moscow in 2010 and is currently the associate professor at the Institute of Nuclear Physics and Engineering. His areas of professional interests include improving fuel burn-up, nuclear fuel cycle, non-proliferation, and fast reactor safety. Currently, his scientific research is supported by the Russian Science Foundation. He lectures on theoretical aspects of nuclear reactors and conducts laboratory works on experimental reactor physics. He is serving on the Gen IV International Forum Education and Training Task Force.

Idea of slow down chain reaction:

This idea is safety improvement by slowing down chain reaction. These requirements to slow down chain reaction are a neutron age as large as possible and a diffusion length larger than square root of neutron age. The idea to slow down chain reaction is a fast neutrons go deeply into reflector and return to the core with essential time delay.



Reactor	Thermal	Fast
Λ	~ ms	~ μs
β	0.65%	0.36%

Safety improvement by slowing down chain reaction

- How we can slow down chain reaction
  - fast neutrons from the core should penetrate deeply into reflector
  - they should have high probability to return to the core as a result of diffusion (in some way "delayed" neutrons)

5

Characteristics of Chain Reaction Rate:

The reflector need to a large atomic mass for a large neutron age, a small absorption cross-section for a large diffusion lengths and a small absorption cross-section for a long lifetime of a thermal neutron. The lead 208 is a good choice for a material of reflector.

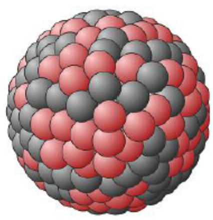
1 Neutron age $\tau(E) \sim A \int_{E_1}^{E_2} \frac{1}{\Sigma_s^2} \cdot \frac{dE}{E}$ $\boxed{\tau \uparrow A \uparrow}$

$\sqrt{6\tau}$ – mean migration of neutrons at slowing down

2 Diffusion length $L \sim \frac{1}{\sqrt{\Sigma_a^{th} \cdot \Sigma_s^{th}}}$ $\boxed{L \uparrow \Sigma_a^{th} \downarrow}$

$\sqrt{6L}$ – mean migration of neutrons at diffusion

3 Lifetime of thermal neutrons $T_{th} \sim \frac{1}{\Sigma_a^{th}}$ $\boxed{T_{th} \uparrow \Sigma_a^{th} \downarrow}$



^{208}Pb

Reflector Properties:

The neutron age and diffusion length of lead 208 are very large. The length of thermal neutron lifetime is very important for safety. The thermal neutron lifetime of lead 208 reflector is longer than in any other material.

Material	$\sqrt{6\tau}$ (cm)	$\sqrt{6L}$ (cm)	Slowing down probability (2 MeV → 0.025 eV)	Lifetime of thermal neutrons (ms)
^{208}Pb	213	843 !	0.993	597 !
Pb_{nat}	213	33	0.304	0.9
Na	227	43	0.297	0.3
Bi	223	96	0.160	4.7
C	49	138	0.998	13

Moderator Properties:

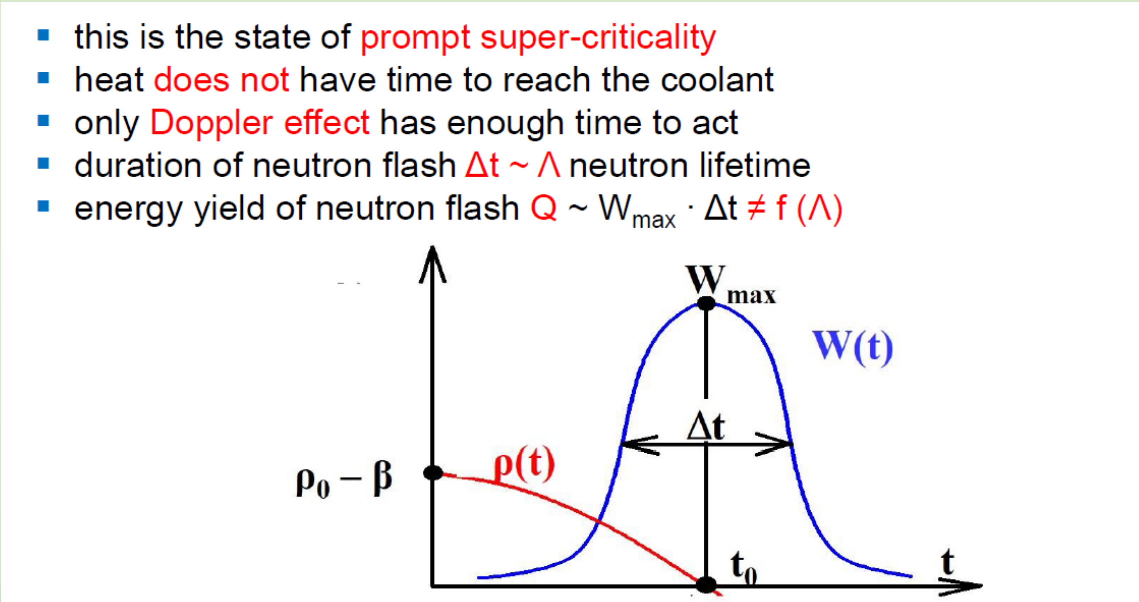
The logarithmic energy decrement is describes average energy loss per a collision. it is not dependent on energy and it depends only on atomic mass. The lead 208 is a low logarithmic energy decrement and low moderating ability. But, the absorption cross-section of lead 208 is very small. As such Moderating ratio of lead 208 is a better than light water or barium oxides or graphite.

Material	Logarithmic energy decrement ξ	Moderating ability $\xi \cdot \Sigma_s$ (cm^{-1})	Moderating ratio $\xi \cdot \Sigma_s / \Sigma_a$
H_2O	0.95	1.39	70
D_2O	0.57	0.18	4590
BeO	0.17	0.12	247
C	0.16	0.063	242
Pb_{nat}	0.01	0.004	0.61
^{208}Pb	0.01	0.004	477

^{208}Pb is an effective moderator

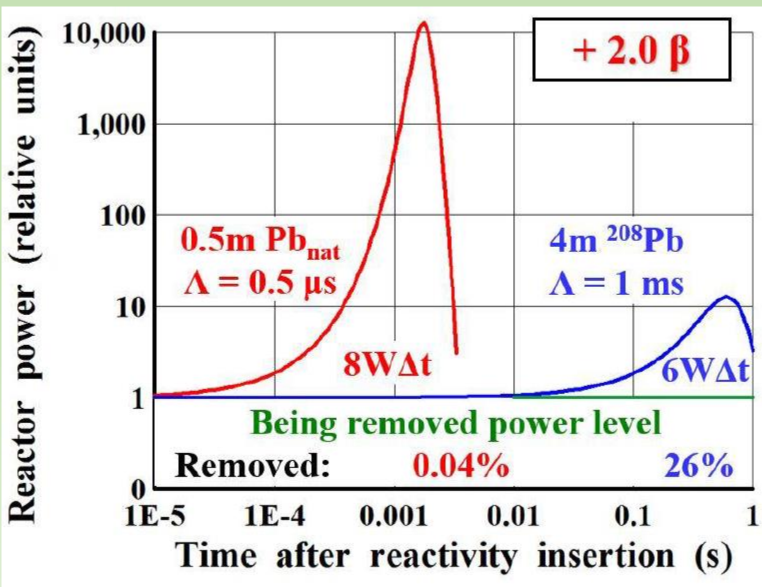
Overview of Neutron Flash model (p0 > B):

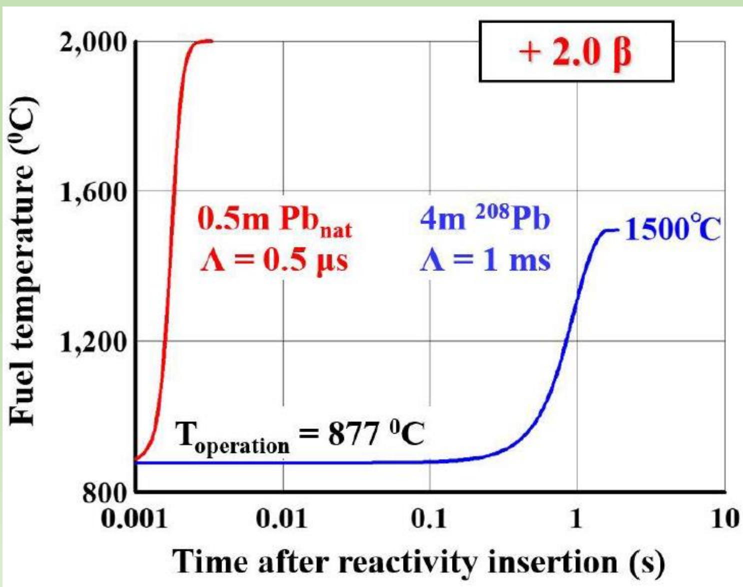
According to the neutron flash model, if a interpret reactivity exceeds delayed neutron fraction is the state of prompt super criticality. And, the doppler effect has enough time to action and duration of neutron flash is proportional to a neutron life to lifetime. while energy yields is in dependence on neutron lifetime.



Reactor Power and Fuel Temperature at the Neutron Flash:

The case of natural leads a neutron lifetime is about one microseconds and the case of lead 208 an about one millisecond. In the reactor power, a peak power is thousand times lower and thousand times slower than natural leads. There is enough time for the heat to be transferred from fuel to coolant. In the fuel temperature, a peak temperature is lower and thousand times slower than natural leads.





6-1. Phenix and Superphenix Feedback Experience

Summary / Objectives:

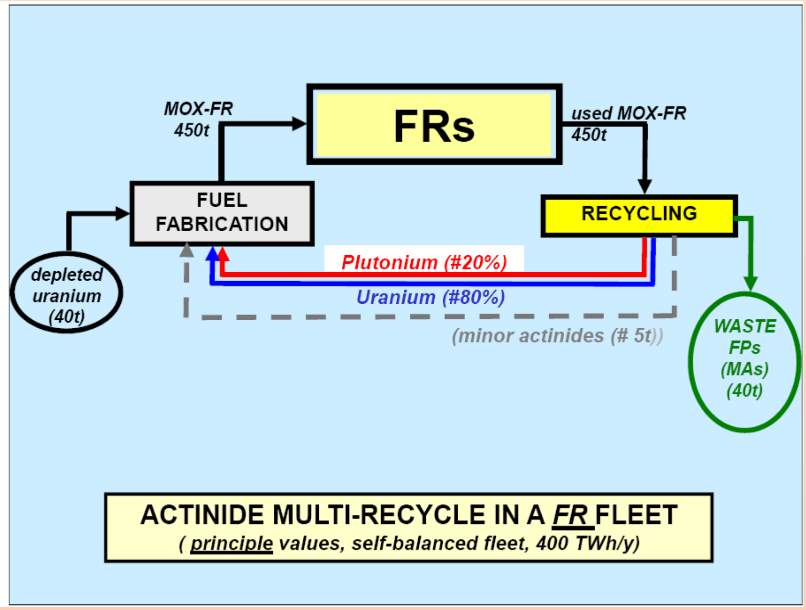
France energy situation is specific : no fossil energy available (oil, coal, gas, etc..), a large fleet of PWR in operation providing about 80% of electricity , and a successful reprocessing activity providing each year about 10 tons of plutonium. In this situation, sodium fast breeder reactors would be very useful for the country, and have been developed with the Rapsodie, Phenix and Superphenix reactors. The feedback experience of these reactors has been analyzed and collected in two books “Phenix: the feedback experience” / EDP sciences 2012, and “Superphenix: Technical and Scientific achievements” / Springer 2016. This thematic analysis was performed on materials, fuel, neutronic, thermal hydraulic, components, water sodium reaction, sodium leaks, safety, and more generally on all the specific technical matters related to this type of reactor. The presentation gives, for each theme cited above, the main results obtained and the main conclusions or recommendations for the future of sodium fast breeder reactors.

Meet the Presenter:

Joël Guidez began his career in the field of sodium-cooled fast reactors, after graduating from the Ecole Centrale de Paris in 1973. He worked at Cadarache for eight years on the design, dimensioning and testing of sodium components for Superphénix. He also followed the initial results, from the Phénix sodium-cooled fast reactor start-up in 1974. Then he joined Phénix where, for five years, he was in charge of measurements and tests on the power plant. In 1987, he returned to Cadarache to lead a thermo-hydraulics laboratory, where many tests were performed for Phénix, Superphénix and the European Fast Reactor (EFR) project. After a period of apparent unfaithfulness to fast reactors, during which he successfully managed the OSIRIS research reactor located in Saclay, and the European Commission’s reactor, HFR located in the Netherlands, he returned to Phénix in 2002, where he managed the reactor until 2008 during his final operating phase. Since 2011, he is considered as international expert in CEA and wrote two books: “Phenix feedback experience” Editor EDP Sciences and “Superphenix. Technical and scientific achievements” editor Springer.

1. Objectives of fast breeder reactors:

- Uranium availability
- Plutonium management
- Management of REP waste
- Transmutation possibilities
- Optimized fuel cycle



2. Sodium fast breeder experience in the world

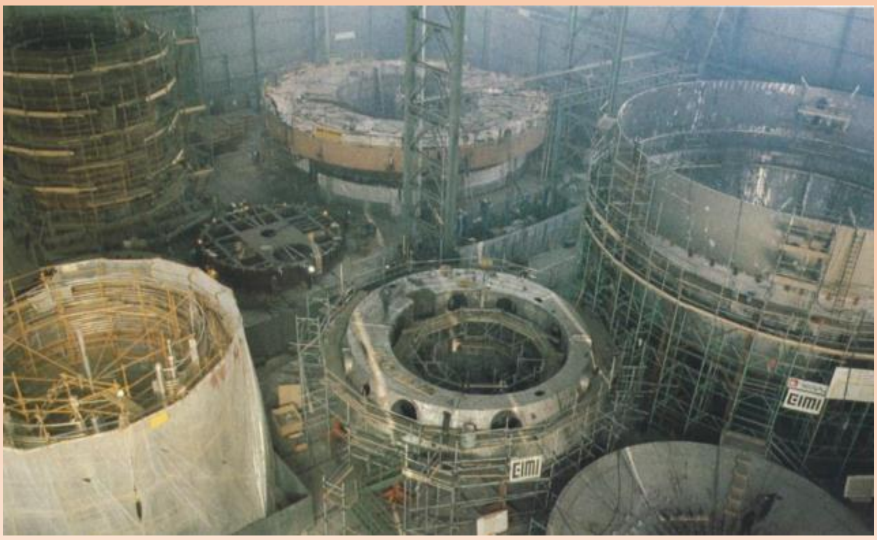
- The first nuclear reactor to produce electricity was a sodium (NaK) reactor in 1951.
- 20 SFR have been built and operated in the world.
- USA/ Russia/ France/ Japan/ India/ China/ UK/ Germany.
- The last one is BN 800 (Russia /800 Mwe) connected to the grid in 2016.
- The PFBR (India /500 Mwe) should start in 2018.

3. Phenix feedback experience

- Built in 1968, by an integrated CEA/EDF/GAAA team, it went critical in 1973 and was co operated with EDF (80% CEA / 20% EDF) from 1974 to 2009.
- During the thirty five year life span, it played its dual role as electricity generator (250 MWe) and experimental research reactor. Thus , it gathered considerable experience for fast breeder reactor systems: demonstration of design and operation , breeder potential, transmutation possibilities, development of all technical fields involved and validation of the technology used.

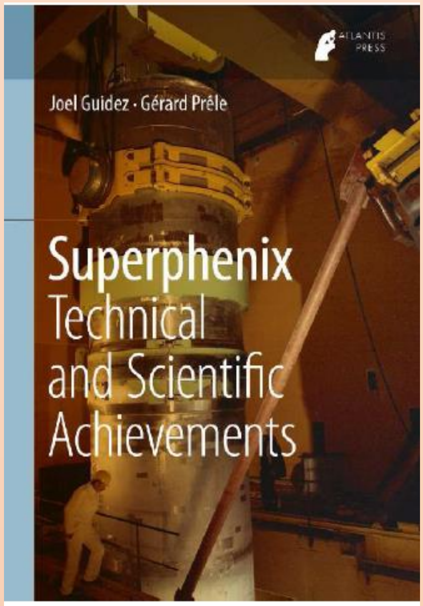
4. Superphenix: technical and scientific achievements

- A huge industrial experience was acquired during the reactor construction.
- The reactor was built in seven years , from 1977 to the beginning of sodium filling sodium in1984.
- The nominal power was reached in December 1986.
- Despite a complicated political life, a big experience on all the technical fields was also acquired until the reactor shut down ten years later.



5. Thematic analysis

- Two books have been written to try to summarize this experience.
- The books are not organized around a chronological experience but with thematic analysis.
- The main themes studied are neutronic , materials , components, thermalhydraulic , fuel, handling, and maintenance.



6. Some examples of accumulated experience

- Reprocessing experience on Phénix (because it is an industrial experience unique in the world)
- SPX construction (impressive industrial work)
- Neutronic of SPX core (the most powerful SFR core ever operated / it remains today a very interesting case for all neutronic studies)


6-2. Astrid - Lessons Learned

Summary / Objectives:

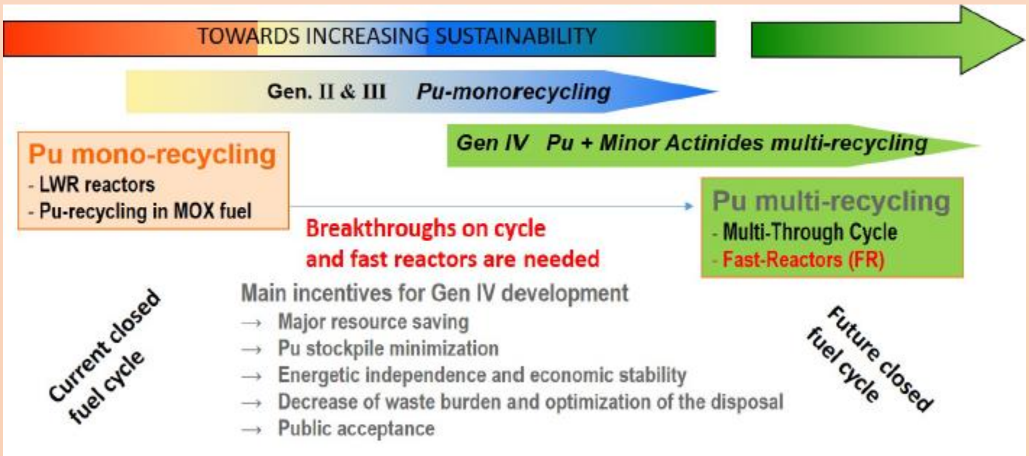
This presentation will first place the context of the choice of Sodium Fast Reactor in the French Nuclear Policy and its rationale for a closed fuel cycle. It will then present the position of the French Sodium Fast Reactor program in the context of Generation IV. The presentation will then focus on the ASTRID (Advanced Sodium Technological Reactor for Industrial Demonstration) project. The technical achievements, major innovation progress and management challenges will be presented. The ASTRID project description will highlight the major use of digital tools (numerical simulation, use of virtual reality, multiscale and multi-physics modeling, PLM: Product Lifecycle Management) used to perform efficiently such a complex project.

Meet the Presenter:

Mr. Gilles Rodriguez is a senior expert engineer at the CEA/CADARACHE (French Atomic Energy Commission/Cadarache center). Since 2016, he has also been the deputy head of the ASTRID Project team, working on Generation-IV Fast Reactor research program. He graduated from the university of Lyon, France in 1990 with an engineering degree in Chemistry and obtained a Master of Science in chemical and process engineering from the Polytechnic University of Toulouse, France, in 1991. His areas of expertise are 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, he joined the CEA project on Sodium Fast Reactor: ASTRID (Advanced Sodium Technological Reactor for Industrial Demonstration), first as responsible of the ASTRID Nuclear Island.

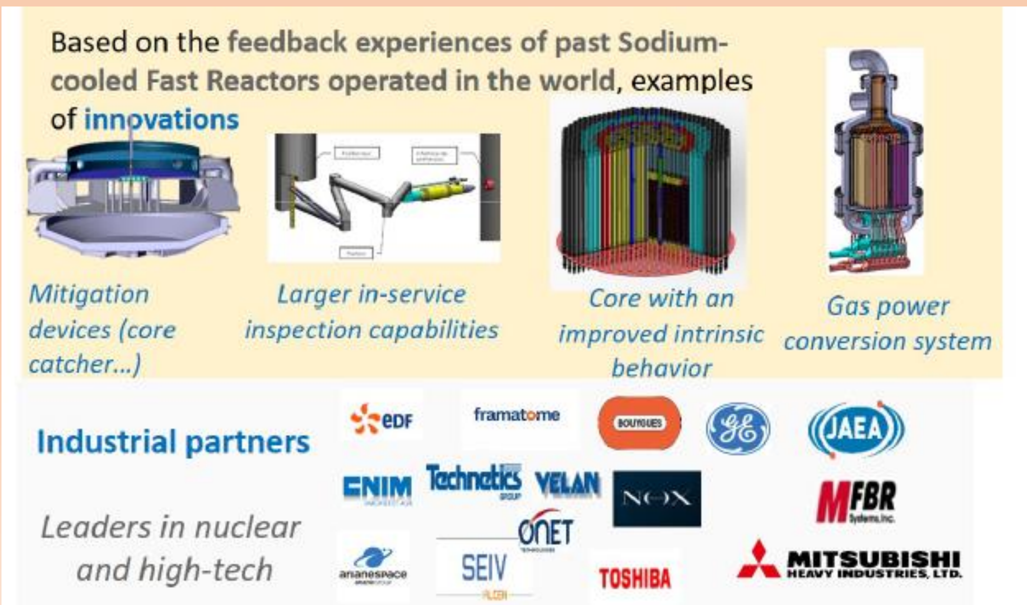
1. French Nuclear Policy:

- 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.



