Overview of Nuclear Reactors – Generation IV

Lecture for the Polish Nuclear Society

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Each illustration, figure, picture and/or table cited from the Handbook, will be associated with figure and page numbers for easy reference

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Contents

Generations of Nuclear Power Reactors

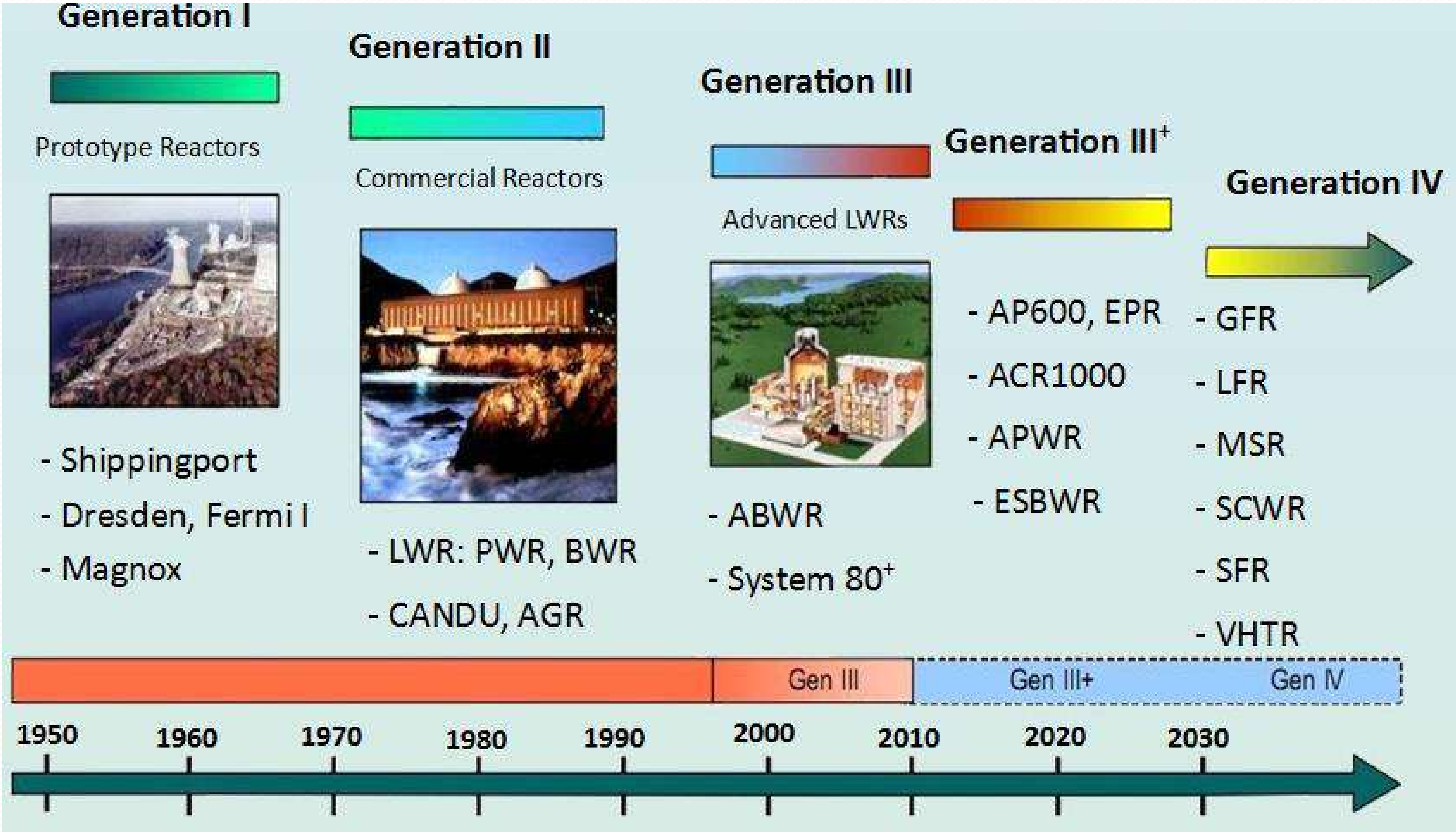
- **Generation I (1950-1970**) experimental and prototype reactors included early prototype reactors such as Shippingport, Dresden, and Fermi I in the USA

- **Generation II (1970-1990)** large, central-station nuclear power reactors included commercial power reactors such as the light water-cooled reactors (LWRs) with enriched uranium including the pressurized water reactor (PWR) and the boiling water reactor (BWR) (104 NPPs built in USA); CANDU in Canada; AGR in the UK, VVER & RBMK – in Soviet Union

- **Generation III and III+ (1990-2025)** evolutionary designs: it has started being deployed in the 1990s and is composed of the advanced LWR: the GE advanced boiling water reactor (ABWR), the Westinghouse Advanced Passive AP1000 and GE economic simplified boiling water reactor (ESBWR) in the USA, Enhanced CANDU6 (EC6) and ACR-1000 in Canada, EPR in France - these are considered as evolutionary designs offering improved safety and economics

- **Generation IV (2025 and beyond)** the next-generation designs

Generations of Nuclear Power Reactors



(Nuclear technology roadmap (U.S. DOE, 2002))

Generation IV International Forum
Origin & Purpose

- **The Generation IV International Forum (GIF)**
  - First meeting - January 2000 by the US Department of Energy's (DOE) Office of Nuclear Energy, Science and Technology with representation from: Argentina, Brazil, Canada, France, Japan, Republic of Korea, South Africa, the UK and the USA
- **Purpose of GIF**
  - International collaboration in the development of Generation IV nuclear energy systems
- The founding document of the GIF signed in July 2001 – it defines the framework for international cooperation in R&D for the next generation of nuclear energy systems

GIF Technology Goals

- Eight technology goals have been defined for Generation IV systems in four broad areas:
  - Sustainability
  - Economics
  - Safety and reliability
  - Proliferation resistance and physical protection
- These ambitious goals are shared by a large number of countries as they aim at responding to the economic, environmental, and social requirements of the 21st century; they establish a framework and identify concrete targets for focusing GIF R&D efforts

GIF Technology Goals

Sustainability

- Generate energy sustainability and promote long-term availability of nuclear fuel and minimize radioactive waste and reduce the long-term stewardship burden

Economics

- Have a life-cycle cost advantage over other energy sources
- Have a level of financial risk comparable to other energy projects

Safety and reliability

- Excel in safety and reliability
- Have a very low likelihood and degree of reactor core damage, and <u>eliminate the need for off-site emergency response</u>

Proliferation resistance and physical protection

- Be a very unattractive route for diversion or theft of weapon-usable materials, and provide increased physical protection against acts of terrorism

Choice of Generation IV Technologies

- The choice of Gen. IV technologies required 10-year efforts of GIF international collaboration
- More than 100 experts evaluated 130 reactor concepts before GIF selected **six** reactor technologies for further R&D as follows
  - Very high temperature reactor (VHTR)
  - Gas-cooled fast reactor (GFR)
  - Sodium-cooled fast reactor (SFR)
  - Lead-cooled fast reactor (LFR)
  - Molten salt reactor (MSR), and
  - Supercritical water-cooled reactor (SCWR)

Countries Developing Gen. IV Reactors

- Very high temperature reactor (VHTR)
  - USA, UK, China, Japan, Russia
- Gas-cooled fast reactor (GFR)
  - USA, Germany, Japan, France, Russia,
- Sodium-cooled fast reactor (SFR)
  - USA, Russia, China, South Korea
- Lead-cooled fast reactor (LFR)
  - Sweden, Belgium, Russia, Japan, Korea, USA, China, EU, ***Romania***, Italy, India
- Molten salt reactor (MSR)
  - Canada, China, Denmark, France, India, Japan, UK, USA, Russia, Australia
- Supercritical water-cooled reactor (SCWR)
  - Canada, Germany, Japan

Very High Temperature Reactors

Very High Temperature Reactors

Very high temperature reactors (VHTRs) - the evolutionary development of high temperature reactors (HTRs) built since the early 1960s as

- **Test HTRs in countries**
  - UK – "Dragon", operated 1963-1976 with outlet temp. 750°C
  - Germany - "AVR", operated 1967-1988 with outlet temp. 950°C
  - Japan – "HTTR", *operated 1988 till today* with outlet temp. 950°C
  - China – "HTR-10", *operated 2000 till today* with outlet temp. 700°C
- **Prototype HTRs in countries**
  - USA – "Peach Bottom", operated 1967-1974 with outlet temp. 725°C
  - USA – "FSV", operated 1976-1989 with outlet temp. 775°C
  - Germany – "THTR-300", operated 1986-1989 with outlet temp. 750°C
  - China - "HTR-PM", *operated 2017 till today* with outlet temp. 750°C

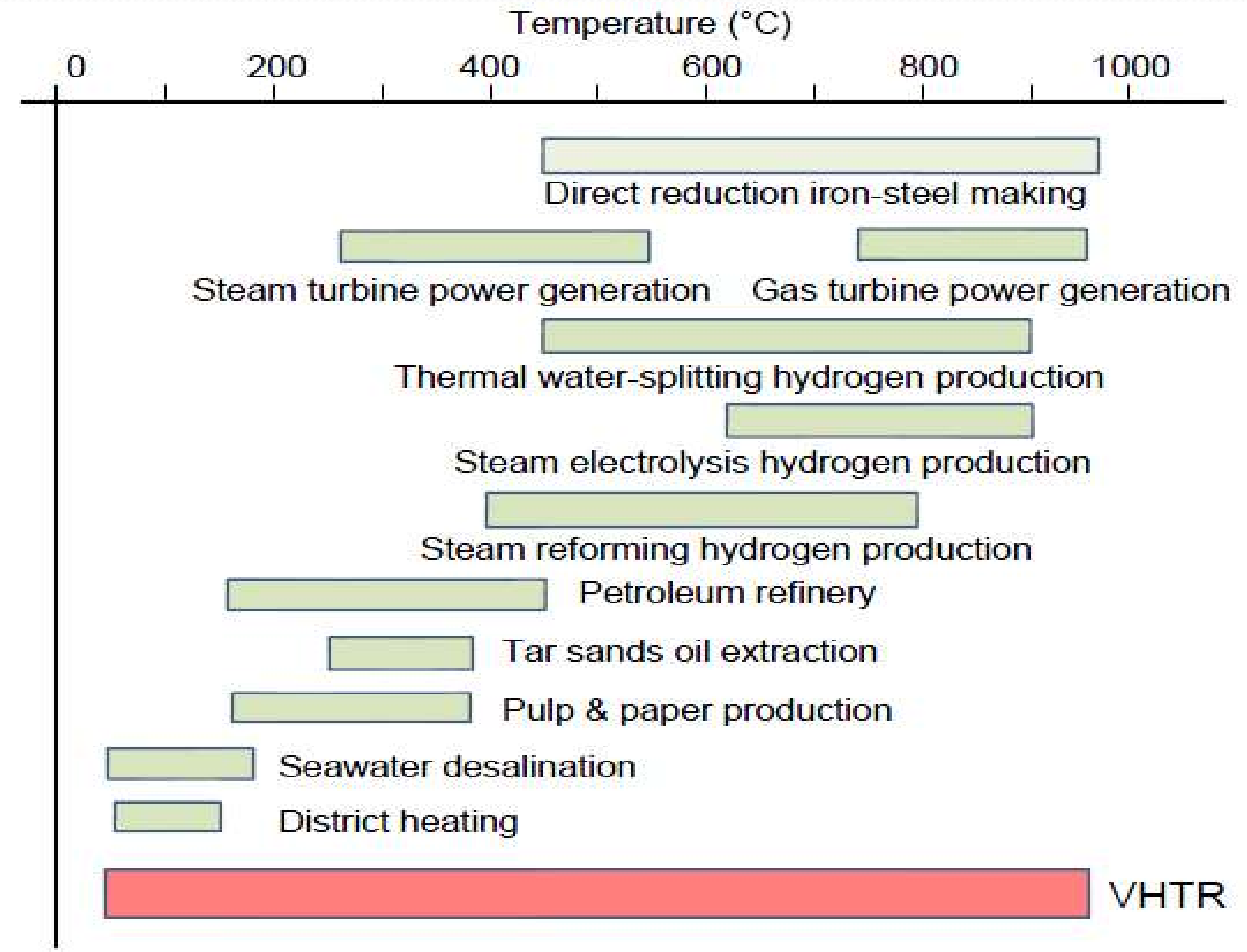
Very High Temperature Reactors

- **Major features of the VHTRs**
  - Helium gas-cooled reactors (at pressure range of 2-7 MPa)
  - Graphite-moderated (thermal neutron spectrum reactors)
  - Core coolant outlet temperature ≥ 900°C
  - The reference thermal power currently estimated ≈ 600 MWth (passive decay heat removal)
  - At first, a once-through low-enriched uranium (<20% U-235) fuel cycle, then closed fuel cycle; VHTR can support alternative fuel cycles such as U-Pu, Pu, mixed oxide (MOX), and U-Thorium (U-Th)
  - The VHTR has two typical reactor configurations
    - Prismatic block type, and
    - Pebble bed type
  - The fuel is the tri-isotropic (TRISO)- the coated particle fuel

