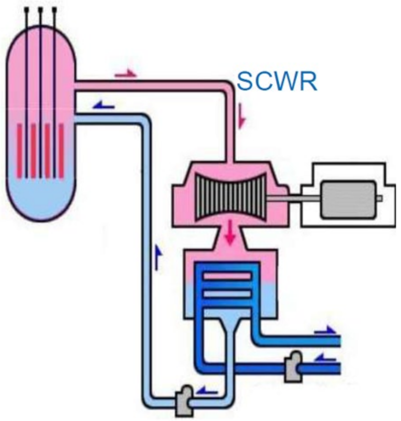
5th Joint ICTP-IAEA Workshop on Physics and Technology of Innovative Nuclear Energy Systems, 13 - 17 July 2026, Trieste

Overview of challenges and advantages of Innovative Large SCWR



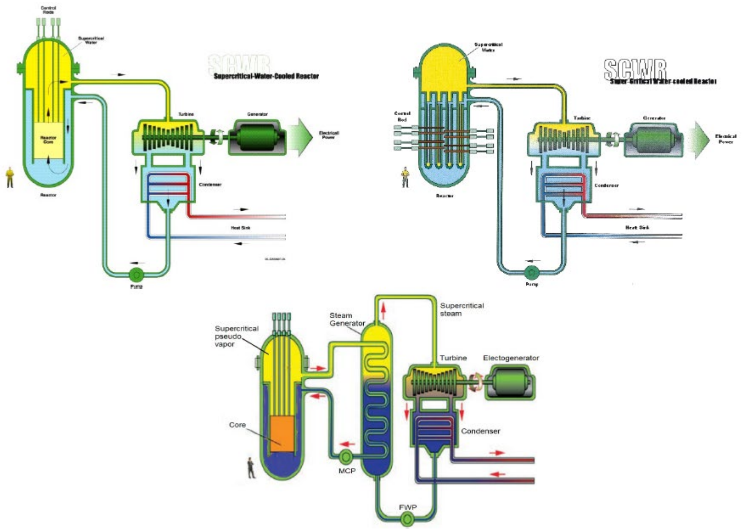
Katya Slavcheva,

Technical Lead - Water Reactor Technology Development, NPTDS

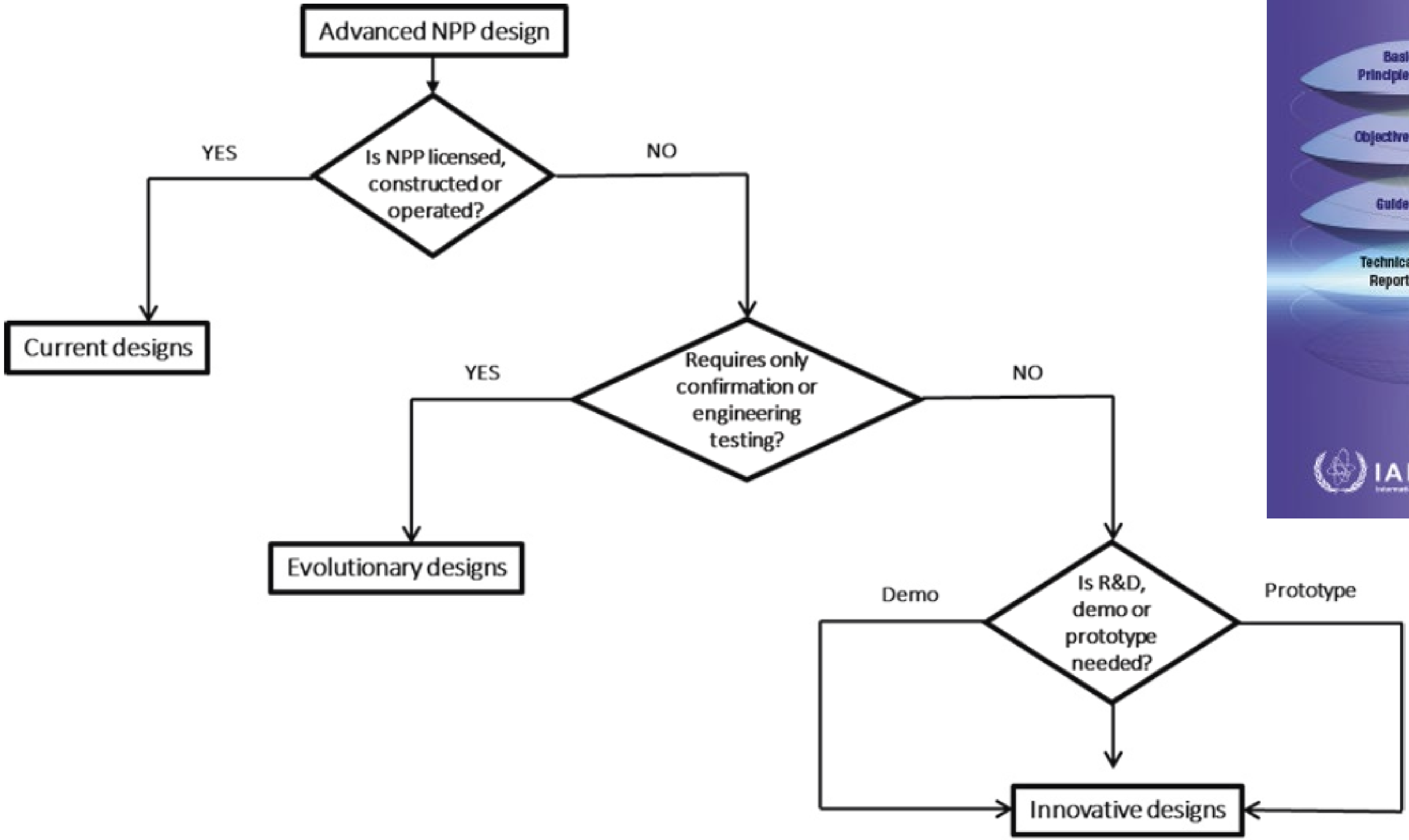