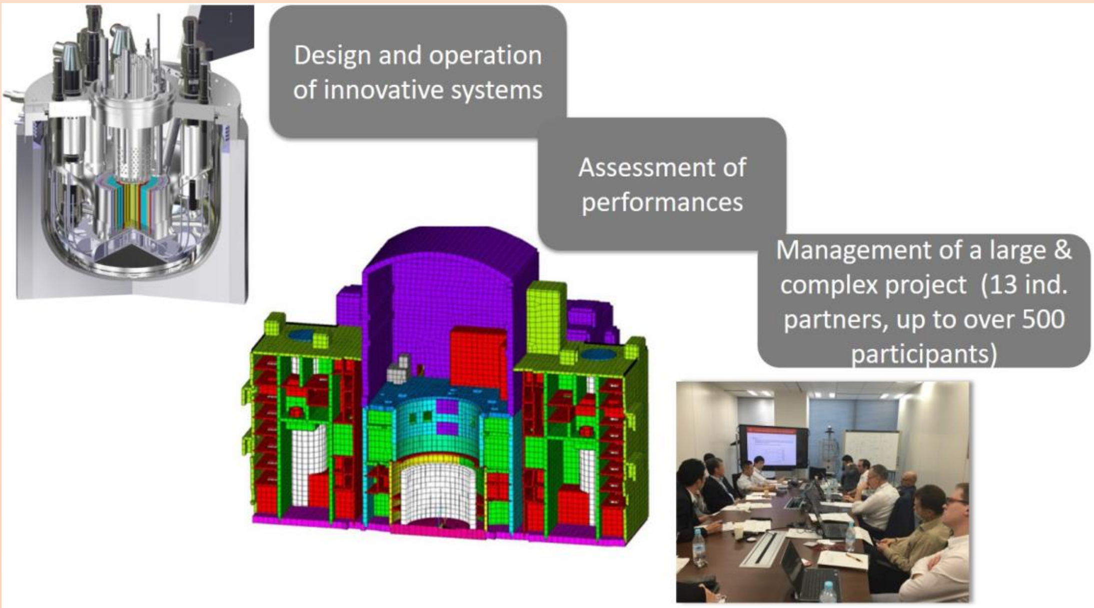
2. The ASTRID Program

- ASTRID is a technological demonstrator and is not a First of a Kind of a commercial reactor.
- The technology of ASTRID allows to have a very resilient design to external events (earthquake, flooding, loss of power, airplane crash…).



3. Use of Digital in ASTRID Project

- Model Complex Phenomena to Consolidate Demonstrations
- Management of a Large Complex Project
- Advantages From the Use of Virtual Reality Description



4. Main Achievements for 2015

- 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.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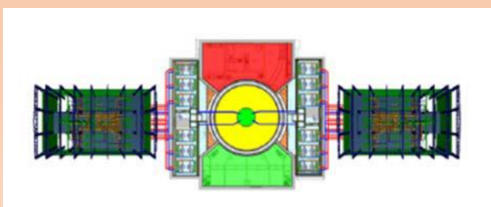
Direction de l'énergie nucléaire

Avancées des recherches sur la séparation-transmutation et le multirecyclage du plutonium dans les réacteurs à flux de neutrons rapides

Juin 2015



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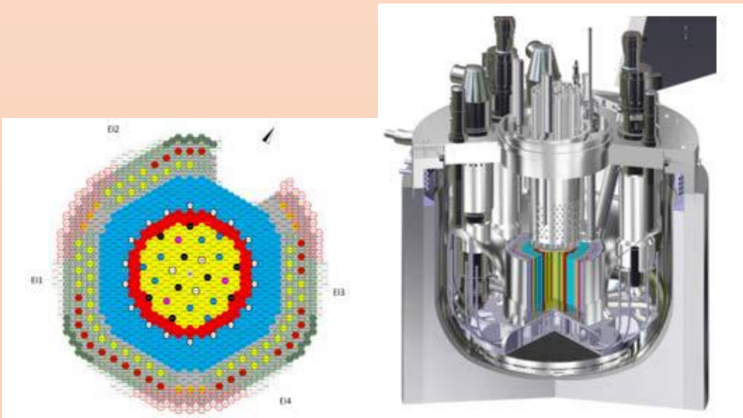
5. ASTRID Main Technical Choices

- 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
- Redundant and diversified decay heat removal systems
- Fuel handling in sodium + combination of internal storage and small external storage



6. Lesson Learned

To make to fulfill the Gen IV requirements, the new safety demonstration that we need to have, and also the cost investments that we have to reduce, it needs to get a close relationship between industry and design teams on one hand and the R&D teams on the other hand.

- 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

32

6-3. BN-600 and BN-800 Operating Experience

Summary / Objectives:

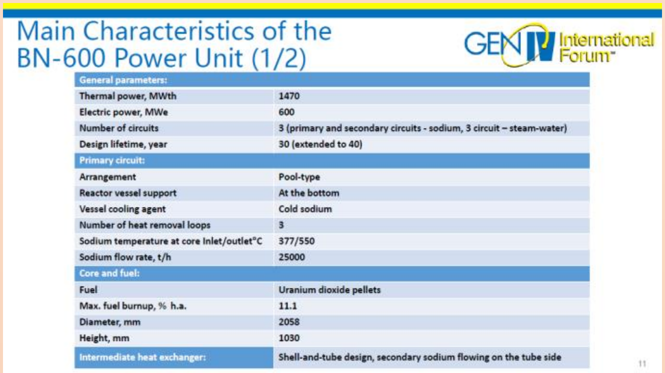
This presentation will first place the context of the choice of Sodium Fast Reactor in the French Nuclear Policy and its rationale for a closed fuel cycle. It will then present the position of the French Sodium Fast Reactor program in the context of Generation IV. The presentation will then focus on the ASTRID (Advanced Sodium Technological Reactor for Industrial Demonstration) project. The technical achievements, major innovation progress and management challenges will be presented. The ASTRID project description will highlight the major use of digital tools (numerical simulation, use of virtual reality, multiscale and multi-physics modeling, PLM: Product Lifecycle Management) used to perform efficiently such a complex project.

Meet the Presenter:

Mr. Ilya Pakhomov is the Head of Laboratory in the State Scientific Center of the Russian Federation - Institute for Physics and Power Engineering named after A.I. Leypunsky (IPPE). Since 2006, he has been charged with developing advanced sodium fast reactors as an engineer, junior researcher and head of laboratory. In 2014, he become a member of the working group on scientific and technical support of the BN-1200 project in IPPE. Currently, he is head of laboratory - management of experiments and engineering safety of fast sodium reactors. He is responsible for research of operability elements of the core, safety issues of sodium fires and safety during interloop leaks in the sodium-water steam generators. He is also involved in the formation of an R&D plan for the Fast Sodium Reactors.

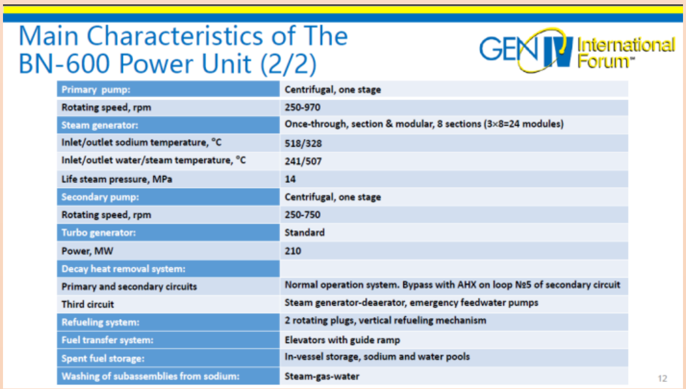
Long-term experiment of SFR in Russia and basic concept of BN-600:

The SFR development has been ongoing for more than 60 years in USSR and Russia, and multiple prototype and experimental reactors and industrial power units have been operated. The fundamental difference of BN-600 from previous SFR in Russia is pool type arrangement of primary coolant. The successful operation of BN-600 has been continued from 1980.


Main Characteristics of the BN-600 Power Unit (1/2)

General parameters:	
Thermal power, MWth	1470
Electric power, MWe	600
Number of circuits	3 (primary and secondary circuits - sodium, 3 circuit – steam-water)
Design lifetime, year	30 (extended to 40)
Primary circuit:	
Arrangement	Pool-type
Reactor vessel support	At the bottom
Vessel cooling agent	Cold sodium
Number of heat removal loops	3
Sodium temperature at core Inlet/outlet°C	377/550
Sodium flow rate, t/h	25000
Core and fuel:	
Fuel	Uranium dioxide pellets
Max. fuel burnup, % h.a.	11.1
Diameter, mm	2058
Height, mm	1030
Intermediate heat exchanger:	Shell-and-tube design, secondary sodium flowing on the tube side

11


Main Characteristics of The BN-600 Power Unit (2/2)

Primary pump:	Centrifugal, one stage
Rotating speed, rpm	250-970
Steam generator:	Once-through, section & modular, 8 sections (3×8=24 modules)
Inlet/outlet sodium temperature, °C	518/328
Inlet/outlet water/steam temperature, °C	241/507
Life steam pressure, MPa	14
Secondary pump:	Centrifugal, one stage
Rotating speed, rpm	250-750
Turbo generator:	Standard
Power, MW	210
Decay heat removal system:	
Primary and secondary circuits	Normal operation system. Bypass with AHX on loop №5 of secondary circuit
Third circuit	Steam generator-deaerator, emergency feedwater pumps
Refueling system:	2 rotating plugs, vertical refueling mechanism
Fuel transfer system:	Elevators with guide ramp
Spent fuel storage:	In-vessel storage, sodium and water pools
Washing of subassemblies from sodium:	Steam-gas-water

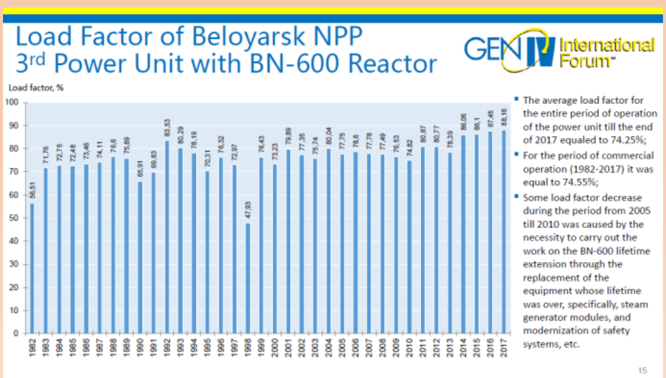
12

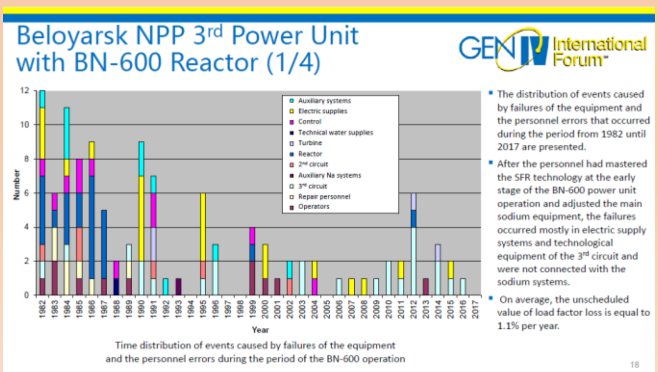
Core and load factor of BN-600:

The burnup design of BN-600 was gradually enhanced with core modification. The successful operation and research made it possible to increase the design value of fuel burnup up to 11.1 % h.a. and change over the longer fuel element life time with 4-hold reactor refueling.

The average load factor is 74.25% by 2017, and during 1982-2004, the load factor slightly decreased due to scheduled maintenance. Only 3 % of whole was due to failure of the equipment or personal errors. The failures mostly occurred in electric supply system and technical equipment of 3rd circuit.

The operating-time of SFR equipment testify to good compatibility of coolant with structural materials used and its low corrosion activity.

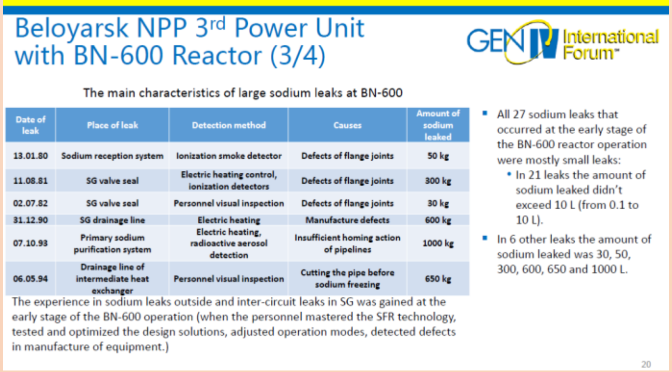




Sodium leaks:

The sodium leaks outside and inter-circuit leaks in SG was gained at the early stage of operation. 27 sodium leaks were detected and there were 14 cases sodium fires. The accumulated leaks experience proved the effectiveness of the protection systems, and no sodium leaks occurred in this 24 years.

Steam generator have demonstrated high performance characteristics and have operated without inter-circuit leaks for 27 years except 12 leaks in early stage of operation.



Beloyarsk NPP 3rd Power Unit with BN-600 Reactor (3/4)

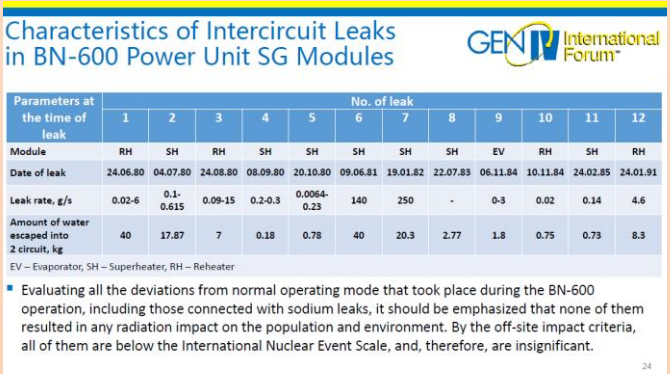
The main characteristics of large sodium leaks at BN-600

Date of leak	Place of leak	Detection method	Causes	Amount of sodium leaked
13.01.80	Sodium reception system	Ionization smoke detector	Defects of flange joints	50 kg
11.08.81	SG valve seal	Electric heating control, ionization detectors	Defects of flange joints	300 kg
02.07.82	SG valve seal	Personnel visual inspection	Defects of flange joints	30 kg
31.12.90	SG drainage line	Electric heating	Manufacture defects	600 kg
07.10.93	Primary sodium purification system	Electric heating, radioactive aerosol detection	Insufficient homing action of pipelines	1000 kg
06.05.94	Drainage line of intermediate heat exchanger	Personnel visual inspection	Cutting the pipe before sodium freezing	650 kg

The experience in sodium leaks outside and inter-circuit leaks in SG was gained at the early stage of the BN-600 operation (when the personnel mastered the SFR technology, tested and optimized the design solutions, adjusted operation modes, detected defects in manufacture of equipment.)

- All 27 sodium leaks that occurred at the early stage of the BN-600 reactor operation were mostly small leaks:
  - In 21 leaks the amount of sodium leaked didn't exceed 10 L (from 0.1 to 10 L).
- In 6 other leaks the amount of sodium leaked was 30, 50, 300, 600, 650 and 1000 L.

20



Characteristics of Intercircuit Leaks in BN-600 Power Unit SG Modules

Parameters at the time of leak	No. of leak 1	2	3	4	5	6	7	8	9	10	11	12
Module	RH	SH	RH	SH	SH	SH	SH	SH	EV	RH	SH	RH
Date of leak	24.06.80	04.07.80	24.08.80	08.09.80	20.10.80	09.06.81	19.01.82	22.07.83	06.11.84	10.11.84	24.02.85	24.01.91
Leak rate, g/s	0.02-6	0.1-0.615	0.09-15	0.2-0.3	0.0064-0.23	140	250	-	0-3	0.02	0.14	4.6
Amount of water escaped into 2 circuit, kg	40	17.87	7	0.18	0.78	40	20.3	2.77	1.8	0.75	0.73	8.3

EV – Evaporator, SH – Superheater, RH – Reheater

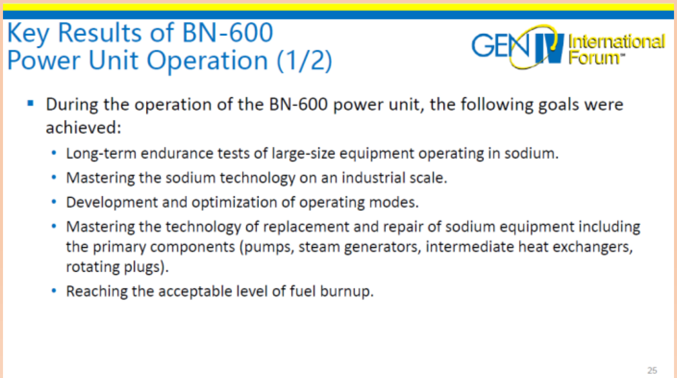
- Evaluating all the deviations from normal operating mode that took place during the BN-600 operation, including those connected with sodium leaks, it should be emphasized that none of them resulted in any radiation impact on the population and environment. By the off-site impact criteria, all of them are below the International Nuclear Event Scale, and, therefore, are insignificant.

24

Key result of BN-600:

During the operation of BN-600, many kind of goals were achieved in addition to more than 147.4 billion kWh of electricity production. On of most important results is the fact that the design parameters for sodium large-scale equipment operation period and life time have been achieved and even exceeded.

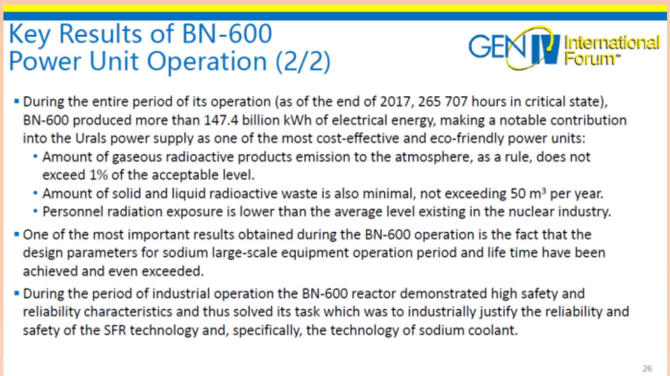
The life time of BN-600 was extended 10 years in 2010 and activities are currently underway to re-extend by 2020.



Key Results of BN-600 Power Unit Operation (1/2)

- During the operation of the BN-600 power unit, the following goals were achieved:
  - Long-term endurance tests of large-size equipment operating in sodium.
  - Mastering the sodium technology on an industrial scale.
  - Development and optimization of operating modes.
  - Mastering the technology of replacement and repair of sodium equipment including the primary components (pumps, steam generators, intermediate heat exchangers, rotating plugs).
  - Reaching the acceptable level of fuel burnup.

25



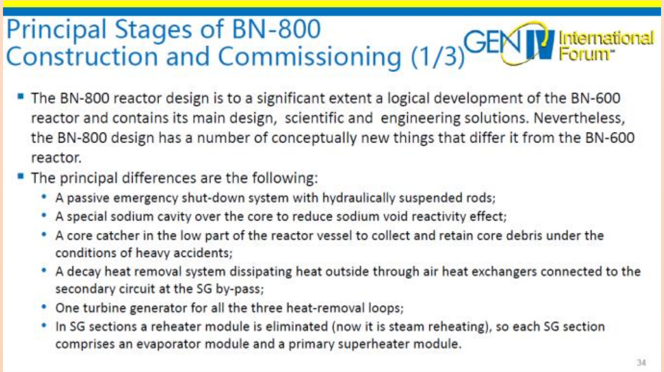
Key Results of BN-600 Power Unit Operation (2/2)

- During the entire period of its operation (as of the end of 2017, 265 707 hours in critical state), BN-600 produced more than 147.4 billion kWh of electrical energy, making a notable contribution into the Urals power supply as one of the most cost-effective and eco-friendly power units:
  - Amount of gaseous radioactive products emission to the atmosphere, as a rule, does not exceed 1% of the acceptable level.
  - Amount of solid and liquid radioactive waste is also minimal, not exceeding 50 m³ per year.
  - Personnel radiation exposure is lower than the average level existing in the nuclear industry.
- One of the most important results obtained during the BN-600 operation is the fact that the design parameters for sodium large-scale equipment operation period and life time have been achieved and even exceeded.
- During the period of industrial operation the BN-600 reactor demonstrated high safety and reliability characteristics and thus solved its task which was to industrially justify the reliability and safety of the SFR technology and, specifically, the technology of sodium coolant.

26

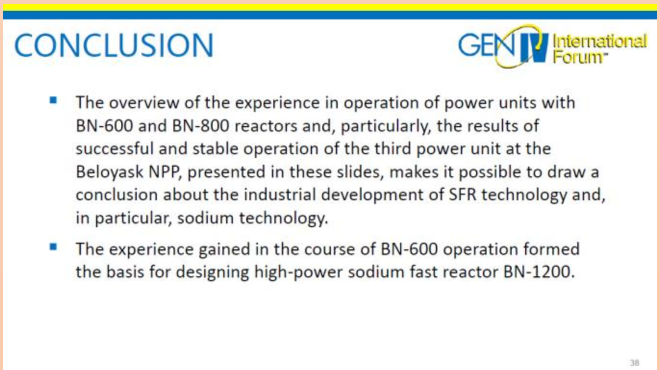
Basic concept of BN-800:

One of main issue of BN-800 is the demonstration of closed fuel cycle. The hybrid core system with both of MOX and enriched uranium fuels are used. BN-800 was designed based on BN600 design but it has number of new things including safety systems. BN-800 has operated 14543 hours and generated 9.4 billion kWh of electricity by the end of 2017.



Prospect for further SFR development in Russia and conclusion:

In compliance with further objectives in development and improvement of SFR technologies, demonstration of closed fuel cycle, commercialization of SFR technology, and development of large-scale SFR technology are highlighted.



7-1. Estimating Costs of Generation IV Systems

Summary / Objectives:

This webinar will provide an overview of the Economic Modelling Working Group's Cost Estimating Guidelines for Generation IV Nuclear Energy Systems (GIF, 2007). Topics include an overview of the Guidelines, a comparison of the Guidelines with other nuclear power plant cost estimating models, and a discussion of benchmarking activities by the EMWG with INPRO.

Meet the Presenter:

Dr. Geoffrey Rothwell since 2013 has been the Principal Economist of the Nuclear Energy Agency (NEA) of the Organization for Economic Cooperation and Development (OECD, Paris, France), where he acts as the Secretariat for the Economic Modelling Working Group (EMWG). For EMWG he wrote the TOR in 2003 as the Chair of the Economics Cross-cut Group of the Generation IV Roadmap Committee. He was active in writing the Cost Estimating Guidelines for Generation IV Nuclear Energy Systems (GIF, 2007). While teaching at Stanford University from 1986-2013, he consulted to Idaho, Oak Ridge, and Pacific Northwest, and Argonne National Laboratories, for whom he updated the University of Chicago's 2004 report, The Economic Future of Nuclear Power, published as The Economics of Nuclear Power, Routledge, London, 2016. Dr. Rothwell grew up in Richland, Washington, and received his PhD in economics from the University of California, Berkeley.