VHTRs – Multipurpose Applications

- The supply temperature range of the VHTRs provides a number of applications
  - Power generation - by steam turbine efficiency at about 40% or by gas turbine at about 50%
  - Industrial heat applications can include the following
    - Hydrogen production (via thermo-chemical process (sulfur-iodine process or hybrid sulfur process), high temperature steam electrolysis, or the steam reformer technology)
    - Steelmaking
    - Desalination and
    - District heating
  - Cogeneration (power & heat) – <u>the reactor thermal power utilization up to 85%</u> (e.g., a 600-MWth reactor could produce 200 MWe by gas turbine, 66 t/day of hydrogen and 40,000 t/day potable water from desalination)
  - The system is expected to be available for commercial deployment by 2020

Very High Temperature Reactors



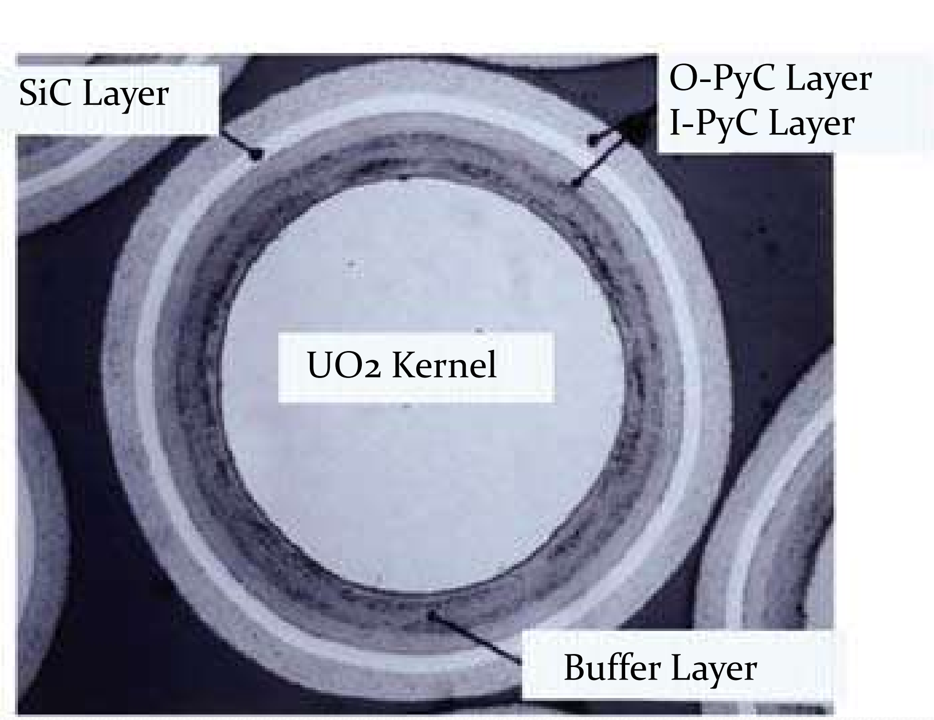
Temperature range of VHTR and heat demand of industries

(Handbook Figure 3.2, page 60)

Fuel & Core Design of VHTRs

- **TRISO** fuel = TRIstructural ISOtropic (three distinct structures have any property the same in every direction = isotropic); it is characterized by
  - The spherical fuel particle ~1 mm dia consists of inner nuclear kernel (<20% enriched UO_2) coated in successive layers of carbon and ceramics (silicon carbide) then carbon (i.e., high-density pyrolytic carbon)
  - Thousands of these particles packed in graphite matrix into a spherical pebble (like tennis ball size ~6 cm dia) or a cylindrical compact (about the size of man's thumb)
- **A prismatic core** contains many hexagonal graphite blocks in which the fuel compacts are embedded
- **A pebble bed** core contains a large number of fuel pebbles

TRISO-coated Fuel Particle



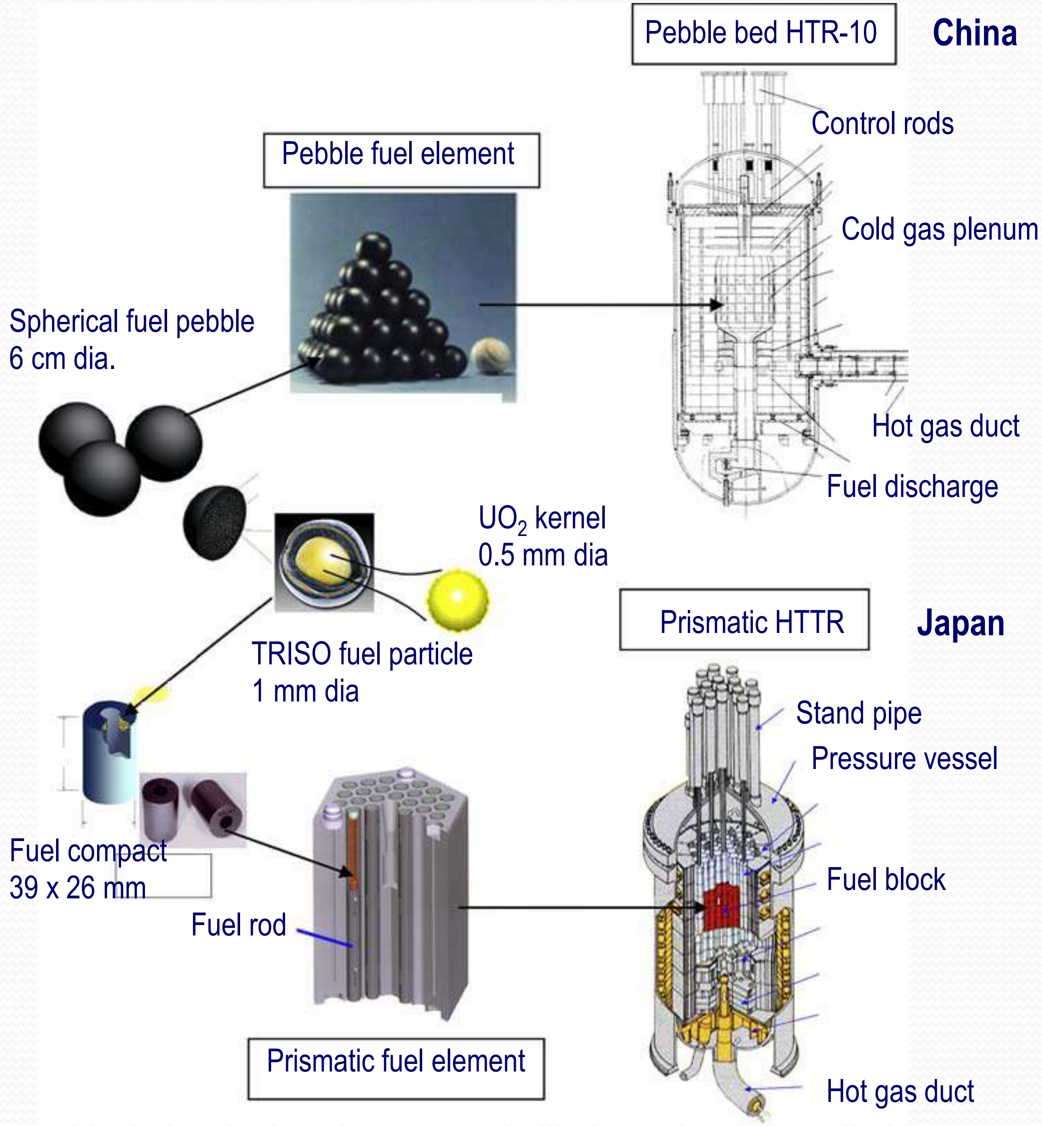
Diameter about 1 mm

SiC – silicon carbide layer
O-PyC –outer layer of high-density pyrolytic carbon
I-PyC – inner layer of high-density pyrolytic carbon
(Handbook Figure 3.3, page 61)

Core Design of VHTRs

- **A prismatic core** contains many hexagonal graphite blocks (150 in the HTTR core) in which the fuel compacts are embedded; helium coolant flows in the channels provided in the block
- **A pebble bed core** contains a large number of fuel pebbles (e.g., 27,000 in the HTR-10 core); helium coolant flows in the void volume formed in the pile of the pebbles
- Both cores are surrounded by **graphite reflector** and enclosed in a **steel pressure vessel**
- Reactivity control rods (RCRs) are inserted from above of the reactor pressure vessel (RPV)

Very High Temperature Reactors

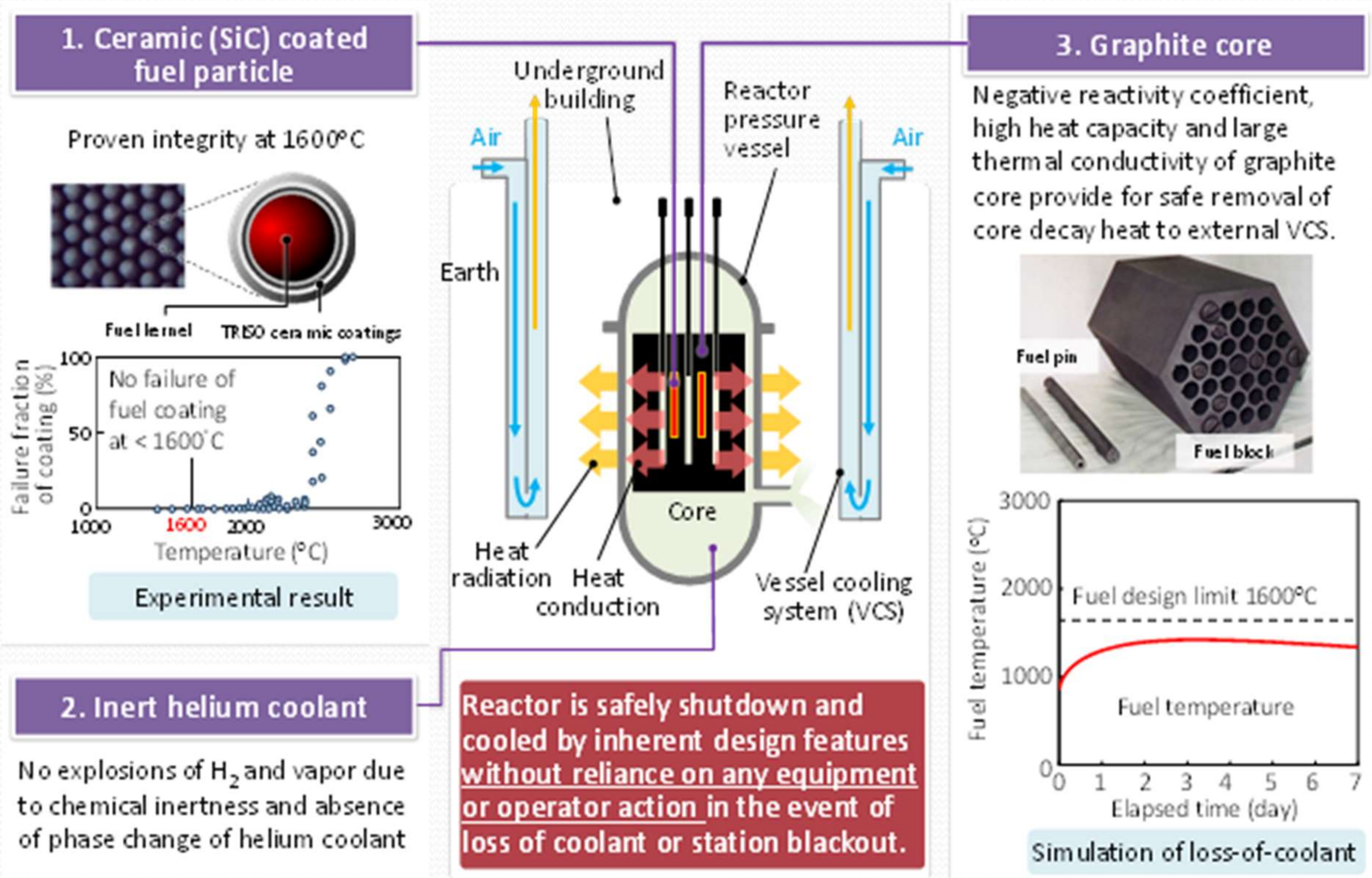


- 10 MWth per reactor unit
- Pile of 27,000 spherical fuel pebbles - continuously recirculated
- Core geometry is maintained by side graphite reflectors and carbon bricks
- Pebbles recirculate downward through the core by pneumatic fuel transport line, until reaching the design burnup of 100 GWd/t
- Spent fuel pebble discharged to spent fuel storage tank