Learning Objectives

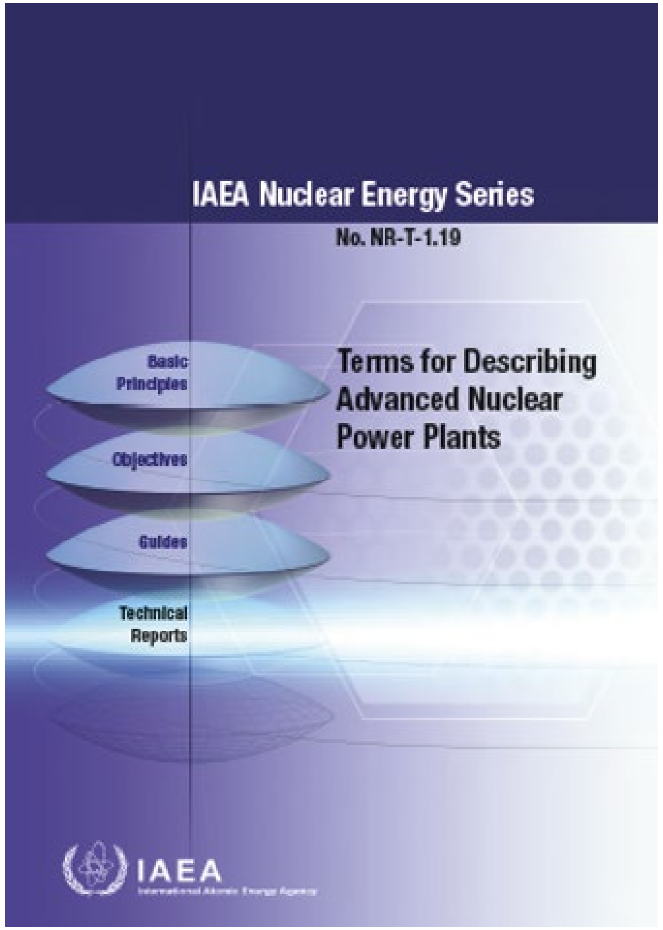
After successful completion of this lecture, participants shall be able to:

- ✓ Understand and distinguish between Evolutionary and Innovative Reactor Designs
- ✓ Outline innovative SCWR designs specifics
- ✓ Comprehend challenges and advantages of innovative SCWR
- ✓ Familiarise with IAEA resources

