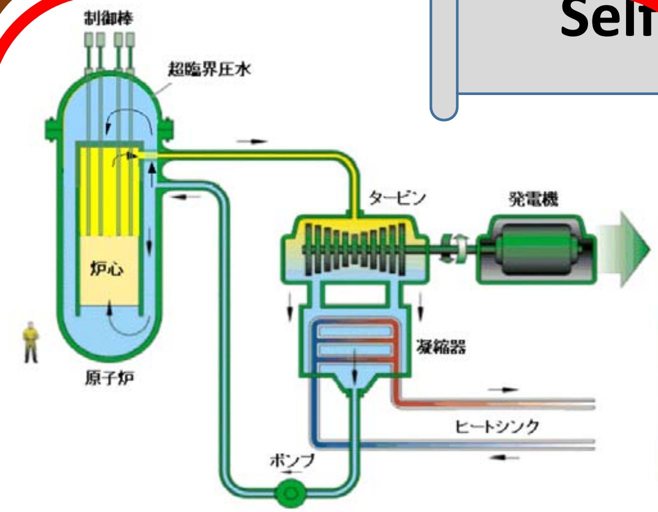
Brief Overview of R&D of Super Critical-Water-cooled Reactor (SCWR)

超臨界圧軽水冷却炉

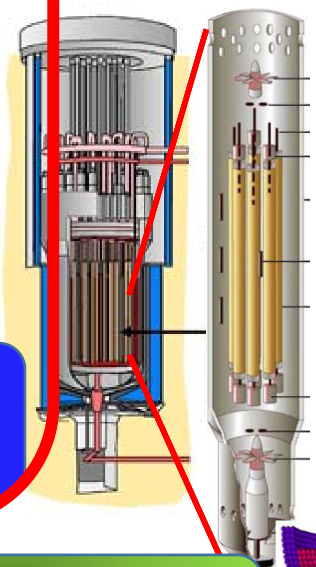
Cooperative Major in Nuclear Energy
Waseda University
Akifumi Yamaji

15th January 2020, Innovative Nuclear System Workshop,
Laboratory for Advanced Nuclear Energy
Institute of Innovative Research, Tokyo Institute of Technology

Self-Introduction (Akifumi Yamaji)

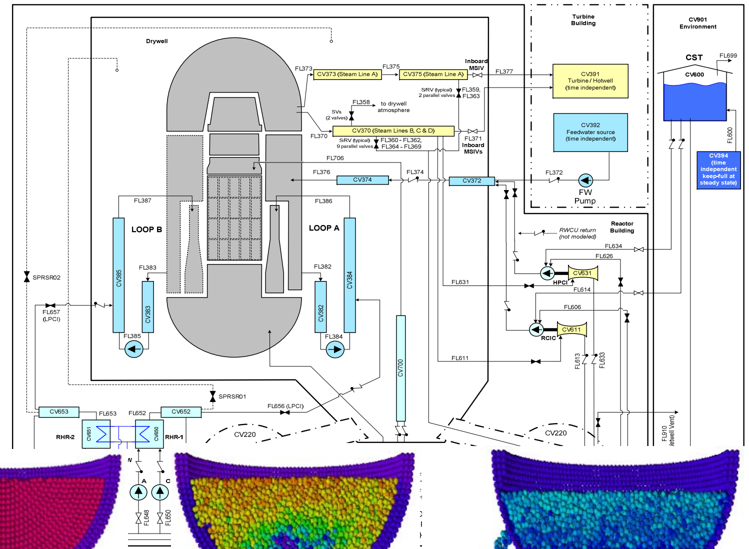


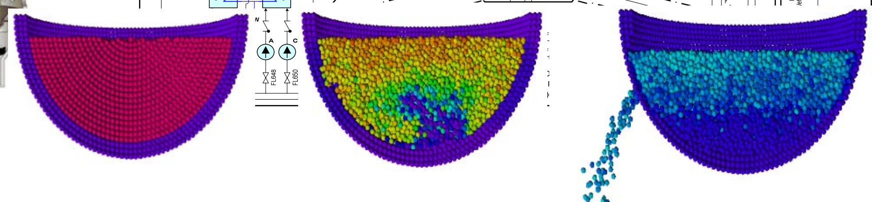
①SCWR



②Fuel Performance Modeling (ATF)

③Severe Accident Analysis (MELCOR)





④Particle method (MPS method)

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Contents

Introduction

- Some issues of nuclear power
  - Reducing initial investment (capital cost)
  - Reducing uncertainty in R&D
  - Reducing spent fuel and spent fuel management

Key issues: Adapting to the Global Free Market Economy（経済性）and Improving Sustainability（持続可能性）