VHTRs: Pebble Bed Reactor design & Prismatic Reactor design
(Handbook Figure 3.1, page 58)

- 30 MWth reactor power

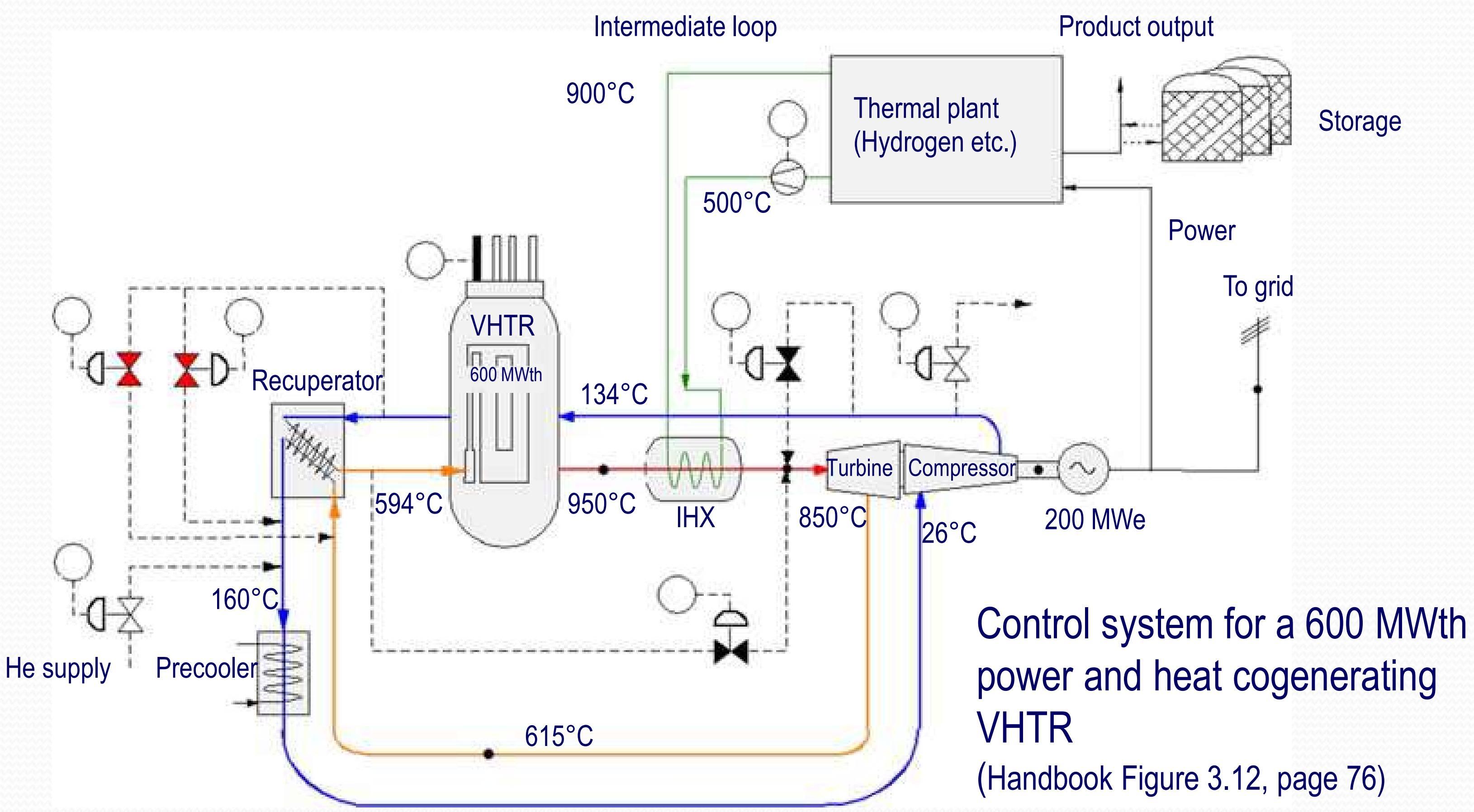
Inherent Safety Features of VHTRs



Inherent safety features of VHTR
(Handbook Figure 3.7, page 71)

Some Examples of Industrial Applications of VHTRs

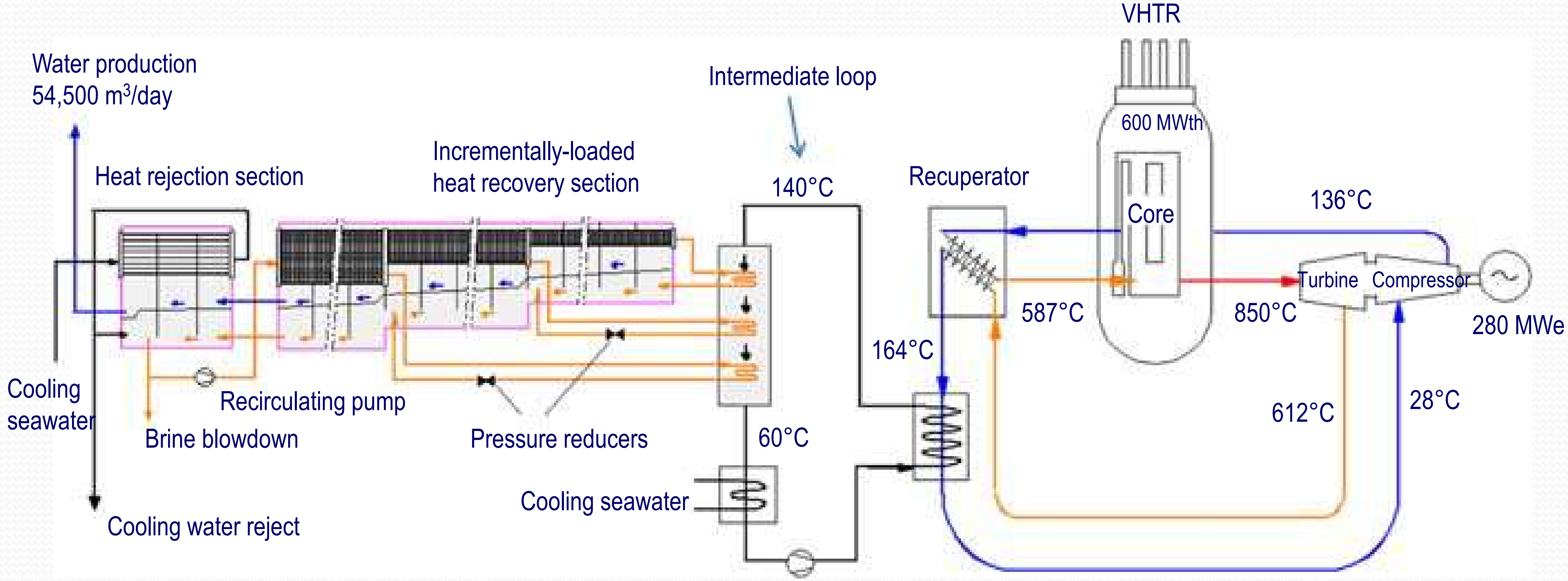
Power & Heat Cogeneration



Control system for a 600 MWth power and heat cogenerating VHTR

(Handbook Figure 3.12, page 76)

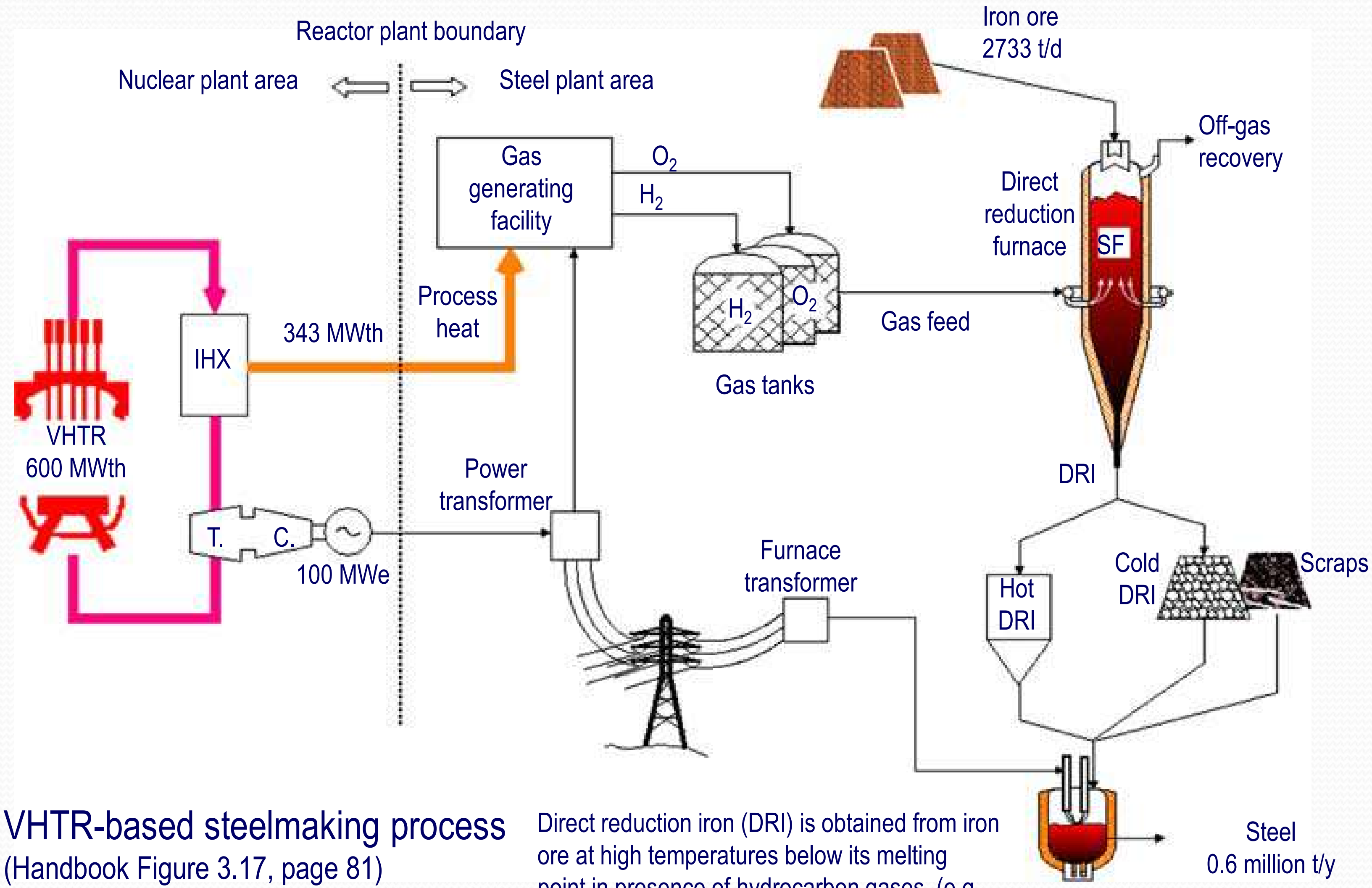
Power Generation & Desalination



Multistage flash system connected via intermediate loop (barrier between VHTR – desalination plant) Overall 600 MWth power utilization up to 87% from 47% in power generation alone