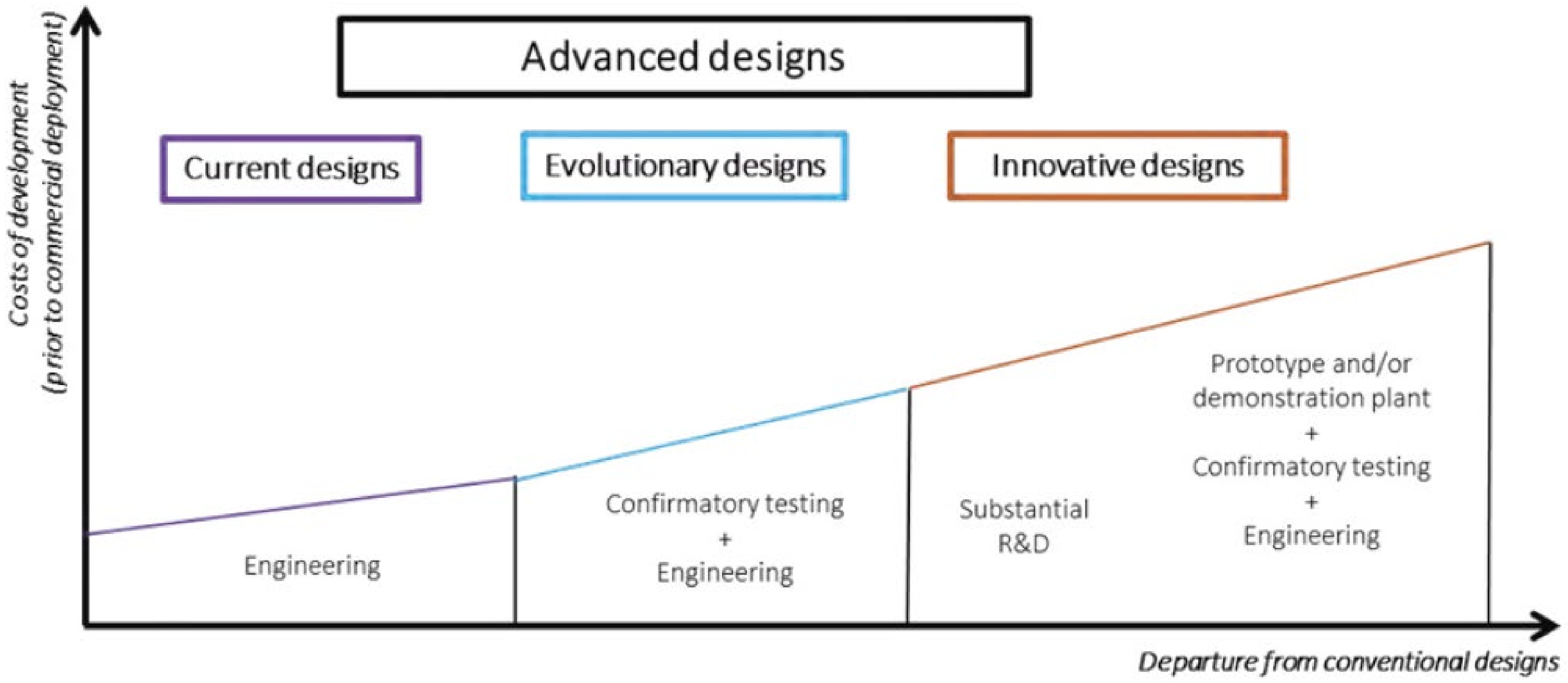


Designs categories





Designs Categories

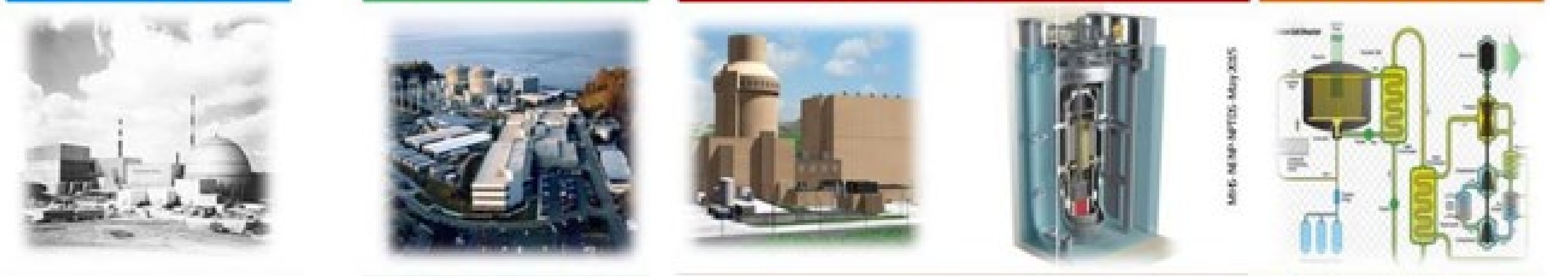


Evolutionary design: an advanced design that achieves improvements over existing designs through **small to moderate mods**, with a strong emphasis on maintaining proven design elements to minimize technological risks.

Innovative design: an advanced design, which incorporates **radical conceptual changes** in design approaches or system configuration in comparison with existing practice.