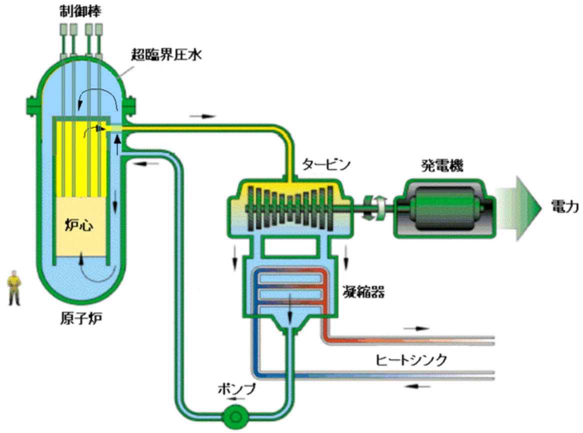
Features of SCWR

***S**uper**C**ritical-**W**ater-cooled **R**eactor*

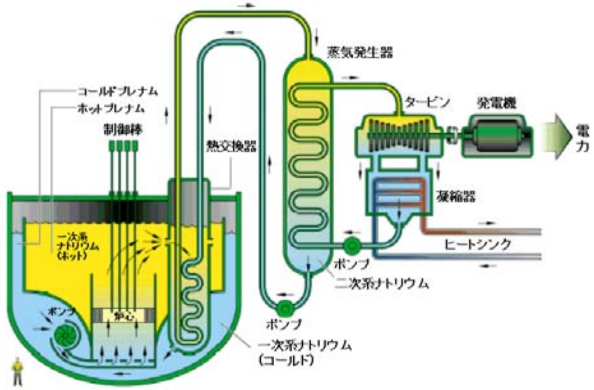
超臨界圧軽水冷却炉

Moving Forward GENERATION IV nuclear energy systems

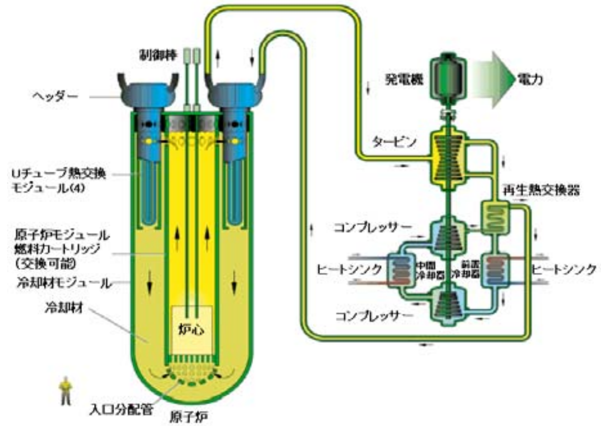
Gen IV Reactors



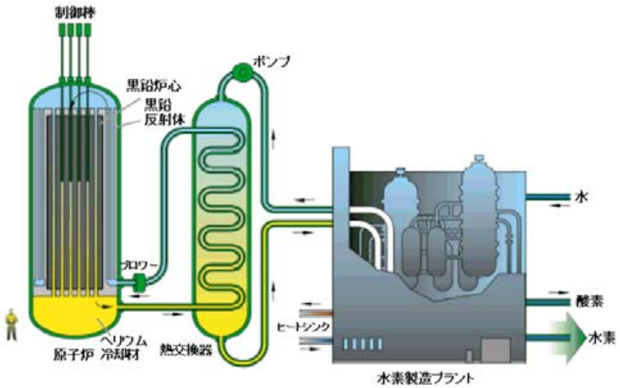
超臨界圧軽水冷却炉（SCWR）
（Thermal／Fast）



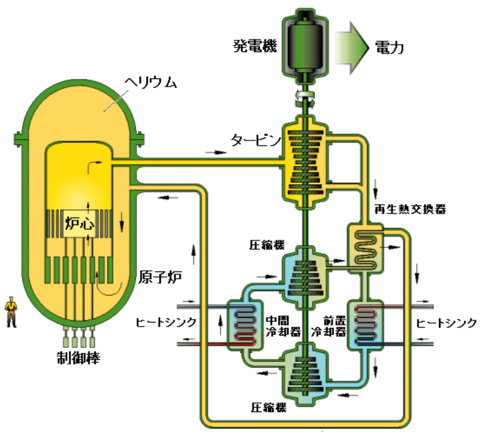
ナトリウム冷却高速炉（SFR）



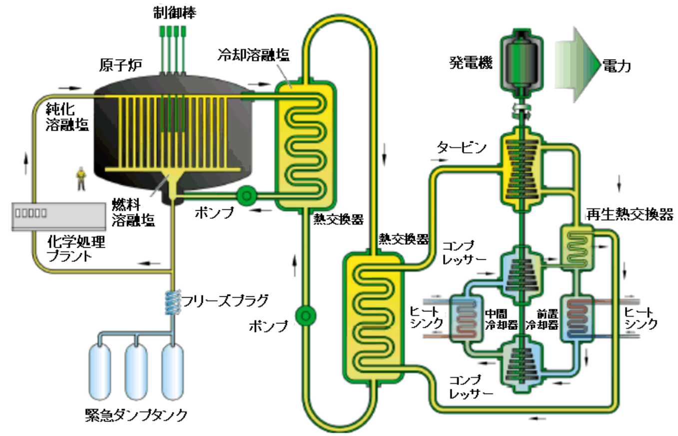
鉛合金冷却高速炉 (LFR)



超高温ガス炉 (VHTR)



ガス冷却高速炉(GFR)



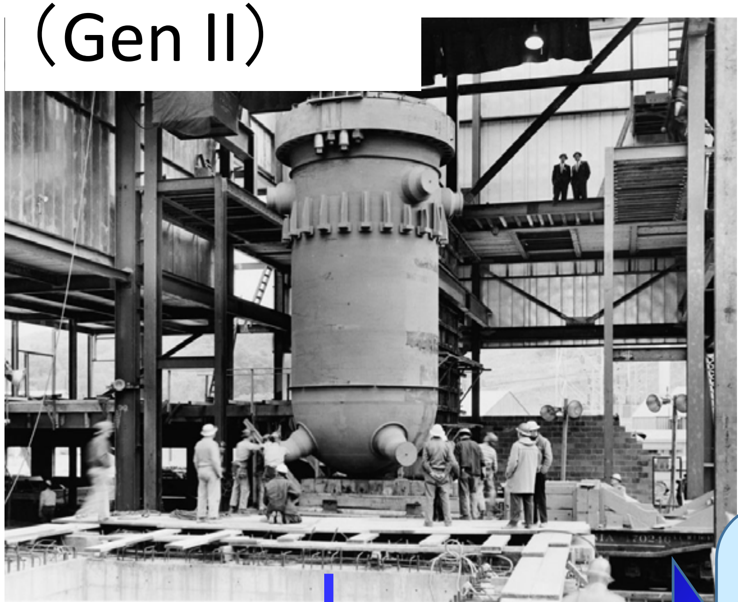
溶融塩炉(MSR)

WASEDA University
早稲田大学

原子力百科事典ATOMICAより （http://sta-atm.jst.go.jp/atomica/）

SCWR is the Gen IV LWR

LWR
(Gen II)



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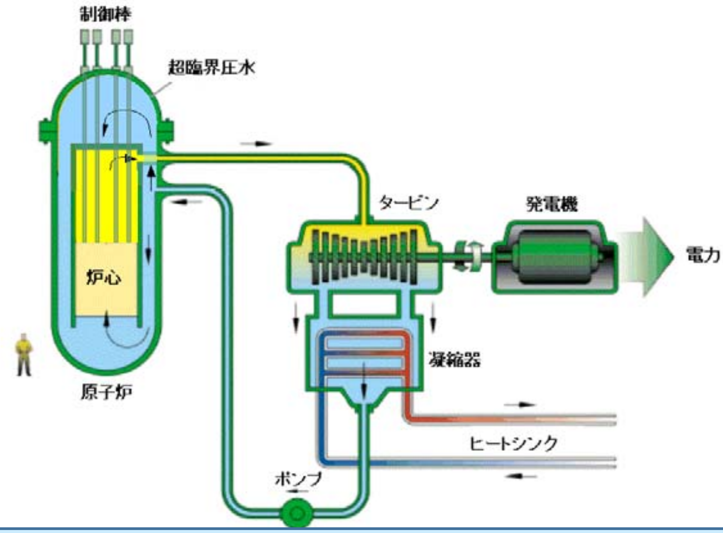
Latest Fossil Fuel
Power Plant (FFPP)