Looking back over the startup phase of the GIF-EMWG:

Economic Modeling Working Group (EMWG) created to define the economic criteria for selecting GIF supported technologies (GIF systems) by the cross-cutting Evaluation Methodology Group (EMG) composing the early Gen-IV Roadmap Committee which selects GIF systems. Two economic criteria: EC-1 low total capital investment cost, and EC-2 low average cost, levelized unit energy costs, LUEC were selected, "Cost Estimating Guideline" and a transparent cost estimating tool, G4-ECONS, were developed by EMWG in 2007.

EVALUATION METHODOLOGY GROUP, EMG (2001-2003)

The EMG was tasked with developing a multi-criteria evaluation to be applied by the technical working groups to some 80 variants of nuclear energy systems for the selection of the most promising technologies.

The EMG developed four sets of criteria:

(1) safety
(2) economic
(3) sustainability
(4) non-proliferation and physical protection

The economic goals were

(1) To have a clear life-cycle cost advantage over other energy sources, and
(2) To have a level of financial risk comparable with other energy projects

5

GIF/EMWG/2007/004

COST ESTIMATING GUIDELINES FOR GENERATION IV NUCLEAR ENERGY SYSTEMS

Revision 4.2

September 26, 2007

Prepared by

The Economic Modeling Working Group Of the Generation IV International Forum

Printed by the OECD Nuclear Energy Agency for the Generation IV International Forum

https://www.gen-4.org/gif/upload/docs/application/pdf/2013-09/emwg_guidelines.pdf

The EMG defined the Terms of Reference for the GIF Methodology Working Groups, one of which was the Economic Modeling Working Group (EMWG), which prepared the *Cost Estimating Guidelines for Generation IV Nuclear Energy Systems* (2007).

The "*Cost Estimating Guidelines*" defined a Code of Accounts (COA) with which the TCIC and LUEC are defined.

7

Code of Accounts and LUEC:

GIF Code of Account (COA) developed for estimated LUEC. COA is bottom-up approach to accumulate the total capital investment cost (TCIC). LUEC composed by annualized TCIC, Operation and Maintenance (O&M), and Fuel costs.

LEVELISED UNIT ENERGY COST (LUEC) in dollars, euros, etc. per megawatt-hour =

KC Capital Cost is equal to the payments each year to the banks and investors, like a annual mortgage payment, to pay down the Total Capital Investment Cost

+

O&M is the *annual* Operations and Maintenance (O&M) expense and Capital Additions, CAPEX

+

FUEL is the *annual* fuel payment, a function of the amount and price of fuel

the sum of which is divided by the annual energy output

E in megawatt-hours (MWh) equal to the product of
MW, the size of the generator in megawatts,
TT, the total number of hours in a year, and
CF, the Capacity Factor

Step 1: Calculate KC from TCIC

Step 2: Calculate O&M and FUEL

Step 3: Divide by E and calculate LUEC

Source: Rothwell, Economics of Nuclear Power (2016, p. 154). London: Routledge.
https://www.routledge.com/Economics-of-Nuclear-Power/Rothwell/p/book/9781138858411

9

The GIF Code of Accounts (COA):

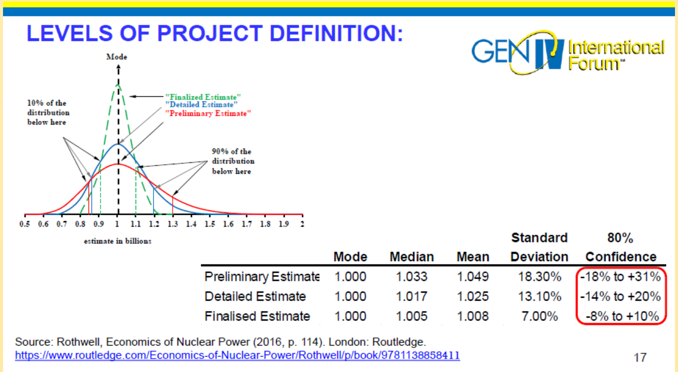
Account Number	Account Title
10	Capitalized Pre-Construction Costs
20	Capitalized Direct Costs
	21 Structures and Improvements
	22 Reactor Equipment
	23 Turbine Generator Equipment
	24 Electrical Equipment
	25 Heat Rejection System
	26 Miscellaneous Equipment
	27 Special Materials
Direct Cost	
30	Capitalized Indirect Services Costs
	35 Design Services Offsite
	36 PM/CM Services Offsite
	37 Design Services Onsite
	38 PM/CM Services Onsite
= Base Construction Cost	
+ 40	Capitalized Owner's Costs
+ 50	Capitalized Supplementary Costs
	+ 55 Initial Fuel Core Load
= Overnight Construction Cost	
+ 60	Capitalized Financial Costs
	+ 63 Interest During Construction
+19+29+39+49+59+69=	Contingencies
= Total Capital Investment Cost	

Account Number	Account Title
70	Annualized O&M Costs
	71 O&M Staff
	72 Management Staff
	73 Salary-Related Costs
	74 Operations Chemicals and Lubricants
	75 Spare Parts
	76 Utilities, Supplies, and Consumables
	77 Capital Plant Upgrades
	78 Taxes and Insurance
	79 Contingency on Annualized O&M Costs
80	Annualized Fuel Cost
	81 Refueling Operations
	84 Nuclear Fuel
	86 Fuel reprocessing Charges
	87 Special Nuclear Materials
	89 Contingency on Annualized Fuel Costs
90	Annualized Financial Costs
	92 Fees
	93 Cost of Capital
	99 Contingency on Annualized Financial Costs

11

TCIC:

TCIC composed by Direct cost, Indirect Services Costs, Owner's Costs, financial cost, interest during construction (IDC) and contingencies. TCIC except financial, interest and contingency costs is called as overnight cost. Some case consider Initial Fuel Core Load cost as fuel cost but this case consider this as TCIC because this cost is significant as initial cost. The overnight cost of Molten Salt Reactor (MSR) estimated by Oak Ridge National Laboratory (ORNL) was \$3350/kWe (2011USD) for example. IDC estimated depend on construction period. Estimation of appropriate contingency is needed. The rate of contingency could be decrease in stage of project definition. TCIC was estimated by ORNL in 2011 as \$3149/kWe for the Advanced High Temperature Reactor (AHTR) System with 9% enriched uranium compare with \$4012 of PWR12 for example.



	Mode	Median	Mean	Standard Deviation	80% Confidence
Preliminary Estimate	1.000	1.033	1.049	18.30%	-18% to +31%
Detailed Estimate	1.000	1.017	1.025	13.10%	-14% to +20%
Finalised Estimate	1.000	1.005	1.008	7.00%	-8% to +10%

Source: Rothwell, Economics of Nuclear Power (2016, p. 114). London: Routledge.
https://www.routledge.com/Economics-of-Nuclear-Power/Rothwell/p/book/9781138858411
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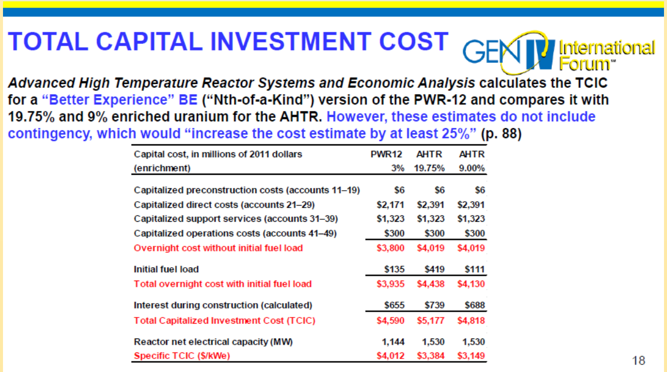


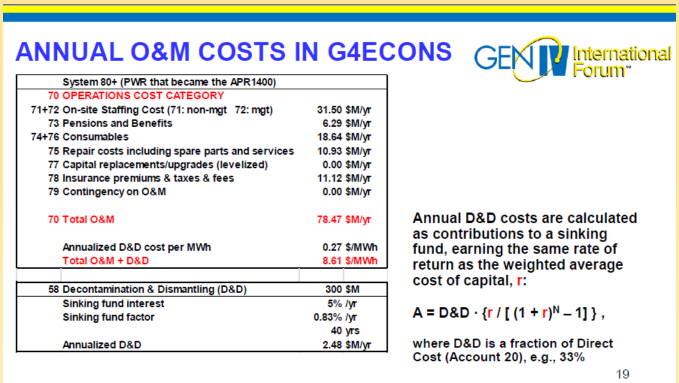
Advanced High Temperature Reactor Systems and Economic Analysis calculates the TCIC for a "Better Experience" BE ("Nth-of-a-Kind") version of the PWR-12 and compares it with 19.75% and 9% enriched uranium for the AHTR. However, these estimates do not include contingency, which would "increase the cost estimate by at least 25%" (p. 88)

Capital cost, in millions of 2011 dollars (enrichment)	PWR12 3%	AHTR 19.75%	AHTR 9.00%
Capitalized preconstruction costs (accounts 11–19)	\$6	\$6	\$6
Capitalized direct costs (accounts 21–29)	\$2,171	\$2,391	\$2,391
Capitalized support services (accounts 31–39)	\$1,323	\$1,323	\$1,323
Capitalized operations costs (accounts 41–49)	\$300	\$300	\$300
Overnight cost without initial fuel load	\$3,800	\$4,019	\$4,019
Initial fuel load	\$135	\$419	\$111
Total overnight cost with initial fuel load	\$3,935	\$4,438	\$4,130
Interest during construction (calculated)	\$655	\$739	\$688
Total Capitalized Investment Cost (TCIC)	\$4,590	\$5,177	\$4,818
Reactor net electrical capacity (MW)	1,144	1,530	1,530
Specific TCIC (\$/kWe)	\$4,012	\$3,384	\$3,149

18

O&M and Fuel Costs:

Such kind of staffing cost and repair cost are estimated as O&M cost. Decontamination & Dismantling (D&D) cost are estimated as contributions to a sinking fund. Fuel cost includes front end and backend cost. Fuel cost was estimated as \$10.74/MWh for AHTR System with 9% enriched uranium compare with \$5.60 of PWR12 for example.



System 80+ (PWR that became the APR1400)	
70 OPERATIONS COST CATEGORY	
71+72 On-site Staffing Cost (71: non-mgt 72: mgt)	31.50 \$M/yr
73 Pensions and Benefits	6.29 \$M/yr
74+76 Consumables	18.64 \$M/yr
75 Repair costs including spare parts and services	10.93 \$M/yr
77 Capital replacements/upgrades (levelized)	0.00 \$M/yr
78 Insurance premiums & taxes & fees	11.12 \$M/yr
79 Contingency on O&M	0.00 \$M/yr
70 Total O&M	78.47 \$M/yr
Annualized D&D cost per MWh	0.27 \$/MWh
Total O&M + D&D	8.61 \$/MWh

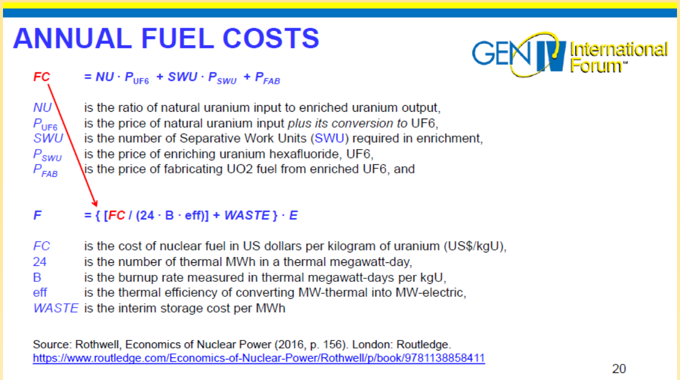
58 Decontamination & Dismantling (D&D)	300 \$M
Sinking fund interest	5% /yr
Sinking fund factor	0.83% /yr
	40 yrs
Annualized D&D	2.48 \$M/yr

Annual D&D costs are calculated as contributions to a sinking fund, earning the same rate of return as the weighted average cost of capital, r:

$$A = D\&D \cdot \{ r / [(1 + r)^N - 1] \},$$

where D&D is a fraction of Direct Cost (Account 20), e.g., 33%

19



$$FC = NU \cdot P_{UF6} + SWU \cdot P_{SWU} + P_{FAB}$$

NU is the ratio of natural uranium input to enriched uranium output,
P_{UF6} is the price of natural uranium input *plus its conversion to* UF6,
SWU is the number of Separative Work Units (SWU) required in enrichment,
P_{SWU} is the price of enriching uranium hexafluoride, UF6,
P_{FAB} is the price of fabricating UO2 fuel from enriched UF6, and

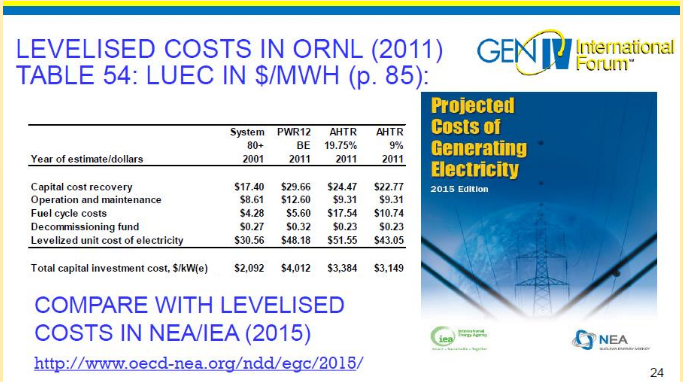
$$F = \{ [FC / (24 \cdot B \cdot eff)] + WASTE \} \cdot E$$

FC is the cost of nuclear fuel in US dollars per kilogram of uranium (US\$/kgU),
24 is the number of thermal MWh in a thermal megawatt-day,
B is the burnup rate measured in thermal megawatt-days per kgU,
eff is the thermal efficiency of converting MW-thermal into MW-electric,
$WASTE$ is the interim storage cost per MWh

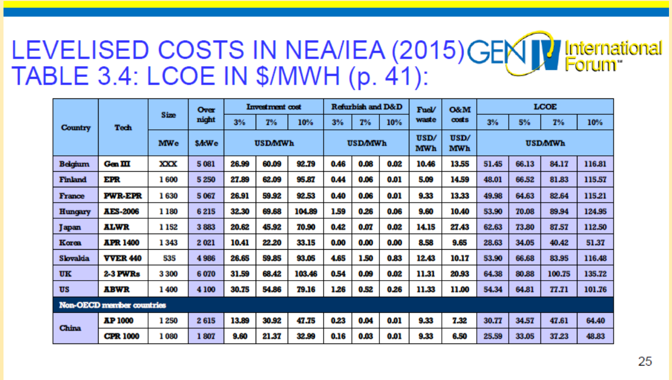
Source: Rothwell, Economics of Nuclear Power (2016, p. 156). London: Routledge.
https://www.routledge.com/Economics-of-Nuclear-Power/Rothwell/p/book/9781138858411
20

Cost estimation of LUEC by ORNL and NEA:

ORNL estimated as \$30.56 /MWh for System 80+, \$48.18/MWh, \$43.05/MWh for AHTR System with 9% enriched uranium. NEA is regularly reporting the estimated levelized cost of each counties. Relatively low overnight cost was estimated for AR1400 in Korea and AP1000/CPR1000 in China.



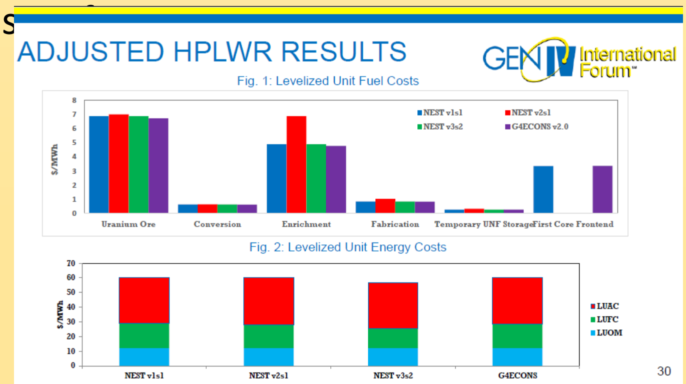
Year of estimate/dollars	System 80+ 2001	PWR12 BE 2011	AHTR 19.75% 2011	AHTR 9% 2011
Capital cost recovery	\$17.40	\$29.66	\$24.47	\$22.77
Operation and maintenance	\$8.61	\$12.60	\$9.31	\$9.31
Fuel cycle costs	\$4.28	\$5.60	\$17.54	\$10.74
Decommissioning fund	\$0.27	\$0.32	\$0.23	\$0.23
Levelized unit cost of electricity	\$30.56	\$48.18	\$51.55	\$43.05
Total capital investment cost, \$/kW(e)	\$2,092	\$4,012	\$3,384	\$3,149

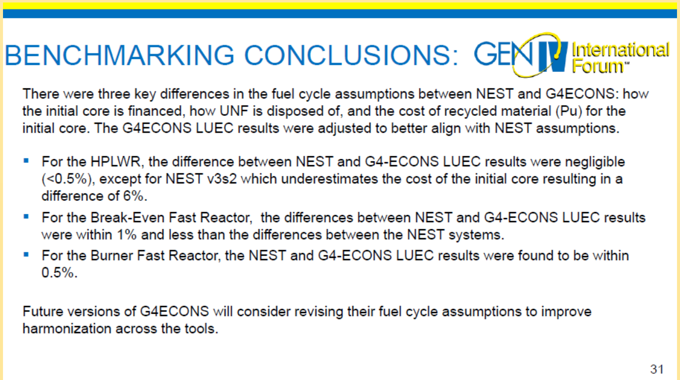


Country	Tech	Size	Over night	Investment cost			Refurbish and D&D			Fuel/ waste	O&M costs	LCOE			
				3%	7%	10%	3%	7%	10%			3%	5%	7%	10%
		MWe	\$/kWe	USD/MWh			USD/MWh			USD/ MWh	USD/ MWh	USD/MWh			
Belgium	Gen III	XXX	5 081	26.99	60.09	92.79	0.46	0.08	0.02	10.46	13.55	51.45	66.13	84.17	116.81
Finland	EPR	1 600	5 250	27.89	62.09	95.87	0.44	0.06	0.01	5.09	14.59	48.01	66.52	81.83	115.57
France	PWR-EPR	1 630	5 067	26.91	59.92	92.53	0.40	0.06	0.01	9.33	13.33	49.98	64.63	82.64	115.21
Hungary	AES-2006	1 180	6 215	32.30	69.68	104.89	1.59	0.26	0.06	9.60	10.40	53.90	70.08	89.94	124.95
Japan	ALWR	1 152	3 883	20.62	45.92	70.90	0.42	0.07	0.02	14.15	27.43	62.63	73.80	87.57	112.50
Korea	APR 1400	1 343	2 021	10.41	22.20	33.15	0.00	0.00	0.00	8.58	9.65	28.63	34.05	40.42	51.37
Slovakia	VVER 440	535	4 986	26.65	59.85	93.05	4.65	1.50	0.83	12.43	10.17	53.90	66.68	83.95	116.48
UK	2-3 PWRs	3 300	6 070	31.59	68.42	103.46	0.54	0.09	0.02	11.31	20.93	64.38	80.88	100.75	135.72
US	ABWR	1 400	4 100	30.75	54.86	79.16	1.26	0.52	0.26	11.33	11.00	54.34	64.81	77.71	101.76
Non-OECD member countries															
China	AP 1000	1 250	2 615	13.89	30.92	47.75	0.23	0.04	0.01	9.33	7.32	30.77	34.57	47.61	64.40
	CPR 1000	1 080	1 807	9.60	21.37	32.99	0.16	0.03	0.01	9.33	6.50	25.59	33.05	37.23	48.83

Benchmarking G4-ECONS and NEST developed by IAEA:

NEST was developed in 4 phases by IAEA, and it was extended to treat designs of break-even closed fuel cycle and multiple conversion rates in Version 4. The benchmark study between G4-ECONS and NEST was carried out with selected thermal reactor (high performance LWR by KIT) and fast reactor (BN-800 by Rosatom) and identified little deference but not s



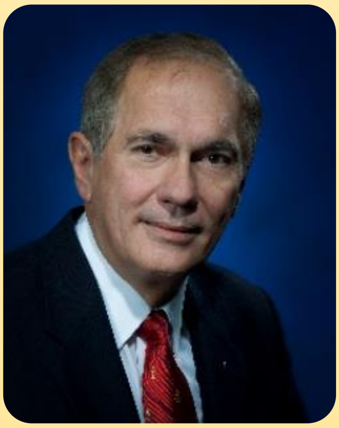


7-2. Proliferation Resistance and Physical Protection of Generation IV Reactor Systems

Summary / Objectives:

This webinar will provide an overview of the activities of the Generation IV Proliferation Resistance and Physical Protection Working Group. Topics include a presentation of the methodology developed by the group, an illustration of the methodology to an example nuclear system, and a summary of ongoing interactions between the group and the designers of the six Generation IV nuclear energy systems. Other outreach activities of the group associated with various national and international organizations will be briefly summarized.

Meet the Presenter:



Dr. Robert A. Bari is Senior Scientist Emeritus at Brookhaven National Laboratory and has over 40 years of experience in nuclear energy research. He has performed studies on safety, security and nonproliferation of advanced nuclear concepts. For 15 years Dr. Bari was co-chairman of the working group on proliferation resistance and physical protection of the Generation IV International Forum. He has served on the Board of Directors of the American Nuclear Society and as President of the International Association for Probabilistic Safety Assessment and Management. Dr. Bari was awarded the Theo J. "Tommy" Thompson Award in 2003 by the American Nuclear Society. In 2004, he received the Brookhaven National Laboratory Award for Outstanding Achievement in Science and Technology. Dr. Bari is a fellow of the American Nuclear Society and of the American Physical Society. He has participated in risk-based standards development for nuclear technologies for more than two decades. He has been a committee member of the U. S. National Academy of Sciences on Lessons Learned from the Fukushima Nuclear Accident for Improving Safety and Security of the U.S. Nuclear Plants. Dr. Bari also chaired a workshop of the U. S. National Academy of Sciences on safety and security culture held jointly between the U.S. and Brazil in 2014. He received his doctorate from Brandeis University (1970) and his bachelor's degree from Rutgers University (1965). He was awarded membership in the Phi Beta Kappa, Sigma Xi, and Sigma Pi Sigma honor societies.

Getting PR&PP Right!

The next Hiroshima/Nagasaki must be prevented.

Peace Statue in Nagasaki Peace Park

Definitions

- *Proliferation resistance* is that characteristic of a nuclear energy system that impedes the diversion or undeclared production of nuclear material, or misuse of technology, *by the host State* in order to acquire nuclear weapons or other nuclear explosive devices.
- *Physical protection* (robustness) is that characteristic of a nuclear energy system that impedes the theft of materials suitable for nuclear explosives or radiation dispersal devices, and the sabotage of facilities and transportation, *by sub-national entities and other non-host State* adversaries.