Reactor Deployment Timeline

Generation I	Generation II	Generation III/III+	Generation IV



Early Prototypes	Commercial Reactors	Advanced 'Current' and 'Evolutionary' Reactors	'Innovative' Reactors

Generation I

- **Calder Hall** GCR
- **Douglas Point** PHWR/CANDU
- **Dresden-1** BWR
- **Fermi-1** SFR
- **Kola 1-2** PWR/VVER
- **Peach Bottom 1** HTGR
- **Shippingport** PWR

Generation II

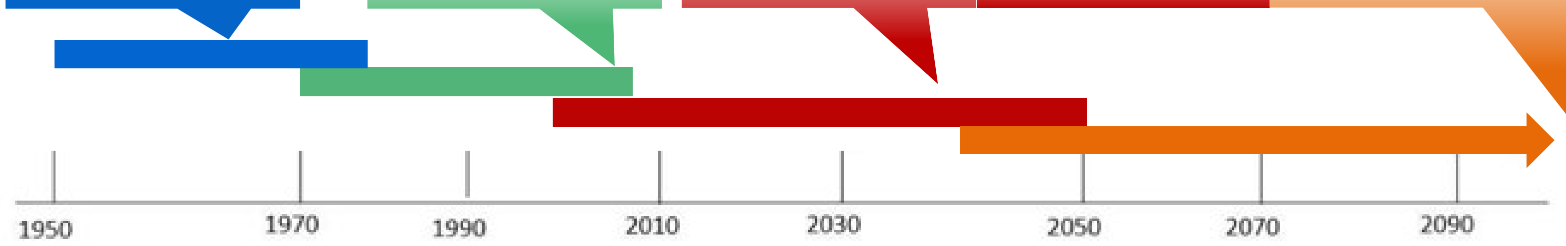
- **Bruce** PHWR/CANDU
- **Calvert Cliffs** PWR
- **Flamanville 1-2** PWR
- **Fukushima II 1-4** BWR
- **Grand Gulf** BWR
- **Kalinin** PWR/VVER
- **Kursk 1-4** RBMK
- **Palo Verde** PWR

Generation III/III+

- **VVER-12000** (AES2006) Gidopress PWR
- **HPR1000** (Hualong One) CNNC & CGN
- **ESBWR** GE Hitachi
- **Small Modular Reactors**
  - **GEH** BWRX-300
  - **Holtec** SMR-300
  - **iSMR** PWR
  - **NuScale** PWR
  - **NUWARD** PWR
- **ABWR** GE Hitachi, Toshiba
- **CANDU MONARK** AtkinsRéalis
- **AP1000** Westinghouse PWR
- **APR1400** KHNP PWR
- **SRZ-1200** Mitsubishi HI PWR
- **EPR** Framatome PWR

Generation IV

- **GFR** Gas cooled fast reactor
- **LFR** Lead cooled fast reactor
- **MSR** Molten salt reactor
- **SFR** Sodium cooled fast reactor
- **SCWR** supercritical water cooled reactor
- **VHTR** Very high temp. reactor



Gen Differentiating Factors

- Cost effectiveness
- Safety improvement
- Security
- Non-proliferation
- Fuel and fuel cycle

Innovative Reactors (Gen-IV)

Established in 2001, the Generation IV International Forum (GIF) was created as a co-operative international endeavour seeking to develop the research necessary to test the feasibility and performance of fourth generation nuclear systems, and to make them available for industrial deployment by 2030:

- Very-High Temperature Reactor (VHTR)
- Sodium-cooled Fast Reactor (SFR)
- **Supercritical Water-cooled Reactor (SCWR)**
- Gas-cooled Fast Reactor (GFR)
- Lead-cooled Fast Reactor (LFR)
- Molten Salt Reactor (MSR)

GIF Portal - Portal Site Public Home (gen-4.org)

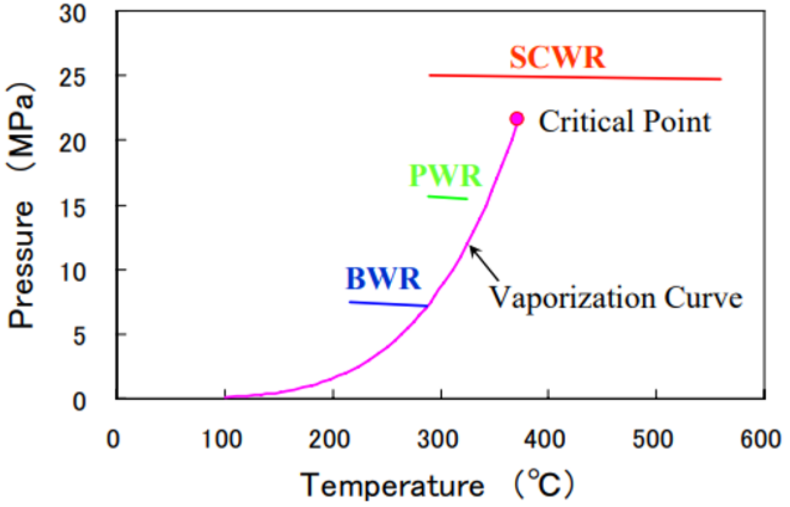
Supercritical Water-cooled Reactor (SCWR)

Operates at 374°C, 22.1 MPa **above water's critical point.**

22.1 MPa, **strict distinction between liquid and steam disappears**, the coolant is considered to be in the supercritical phase.