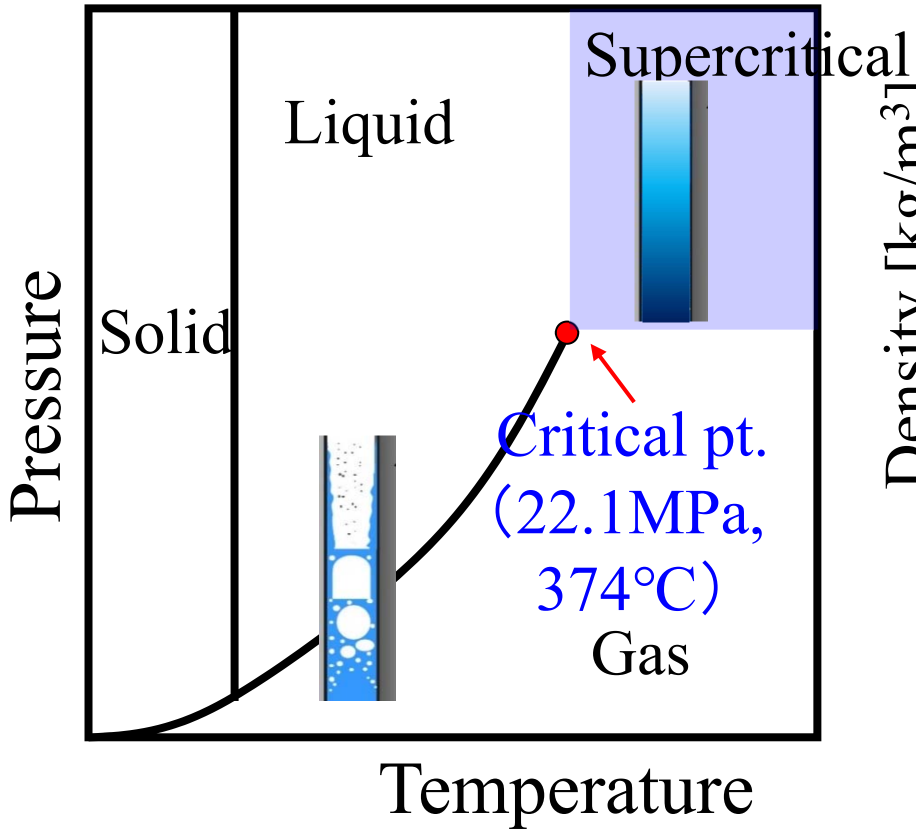


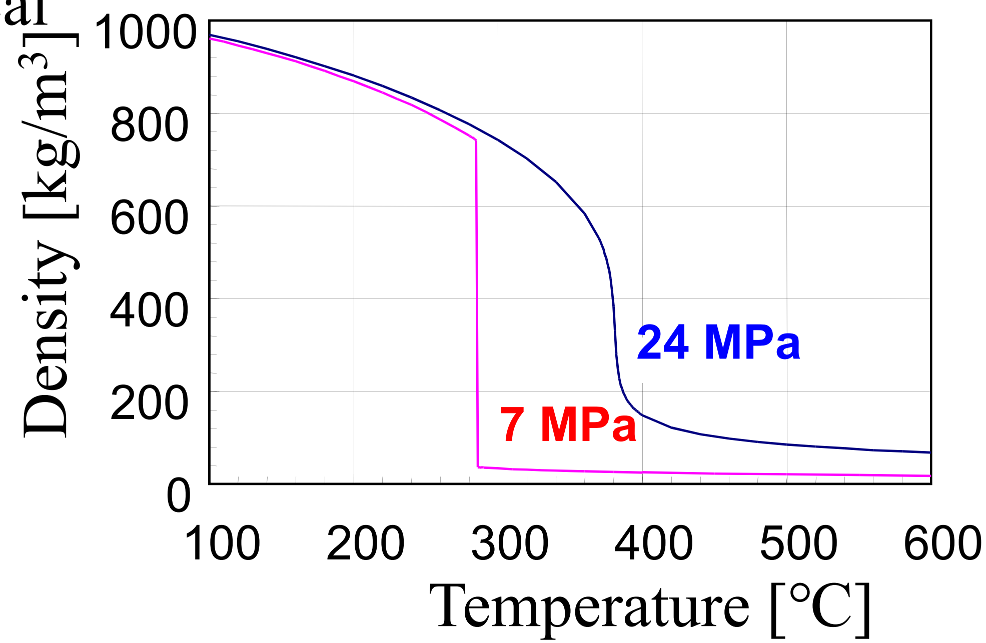
1. Simple and compact plant system
2. Higher thermal efficiency
3. Utilization of the current technologies
4. Flexible fuel management (thermal to fast)

High Economy Beyond the Scope of the Current LWR

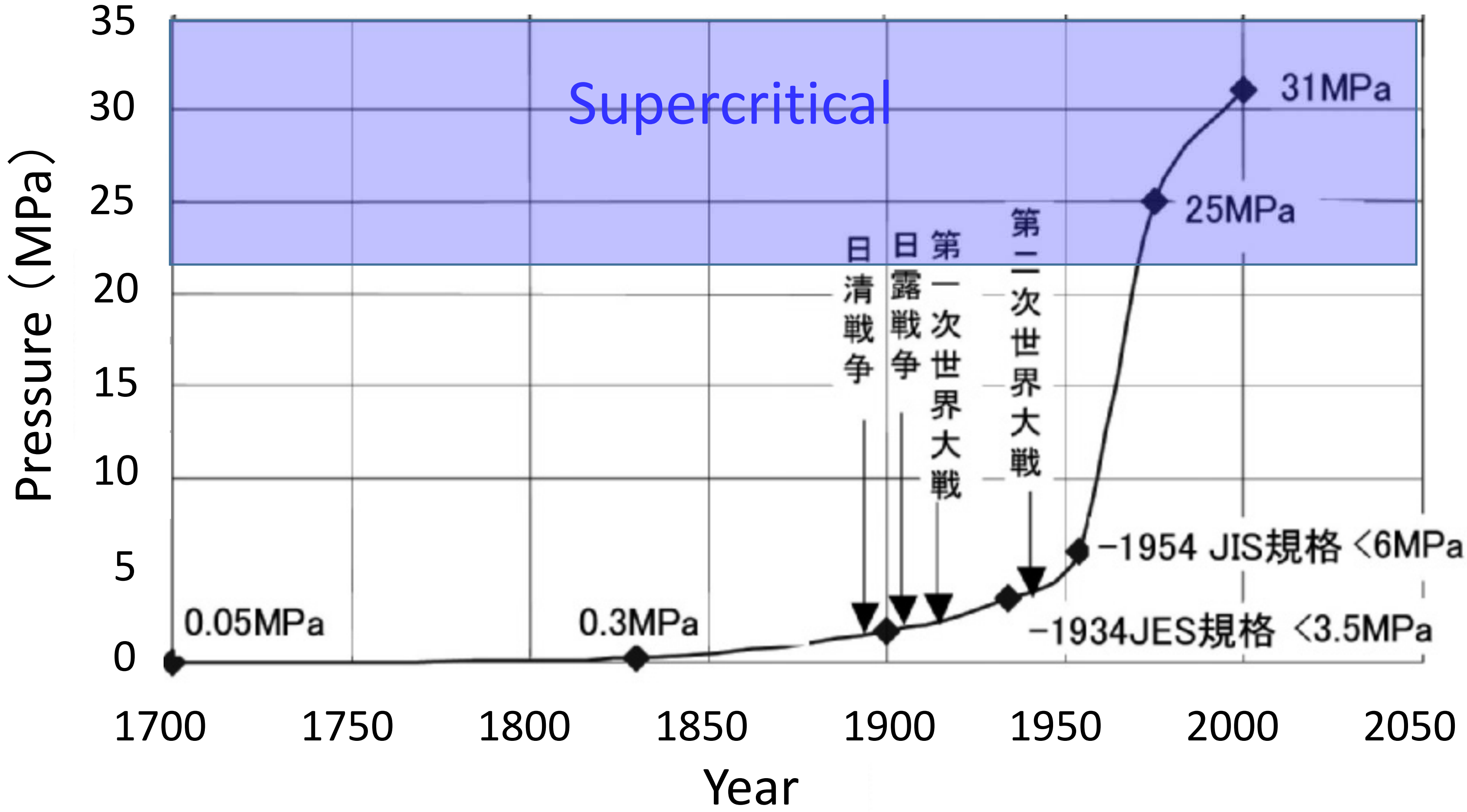
Features of Supercritical Water

- Continuous changes of properties from "Water-like" to "steam-like" without boiling.