VHTR desalination cogeneration process (Reactor power 600 MWth)
(Handbook Figure 3.16, page 80)

VHTR-based Steelmaking Process



VHTR-based steelmaking process
(Handbook Figure 3.17, page 81)

Direct reduction iron (DRI) is obtained from iron ore at high temperatures below its melting point in presence of hydrocarbon gases (e.g., H_2 and CO) – reducing iron ore to metallic iron

Other Concepts & Designs of Generation IV Nuclear Reactors

Other Generation IV Nuclear Reactors

- Presented VHTRs operate on neutrons within the **thermal spectrum of energy** (i.e., brought to energy level of ~0.025 eV that is needed to fission U-235; they require a neutron moderator
  - 0.7% of NU will fission by thermal neutrons
- Other reactors proposed as Gen. IV technologies are operated on **fast neutrons** (these reactors, for short, are called "<u>fast reactors</u>") – no moderator needed
  - 99.3% of NU will fission by fast neutrons of > 5 MeV

Burners versus Breeders

- **Fast reactors can play a role as**
  - Burners - the case when the ratio of final to initial fissile content is <1

    = reactors consume more fissile material (U-235 and minor actinides) than they produce (fissile Pu)
  - Breeders – it is the case when this ratio is >1; if the ratio =1 the reactor is called iso-breeder
- If the core of the fast reactor is surrounded by a steel reflector then only **burning** is possible, but **breeding** is possible with U-238 fertile blanket around the core
- **Gen. IV reactors** are designed to (among other features) effectively utilize fuel and reduce radioactive waste by **burning** actinides and transuranium elements*

* These radioactive elements are created from uranium U235 by absorbing additional neutrons (atomic no > 92)

General Advantages of Fast Reactors

- **Advantages of nuclear reactors on fast neutrons**
  - Lack of moderator
  - Reactor size, weight & complexity greatly reduced (used e.g., on submarines)
  - Running reactor on Pu-239 – more neutrons available than from U-235 fission (this surplus breeds more Pu-239 that it consumes)
  - Have a strong negative temperature coefficient of reactivity – inherent safety feature
  - A single fast reactor can extract more energy from NU than sseveral thermal reactors (<1% in a normal once-through cycle to as much as 60% in the best fast reactor cycles)

Advantages of Fast Reactors for Ontario

- **Advantages of fast reactors among Ontario's CANDU fleet***
  - Productive elimination of 54,000 tons of used CANDU fuel (elimination of the "million-year" radiotoxic TRUs (Pu, Am, Cm, etc.); fission products as residue - would required short-term safe storage (~300 years)
  - From this radioactive "waste" 134 x more non-carbon electricity can be generated than from original fresh fuel
  - In Canadian context, it would be worth of $81 trillion ($1.5 billion per ton of used fuel)
  - This avoids emissions of 510 billion tons of CO_2 to the atmosphere
  - It replaces use of natural gas (reduce GHG emissions even further)
  - Allows local fuel recycling; no transportation of radioactive materials on roads, through towns (today's example: GE-Hitachi PRISM FNR plus ANL fuel cycle facility)
  - It is "green" emission-free fuel (this fuel exists, no mining required)
  - For Ontario, it would provide energy for > 4,000 years from nuclear "waste" only!

*By permission of Prof. Peter Ottensmeyer of University of Toronto (2018)

General Disadvantages of Fast Reactors

- **Disadvantages of nuclear reactors on fast neutrons**
  - Expensive fuel due to a higher enrichment
  - MOX as a product from fast reactors still very expensive fuel
  - Nuclear proliferations and security issues
  - Liquid metals as coolants become radioactive, boil during accidents – positive void coefficient of reactivity
  - <u>Small size of the core with high power density requires very efficient heat transfer to avoid core damage</u>

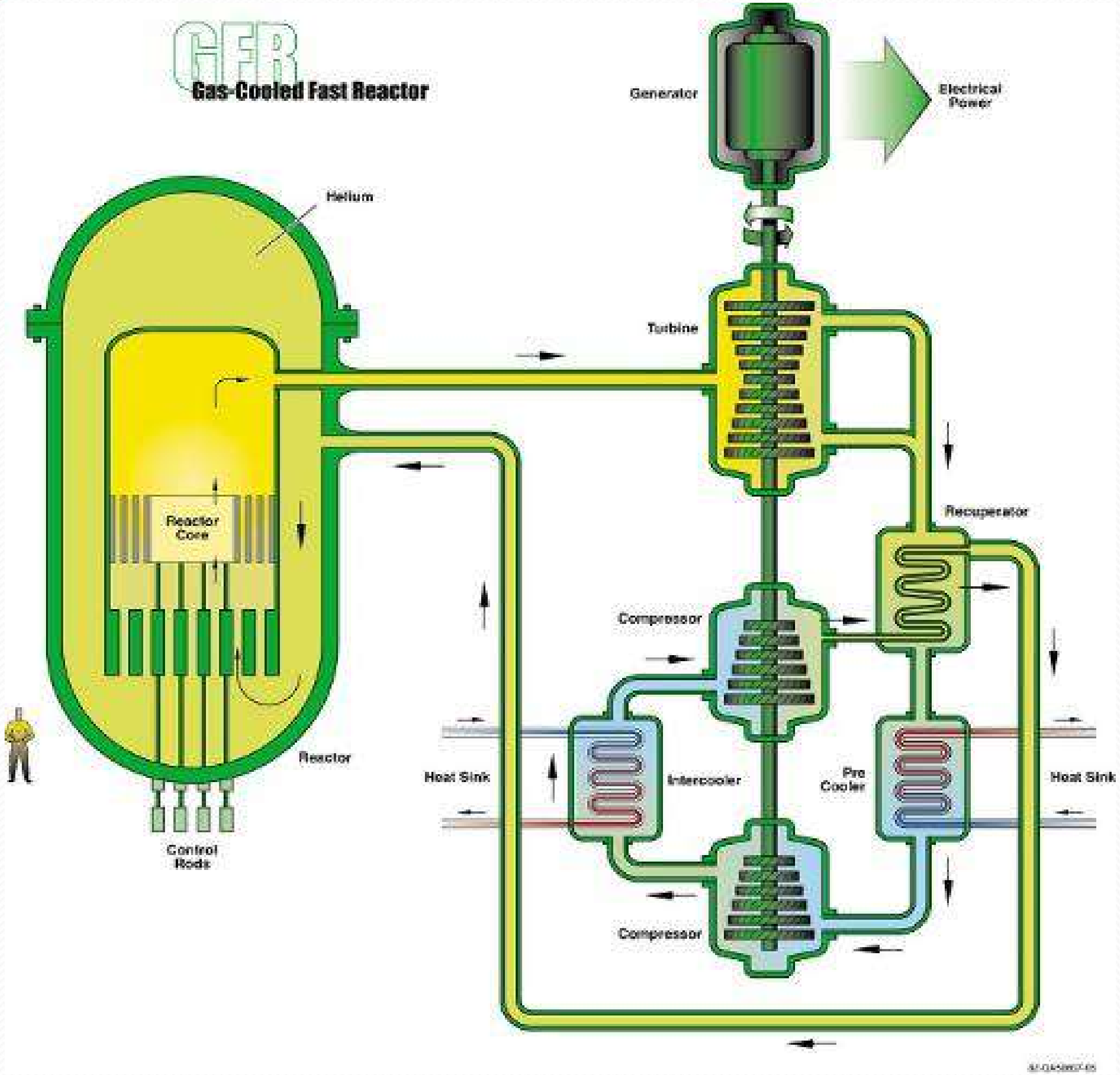
Gas-cooled Fast Reactors

Gas-cooled Fast Reactors

General features of gas-cooled fast reactors (GFRs)

- Coolant – helium (pressure above 7 MPa to compensate for low heat capacity of He)
- High-temperature (core outlet temperature of 850°C) - high thermal cycle efficiency and industrial use of the generated heat
- Fast neutron spectrum (ensures long-term sustainability of uranium resources and waste minimization)
- Closed fuel cycle (through fuel multiple reprocessing and fission of long-lived actinides)
- Fuel of high density such as mixed silicone carbide-fueled pins in ceramic-clad that provides good performance regarding plutonium breeding and minor actinide burning

Gas-cooled Fast Reactor



GFR shown with direct gas turbine Brayton power cycle and heat for industrial applications (e.g., second steam Rankine cycle or hydrogen production)
(Handbook Figure 2.4, page 43)

The GFR reference design based on 2400-MWth reactor core in a steel pressure vessel consisting of assembly of hexagonal fuel elements, each consisting of ceramic-clad, mixed carbide-fueled pins.
The favored material at the moment for the pin clad is silicon-carbide and for hex tubes is fiber-reinforced silicon carbide.

Sodium-cooled Fast Reactors

Sodium-cooled Fast Reactors

Glimpse on History

- **Basic SFR technology has been established in former fast reactor programs, and was confirmed by**
  - Phenix demonstration tests in France (250/563 MWe/MWth) - start1973 till 2010
  - Operation of the Monju demonstration reactor in Japan (280/714 MWe/MWth), and
  - Lifetime extension of the BN-600 demonstration reactor in Russia (600/1470 MWe/MWth) – started in 1980
- **New programs involving SFR technology include**
  - France & Japan – Advanced Sodium Technical Reactor for Industrial Demonstration (ASTRID)
  - China Experimental Fast Reactor, which was connected to the grid in July 2011
  - India’s Prototype Fast Breeder Reactor, and
  - The latest success in Russia with putting into operation the BN-800 reactor
- Owing to the significant past experience with sodium-cooled reactors in several countries, the deployment of SFR systems is targeted for 2020

Advantages of Liquid Sodium as Coolant

- Sodium-23 (the only stable sodium isotope) - a very weak neutron absorber and moderator (ideal to maintain fast neutrons in the reactor core)
- **Major features of liquid sodium**
  - High thermal conductivity and heat capacity = thermal inertia against overheating
  - It does not to be pressurized – its boiling point* (>880°C) is much higher than reactor's operating temperature by a large margin
  - High operating temperature (>500°C) = high cycle efficiency
  - It does not cause corrosion of steel reactor parts
  - It is electrically conductive – can be pumped by electromagnetic pumps