Distinction is important to articulate

PR&PP Group Major Products

- Methodology for PR&PP Evaluation
- Example Case Study
- Gen IV System Comparison Study
- Supporting Products:
  - PR&PP bibliography
  - PR&PP FAQ
- …and ongoing interactions with Gen IV designers

For reports see: https://www.gen-4.org/gif/jcms/c_9365/prpp

Value of PR&PP Evaluations for Future Designs

- Introduce PR&PP features into the design process at the earliest possible stage of concept development
- As the design matures, increasing detail can be incorporated in the PR&PP model of the system: progressive refinement
- PR&PP results can inform choices by policy makers

System Response

- Pathway analysis: Intuitive way to describe & analyze proliferation, theft, or sabotage scenarios and to identify vulnerabilities
- Segmentation & decomposition, then re-aggregation

System Response (cont'd)

- Pathways: Potential sequences of events followed by the proliferator or adversary to achieve its objectives
  - Along any pathway the proliferant state or adversary will encounter various difficulties, barriers, or obstacles, all of which are collectively called "proliferation resistance" or "physical protection robustness"
- Considers time-dependent aspects and uncertainty

CASE STUDY: EXAMPLE SODIUM FAST REACTOR (ESFR)

Case Study Objectives

- Demonstrate the Methodology for an entire system
- Confirm applicability at different levels of design detail
- Provide examples of PR&PP evaluations for future users of the Methodology
- Determine the needs for further methodology development

GEN IV International Forum℠
Expertise | Collaboration | Excellence

7-3. Materials Challenges for Generation IV Reactors

Summary / Objectives:

The Generation IV reactors offer significant advantages over typical light water reactors including increased power conversion efficiency, passive safety features and in some cases process heat for other applications (e.g. hydrogen production). These families of reactors include 3 fast reactors [sodium fast reactor (SFR), lead fast reactor (LFR) and gas-cooled fast reactor (GFR)], one thermal reactor [very high temperature reactor (VHTR)] and two fast or thermal reactors [supercritical water reactor (SCWR) and molten salt reactor (MSR)]. The extreme environments in these families of reactors create significant challenges to materials ranging from high doses from a fast neutron flux (SFR, LFR, GFR, SCWR and MSR), more corrosive environments from molten salt (MSR) or lead coolants (LFR) and high temperatures in the helium-cooled reactor concepts (e.g. GFR and VHTR). This presentation will discuss the materials challenges in Generation IV reactor concepts and summarize radiation effects in reactor metals proposed for these concepts over prototypic irradiation conditions

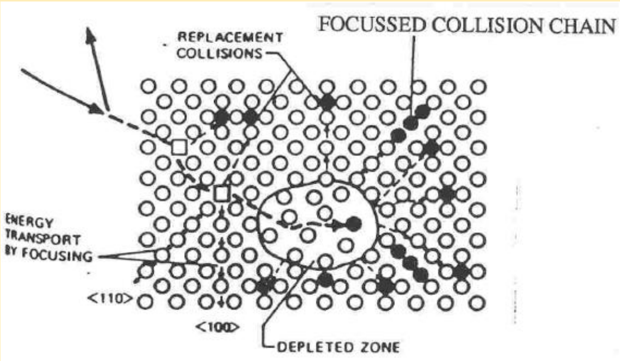
Meet the Presenter:

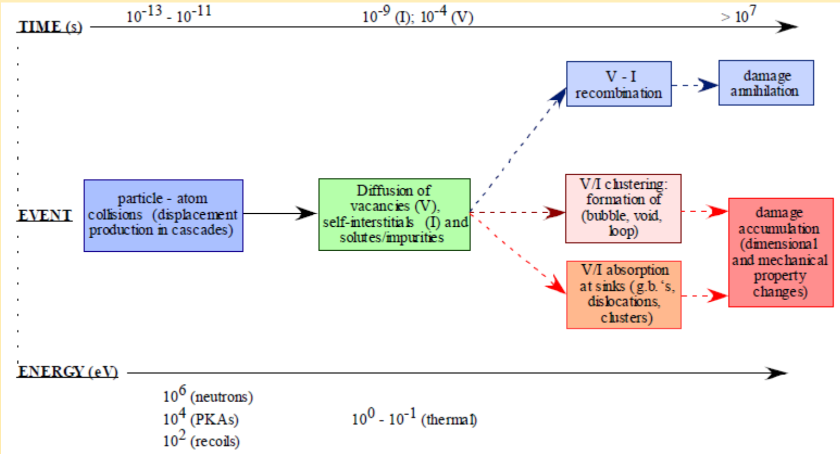
Stuart Maloy is a Team Leader for MST-8 (materials at radiation and dynamic extremes) at Los Alamos National Laboratory and is the advanced reactor core materials technical leader for the Nuclear Technology Research and Development's Advanced Fuels campaign and the NEET Reactor Materials Technical Lead for DOE-NE.

He has applied his expertise to characterizing and testing the properties of metallic and ceramic materials in extreme environments such as under neutron and proton irradiation at reactor relevant temperatures. This includes testing the mechanical properties (fracture toughness and tensile properties) of Mod 9Cr-1Mo, HT-9, 316L, 304L, Inconel 718, Al6061- T6 and Al5052 after high energy proton and neutron irradiations using accelerators and fast reactors.

Radiation Damage :

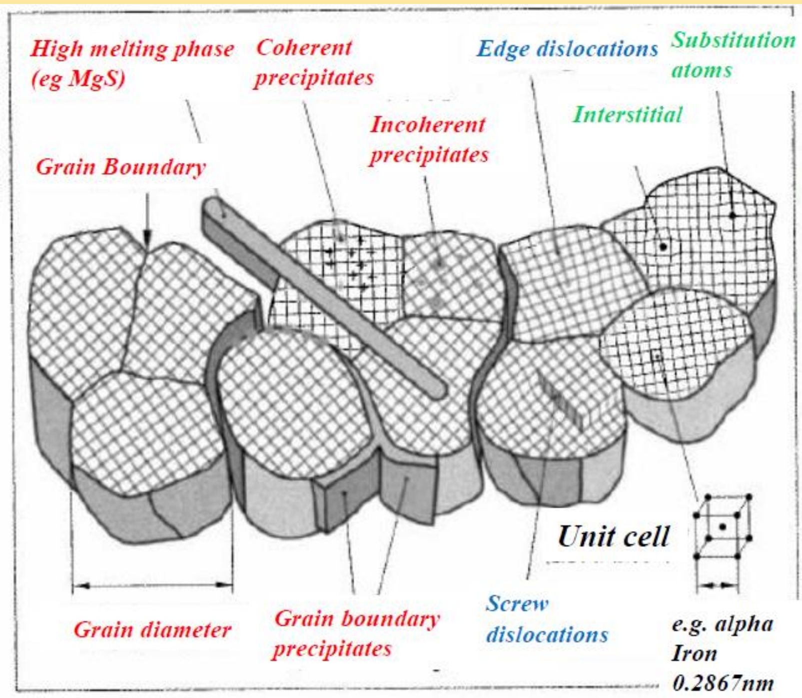
Displacement damage occurs when enough energy (approximately 25 eV) is transferred to an atom producing a or many Frenkel defects. Though a large number of Frenkel defects (vacancy / self-interstitials) annihilated in short time, some defects remain and make cluster.





A wide range of materials properties are determined on the mesoscale :

As the result of the clustering, the accumulated defect grows to mesoscale. Unlike with nanoscale defects, mesoscale defects affect the various material properties. This is the mechanism of the radiation damage.



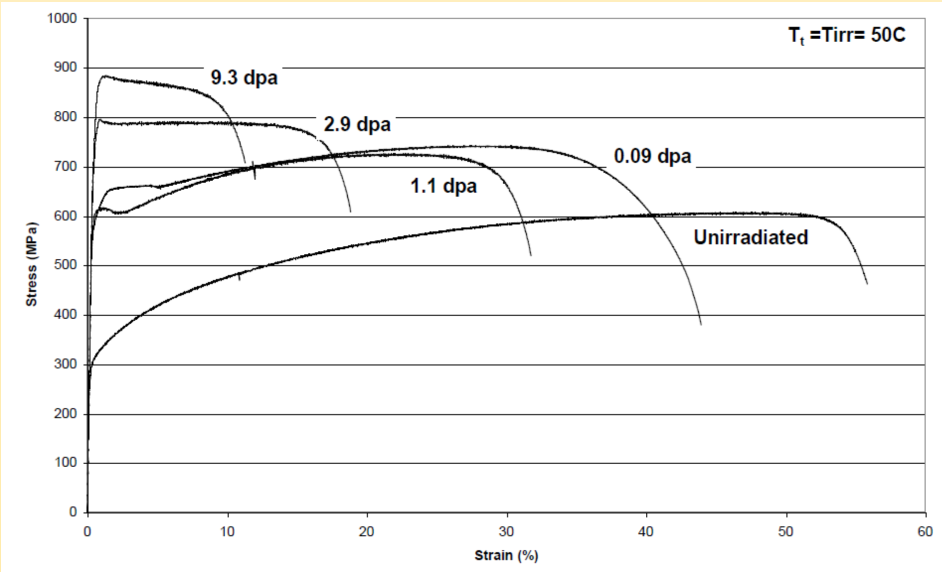
Green=nanoscale defects

Blue= nanoscale-meso scale

Red=mesoscale defects (actually defect dependent

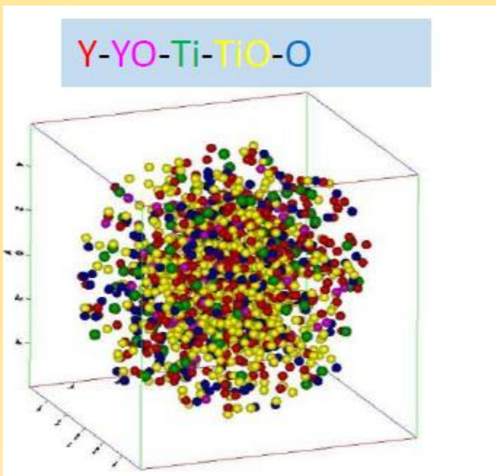
Stress/Strain curves of 316L stainless steel after irradiation :

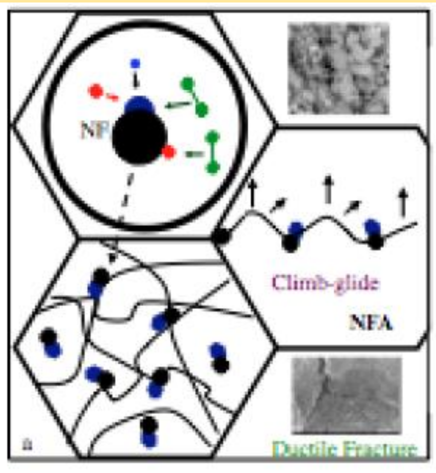
By the irradiation, yield stress of 316L stainless steel is increased (hardening) and elongation is decreased (embrittlement).

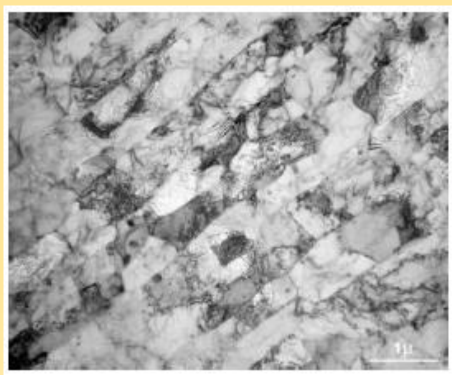


Nanostructured Ferritic Alloys :

Nanostructured ferritic alloys (or Oxide Dispersion Strengthen alloys, ODS) , which is made by mechanical alloying, have a fine distribution of oxide particles nano features within the material. This nanostructure brings increase of the strength, creep resistance, irradiation resistance. Therefore, these alloys show promise as advanced radiation tolerant materials.







Reactor operating conditions :

Each GIF systems have particular operating conditions:

- Coolant
- Temperature
- Lifetime Dose

Reactor Type	Fuel Materials	Fuel Temperature	Pellet to Clad bond	Coolant Type	Structural Materials for Core Internals	Lifetime Dose (dpa)	Structural Temperatures
Gen IV/ Lead Fast Reactor LFR	U/PuN; TRUN (enriched to N^{15})	500-600C	Lead	Pb or LBE	Ferritic/Martensitic Steel alloys	150-200	400-600C
Gen IV/ Sodium Fast Reactor SFR	Metal(U-TRU-10%Zr Alloy), MOX(TRU bearing)	600-800C (metal fuel) 800-2000C (Oxide fuel)	Sodium	Sodium	Ferritic/Martensitic Steel alloys	150-200	400-550C
Gen IV/ Gas cooled Fast Reactor GFR	UPuC/SiC (50/50%) with 20% Pu content ; Solid Solution fuel with SiC/SiC cladding	2000 +	Helium	Helium	Nickel Superalloys /Ceramic Composites	80	500-1200C
Fusion Energy	N/A	N/A	N/A	Pb-Li	F/M steels; Vanadium alloys; Ceramics	150	300-1000C
LWR – PWR, BWR	UO2	800-1600C	Helium	Water	316L.ferritic pressure vessel, Zircalloy cladding	Cladding ~10 dpa, Internals up to 80 dpa	200-300C
Very High Temperature Reactor (VHTR, NGNP)	TRISO	800-2000C	Intimate contact	Helium	Ni-based alloys, ceramics and graphite	~10 dpa	700-1000C
Supercritical Water Reactor (SCWR)	UO2	800-2000C	Helium	Water	F/M steels, austenitic steels	10-30 thermal 100-150 Fast	300-600C
Molten Salt Reactor (MSR)	Na, Zr, U, Pu fluorides	700-800C	N/A	N/A	Ni-based alloys, graphite	100-150 dpa	600-800C

Materials Performance Issue :

Because of the difference of operating condition, each GIF systems have particular material performance issues.

Reactor type	Primary Materials	Performance Issues
Light Water Reactors (PWR/BWR)	Ferritic pressure vessel steels, Fe-based austenitic stainless steels, zirconium alloys	IGSCC, IASCC, Fuel clad mechanical interaction, hydriding, Radiation embrittlement (DBTT), hydrogen embrittlement
Very High Temperature Reactor (VHTR)	Ni-based superalloys, Graphite, ferritic/martensitic steels, W/Mo Alloys, SiC/SiC composites	Helium embrittlement, creep strength, swelling, RIS, transmutation, toughness, oxidation
Sodium Fast Reactor (SFR)	Fe-based austenitic SS, Ferritic/martensitic steels,	Radiation Embrittlement (DBTT), toughness, helium embrittlement, swelling, RIS, corrosion, FCCI
Lead Fast Reactor (LFR)	Fe-based austenitic SS, Ferritic/martensitic steels,	Radiation Embrittlement (DBTT), toughness, helium embrittlement, swelling, RIS, corrosion, FCCI, liquid metal embrittlement
Supercritical Water Reactor (SCWR)	Ferritic pressure vessel steels, Fe-based austenitic stainless steels, zirconium alloys, ferritic/martensitic steels	IGSCC, IASCC, Fuel clad mechanical interaction, hydriding, Radiation/helium embrittlement (DBTT), swelling, RIS, corrosion, toughness
Gas Fast Reactor	Ceramics (carbides, nitrides), ceramic composites, nickel superalloys	Helium embrittlement, creep strength, swelling, RIS, transmutation, toughness, oxidation
Molten Salt Reactor	Ni-based alloys, graphite, coatings	Corrosion, Helium embrittlement, creep strength, swelling, RIS, transmutation, toughness, oxidation

7-4. Energy Conversion

Summary / Objectives:

The rotary motion, high pressure steam engine was patented by James Watt in 1781. The evolution of steam engines and high pressure boiler technology led directly to the development of the steam turbine coupled to an electrical generator by Charles Parsons in 1884. Since then, over the last 133 years, the world has been using steam turbines to convert heat into electricity in almost all of the world's thermal power stations and in all of the world's nuclear power stations. Specifically for the latter, steam turbines and the Rankine thermodynamic cycle in which they operate offer high efficiency for moderate steam temperatures, temperatures typical of first, second and third generation nuclear reactors. Generation IV reactors offer the potential to move away from the steam Rankine cycle to systems such as helium (or nitrogen) Brayton or supercritical CO2 gas turbine cycles to exploit the higher temperatures that some of the systems generate, to offer plant simplification and potentially higher conversion efficiencies. Non-steam cycles offer other advantages, particularly in connection with the sodium cooled fast reactor, such that the risk of sodium water reactions is massively reduced. Within this webcast, the basic thermodynamics and performance limits of energy conversion systems will be explained and each of the technological options proposed for the energy conversion systems of Generation IV reactors will be presented..

Meet the Presenter:

Dr. Richard Stainsby is a mechanical engineer with a PhD in computational fluid dynamics and heat transfer. He is Chief Technologist for Advanced Reactors and Fuel Cycles at the UK's National Nuclear Laboratory, having worked both in research facilities and industry before joining NNL. He has spent the last 32 years working on light water, high temperature gas (HTGR) and liquid metal and gas fast reactors. He has worked on contracts for PBMR in South Africa on core design and whole plant simulation, for the National Nuclear Regulator, also in South Africa, and for the USNTRC on the development of licensing tools for HTGRs. He is a past Chair of the GIF GFR System Steering Committee and a current Euratom member of the GIF SFR System Steering Committee. He has led two European projects (GCFR-STREP and GoFastR) on gas cooled fast reactors (GFR) and was a leader of the innovative architecture and balance of plant sub-project within the Euratom CP-ESFR project between 2009-2013.

The linkage between a nuclear reactor and its power conversion system : The reactor must supply a flow of heat that is controllable and of sufficient quality to match the requirements of the power conversion system (or engine). The engine must supply a stable flow of coolant to the reactor inlet that respects its material limits and neutronic requirements. A reactor is a temperature dependent heat source not fuel flow dependent as in a fossil fueled plant.

Why are Gen IV reactors different from other nuclear reactors ?

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- At least 3 concepts are intended to operate at high-temperature – so we need heat engines that can exploit high temperature heat sources efficiently.
  - A conventional Rankine (steam) cycle will not make best use of heat of such high quality.
- The architecture of some high-temperature systems is based on using the fluid returning from the power conversion system to cool the reactor pressure vessel (RPV).
  - This places an upper limit on the amount of waste heat recovery (recuperation) we can employ.
- Two of the concepts are gas cooled. All gas-cooled reactors use a low density coolant that consumes a lot of power to circulate.
  - The coolant circulation power can consume a significant fraction of the power output,
  - It is important to minimise the core pressure drop and to minimise the primary flow rate ($P_c \propto Q^3$).