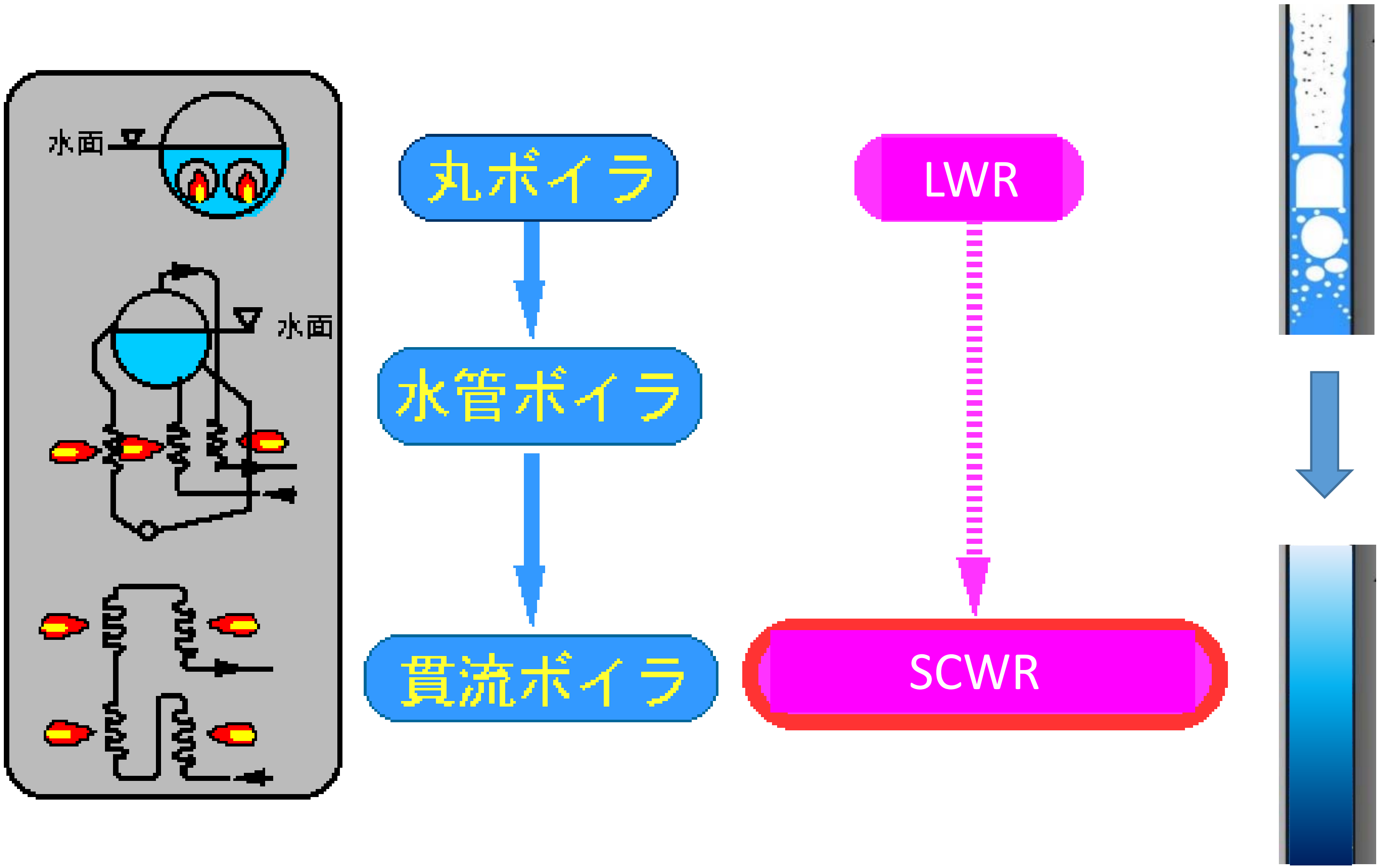


(FYI) Development of Boilers



寺本憲宗"ボイラー技術の系統化調査" 技術の系統化調査報告 第7集 2007年3月
産業技術史資料情報センター （http://sts.kahaku.go.jp/index.php）（accessed 2016/10/27）

Evolution of Boilers and LWR



Features of SCWR

1. Simple and compact once-through direct cycle plant system（貫流・直接サイクル）
2. High thermal efficiency
3. Utilization of matured LWR and FFPP（火力）
4. Flexible fuel management with thermal / fast neutrons

（1）: Simple and Compact Once-through Direct Cycle System

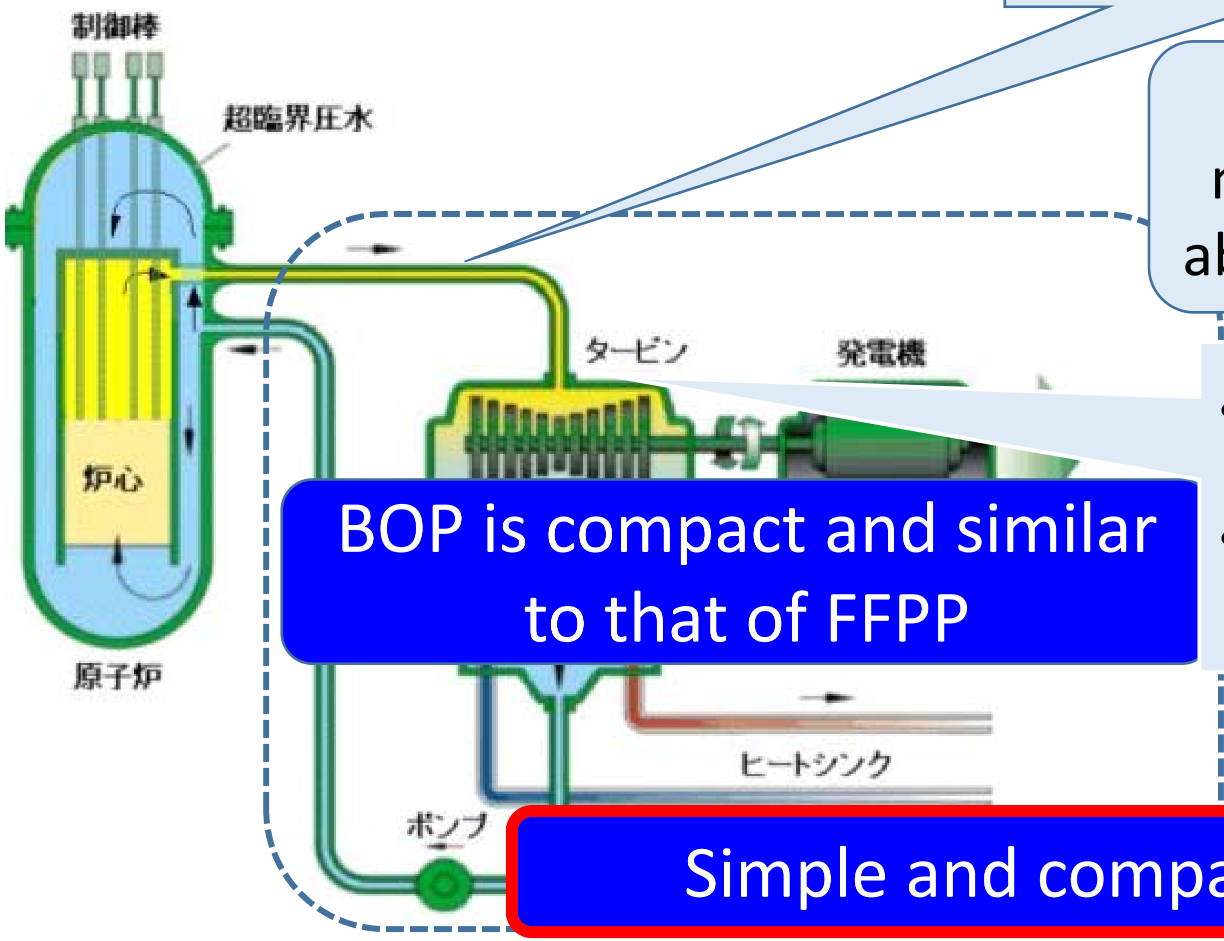
- No "boiling"→ No need to separate water / steam
- Entire supercritical "*steam*" is fed to the turbine directly

Once-through direct cycle

Reduction in number of main steam lines

High specific enthalpy, main steam flow rate is about 1/8 to 1/10 of LWR

- High speed high pressure turbines can be used.
- Number of low pressure turbines can be reduced.



BOP is compact and similar to that of FFPP

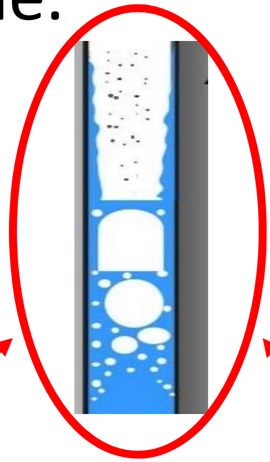
Simple and compact plant system

(FYI) : BWR and PWR

- "steam has to be separated from "water" before feeding the turbine.