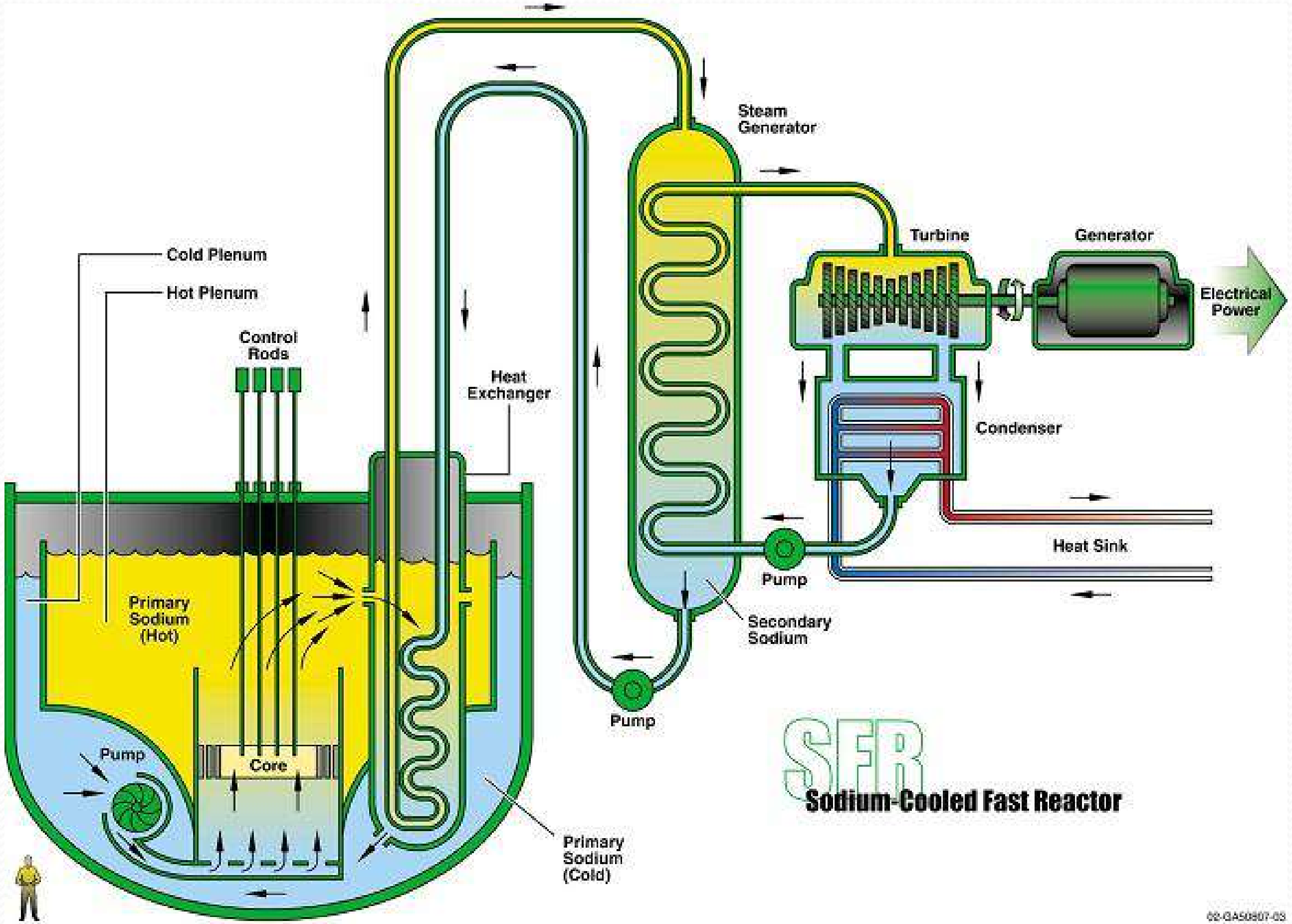
*Melting point 97.8°C

Disadvantages of Liquid Sodium as Coolant

- Liquid sodium is very much chemically reactive
  - If comes in contact with water – it explodes
  - If comes in contact with air – it burns
- Under neutrons collisions it becomes radioactive (however, its half-life time is only 15 hrs)

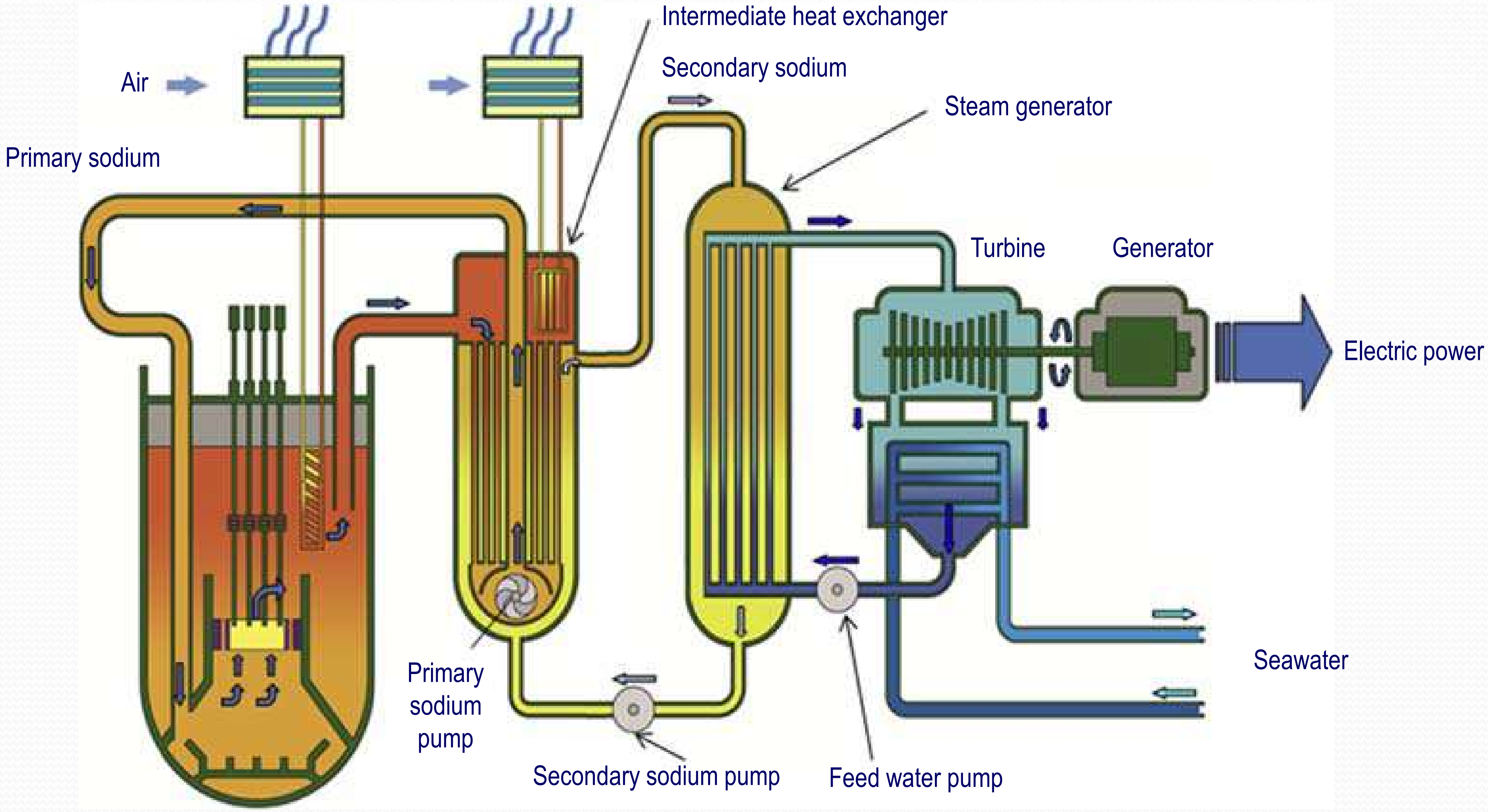
Sodium-cooled Fast Reactors

- SFR - long thermal response time
- Power conversion system - high thermal efficiency, safety, and reliability
- Primary system operates near atm. pressure
- Intermediate sodium system links power conversion system which can be as
  - Rankine cycle (water as working fluid) or
  - Brayton cycle (supercritical carbon dioxide or nitrogen as working fluids)



<u>Pool-type</u> sodium-cooled fast reactor system
(Handbook Figure 5.3, page 103)

Sodium-cooled Fast Reactors



Loop-type of sodium-cooled fast reactor system
(Handbook Figure 5.4, page 104)

Lead-cooled Fast Reactors

Lead-cooled Fast Reactors

Countries & Designs - History

- **Soviet Union**, in the 1950s lead (LBE)-cooled reactors built for 15 submarines (of 73 & 155 MWth) (the last submarine decommissioned in 1995) = 80 reactor years of operating experience
  - This operational experience = strong base to understand the technology and identifying solutions to the technical challenges for currently considered Gen. IV LFR systems
- **Russia**, from the year 2000 - two LFRs initiatives are currently being pursued (a small SVBR-100 cooled by LBE and a medium-size BREST cooled by lead)
- **Europe**
  - From 2006 the ELSY (**E**uropean **L**ead-cooled **SY**stem) project initiated to define an LFR of industrial size of 1500 MWth and 600 MWe
  - From 2010 the LEADER (**E**uropean **A**dvanced **L**ead-cooled **R**eactor **D**emonstration) project initiated for an industrial-sized reactor under the name ELFR (**E**uropean **L**ead **F**ast **R**eactor) and also is examining a demonstrator LFR of power 100 MWe called ALFRED (**A**dvanced **L**ead **F**ast **R**eactor **E**uropean **D**emonstrator) that is under consideration for construction in **Romania**

Lead-cooled Fast Reactors

Countries & Designs

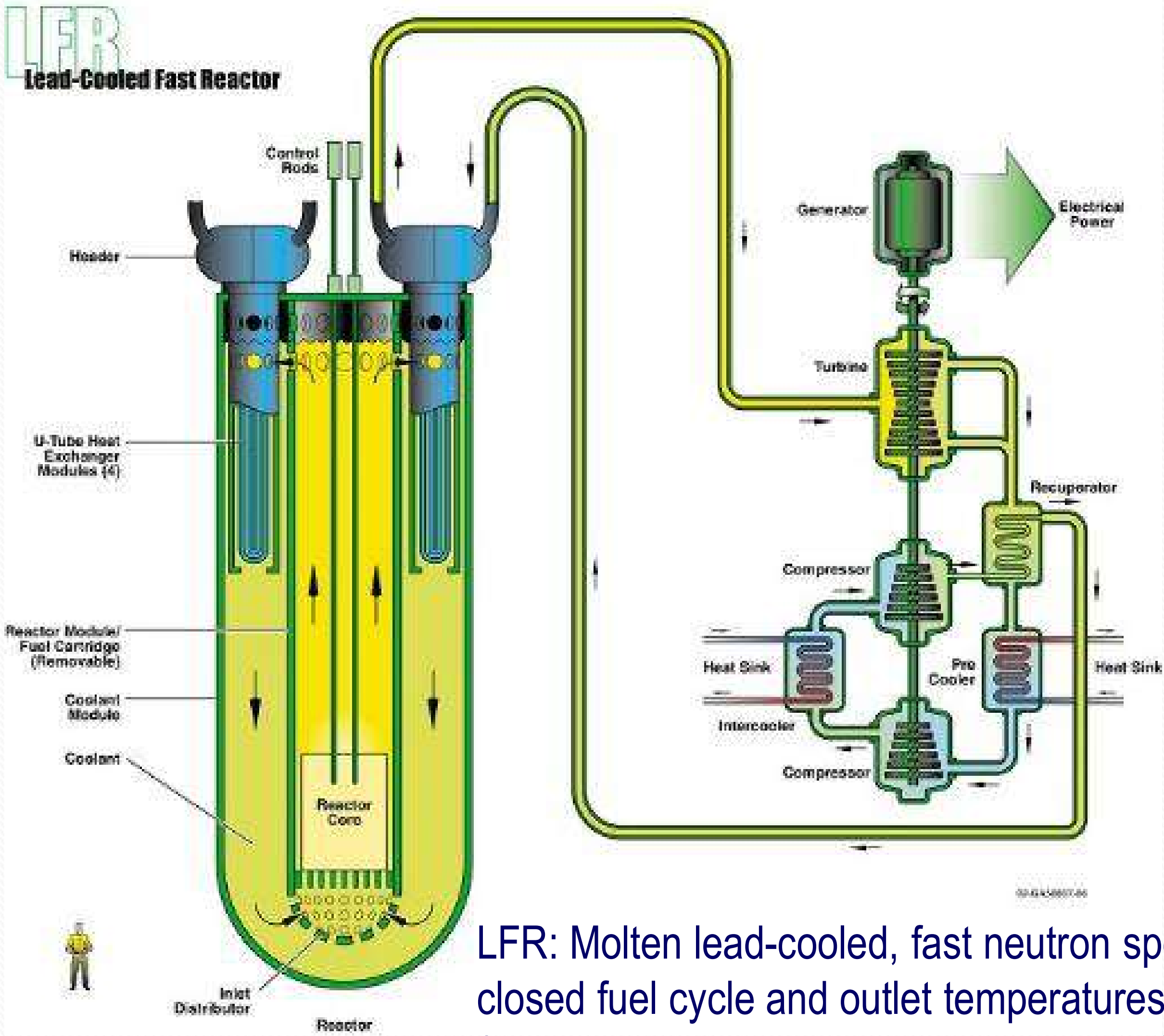
- **Europe**
  - **Belgium**, SCK-CEN intends to build an ADS (**A**ccelerator **D**riven **S**ystem) demonstrator, called MYRRHA (**M**ulti-purpose h**Y**brid **R**esearch **R**eactor for **H**igh-tech **A**pplications) coupling a particle accelerator with a reactor; MYRRHA would be cooled by LBE
- **USA**, from 2008 as a reference design of a small LFR known as SSTAR (**S**mall **S**ecure **T**ransportable **A**utonomous **R**eactor)
- **China**, from 2012 **C**hina **LEA**d-based **R**eactor (CLEAR) is the reference reactor for China's Lead-based Fast Reactor Development Plan
- Several additional design studies have been or are being carried out in a number of other countries including the **South Korea** (URANS-40)**,** **Japan (**BWFR) and **Sweden** (SEALER)