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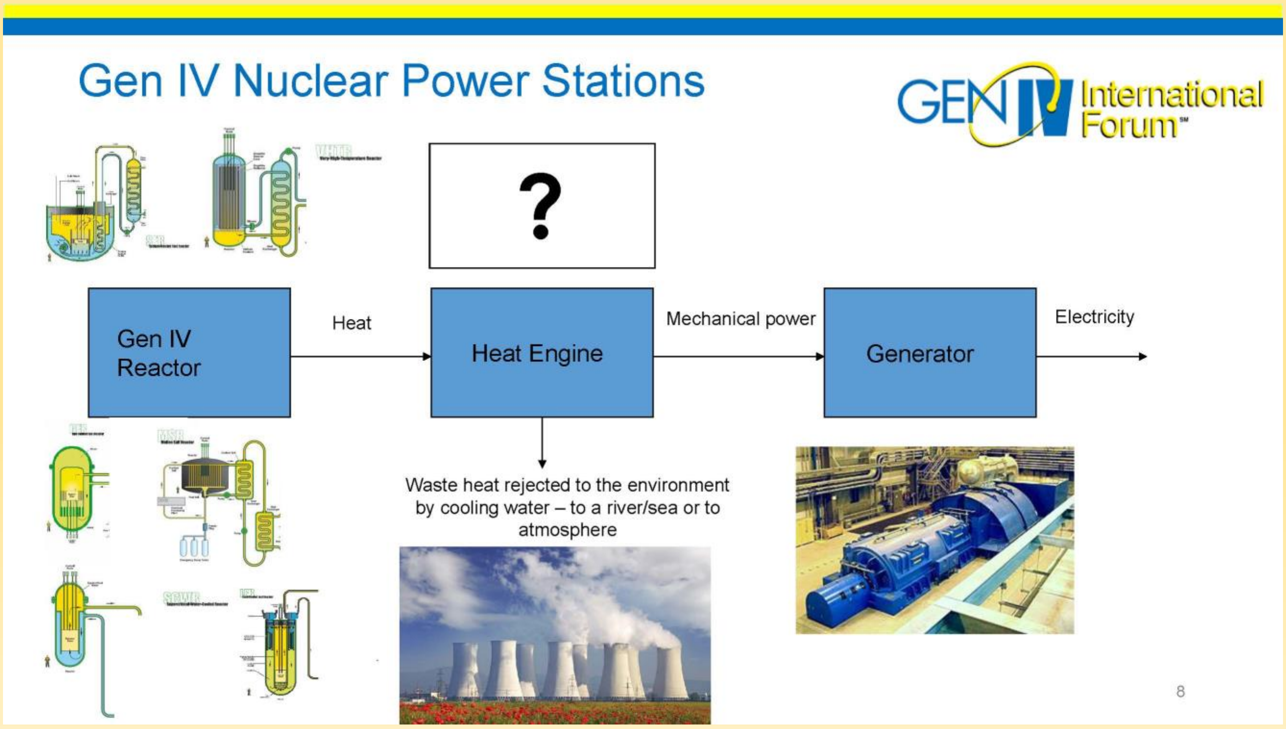
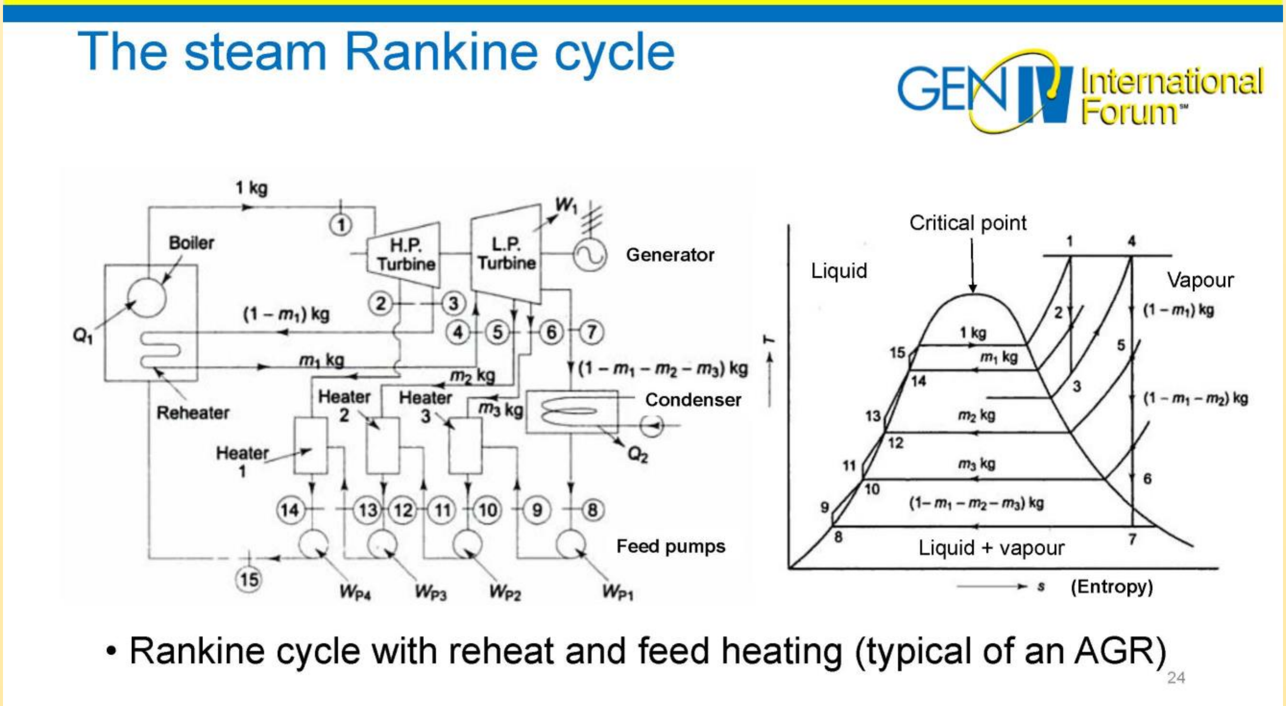
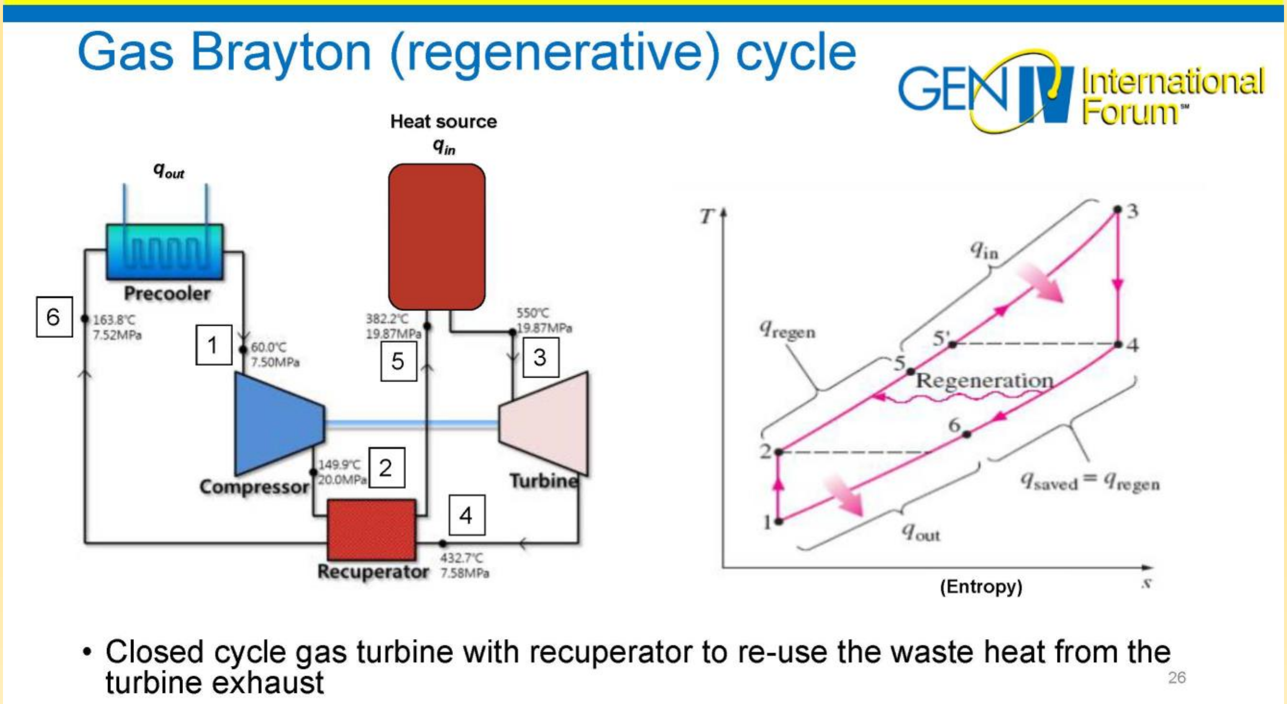
Heat engine for Gen IV reactors: There is no single optimal heat engine for all six types of Gen IV reactors. We need to consider how much mechanical power do we get for a given amount of thermal power, rejecting heat to the environment, and maximize the efficiency of the whole system.



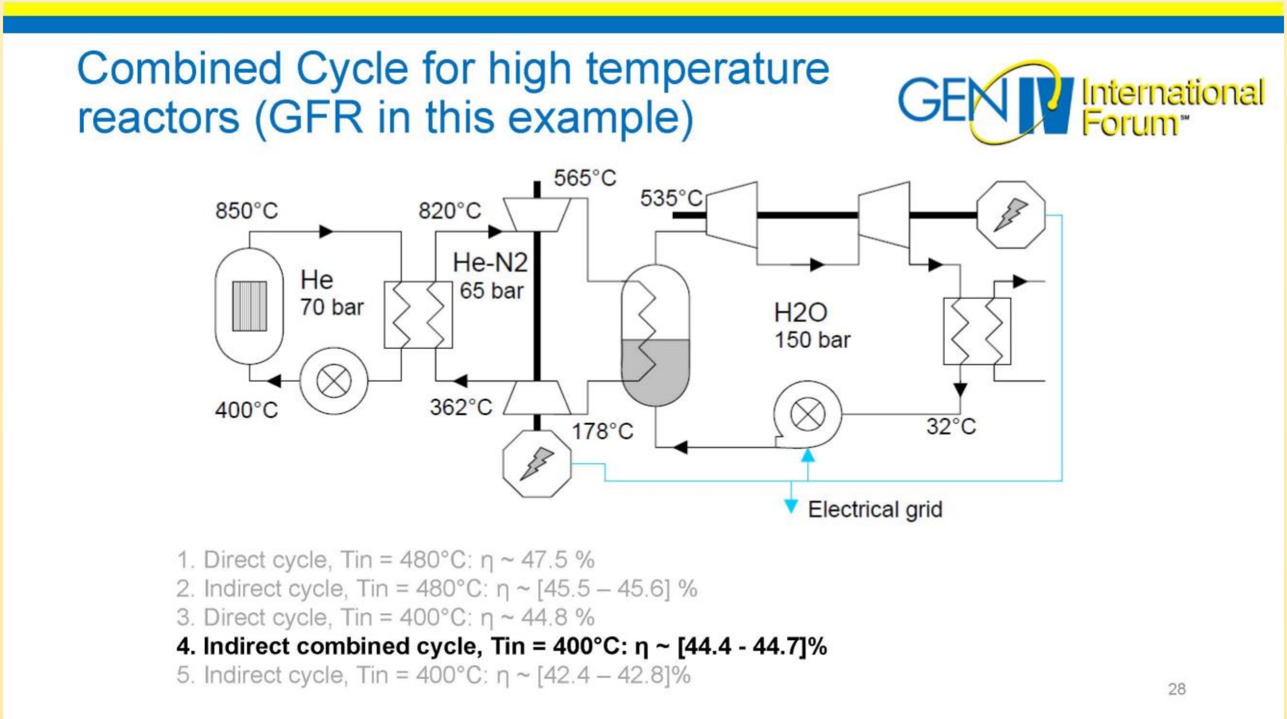
Rankine cycle: Rankine cycle is well known for over 120 years now and it is used as the way of generating electricity in the world power plant. High efficiency is achieved because of excellent work ratio and bulk of heat addition and heat rejection both occur as constant temperature processes.



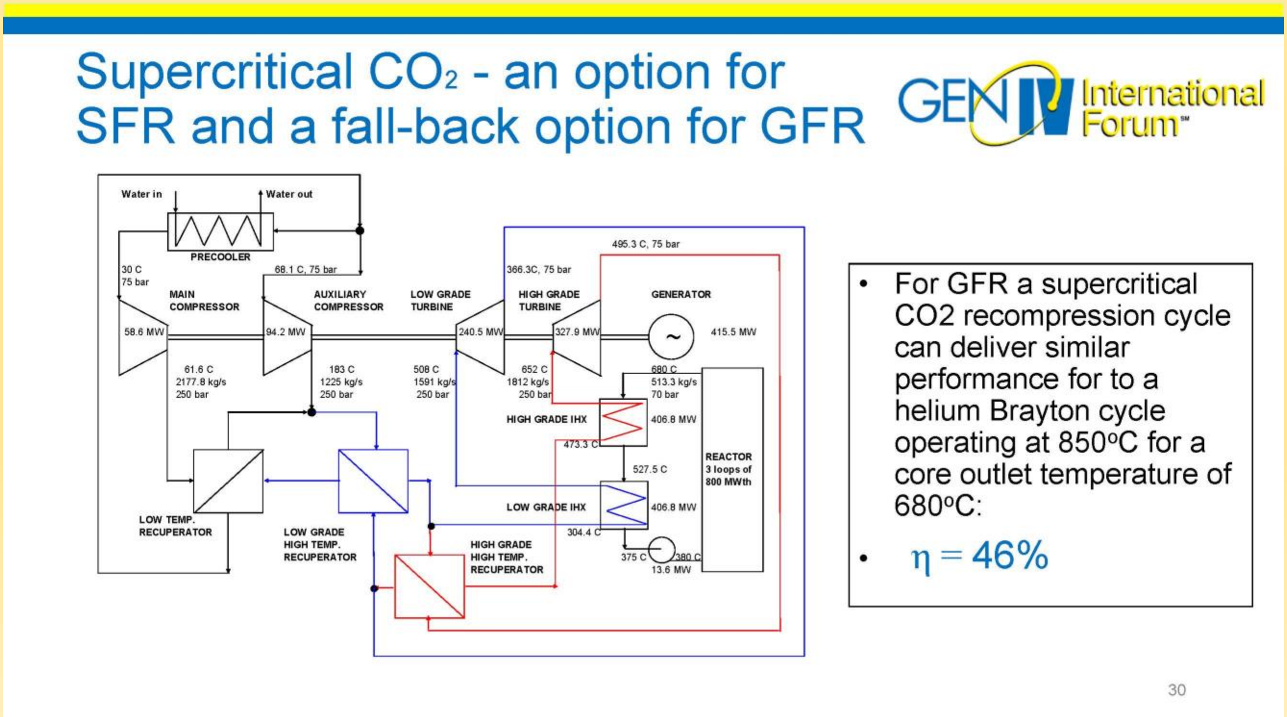
Brayton cycle: In the case of high temperature power generation, turbine technology can be applied to power generation. For a good gas turbine cycle, the difference in height between 4 and 3 should be as large as possible between 1 and 2.



Combined cycles : Combined cycles have a good track record of use in many fossil fired CCGT power plants. Gas turbines and high-efficiency gas-to-gas recuperators are expensive. On the other hand, steam turbines are cheap and heat recovery steam generators are a low-risk technology.



Supercritical CO2 : This cycle is a gas turbine cycle using a supercritical fluid. This cycling technology is very well understood thermochemically but needs to be checked for practicality in engineering. One of the biggest problems we face is that we must operate under very high pressure.



7-5. Thermal Hydraulics in Liquid Metal Fast Reactors

Summary / Objectives:

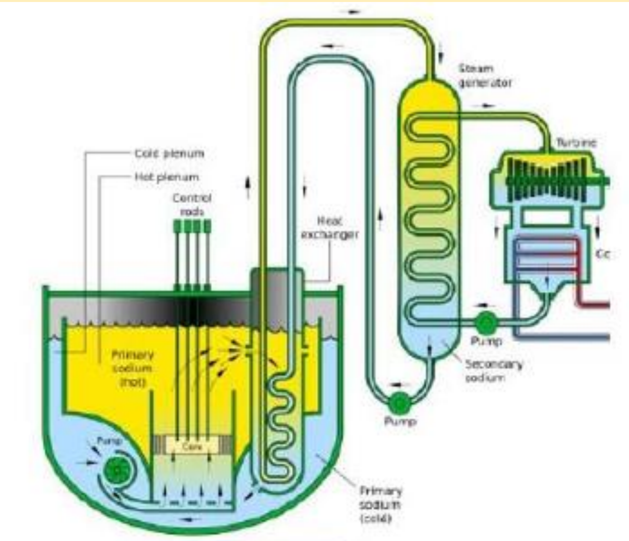
Thermal-hydraulics play a determining role in the design, operation and safety of liquid-metal reactors (LMRs) cooled by sodium, lead or lead-bismuth eutectic. The strong heat transfer performance and high boiling point of liquid metal enable the use of high working temperatures without pressurization. Because no pressure vessel is needed, most reactor designs then adopt a "pool-type" primary circuit, which minimizes the potential consequences of a primary leak and provides a large reserve of thermal inertia in accidental scenarios. While these common design characteristics of LMRs have direct advantages, they are also the source of complex thermal-hydraulic phenomena with potential high impact: strong temperature gradients must be controlled to avoid thermal fatigue on reactor structures, decay heat removal in pool-type designs depends on complex natural convection patterns. In this way, many key aspects of the justification of LMRs depend on understanding and simulating complex thermal-hydraulic phenomena. This webinar provides an overview of these phenomena and the current state-of-the-art for simulating them.

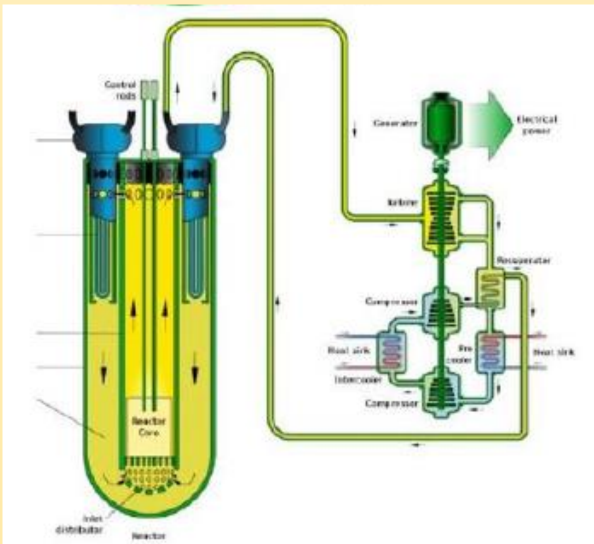
Meet the Presenter:

Dr. Antoine Gerschenfeld obtained his PhD from Ecole Normale Supérieure, France, in 2012, and has been coordinating R&D on the thermal-hydraulics of Sodium Fast Reactors at the Commissariat a l Energie Atomique et aus Energies Alternatives (CEA)'s Thermal-Hydraulics and Fluid Mechanics Section (STMF) since 2013. In that capacity, he has led the development of a subchannel thermal-hydraulics code (TrioMC) as well as the development of a tool for coupling coarse and fine models in a single reactor-scale simulation (MATHYS). He has also been involved in a number of collaborations : bilateral exchanges with DOE, JAEA and IPPE, as well as EURATOM projects.

Introduction on Thermal Hydraulics of LMFR:

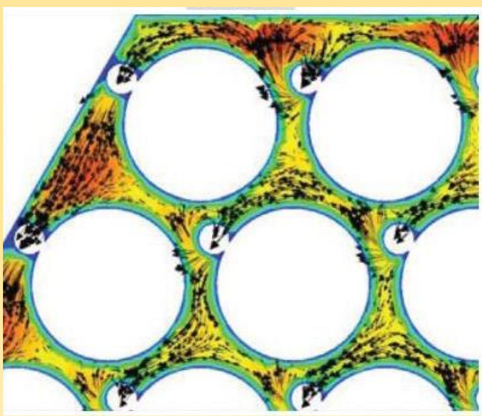
Liquid metal coolants have advantages such as little neutron moderation, large working temperature at ambient pressure and good to excellent thermal conductivity. However, they are also the source of complex thermal-hydraulic phenomena with potential high impact.

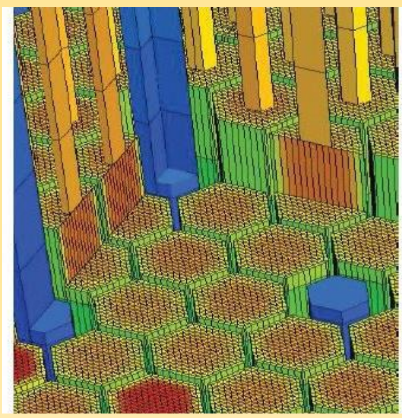


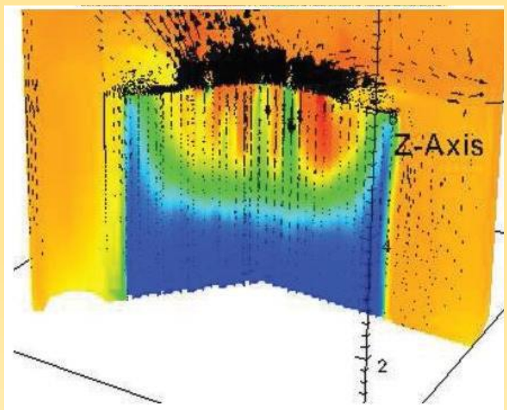


Issues / Core:

Subassemblies (S/As) have quite complex structures such as pins and wires (or grids). Issues of interests inside S/As are to know cladding temperatures both in nominal states (≤ 620°C) and in accidents (≤ 1200°C). There are issues from the point of overall behavior of core both in normal operations and accidental scenarios, which includes the coupling problem with neutronics and fuel thermal mechanics.

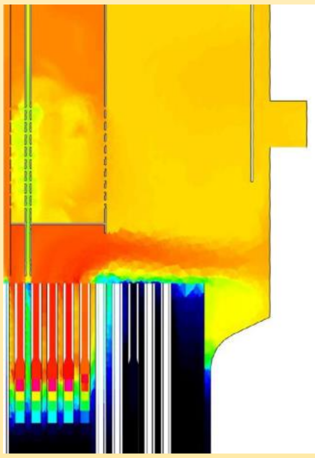


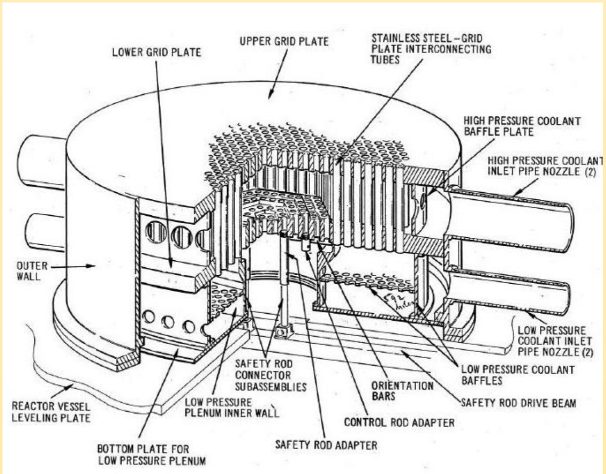


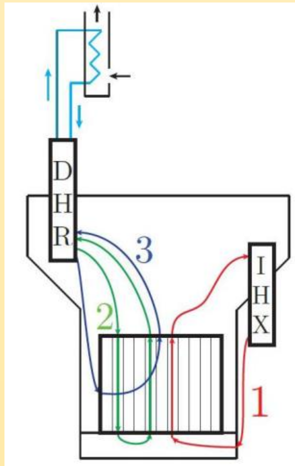


Issues / Pool, Component and Global:

In hot or cold pools, main issues on thermal hydraulics are on thermal load such as thermal fluctuation due to jet mixing, thermal stratification and hot/cold shocks in accidents. Issues on components are about its performance in normal or steady states and accidental aspects such as the pump trip situation. Gas transport in the primary circuit and decay heat removal system are issues involving the complete reactor.

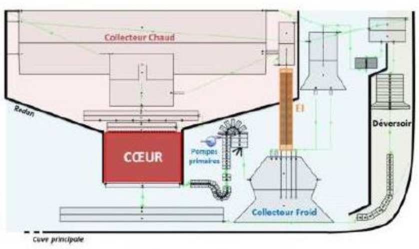
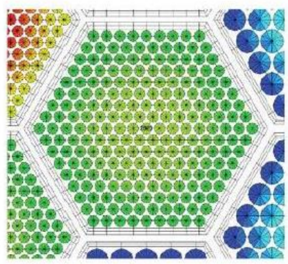
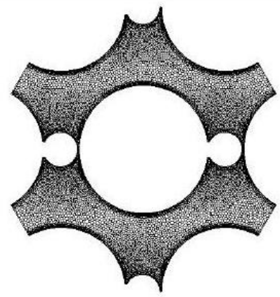






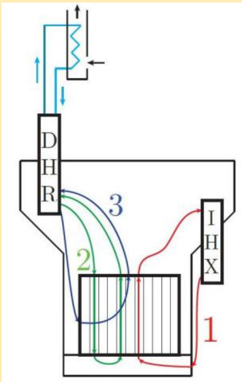
Modeling thermal hydraulics:

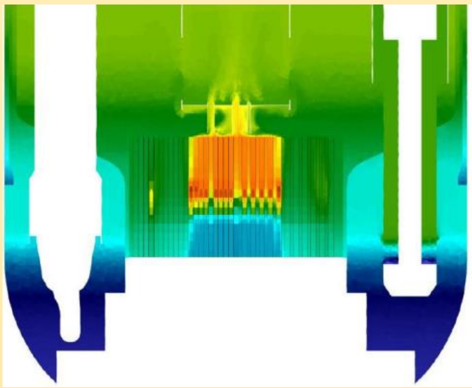
Thermal hydraulics has highly non-linear behavior and problem of scales. Ab initio modelling is very difficult and a cut-off scale is needed. There are various thermal-hydraulics codes according to the choice of cut-off. Those codes will be used according to the issues to be evaluated.