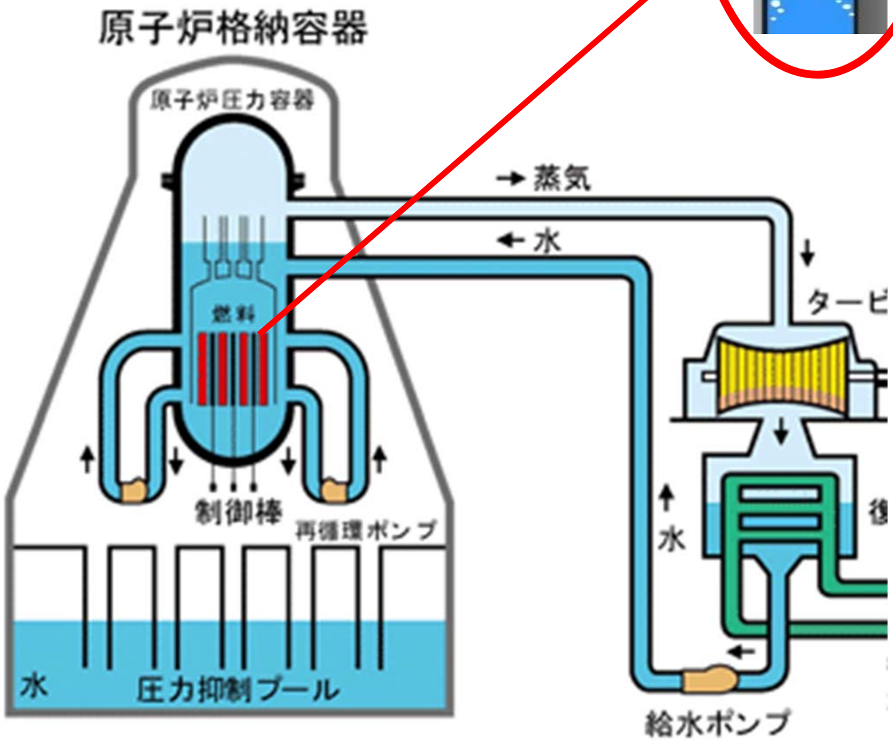
BWR

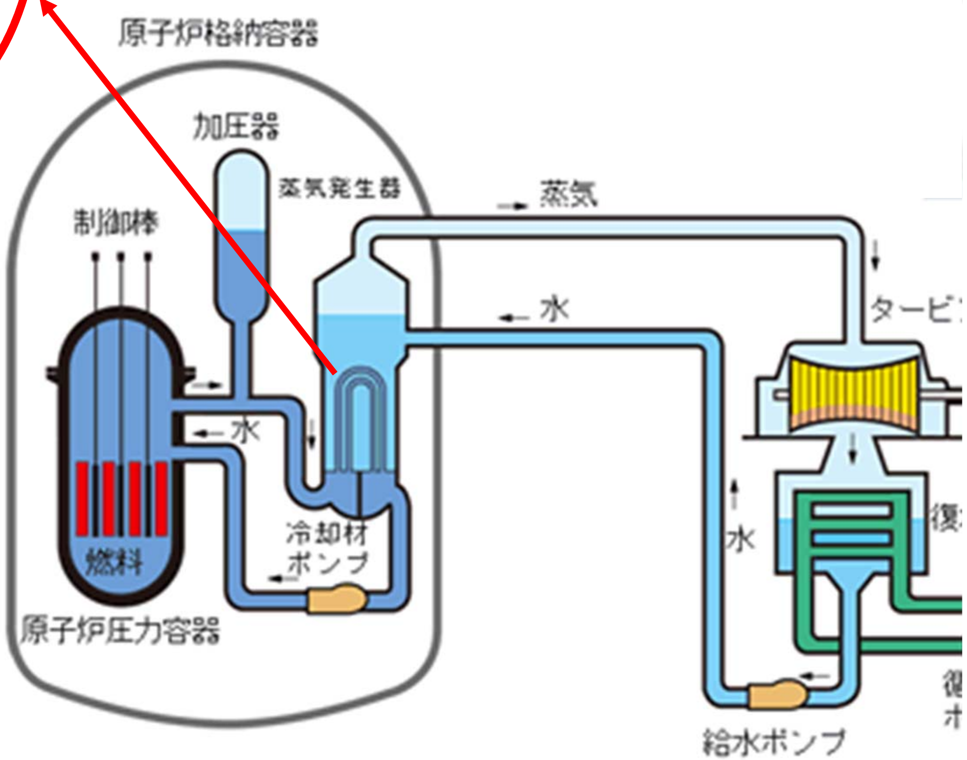
- Separator
- Dryer
- Re-circulations



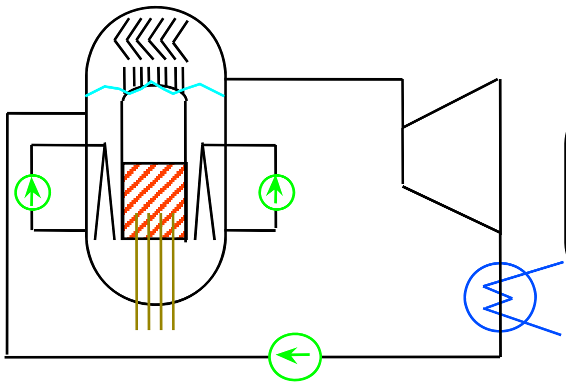
PWR

- Primary, secondary
- Steam generator
- Pressurizer

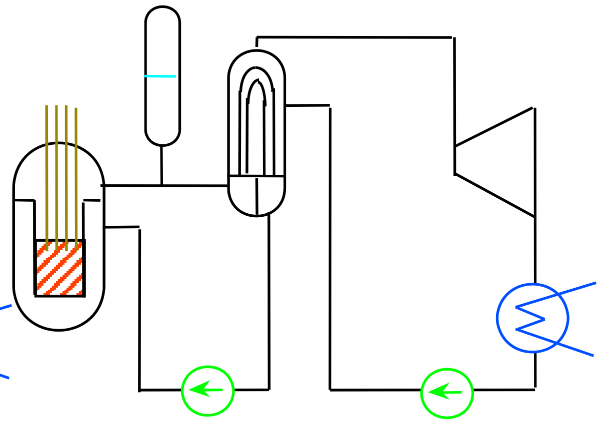




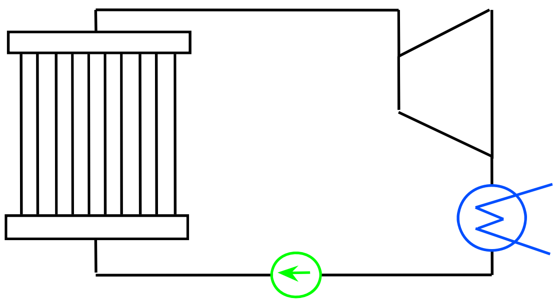
Comparison of Plant Systems



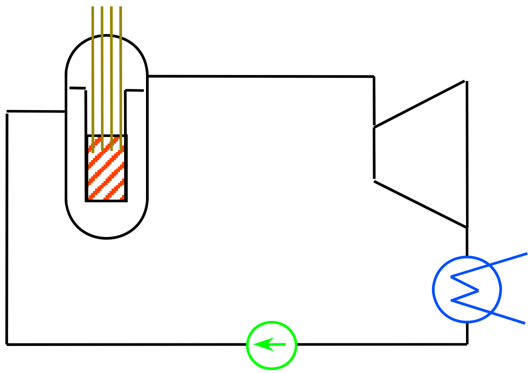
BWR



PWR



Supercritical FFPP



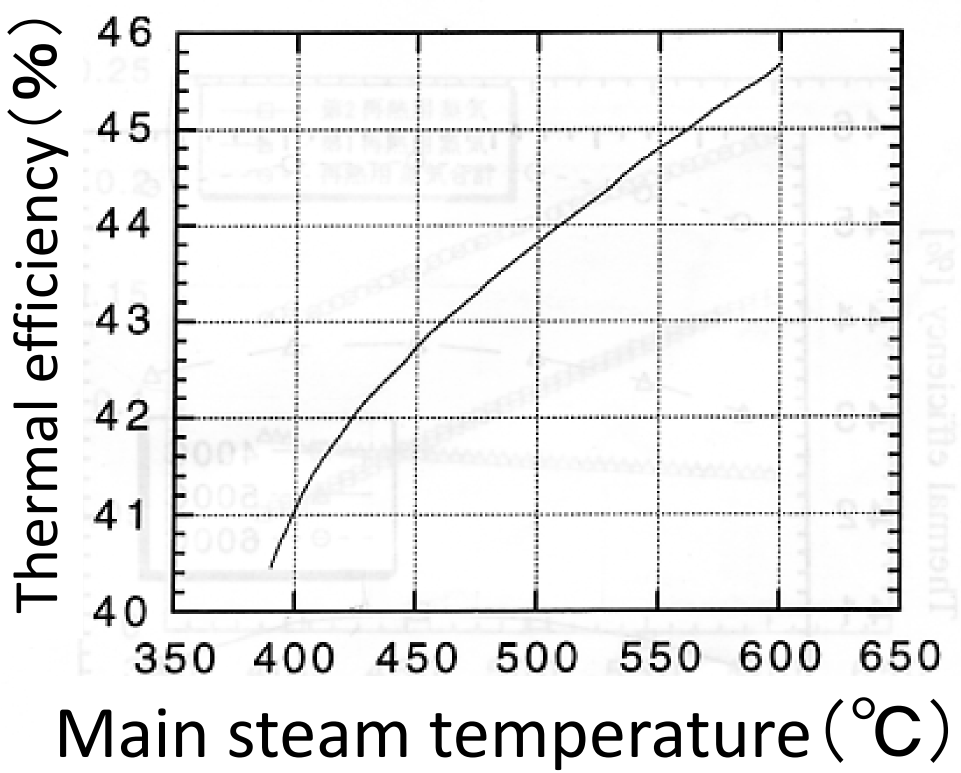
SCWR

(Once-through direct cycle)

(2): High Thermal Efficiency

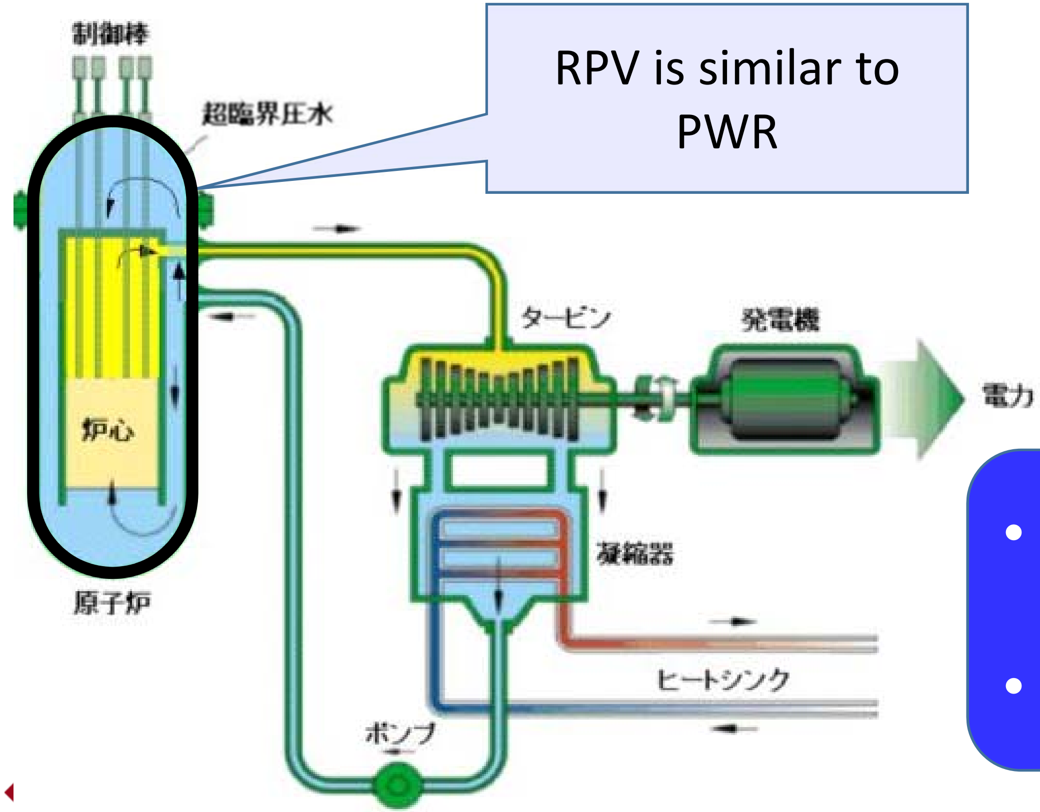
- "Steam temperature" is not limited by the "saturation temperature"
- Typical design
  - Main steam T: 500 - 625℃／Efficiency: 44 - 48%