Can feature **thermal, fast, or mixed neutron spectra** and are designed in two configurations: **pressure vessel** (similar to BWRs and PWRs) and **pressure tubes** (like CANDU reactors).

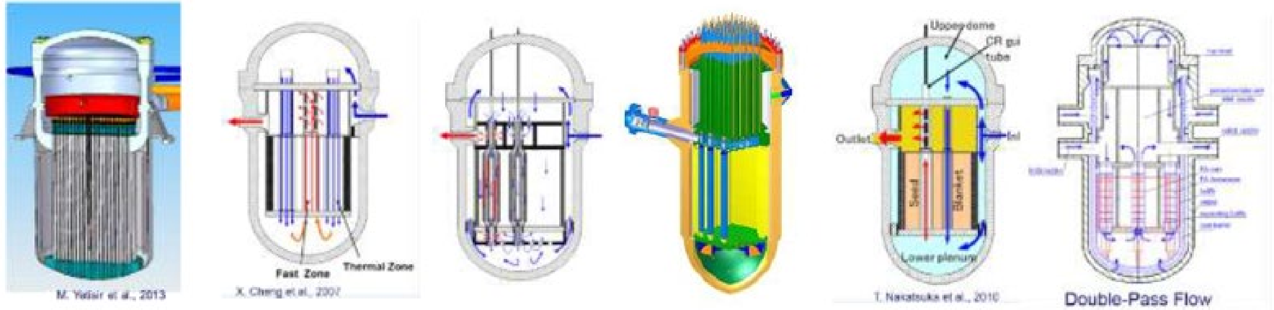
Thermal efficiencies of **44-48%**, significantly improving over current **34-36%**.



Operating pressure and temperature ranges in SCWR, Pressurized Water Reactor (PWR), and Boiling Water Reactor (BWR) designs

SCWR designs

- Strong interest in SCWR designs among the IAEA Member States (MSs)
- Several conceptual designs of SCWR developed to generate greater than 1000 MWe in **Canada, China, European Union, Japan and the Russian Federation.**
- Development expanded to include designing of **super-critical small modular reactors (SCSMR)** for generating capacity of up to 350 MWe (prototypes for the reference SCWR design).
- The design process requires advances in **key technology areas such as neutronics, fuel, materials, chemistry, thermal-hydraulics, control and safety**.



Advantages of SCWR designs to current WCRs

➢ **High thermal efficiency**

High temperature and pressure of the turbine inlet steam lead to a thermal efficiency of **1.2–1.4 times the one in current WCRs**

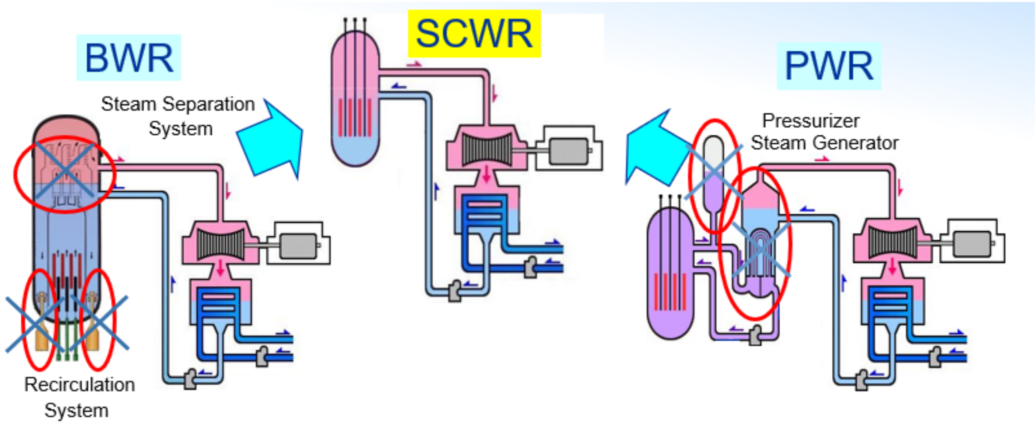
	WCR	SCWR
coolant	H_2O	H_2O
outlet T, C	288-329	500
efficiency, %	35	45
max P, MPa	17	25
spectrum	thermal	thermal/fast