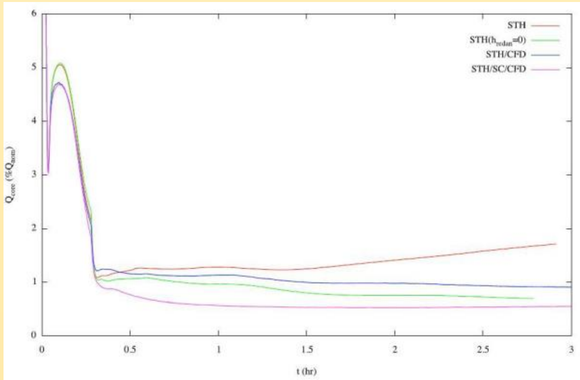
Scale	System (STH)	Subchannel (SC)	CFD
			
Simulation scale	channel (1D) volume (0D, 3D)	subchannel (between pins)	microscopic (DNS) fine (LES, RANS)
Physical models	every phenomenon (heat transfer, pressure drop)	fine geometry (wires, grids...)	nothing (DNS) turbulence (LES/RANS)
Code used at CEA	CATHARE	TrioMC	TrioCFD

Application / Natural convection in LMRs:

Natural convection is a global phenomenon in a reactor. Modelling based on STH is a natural choice. However, there are problems how to evaluate local issues which give feedbacks to the global behavior. On the other hand, modelling everything in CFD is not a reasonable approached because of the problem of extra computational cost. Combinations of STH and CFD (or SC) based on code coupling are prospected approaches.

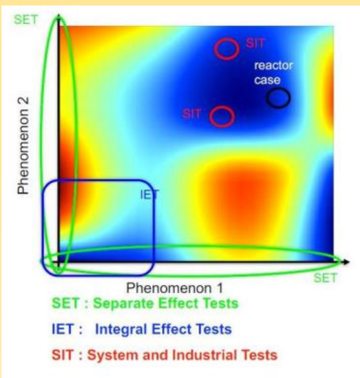


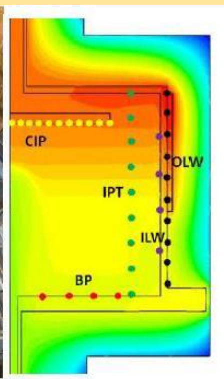




Application/ Validation(Natural convection):

All physical models introduced must be established experimentally. Then, validation of the physical models are important. Because of the non-linearities, combined effects resulting from the interactions of separate phenomena must also be validated. Therefore, validation experiments will be performed with a hierarchy. There are some examples on combined effects tests and integral scale tests using actual reactors.





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Expertise | Collaboration | Excellence

7-6. Generation IV Coolants Quality Control

Summary / Objectives:

The quality of coolant in Fast Neutron Reactors must be controlled due to the potential impact of impurities on the structural material, on the dosimetry and subsequently on the operation. Liquid metals (sodium, lead-bismuth eutectic, pure lead) and gas (He) need to be purified in order to avoid deleterious effects and satisfy several safety requirements. Several purification systems and dedicated instrumentation have been developed for this purpose, taking into account the specific properties of each coolant.

Meet the Presenter:

Dr. Christian Latgé graduated in Chemical Engineering (1979) and earned his PhD from the Institut National Polytechnique in Toulouse (France). His PhD in CEA Cadarache was dedicated to Na chemistry and purification systems. He participated in the start-up and then operation of Superphenix and operational feedback analysis (Phenix, Superphenix and foreign reactors), in the field of chemistry, radiochemistry and technology. He was also involved in design activities in EFR & SMFR. As Head of Service, he coordinated activities dedicated to process studies for decontamination and nuclear waste conditioning in Cadarache. He carried out studies dedicated to tritium systems and hydrogen risk mitigation for the ITER project. As Director of the International Project Megapie, Dr. Latgé led a team dedicated to the development of a Lead-Bismuth Eutectic Spallation target for nuclear waste transmutation. He served as the Head of Sodium School in Cadarache and now teaches at CEA-INSTN and several French Universities. He has been involved in several Educational Sessions organized by the IAEA on Fast Reactors, in Argentina, Mexico and Trieste ITCP and is the CEA representative on the GEN-IV International Forum Education & Training Task Force. He is currently involved in SFR and recently in ASTRID project as expert and he is involved in several international collaborations (Russia, India, Japan, Latvia, EU, IAEA, NEA-OECD….) related to the development of Fast Neutron Reactors.

In the XFR, X means the kind of coolant. SFR is sodium cooled fast reactor and LFR is lead cooled fast reactor. The coolant must be able to extract heat from the reactor efficiently. It is also required to transfer heat efficiently to the energy conversion system. They are also required to ensure the safety structural and operational conditions.

Coolant Functions for the Primary Circuit of XFR

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- The coolant(s) must accomplish the following key tasks
  - Extract heat from the core: high specific heat and thermal conductivity ensure good extraction
  - Transfer heat to an energy conversion system (steam generator or exchanger + turbine) or to a system which directly uses the heat: heavy oil extraction (oil shales), thermochemical production of hydrogen, desalination of sea water
  - Assure safety by providing the system with a degree of thermal inertia
- In a Fast Neutron Reactor, the coolant must NOT
  - Significantly slow neutrons
  - Activate under flux, producing compounds which create unacceptable dosimetry
  - Change the behavior of structural materials
  - Induce unacceptable safety conditions
  - Induce insurmountable operating problems
  - Lead to wastes which can't be processed during operation or dismanting

3

Impurities in the coolant may adversely affect the operation of SFR and LFR. It can cause corrosion, reduction of heat transfer coefficient and formation of an obstruction in a narrow space.

Why is it necessary to control quality and to purify the coolant?

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- Primary coolant of XFR:
  - [O] is a key parameter of corrosion
    - For SFR→ contamination→dosimetry → necessity to decontaminate (handling, repair, ISI,..): [O]<3ppm
    - For HLM-FR (or ADS)→ necessity to master dosimetry and to eliminate corrosion particles (filtering)
  - [O] well mastered can help to maintain oxide layer stable (protection against hard corrosion in heavy liquid metals HLM). It also allows enhancement of tribology.
  - [O] can induce precipitation of coolant oxide : <u>issue for HLM: PbO particles</u> due to very low dissolution rate; in case of very large O ingress, it can modify the composition of binary alloys ie Pb-Bi... (it is not a problem for Na),
- For Intermediate circuits of SFR (Na) :
  - [H] has to be maintained as low as achievable in order to detect as soon as possible a water ingress in Na (Na-H_2O reaction generates H_2): [H] <0.1 ppm
  - In steady-state operation, aqueous corrosion in SGU produces Fe_3O_4 and H: H diffuses towards intermediate Na.
  - Moreover, Na purification allows to minimize tritium release. (Nota: Tritium release is a common issue for all nuclear systems, including HLM cooled FRs
- For all the circuits :
  - Control plugging hazards in narrow gaps, tubing, openings, seizing of the rotating parts, reduction of heat transfer coefficient in IHX (Intermediate Heat Exchanger)...

  → to limit the plugging hazard, necessity to maintain [O] < [O]* and [H] < [H]* at the coldest point of the circuits, for all operating conditions ; value recommended in SFRs: Tsat < Tcp - 30°C

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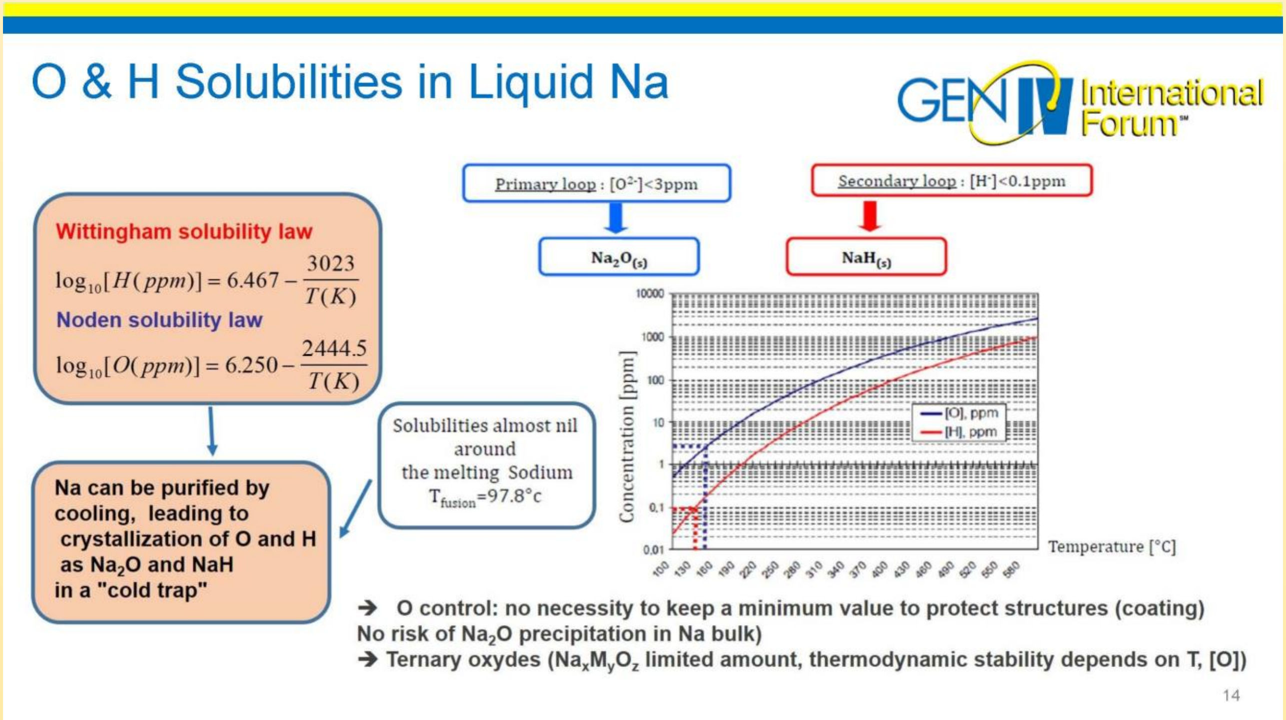
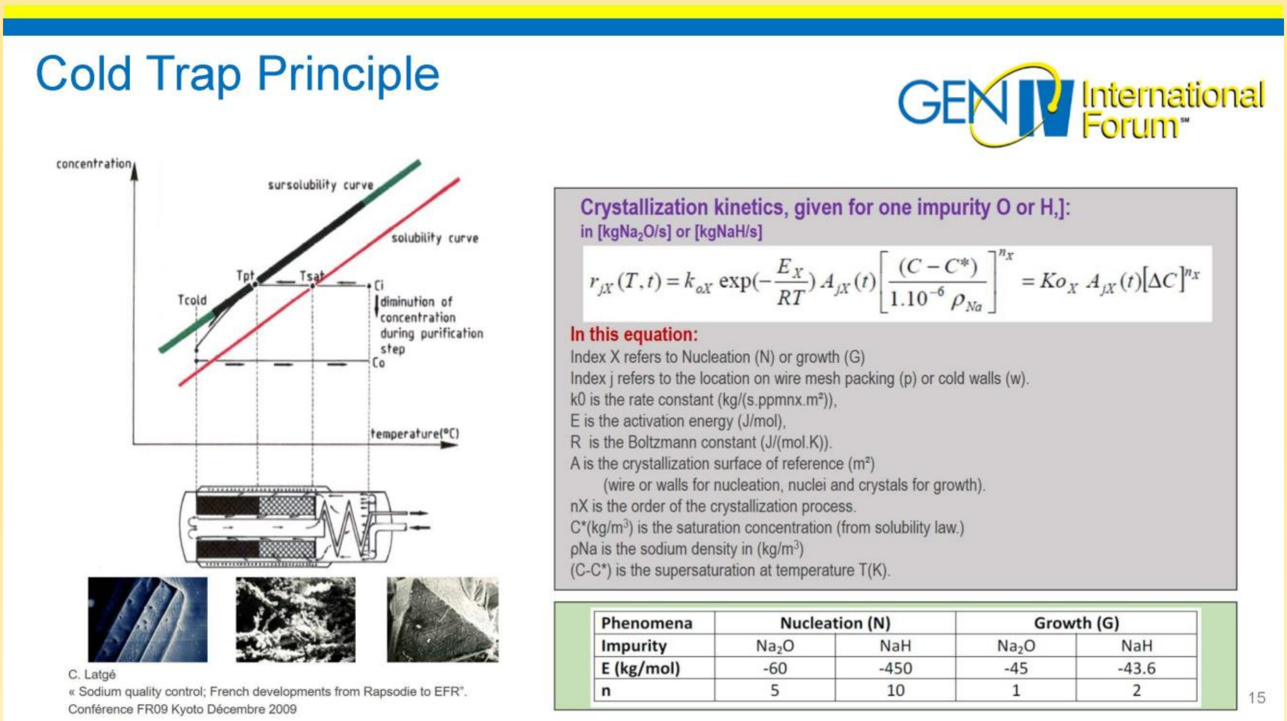
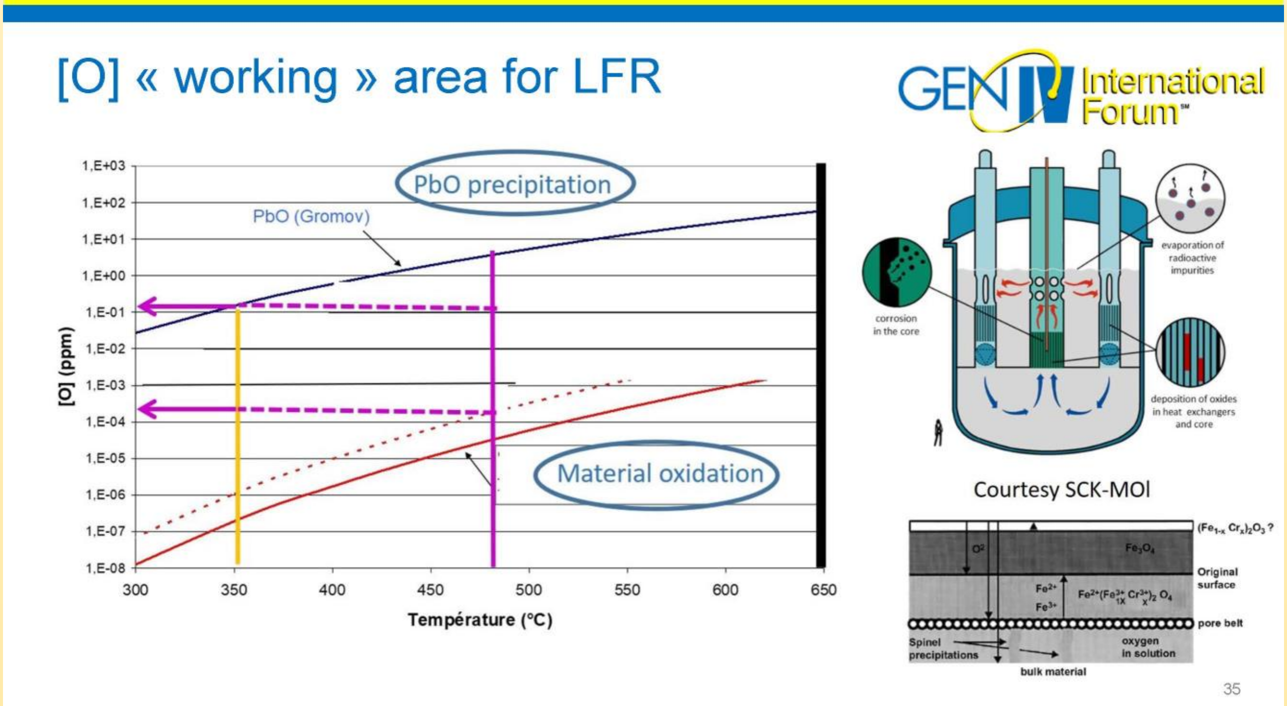
The concentration of impurities such as oxygen and hydrogen that can be dissolved depends on the temperature of the coolant sodium in the case of SFR.



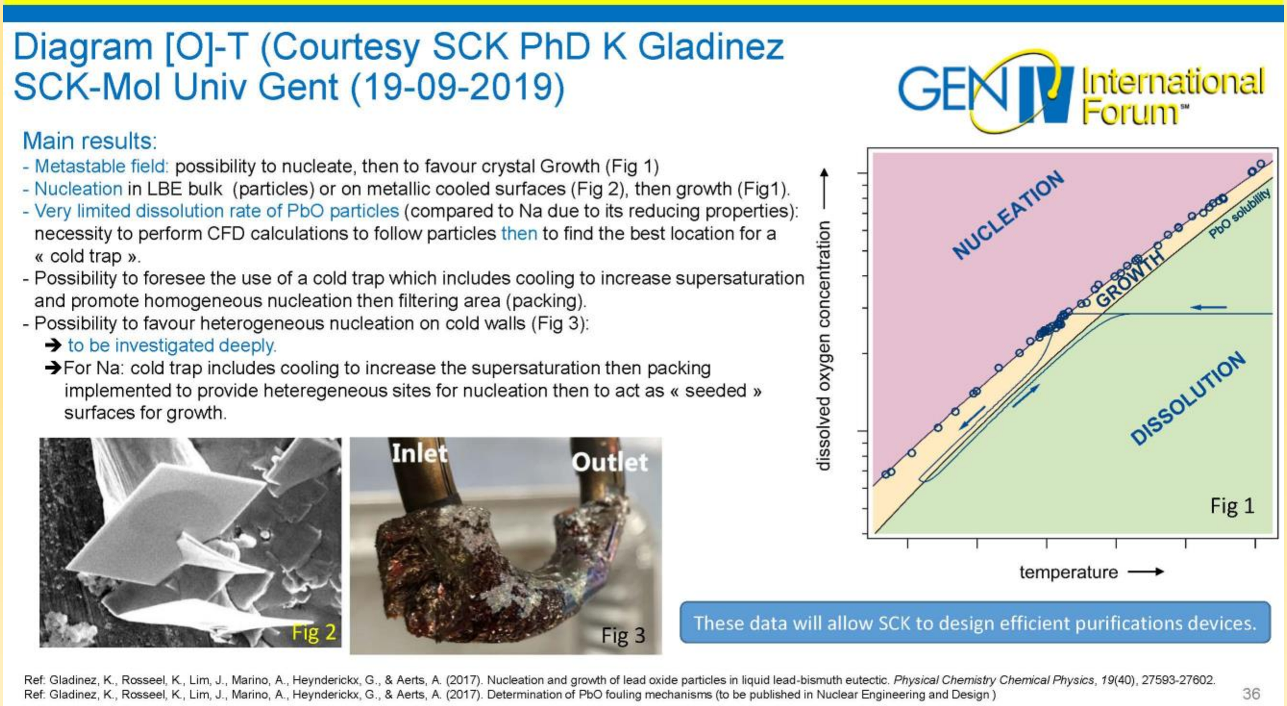
The principle of purification in a cold trap is explained. Sodium can be purified by cooling, leading to crystallization of O and H as Na_2O and NaH in a “cold trap”. The cooled sodium is then heated up again for operation.



In the case of LFR, if the working area of the coolant is not properly maintained, it will cause corrosion and oxide deposition, which will damage the reactor.



The method of removing impurities in cold traps and filters is being carefully investigated because it is different from the case of sodium. Examples of recent research results are given, and these results can be used to design efficient purification devices.



8-1. Cement Matrix for Nuclear Waste

Summary / Objectives:

This webinar discusses the formulation of an alternative cement matrix for solidification/stabilization of nuclear waste. The presentation provides an overview of the multiple complexities of waste management, and the many challenges that arise from it. Topics include a presentation of the French nuclear waste management methods, specific examples on solidification/stabilization of nuclear waste, the physico-chemical aspects of the interactions between the containment matrix and the waste, and the miniaturization of samples for the development of new matrices allowing human radiation protection. The webinar also highlights current experimental research focused on Portland cement and a magnesium potassium phosphate cement matrix. The latter is a promising cement for the stabilization/solidification of heavy metals. Other potential cementitious matrices will also be discussed.

Meet the Presenter:

Mr. Matthieu De Campos is a second year PhD student at the University of Lille, more specifically within the Solid Chemistry axis of the UCCS laboratory (Catalysis and Solid Chemistry Unit). He is a member of the research team CIMEND («ChImie, Matériaux Et procédés pour un Nucléaire Durable» i.e. «Chemistry, Materials and Processes for Sustainable Nuclear Activities”). This research team is involved in a joint laboratory between the University of Lille and Orano, the Laboratoire de Recherche Commun Cycle du Combustible et Chimie de l’Uranium (LR4CU) (for Joint Research Laboratory on Fuel and Uranium Chemistry). The LR4CU is focused on generating added value to fuel cycle by-products and optimizing nuclear processes. The aim is to increase the TRL levels for futures industrial applications. His PhD research aims at adding value to low-radioactive metallic materials, by considering them as reagents for the synthesis of cementitious matrix. His research activities, funded by Orano, are based on a multidisciplinary approach combining Civil Engineering and Solid State Chemistry. In 2017, he graduated from Artois University with a Masters’ Degree in Materials Chemistry for Energy and the Environment.

1. French Classification of Nuclear Waste:

Separation of nuclear waste into 6 categories based on its radioactivity level and life span.

Dismantling generates many different type of wastes.

The chemical nature of this waste is the main difficulty in managing it during dismantling.

This is why the development of new adapted cementitious matrices is important to ensure safe handling & protect humans from their toxicity.

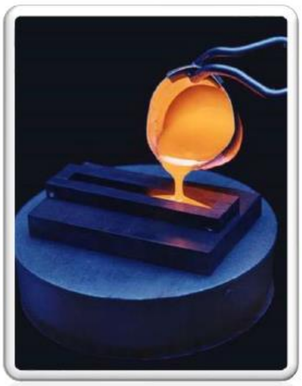
Category	Very short-lived waste	Short-lived waste	Long-lived waste
Very low-level waste (VLLW)	VSLW Management through radioactive decay	VLLW Surface disposal (Industrial facility for grouping, storage and disposal)	VLLW Surface disposal (Industrial facility for grouping, storage and disposal)
Low-level waste (LLW)	VSLW Management through radioactive decay	LILW-SL Surface disposal (Aube and Manche disposal facilities)	LLW-LL Near-surface disposal under development
Intermediate-level waste (ILW)	VSLW Management through radioactive decay	LILW-SL Surface disposal (Aube and Manche disposal facilities)	ILW-LL Deep geological repository at the project phase
High-level waste (HLW)	Not applicable		HLW Deep geological repository at the project phase

ANDRA. *National Inventory of Radioactive Materials and Waste*. 2018.

2. The Conditioning Routes for Radioactive Waste:

The common point of these conditioning routes is storage.