  - Utilization of high temperature heat (e.g., hydrogen production)

Duffey, R.B. and Leung, L.K.H., "Advanced Cycle Efficiency: Generating 40% More Power from the Nuclear Fuel", Proc. World Energy Congress (WEC), Montreal, Canada, September 12-16, 2010



(3) : Utilization of the Current Technology

- Separate "pressure boundary" from "temperature boundary" (i.e., structures with large mechanical load is cooled with inlet coolant)
- Uses of main components (turbine, pump, main steam line, RPV (except nozzles), CR drive mechanisms, etc) are within the experiences of supercritical FFPP／LWR



RPV is similar to PWR

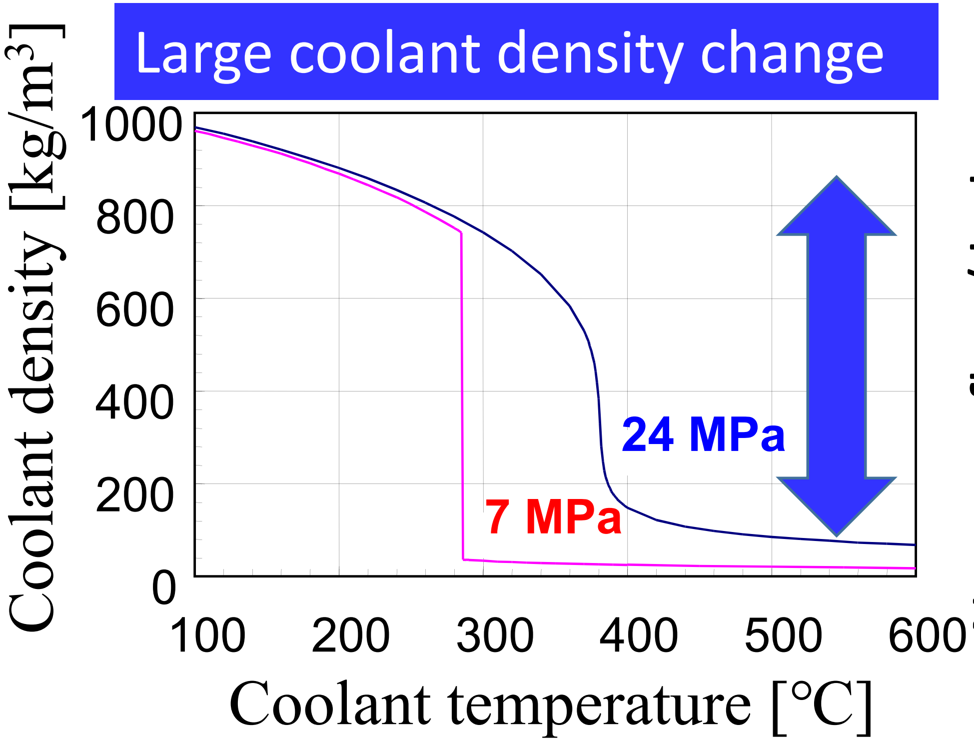
- Main R&D is limited to the core and in-core structures
- Limited uncertainty in R&D