Advantages of SCWR designs to current WCRs

➢ **Simplification of the plant system**

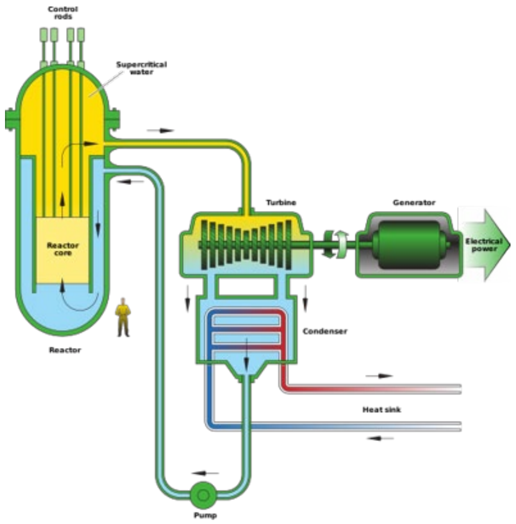
Recirculation system, steam-water separation system, steam generators, and pressurizer of other designs can be **eliminated** because there is no phase change

Offer economic benefits through increased efficiency and simpler designs.



Advantages of SCWR designs to current WCRs

- The **higher steam enthalpy** allows to **decrease** the size of the **turbine system** and thus to **lower the capital costs** of the conventional island



Main Challenges for the Deployment of SCWR

Technological challenges associated with materials and thermal-hydraulics:

- Supercritical water creates a harsh environment for structural materials and thus significantly influences their lifetime.
- The most demanding conditions are for the fuel cladding. While there are numerous structural materials that can withstand the SC conditions, these usually have a large neutron capture cross-section impacting the neutron economy.
- Testing under simulating conditions involving neutron radiation.
- Optimization of water chemistry to limit the corrosion of structural materials and radiolysis (availability of testing devices)
- Validation of computational tools used for the conceptualization of nuclear reactors.
- Demonstration of the proposed passive safety systems.

SCWRs have yet to be built and operated.

CRP “Advancing Thermal-Hydraulic Models and Predictive Tools for Design of SCWR Prototypes” (2022-2026)

- Thermal-hydraulics is one of the remaining critical areas important for maintaining coolability of the fuel and hence safe operation of the reference and prototype SCWRs.
- A significant amount of thermal-hydraulics studies was performed to support design, licensing and operation of the current fleet of nuclear reactors operating at subcritical pressures.
- Information and experimental data remain scarce on thermal-hydraulics behaviours for fluids at supercritical pressures.

CRP “Advancing Thermal-Hydraulic Models and Predictive Tools for Design of SCWR Prototypes” (2022-2026)

The objective of this CRP is to therefore, establish a coherent body of knowledge about fluids at supercritical pressures and/or temperatures needed to prototype the SCWR designs.

Deliverable:

TECDOC on “Advancing Thermal-Hydraulic Models and Predictive Tools for Design of SCWR Prototypes”

Final Report of a Coordinated Research Project

SCWR design concepts

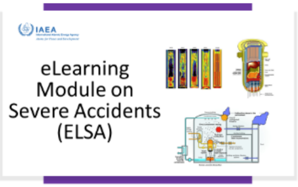
	CANADIAN SCWR	CSR1000	EU HPLWR	JSCWR	VVER-SCP
ARIS Database	Not in ARIS	**In ARIS**	**In ARIS**	**In ARIS**	Not in ARIS
Reactor type	pressure tube type (PT)	pressure vessel type (PV)	PV	PV	PV
Neutron spectrum	Thermal	Thermal	Thermal	Thermal	Fast
Coolant	H2O	H2O	H2O	H2O	H2O
Moderator	D2O	H2O	H2O	H2O	none
Normal operation pressure (MPa)	25	25	25	25	24.5
Core outlet coolant temperature (°C)	625	500	500	510	540
Plant thermal efficiency (%)	48	43.5	43.5	42.7	43-45
Core flow pass	single	double	triple	single	double (single)

E-learnings

[All courses | OPEN-LMS](#)

Courses / Nuclear Technology & Applications / Nuclear Energy / Advanced Water Cooled Reactor Technologies

Advanced Water Cooled Reactor Technologies



eLearning Module on Severe Accidents

Nuclear Technology & Applications

Self-directed | NO Certificate

ENGLISH

This eLearning module provides knowledge about the complex physics of severe accidents in water cooled reactors. It is focused on accident progression in light water reactors, considering the design s...