Vitrification of fission elements



Stabilization by solidification of nuclear waste

Packing

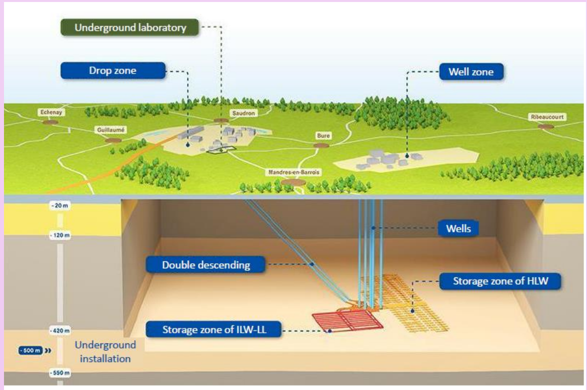
3. Types of Storage:

The French National Radioactive Materials and Waste Management Plan (PNGMDR) describes the prescribed management solutions for the different categories of radioactive waste.

Surface Storage: VLLW Disposal

Surface Storage: LILW Disposal

Deep Storage



4. Stabilization/solidification (S/S):

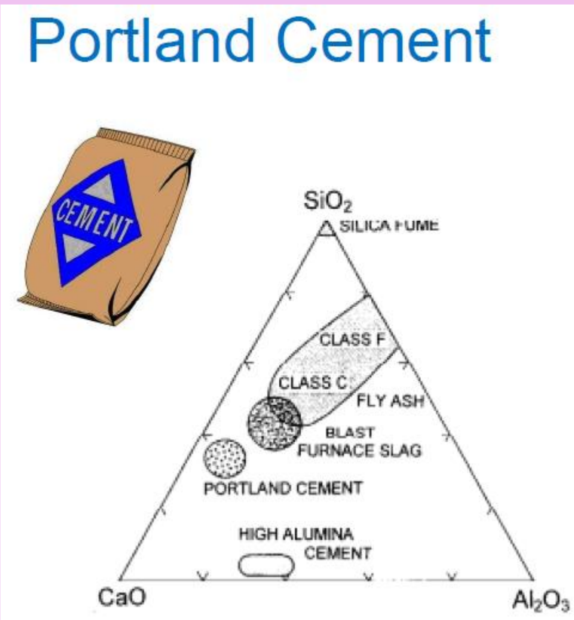
OPC(Ordinary Portland Cement)-based S/S of soluble Pb

→ Physical encapsulation by calcium-silicate-hydrate (C-S-H) gels (present in Portland cement)

MKPC(Magnesium Potassium Phosphate Cement)-based S/S process

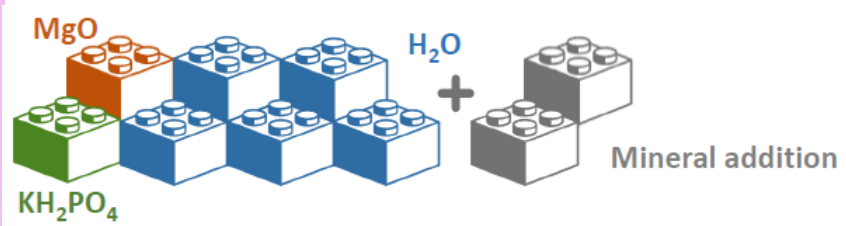
→ Chemical stabilization with residual phosphate and physical fixation by K-struvite cement.

MKP is a more efficient and chemically stable inorganic binder for the Pb S/S process (compared to Portland cement)



Formation of MKP cement:

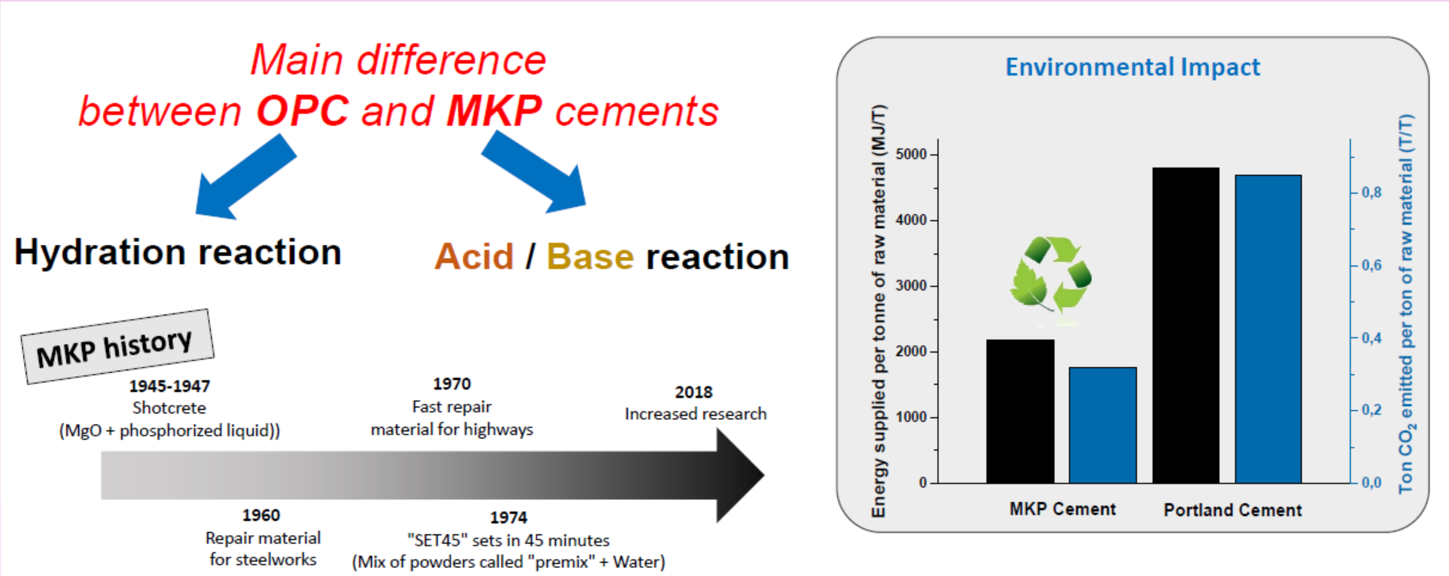
$$« KH_2PO_4 + MgO + 5\,H_2O \rightarrow MgKPO_4 \cdot 6H_2O »$$



5. Difference Between MKP & OPC:

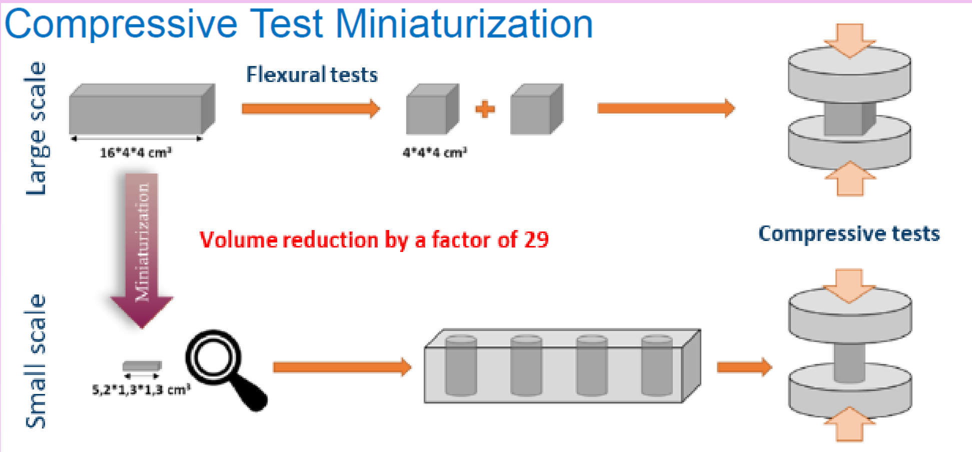
The formulation of innovative matrices requires:

- Implementation of specifications according to the intended use
- Use of a cementitious matrix appropriate to the waste
- Formulation tests
- Performance optimization(physical, leaching…)
- Understand the physico-chemical phenomena involved



6. Physical Integration of Nuclear Waste:

To Demonstrate feasibility to enable to scale-up while unlocking the technological locks



8-2. Interactions between Sodium and Fission Products in Case of a Severe Accident in a Sodium-cooled Fast Reactor

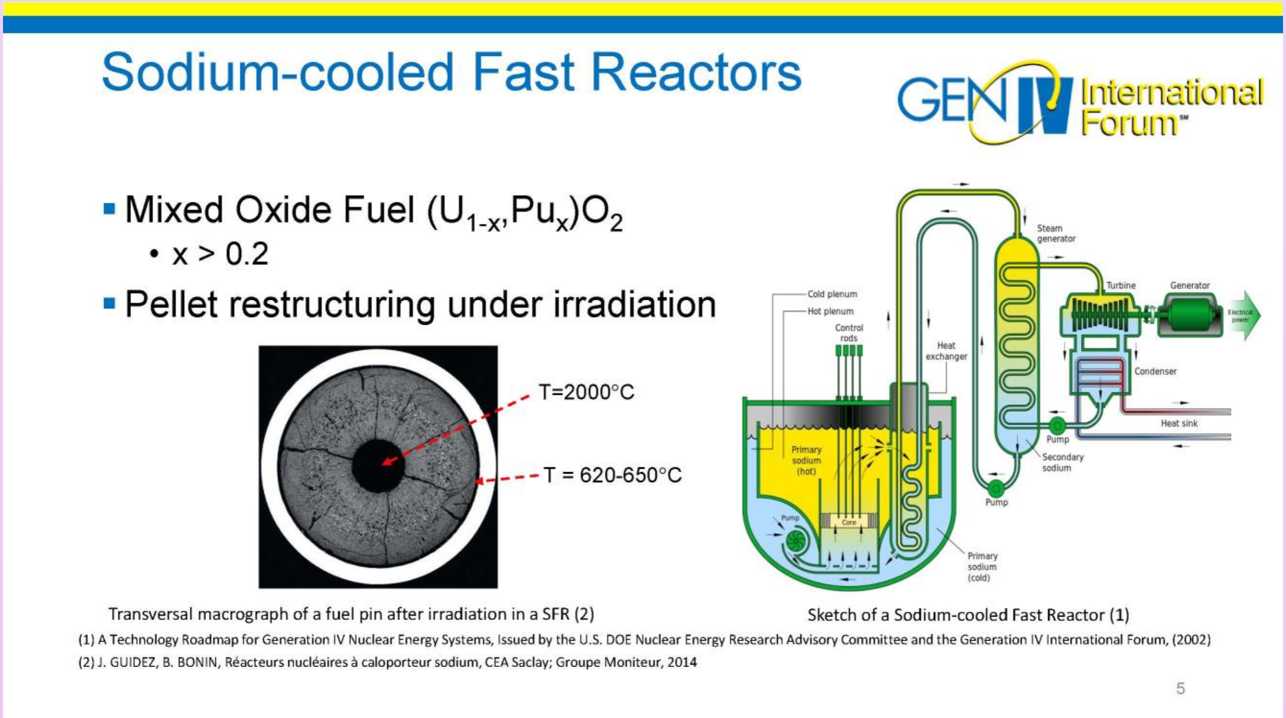
Summary / Objectives:

An overview of severe accident scenarios in Sodium-cooled Fast Reactors will be presented, focusing on the thermochemistry aspects and how the CALPHAD method could be used to enhance the prediction of the different phases that could form depending on the conditions of the system. CALPHAD, which stands for CALculation of PHAse Diagram, is a semi-empirical method that enables to develop a thermodynamic model based on the Gibbs free energy of the gas, liquid and solid phases as a function of temperature, pressure and composition of the system. Experimental measurements of the thermodynamic properties of some fission product compounds formed in the Joint Oxide Gain after interaction with sodium will be presented. These data will be used as input for the thermodynamic modeling.

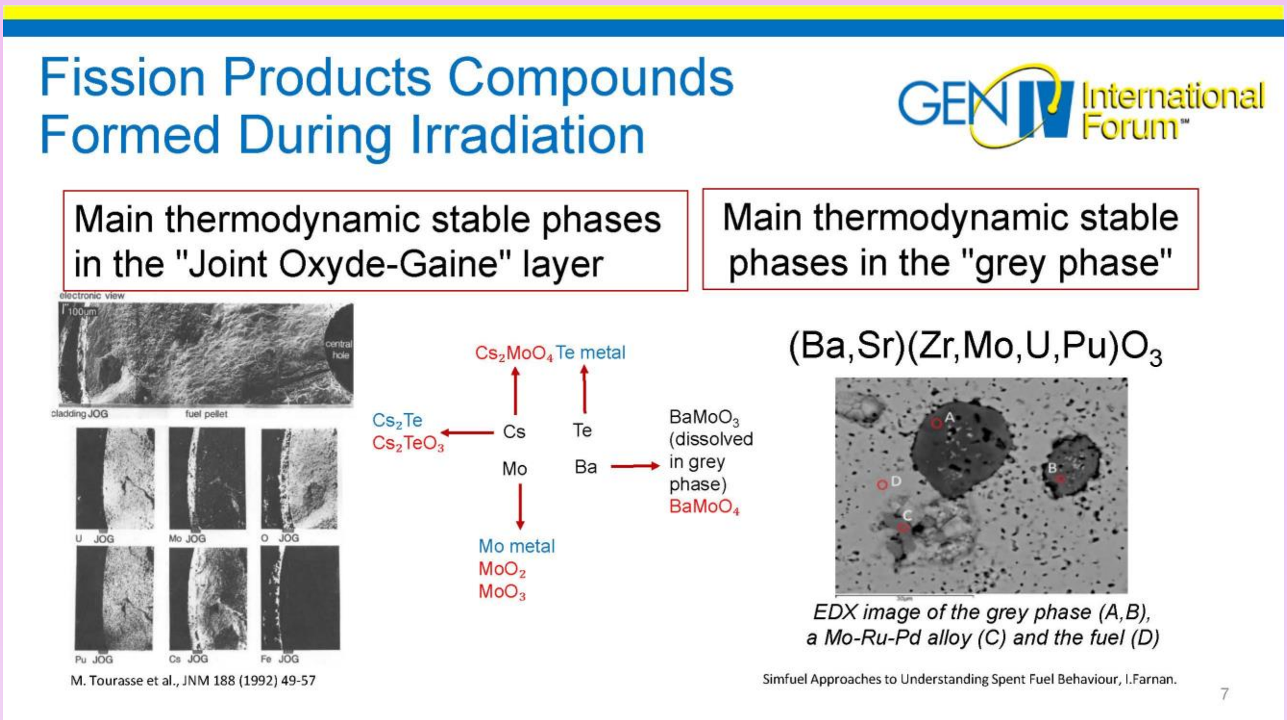
Meet the Presenter:

Mr. Guilhem Kauric is a second year PhD student at CEA Saclay in the "Service de la corrosion et du comportement des matériaux dans leur environnement" (SCCME) in the "Laboratoire de Modelisation de Thermodynamique et de Thermochimie (LM2T)". His PhD research aims at investigating the chemical interactions between MOX fuel, fission products and sodium for the safety assessment of the Sodium-cooled Fast Reactor in case of severe accident. As the chemical system contains many elements, the CALPHAD method approach is the most suitable to develop a model for this study. His research activities, funded by CEA and the ENEN + program, are based on a multidisciplinary approach combining experimental work and modelling. In 2017, he graduated from Chimie Paristech ENSCP (diplome d'ingenieur option chimie des materiaux) and from INSTN with a Master's Degree in Nuclear Engineering option Fuel Cycle.

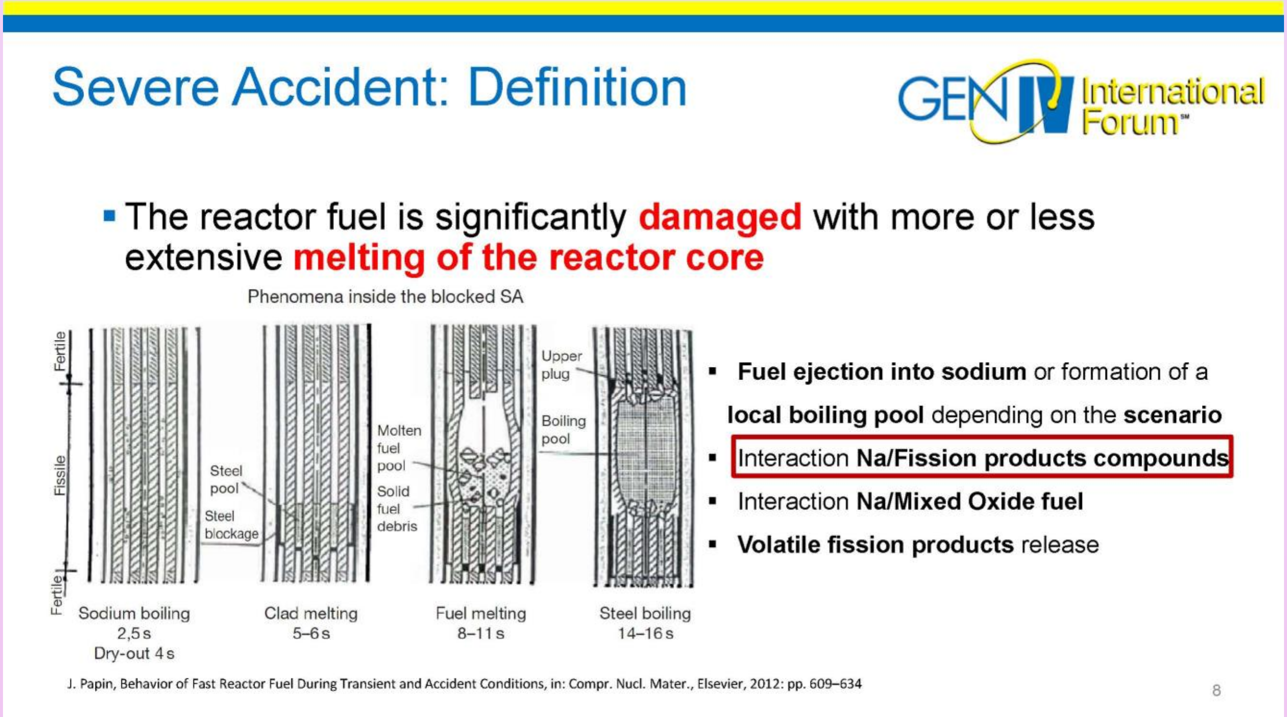
The target is mixed oxide fuel, which is the fuel of SFR.
The mixed oxide fuel is in the cladding as a fuel pellet and the cladding is cooled by liquid metal sodium.



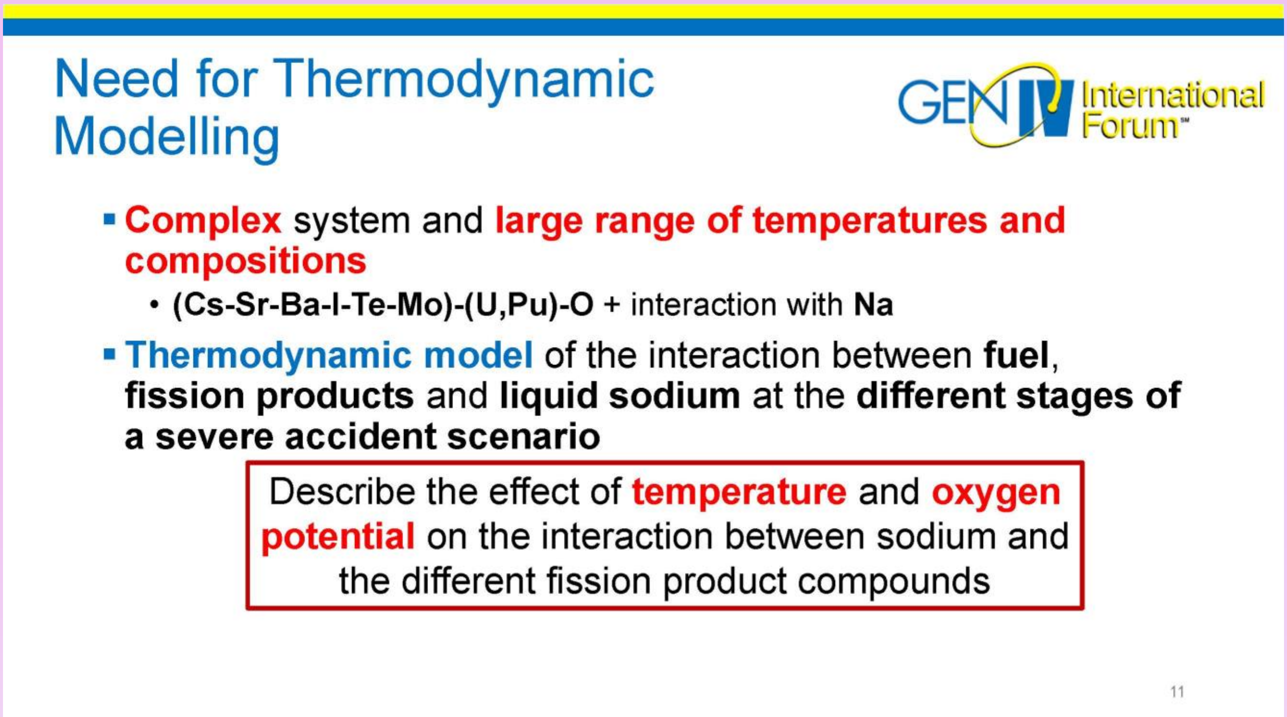
Knowing the stable chemical species produced under irradiation is important, because it affects the assessment of the accident. In addition, it is necessary to consider that SFR is characterized by sodium coexistence.