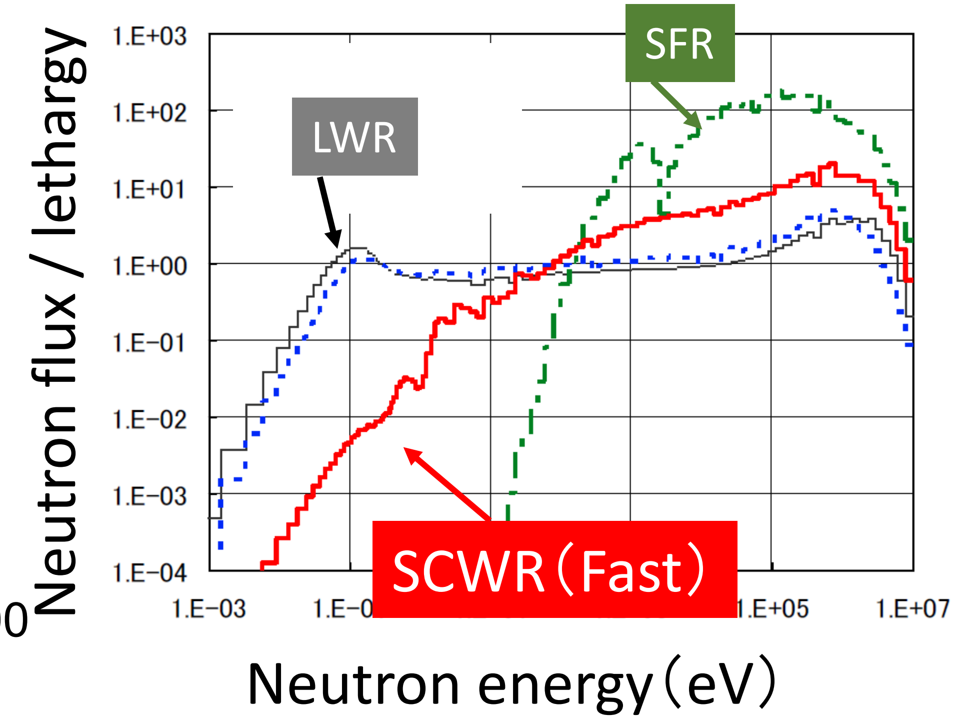
(3) : Utilization of the Current Technology

- Most venders of LWR also make supercritical FFPP
- Most utilities that operate LWR also operate supercritical FFPP

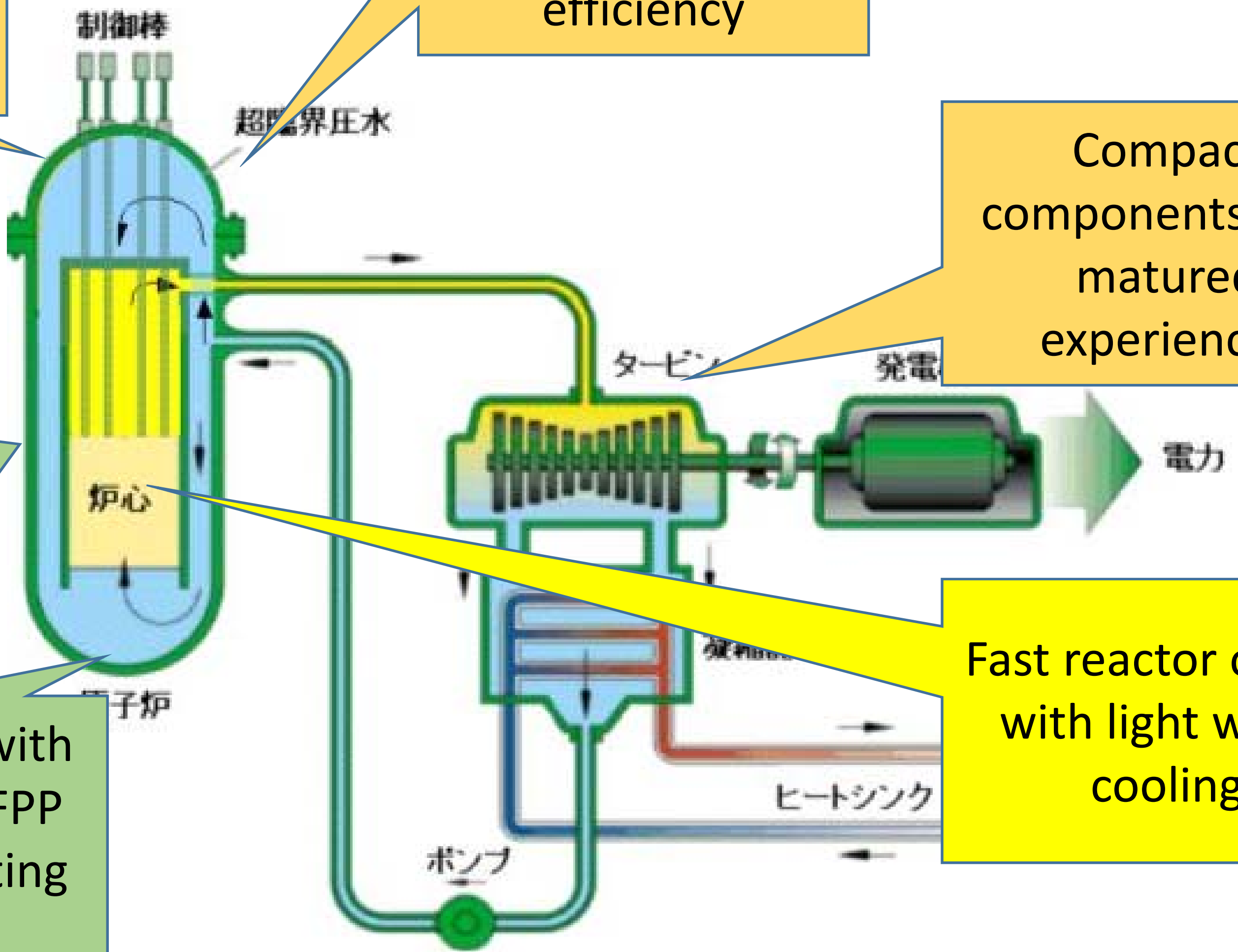
(4): Flexible Fuel Management

- Both thermal and fast spectrum reactors can be designed with the same plant system
  - Thermal: Neutron moderation with water rods
  - Fast: Tight hexagonal fuel bundle





SCWR Features（Summary）



Low capital cost due to simple and compact plant system

High economy and low waste due to high thermal efficiency

Compact components with matured experiences

Reliable safety with simple water cooling system.

Fast reactor option with light water cooling

High reliability with matured fossil FPP and LWR operating experiences.