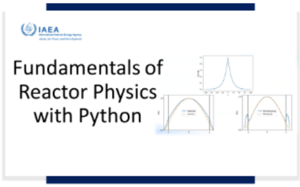
Nuclear Reactor Technology Assessment for Near Term Deployment

Nuclear Technology & Applications

Self-directed | NO Certificate

ENGLISH

This module describes Nuclear Reactor Technology Assessment for Near Term Deployment, and includes an introduction to RTA, methodology description, Detailed Description of Key Elements and Key Topics,...



Fundamentals of Reactor Physics with Python

Nuclear Technology & Applications

Self-directed | NO Certificate

ENGLISH

This module gives a brief overview ofÂ basics of nuclear physics,Â neutron slowing down,Â neutron transport,Â neutron diffusion,Â reactor kinetics andÂ fuel depletion using python as a tool for the an...

Pressurized Heavy Water Reactors

Nuclear Technology & Applications

Self-directed | NO Certificate

ENGLISH | ARABIC

This module introduces an overview of pressurized heavy water reactors: introduction, history, advanced designs, and global deployment. It also provides a thorough review of the presented materials.Â ...

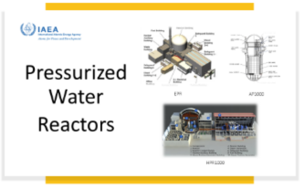
Supercritical Water Reactors

Nuclear Technology & Applications

Self-directed | NO Certificate

ENGLISH | FRENCH | ARABIC

This module introduces an overview of supercritical water reactors: background, different designs, advancements, and ongoing R&D. It also provides a thorough review of the presented materials.Â AVAIL...



Pressurized Water Reactors

Nuclear Technology & Applications

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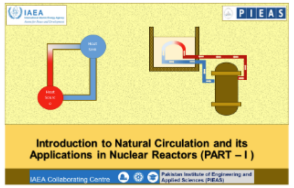
ARABIC

This module introduces an overview on Pressurized Water Reactorsâ (PWR) history, as well as different designs and advancements.Â It also provides a detailed



Introduction to Natural Circulation and its Applications in Nuclear Reactors (PART - I)

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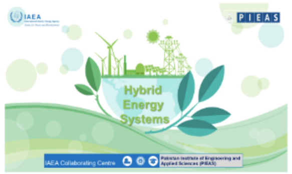
This module introduces the phenomenon of natural circulation and its common examples.Â It discusses the stepwise development of natural circulation loop, its



Hybrid Energy Systems

Nuclear Technology & Applications

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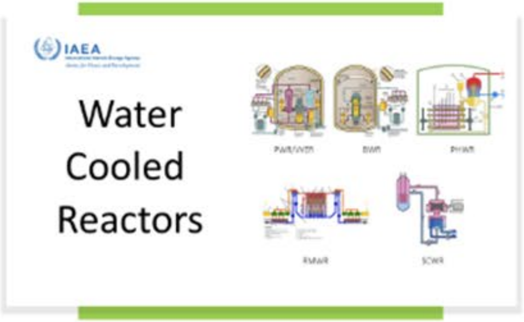
ENGLISH

This module introduces global energy scenario and the role of Hybrid Energy Systems.Â Detailed technical descriptions about the NuclearâRenewable Hybrid Energy Systems with case studies are provided

E-learnings



Overview of Water Cooled Reactors

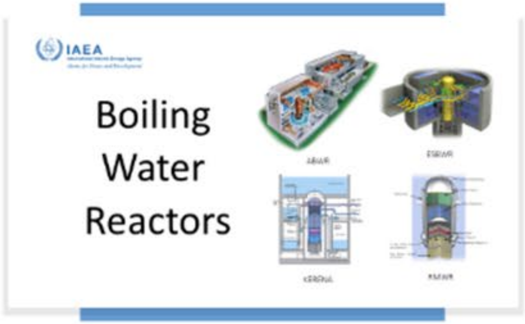
Nuclear Technology & Applications

Self-directed NO Certificate

SPANISH ENGLISH FRENCH

ARABIC RUSSIAN

This module describes WCRs including basic concepts, history, types & IAEA databases & basic education & training simulators. AVAILABLE IN ADDITIONAL UN LANGUAGES Target audience:Â Young professionals...



Boiling Water Reactors: An Introduction to History, Advanced Designs, and Global Deployment

Nuclear Technology & Applications

Self-directed NO Certificate

SPANISH ENGLISH ARABIC

This module introduces the history, advanced designs, and global deployment of boiling water reactors. AVAILABLE IN ADDITIONAL UN LANGUAGES Target audience: Young professionals, stakeholders, and new ...