Lead-cooled Fast Reactors

General features of Lead-cooled Fast Reactors (LFRs)

- Coolant may be either **lead (preferred option)**, or lead-bismuth eutectic (LBE)
- Lead and LBE are relatively inert liquids with very good thermodynamic properties
- Coolants operate at high temperatures (500-800°C) and low pressure (near atmospheric); hence, multiple applications including production of electricity, hydrogen, and process heat
- Molten lead - very high margin to boiling (at 1737°C) and <u>benign interaction with air or water</u> = enhanced safety of LFRs

Lead-cooled Fast Reactor



LFR: Molten lead-cooled, fast neutron spectrum reactor with closed fuel cycle and outlet temperatures within 500-550°C (shown with indirect Brayton power cycle)
(Handbook Figure 2.6, page 46)

Molten Salt Cooled Reactors

Molten Salt Cooled Reactors
Glimpse on History

- Molten salt cooled reactor (MSR) technology was first studied more than 50 years ago
- Significant experimental studies were performed at the Oak Ridge National Laboratory (ORNL), USA, in the 1950s and 1960s
- Two demonstration reactors were built and tested providing an experimental basis for their feasibility
- These MSRs ran **on thermal neutron spectrum** and were graphite-moderated reactors; they used liquid fuel in the mixture of fluoride salts based on lithium and beryllium (i.e., LiF-BeF_2-UF_4) reaching temperatures above 600°C

Molten Salt Cooled Reactors Types

- Molten salt reactors can be divided into two categories as those using
  - Molten salt as a **coolant only**, but nuclear **fuel is solid** (e.g., ceramic fuel dispersed in a graphite matrix)
  - The nuclear **fuel** (e.g., uranium tetrafluoride - UF_4) **dissolved** in the coolant molten salt – two functions performed by one medium
  - **Thermal neutrons** – with potential to breed U-233 from thorium Th-232
  - **Fast neutrons** – to breed new fissile materials and burn nuclear "waste" – as a new development of molten salt cooled fast reactors (MSFRs) within the Gen. IV initiative
- Molten salts provide low pressure, high temperature cooling, efficiently removing heat from the core (as compared to gas coolants) thus reducing pumping power, simplifying piping system and also reducing the core size

MSR Advantages

Molten salt fuel mixture offers potential advantages over solid-fueled systems such as

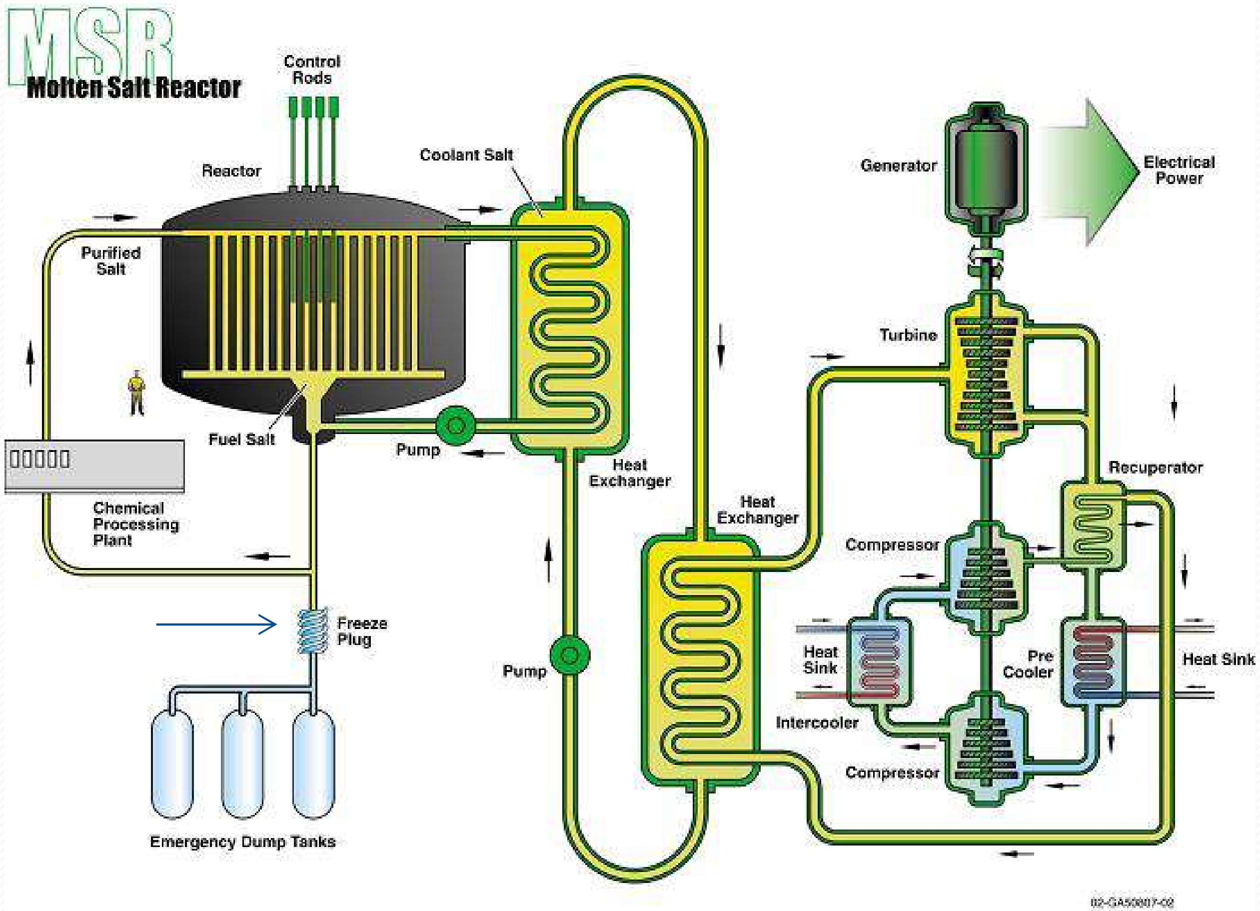
- Possibility of fuel composition (fertile/fissile) adjustment
- Fuel reprocessing without shutting down the reactor
- Overcoming the difficulties of solid fuel fabrication and handling
- Better resource utilization by high fuel burn-ups (with TRUs remaining in the liquid fuel to undergo fission)
- MSFR systems have lower fissile inventories, no reactivity reserve, strongly negative reactivity coefficients, and a homogeneous isotopic fuel composition in the reactor (no loading plan required)

MSFR Further Advantages

A circulating liquid fuel also playing the role of the coolant presents some more advantages such as

- Rapid, passive, fuel geometry reconfiguration via gravitational draining (quick fuel removal from the core to drainage tanks via cold plugs = passive safety feature)
- Molten fluorite salts do not generate burnable hydrogen
- Heat production directly in the fuel (no heat transfer delay to the coolant)
- Due to continuous removal of noble fission products (e.g., Xe, Kr) no possibility of Xe poisoning of the reactor core (as the case in LWRs when power load changes or at shutdown)

Molten Salt Cooled Reactors



MSR: Molten salt-cooled reactor with outlet temperatures within 700-800°C (shown with indirect Brayton power cycle)
(Handbook Figure 2.7, page 48)

Supercritical Water-cooled Reactors

Supercritical Water-cooled Reactors
Glimpse on History

- In the 1950s the attractive idea - using water at supercritical conditions (i.e., above its critical point of 22.1 MPa and 374°C) was investigated for **steam generators** in fossil-fired power plants
- This work was done in the USA and the Soviet Union
- In the early 1960s this idea was developed further and implemented in designs of **nuclear reactors** (in the USA, the Soviet Union, the UK, France)
- However, the idea was abandoned for almost 30 years with emergence of LWRs and only regained interest in the 1990s following LWR's maturation
- Since 2010 GIF indicated the SCWR technology as one of Gen. IV technologies to be developed and implemented (N.B. as of 2012 this concept was investigated in 13 countries)

Advantages of SCWRs

- Water above its critical point becomes like a dense gas (steam & liquid of the same density = single-phase coolant)
- This coolant feature eliminates the need for
  - Steam generators and pressurizers - as needed in PWRs
  - Jet and/or recirculation pumps, steam separators and dryers - as needed in BWRs
  - (Thus reactor coolant pumps are not required, the only pumps driving SCW under normal operating conditions are feed water pumps and condensate extraction pumps)
- By avoiding coolant boiling in the core there is no danger of occurrence of boiling crises or critical heat flux (CHF) - the most limiting factor in operation of LWRs

Advantages of SCWRs

- Coolant experiences a significantly higher enthalpy rise in the core (in contrast to LWRs by factor of 8) which reduces the core mass flow for a given thermal power and increases the core outlet enthalpy to superheated conditions
- SCWR has the simplest type of cycle possible
  - Direct-cycle design: SCW ("ssuperheated steam") is supplied directly to a high-pressure steam turbine and the feed water from the steam cycle is supplied back to the core (as in current BWRs) (= a supercritical Rankine cycle)
  - Once-through design: a once-through steam cycle - no coolant recirculation inside the reactor (as in current BWRs)
- Higher operating coolant temperatures allow for higher thermodynamic cycle efficiency (about 45% vs 33-36% of current LWRs)

Advantages of SCWRs

- SCW has excellent heat transfer characteristics thus allowing for a high power density, a small core and containment size = NPP simplification and improved economics
- Containment, pressure suppression pools, emergency cooling and residual heat removal systems can be significantly smaller than those of current LWRs
- The higher steam enthalpy allows to decrease the size of the turbine system – thus lowering the capital costs of the BOP (i.e., Balance of the Plant – the conventional part of an NPP)

Disadvantages of SCWRs

- Compact primary cooling system (as compared to LWRs) with a low coolant inventory = less heat capacity to deal with transients and loss-of-coolant accidents (too high temperatures of conventional metallic fuel cladding)
- Higher coolant pressure & temperature and a higher temperature rise across the core (as compared to LWRs) = increased mechanical and thermal stresses of the core structure
- Higher coolant temperature at the core end = lower coolant density that requires an addition of extra moderator in this core region
- Need for special SCW chemistry to prevent stress corrosion cracking, embrittlement, and retain strength and creep resistance under neutron radiation
- Special start-up procedures have to be established to avoid flow instabilities before water reaches supercritical conditions