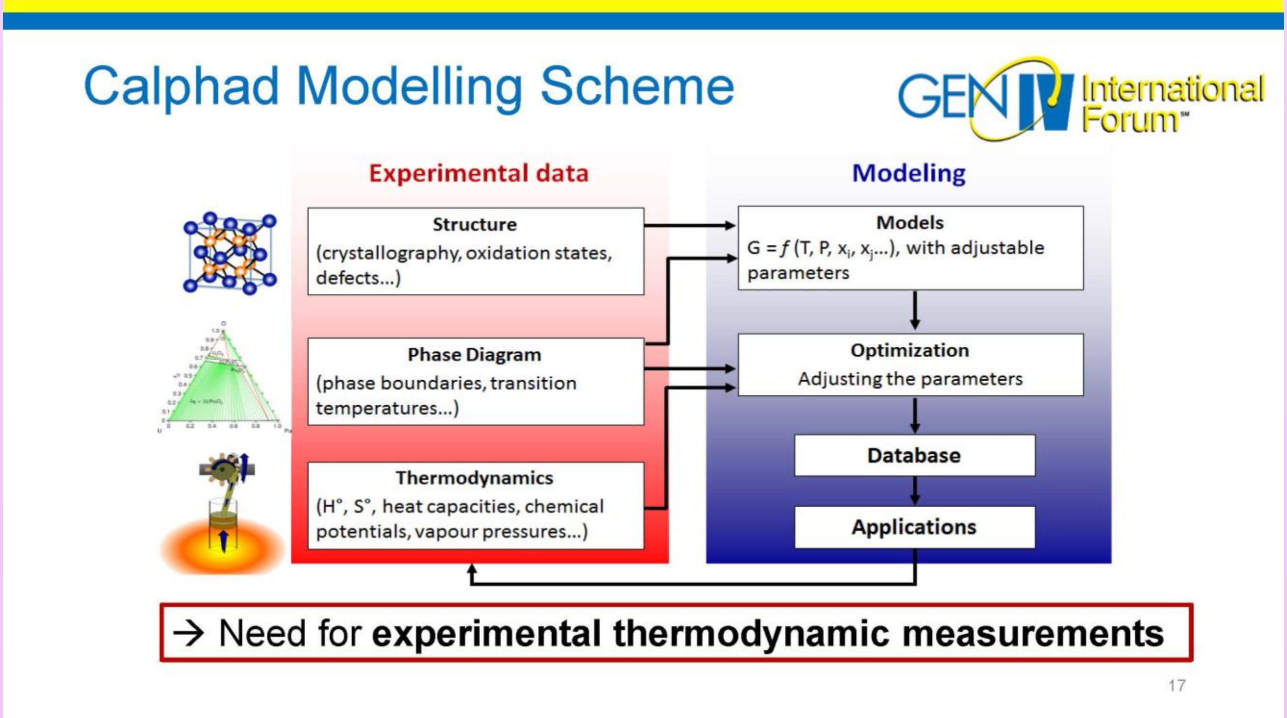
Assuming a severe accident, there is an interaction between FP or mixed oxide fuel and sodium. In this study, a thermodynamic study has been carried out focusing on this interaction.



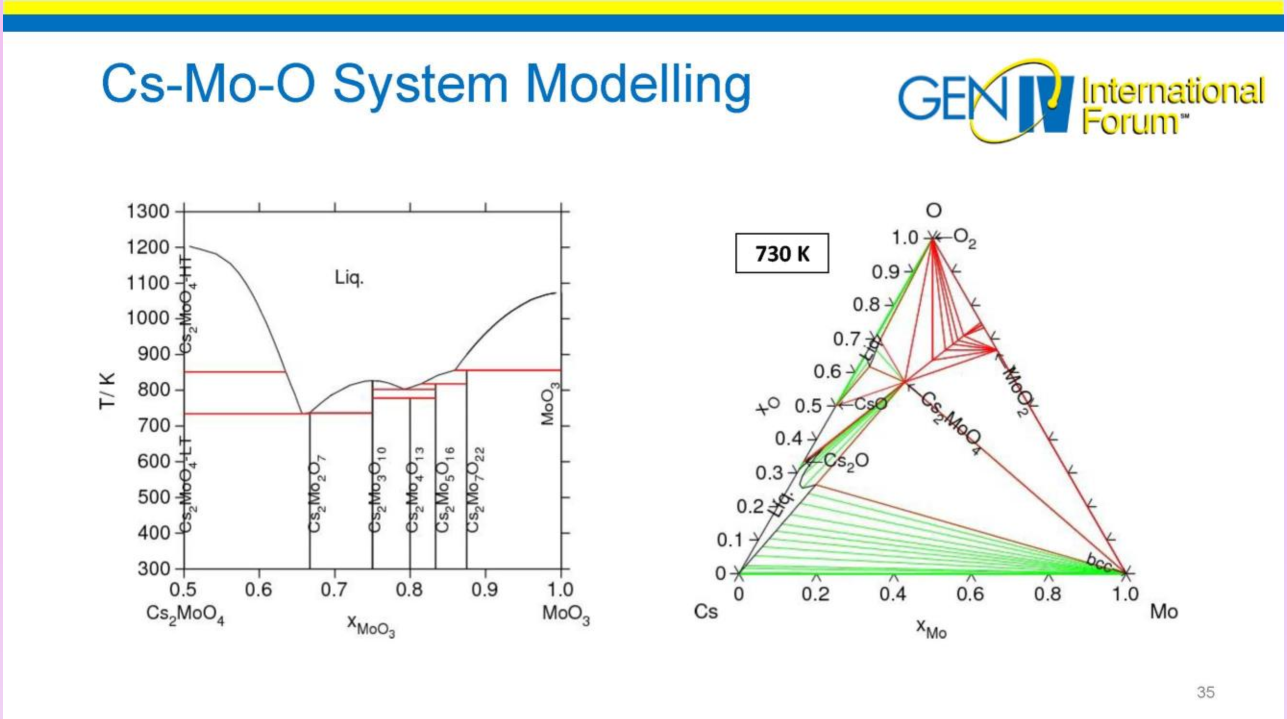
This interaction is different depending on the temperature and oxygen potential, and the stable compounds to be produced will be different. Thermodynamic models that can be applied over a wide range of temperatures and components are needed for severe accident evaluation.



Using the Calphad modelling scheme, we can know which compounds are thermodynamically stable. This model requires some experimental thermodynamic data. Prediction accuracy will continue to improve as the data is expanded. The study is being carried out in a multilateral collaboration as The TAFID Database Project.



An example of Cs-Mo-O is presented as an application result of these projects. As a function of their respective compositions, compounds that are stable at a given temperature can be identified. It is very useful for severe accident assessments.



8-3. Security Study of Sodium-Gas Heat Exchangers in Frame of Sodium-cooled Fast Reactors

Summary / Objectives:

This webinar provides an overview of a Sodium Fast Reactor system and presents an accident scenario in Compact plates Sodium-Gas heat Exchangers (ECSG) of SFR. The overpressure (180 bar in the nitrogen loop while 5 bar in the sodium loop) could result in nitrogen leaking into the liquid sodium. The present work focuses on the analysis of the predominant physical phenomena in the jet (the viscous diffusion, the momentum exchange between the two fluids) and supersonic gas jet, the development of the compressible multiphase flow model (Baer-Nunziato model) and its numerical schemes. In addition, the model is implemented using the numerical tool CANOP that enables researchers to generate the Adaptive Mesh Refinement and to calculate in parallel.

Meet the Presenter:

Dr. Fang Chen recently earned her PhD titled: “Numerical study of the under-expanded nitrogen jets submerged into liquid sodium in the frame of sodium-cooled fast reactor (SFRs)” from the university of Aix Marseille, France. She pursued her research at the CEA Cadarache, Service de Technologie des Composants et des Procédés (STCP), Laboratoire de Technologie, Procédés et Risques Sodium (LTPS). In 2016, she double majored as an Engineer in Energetics, Mechanics and received a Master in Physics of Multiphase Flow from the University of Aix-Marseille, France.

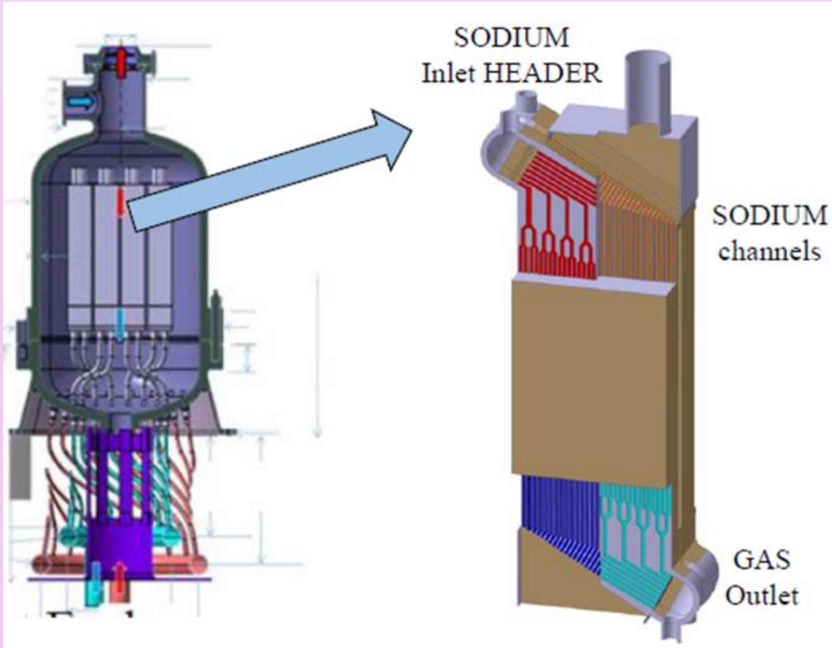
1. SGHE (Sodium Gas Heat Exchangers) design of French SFR ASTRID :

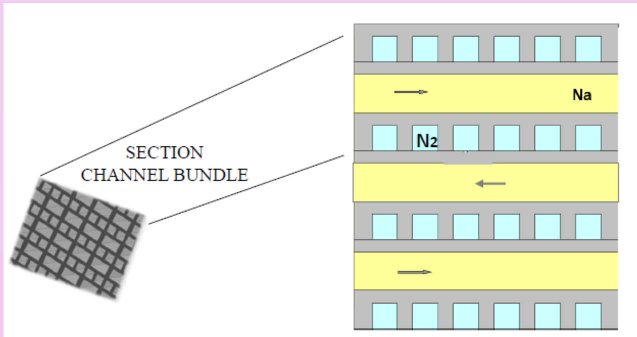
Pressure difference between the secondary & tertiary loop:

–180 bar in gas loop, 5 bar in sodium loop.

Accident scenario (wall crack): gas leak into sodium, under-expanded gas jet.

Safety analysis : acoustic detection of gas leak

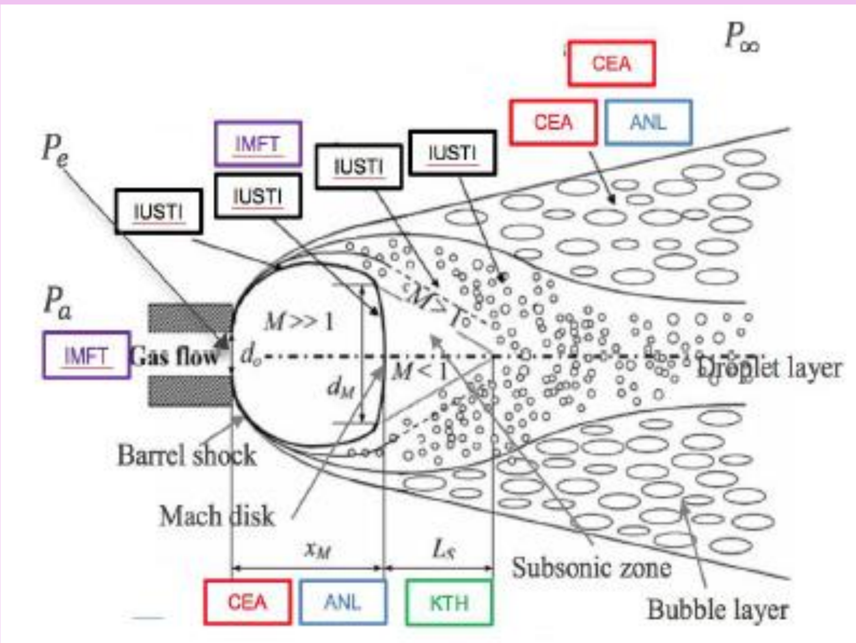


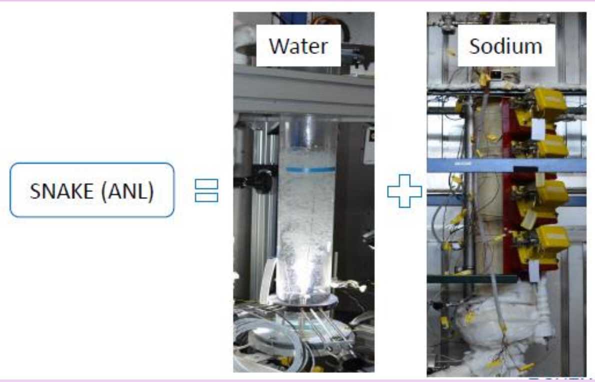


2. Objective of present work :

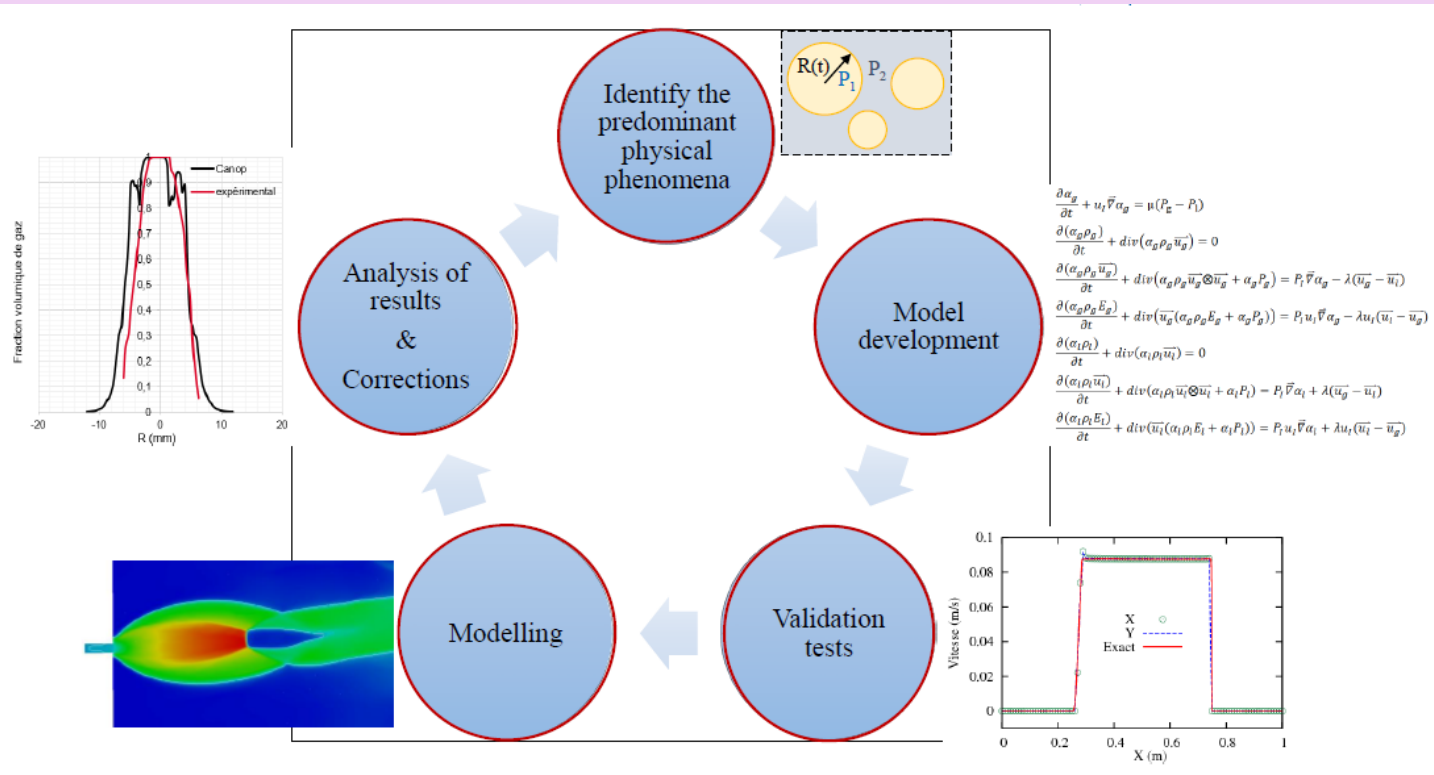
Provide a numerical tool to find the structure of under-expanded gas jet as a function of the flowrate of the gas leak

Many organizations including IMFT, CEA, ANL, IUSTI, KTH are in cooperation.

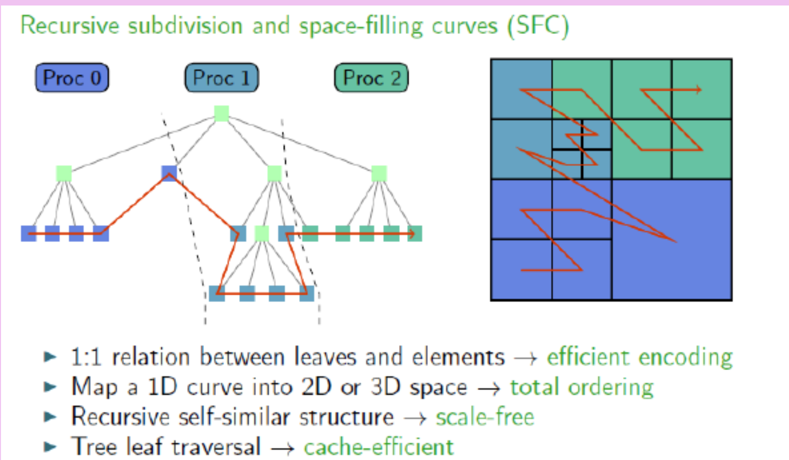


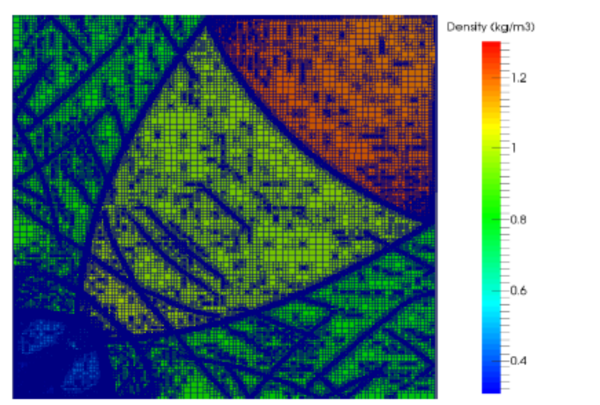


3. Development process :
Model development, Validation tests, Modelling , Analysis of results & Corrections, Identify the predominant physical phenomena



4. Numerical tool –CANOP (Two layers in CANOP) **:**
- Low-level layer:
  Cell-based Adaptive Mesh Refinement (P4estlibrary),
  Efficient parallel computation





An AMR example controlled by the gradient of density.

- High-level layer, for implementing numerical schemes:
  Finite volume method,
  PDF problems in Fluid Dynamics (for astrophysics, multiphase flows, *etc*)

5. Model Validation:

Validation of convective part :

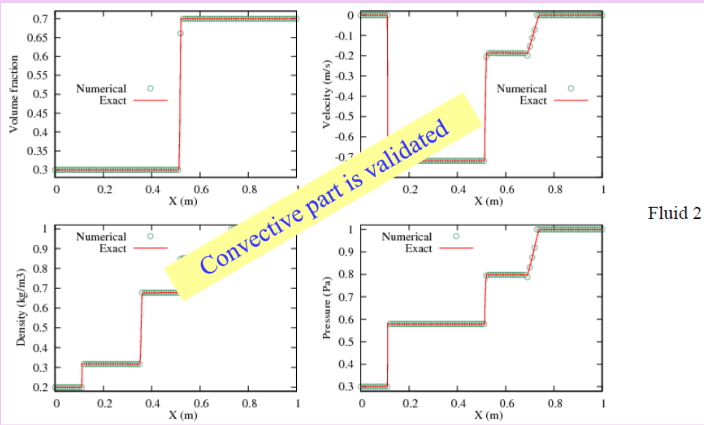
- Two-phase shock tube tests: analytical cases of the literature

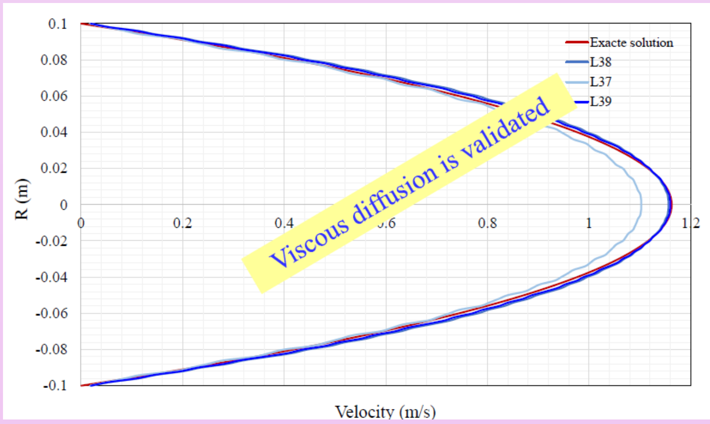
Viscous diffusion :

- Viscous diffusion: Poiseuille flow
- Momentum exchange: mixing layer between two fluids

Modelling of under-expanded gas jets

- Comparison between the numerical results & experiments
- Under-expanded gas jets in SGHE channel

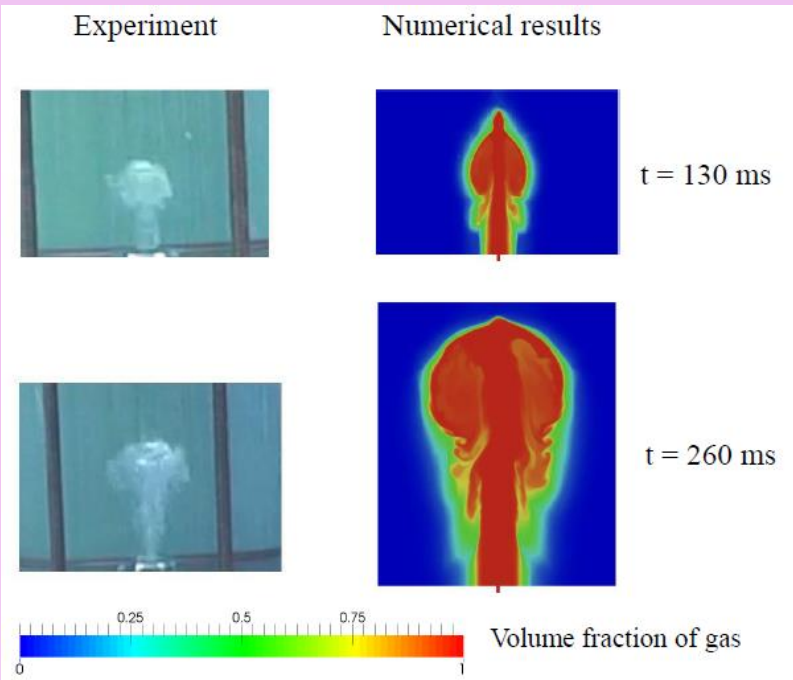




6. Under-expanded gas jets :

Left : Comparison with experiments (Colleoc1990)

Right : Gas jets submerged into sodium liquid in SGHE



Further experimental validation on IKHAR 2 facility in CEA Cadarache

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