Thorium-Based Reactor Designs

Nuclear Technology & Applications

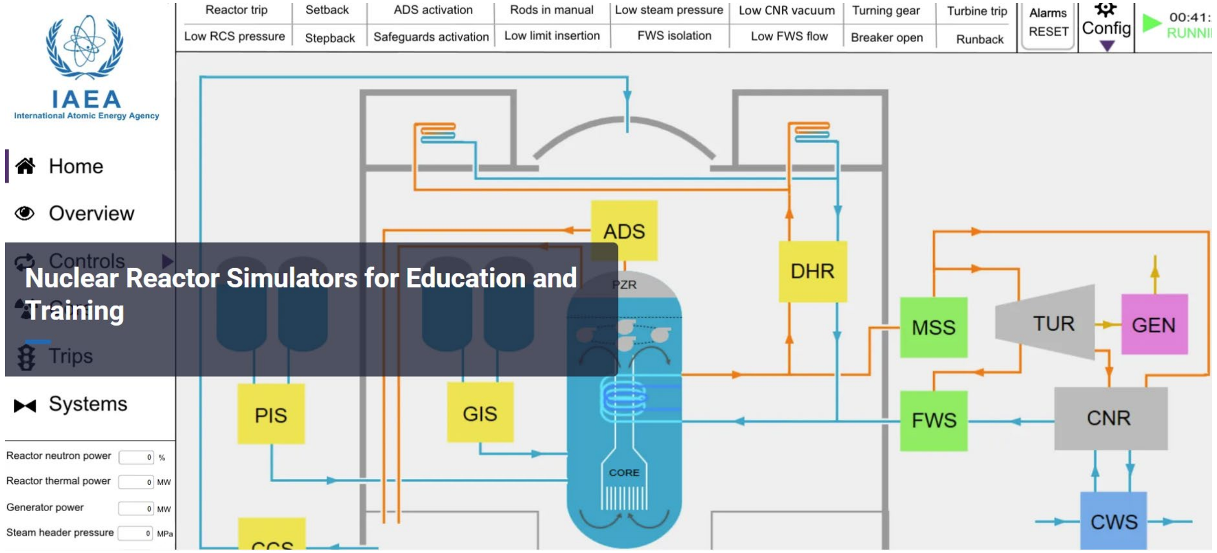
Self-directed NO Certificate

ENGLISH

This module delves into the unique properties of thorium and its potential as a safer and more sustainable alternative in nuclear reactor technology. introduces an overview on thorium-based reactorsâ...

Nuclear Reactor Simulators for Education & Training

https://www.iaea.org/topics/nuclear-power-reactors/nuclear-reactor-simulators-for-education-and-training



- Aid in understanding NPP fundamentals.
- **The IAEA sponsors training courses and workshops on the topic at the request of Member States.**

Pressurized Water Reactor (PWR) Simulators:
Advanced PWR: Two-Loop Large PWR (Korean-OPR 1000)
Russian-type PWR (VVER-1000)
Advanced Passive PWR (AP-600)
Integral Pressurized Water Reactor (SMR)

Boiling Water Reactor (BWRs) Simulators:
Conventional Boiling Water Reactor with Active Safety Systems (BWR)
Advanced BWR with Passive Safety Systems (ESBWR)

Pressurized Heavy Water Reactor (PHWR):
Conventional Pressurized Heavy Water Reactor (PHWR)
Advanced PHWR (ACR-700)

Part-Task Simulator:
Micro-Physics Nuclear Reactor Simulator

Under Development:
High Temperature Gas Cooled Reactor (HTGR)

Simulators.Contact-Point@iaea.org.

Since 1997, NPTDS has built a library of basic education and training simulators available to the Member States for access and use in their development of national nuclear power education and training curricula.

How to Obtain These Simulators?

STEP 1: CREATE A NUCLEUS LOGIN ID BY REGISTERING ON THE IAEA'S NUCLEUS WEBSITE.

STEP 2: DOWNLOAD AND FILL OUT THE SIMULATOR REQUEST FORM.

STEP 3: SCAN AND SEND THE COMPLETED FORM TO THE COMPETENT OFFICIAL AUTHORITY FOR THEIR APPROVAL AND SIGNATURE (STAMP).

Conclusions

Joint efforts for transitioning from research and development results to demonstration facility to SCWR in operation are needed

IAEA role to serve as platform facilitating the above is crucial

New CRP to be formulated shortly.

Need of e-trainings, PC basic simulator, …

Have an idea share it with us!

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