Supercritical Water-cooled Reactors

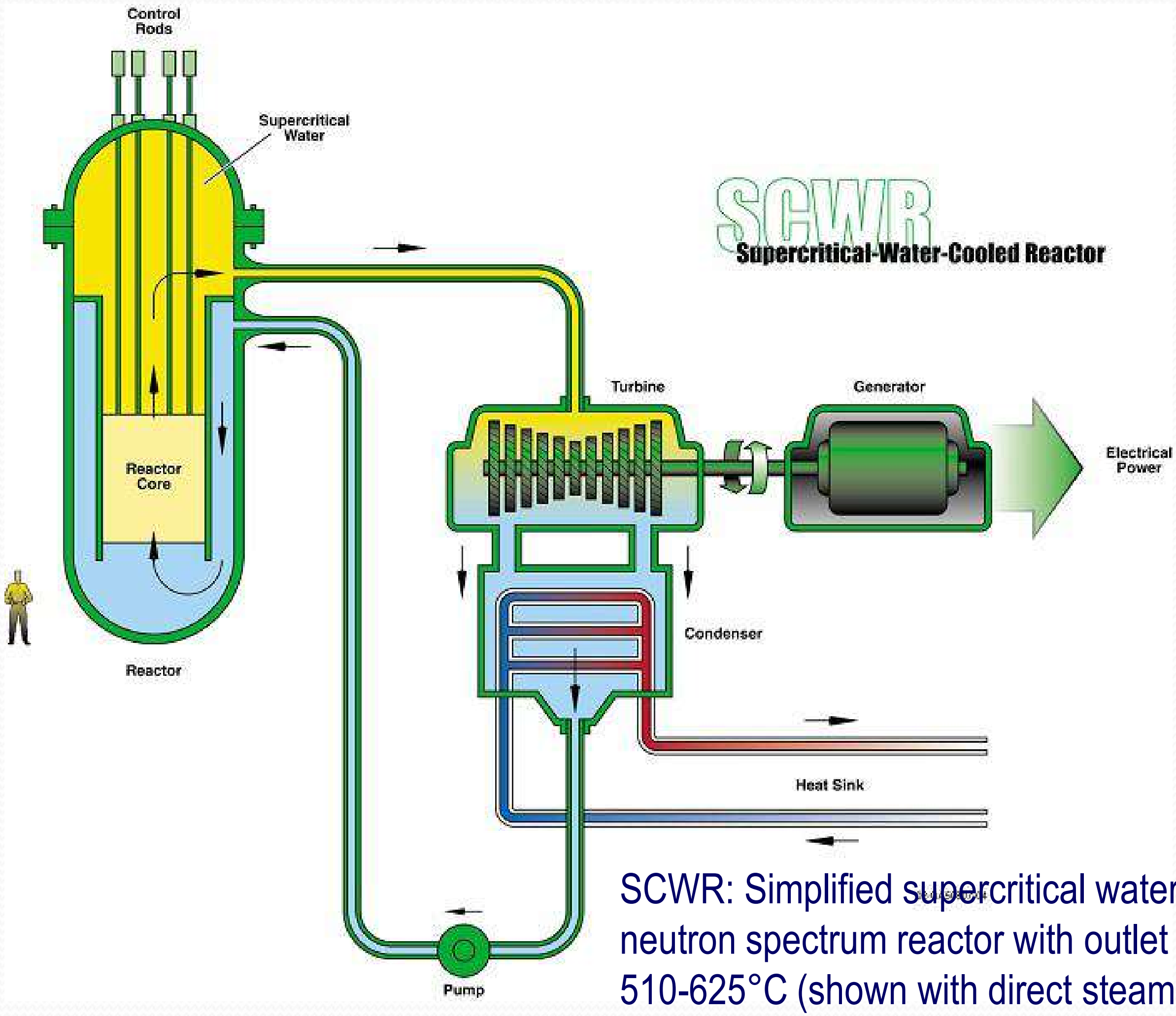
- Main features of proposed SCWRs:
  - Both pressure-vessel and pressure-tube configurations designs are considered
  - Both thermal neutron and fast neutron spectra are considered
  - Both the use of light water or heavy water as moderators are considered, and
  - For both pressure-vessel and pressure-tube designs, a direct-cycle and a once-through steam cycle are considered
- The operation of a 30 to 150 MWe technology demonstration SCWR is targeted for 2022

Supercritical Water-cooled Reactors

Pressure Vessel-type Reactors

- **Japan** - pre-conceptual core design studies of 1700 MWe based on a pressure vessel-type reactor performed (core outlet temperature of > 500°C) assuming both thermal & fast neutron spectra; both options based on coolant heat-up in two steps with intermediate mixing underneath the core; study confirms - net efficiency of 44% and a cost reduction of 30% (vs current LWRs)
- **Europe** - pre-conceptual design of a pressure vessel-type reactor (500°C core outlet temperature) and 1000 MWe power has been developed: core design is based on coolant heat-up in three steps with additional moderator for the thermal neutron spectrum is provided in water rods and in gaps between assembly boxes; study confirms results obtained in Japan wrt efficiency and a cost reduction

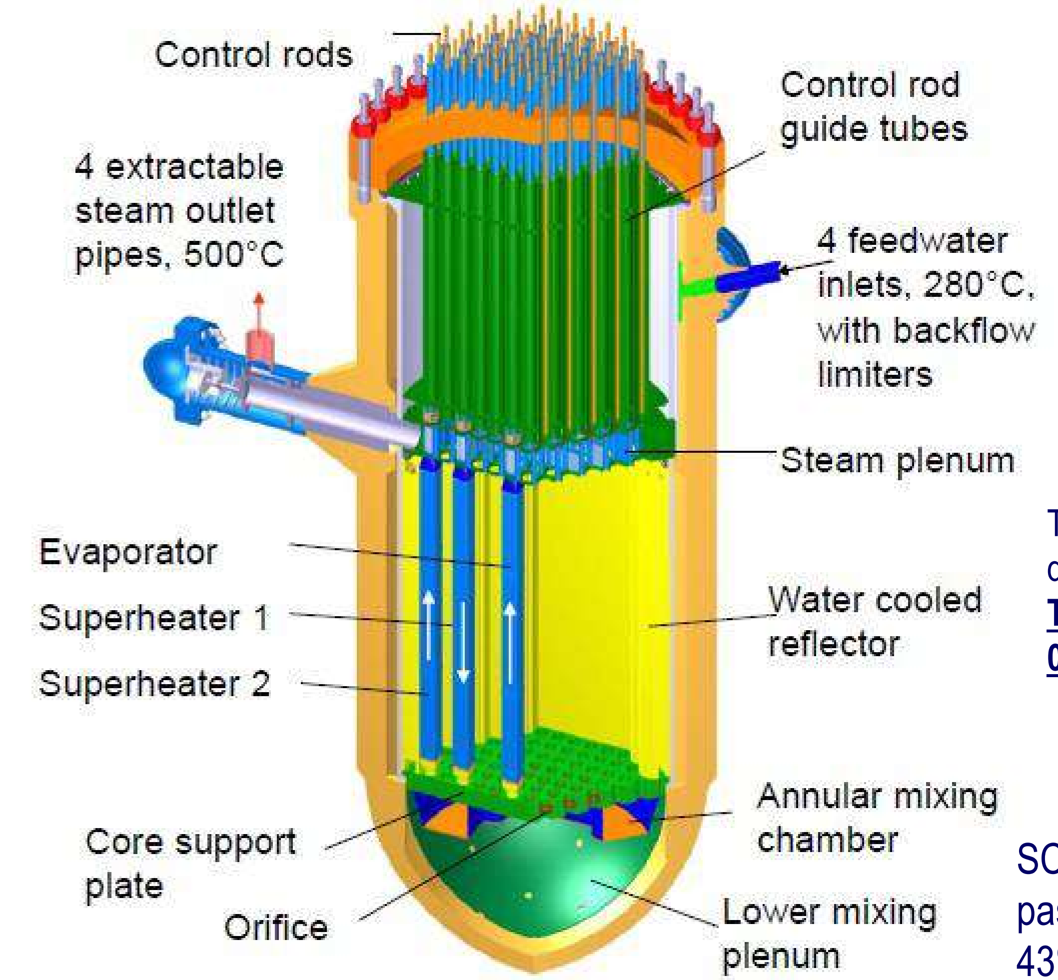
Supercritical Water-cooled Reactors
Pressure Vessel-type Reactor



SCWR: Simplified supercritical water-cooled, thermal neutron spectrum reactor with outlet temperatures within 510-625°C (shown with direct steam turbine Rankine power cycle)
Handbook Figure 2.8, page 50)

Supercritical Water-cooled Reactors

Pressure Vessel-type Reactor



Backflow limiters to minimize loss of coolant in case a feed-water line breaks

The reactor total height of 14.29 m and an inner diameter of 4.46 m

The wall thickness of the cylindrical shell is 0.45 m

SCWR of pressure-vessel design with a three-pass core concept; 2300 MWth/1000 MWe, 43% net efficiency; peak coolant temp <600°C

(Handbook Figure 8.3, page 193)

Supercritical Water-cooled Reactors

Pressure Tube-type Reactors

- **Canada** – development of a pressure tube-type SCWR concept with a 625°C core outlet temperature at the pressure of 25 MPa
- It adopts the direct cycle, which includes a 2540-MWth core that receives feed water at 315°C and 1176 kg/s: the concept is designed to generate 1255 MWe with 49.4% efficiency (300-MWe concept is also considered)
- (The cycle includes steam reheat using a moisture separator reheater (MSR) between the IP turbine and LP turbine; the MSR separates the moisture from the steam and reheats the steam to ensure an acceptable moisture level at the outlet of the LP turbine; four LP condensate heaters are included in the cycle as well as a deaerator and four HP feed-water heaters)

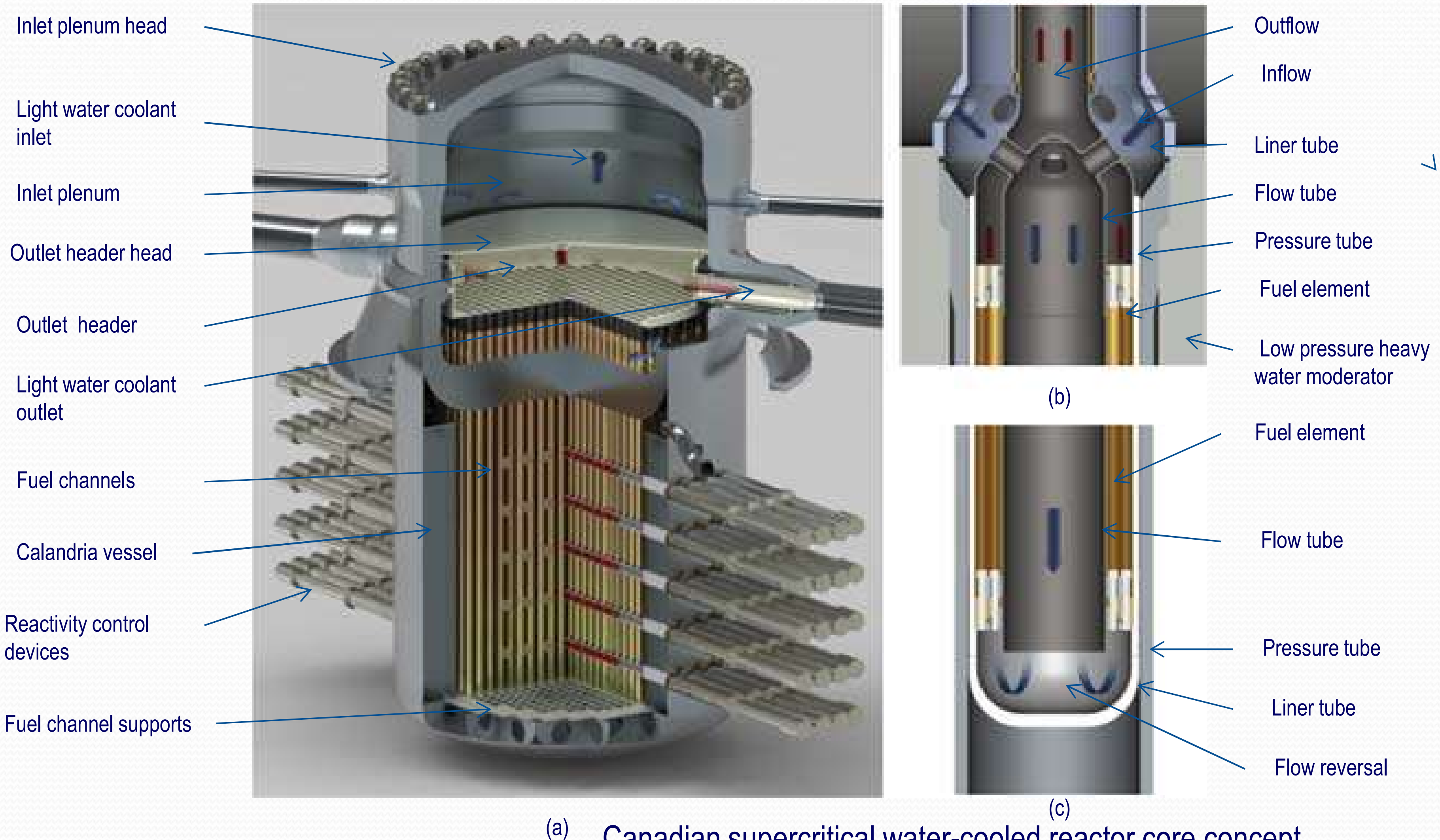
Supercritical Water-cooled Reactors

Pressure Tube-type Reactors

- The design has a modular fuel channel configuration with separate coolant and moderator
- A high-efficiency fuel channel is incorporated to house the fuel assembly
- The heavy water moderator directly contacts the pressure tube and is contained inside a low pressure calandria vessel
- SCWR may use UO_2 in a once-through fuel cycle, with an enrichment of 5-7%, or mixed oxide (MOX) fuel if plutonium should be recycled in a closed fuel cycle; in the case of a thermal neutron spectrum, the use of MOX fuel is optional
- However, because the higher temperatures of the SCWR require stainless steel fuel claddings instead of Zircalloy claddings

Supercritical Water-cooled Reactors

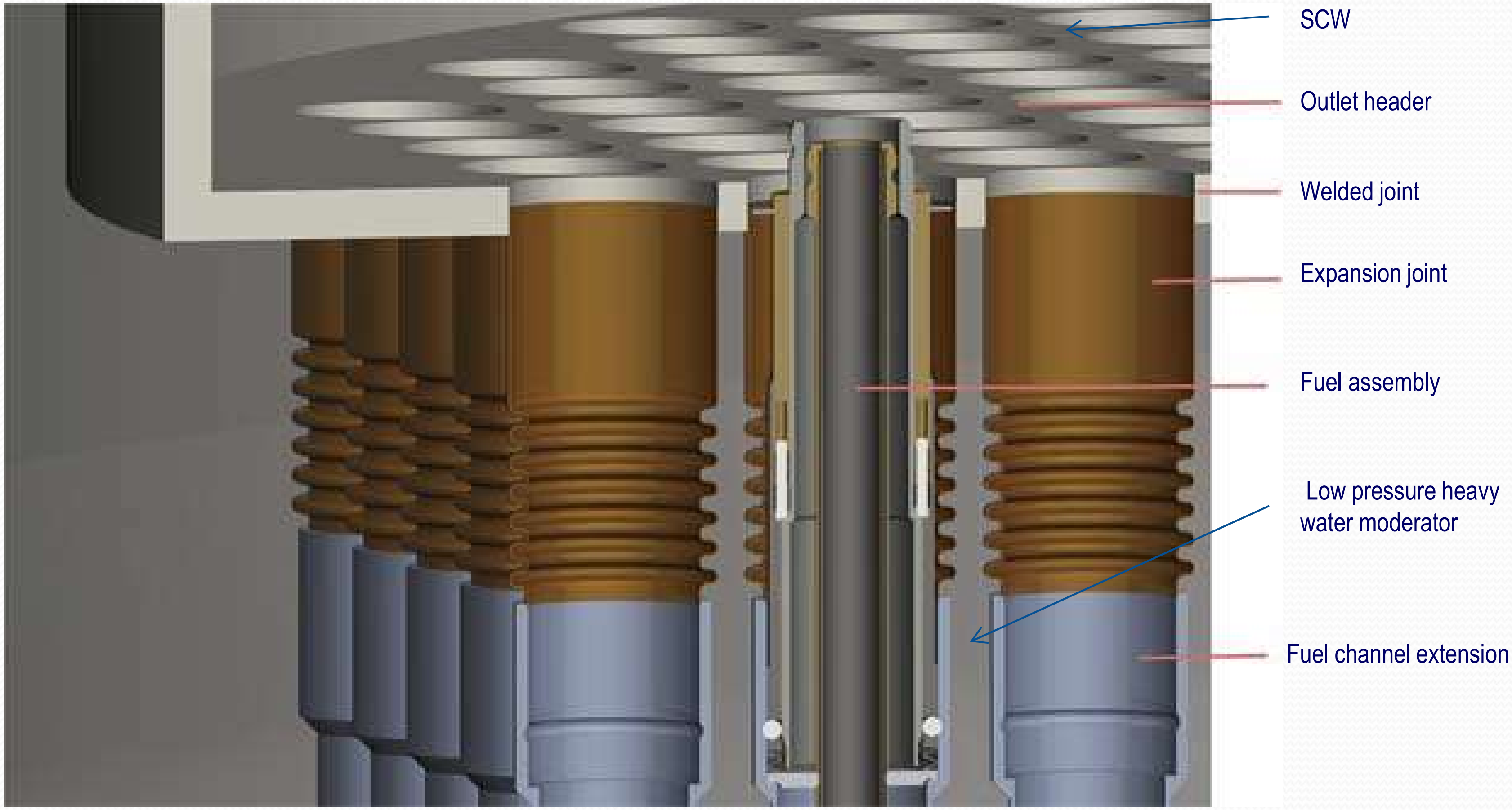
Canadian Pressure Tube-type Reactor



Canadian supercritical water-cooled reactor core concept
(a) Reactor core, (b) crossover piece, and (c) bottom of fuel channel
(Handbook Figure 8.5, page 196)

Supercritical Water-cooled Reactors

Canadian Pressure Tube-type Reactor



Canadian SCWR - Fuel channel connection to the outlet header
(Handbook Figure 8.7, page 197)

Summary of Gen. IV Technologies

Reactor Type	Coolant			Primary Cycle	Secondary Cycle	Efficiency [%]	Expected Deployment [Year]
	Type	Pressure [MPa]	Tin/Tout [°C]				
VHTR	Helium	7	640/1000	Direct Brayton	Indirect Rankine	≥55	2020
GFR	Helium	9	490/850	Direct Brayton	Indirect Rankine	≥50	2030
SFR	Liquid Sodium	~0.1	Primary 380/550 Secondary 320/520	Indirect Rankine	Brayton – supercritical CO2	~40	2020
LFR	Liquid Lead	~0.1	420/540	Indirect Rankine at subcritical water pressure 17	Brayton – supercritical CO2	41-43	2025
MSR	Sodium-fluoride salt with dissolve Uranium	~0.1	Out 700/800	Indirect Brayton – supercritical CO2	Indirect Rankine steam cycle	~50	2025
SCWR	water	25	350/625	Direct cycle	Indirect Rankine	45-50	2022

Deployment of Gen. IV Technologies

- A number of stages are required before Gen. IV nuclear reactors are practically deployed to serve communities
  - Get approvals and licences from regulatory authorities for each step of their development (test facilities, prototypes, final commercial design, then for their site, construction, commissioning, operation, decommissioning, and licence to abandon)
  - Overcome challenges with new fuel types, their production, reprocessing, repositories etc.
  - Build and test prototypes etc.
  - Time predictions indicated by some vendors of these Gen. IV designs seem to be too optimistic, but the progress is ongoing...

SMR Technologies – Today's Picture

- Nowadays a number of companies worldwide are developing SMRs aiming to deploy them by 2030
- The examples are
  - Moltex Energy SSR-W300 (UK & Canada – Stable Salt Reactors*)
  - Terrestial Energy IMSR-400 (Canada, UK & USA – Molten Salt Reactors)
  - GE & Hitachi BWRX-300 (USA & Japan – Water-cooled Reactors with passive safety)
  - NuScale SMR (USA – Water-cooled Reactors cooled by natural circulation)
- Pre-licensing vendor design reviews of SMRs by the CNSC in Canada (the next slide from the CNSC website) – why these companies request the reviews by the CNSC (not by other world nuclear regulatory authorities) ?
- In August 2019, the CNSC and the USNRC have signed an agreement of mutual cooperation on licensing of the SMRs!

CNSC Vendor Design Reviews of SMRs

Vendor design review service agreements in force between vendors and the CNSC					
Vendor	Name of design and cooling type	Approximate electrical capacity (MW electrical)	Applied for	Review start date	Status
Terrestrial Energy Inc.	IMSR Integral Molten Salt Reactor	200	Phase 1	April 2016	Complete
			Phase 2	December 2018	Assessment in progress
Ultra Safe Nuclear Corporation	MMR-5 and MMR-10 High-temperature gas	5-10	Phase 1	December 2016	Complete
			Phase 2	Pending	Project start pending
LeadCold Nuclear Inc.	SEALER Molten Lead	3	Phase 1	January 2017	On hold at vendor's request
Advanced Reactor Concepts Ltd.	ARC-100 Liquid Sodium	100	Phase 1	September 2017	Assessment in progress
Moltex Energy	Moltex Energy Stable Salt Reactor Molten Salt	300	Series Phase 1 and 2	December 2017	Phase 1 assessment in progress
SMR, LLC. (A Holtec International Company)	SMR-160 Pressurized Light Water	160	Phase 1	July 2018	Assessment in
NuScale Power, LLC	NuScale Integral pressurized water reactor	60	Phase 2*	Pending 2019	Project start pending

Deployment of SMR Technologies

Very recent news:

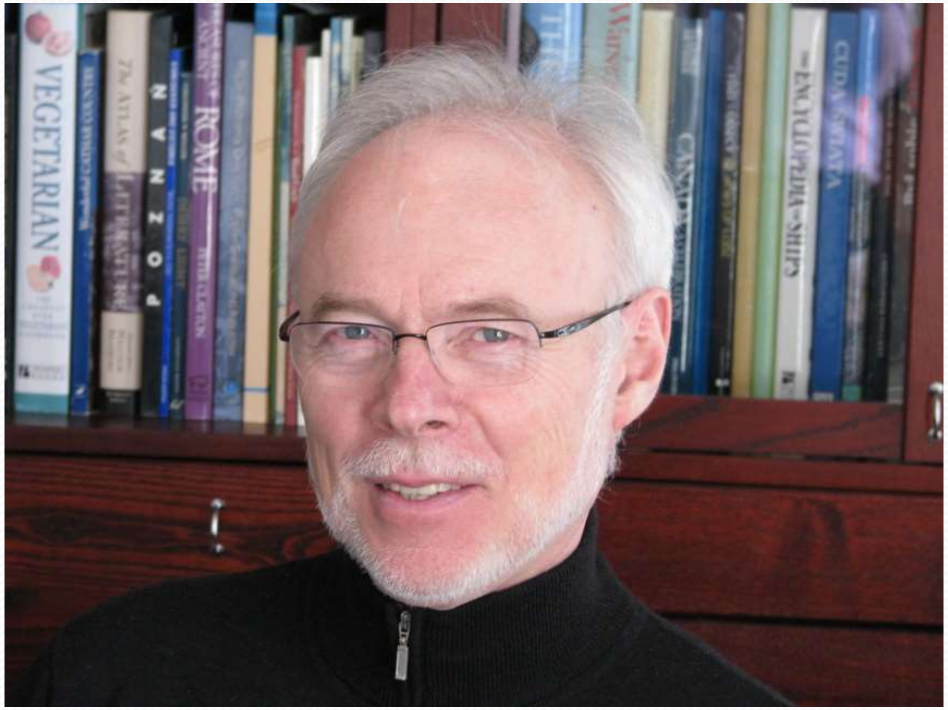
- **<u>The first</u>** application for the site licence in Canada has been submitted to the CNSC (July 2019) by Global First Power Ltd. (Mississauga, ON) partnering with Seattle-based Ultra Safe Nuclear Corporation & OPG for Micro-nuclear power plant (15 MWth transferred via molten salt to non-nuclear plant for electricity & heating) – at the Chalk River site of the Canadian Nuclear Laboratories
- On 18 December 2019 the US NRC has issued **<u>the first</u>** early site permit (ESP) to Tennessee Valley Authority for the potential construction of SMRs at its Clinch River site near Oak Ridge, Tennessee. The ESP valid for 10 to 20 years. The 8000-page ESP application submitted to the NRC in May 2016 was for two or more small modular reactor modules of up to a combined capacity of 800 MWe.
- On 19 December 2019 the floating NPP *Akademik Lomonosov* was connected to the grid generating electricity for **<u>the first</u>** time in the remote Chaun-Bilibino network in Pevek, in Russia's Far East. It is a pilot project/prototype for a future fleet of floating NPPs and on-shore installations based on Russian-made SMRs. This is the world's first nuclear power plant based on SMR technology to generate electricity.

The End

Overview of Generation IV Nuclear Reactors

KSD Professionals is ready to support Poland in the successful realization of its nuclear program

ksdprofessionals@gmail.com



All the best – SD