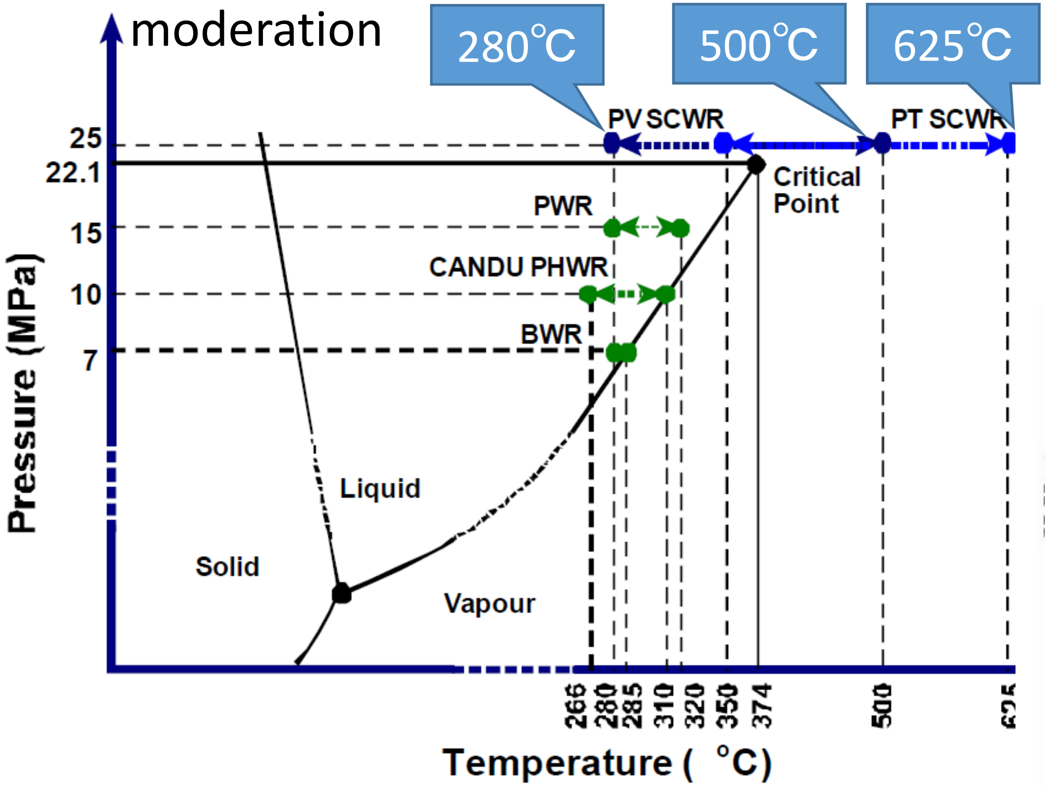
Research and Development Fields

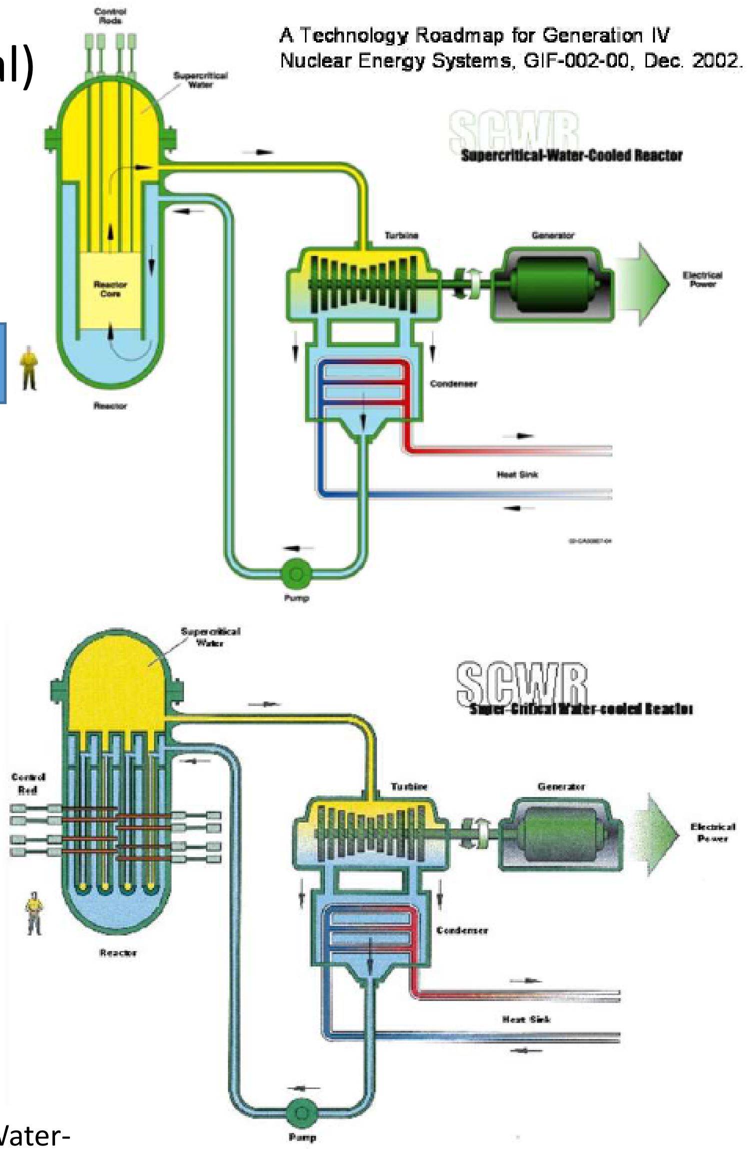
- Plant concept
- Core thermal-hydraulics
- Materials and corrosions

Plant Concept

Pressure Vessel & Pressure Tube

- Pressure vessel type
  - Light water cooling & moderation (thermal)
  - Light water cooled fast reactor
- Pressure tube type
  - Light water cooling & heavy water moderation

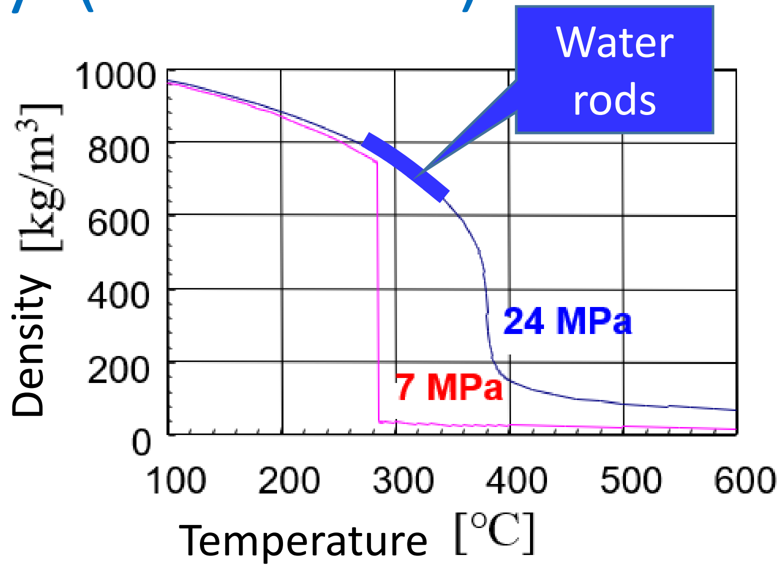


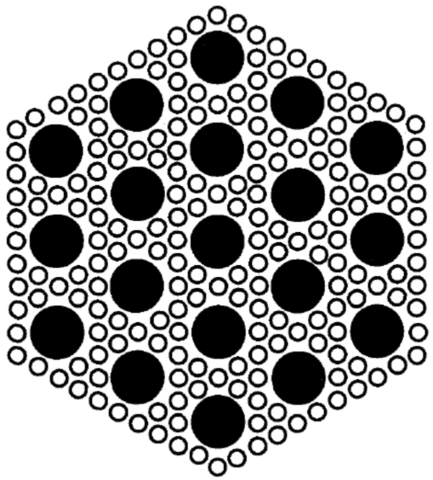


L.K.H. Leung et al., "An Update on the Development Status of the Super-Critical Water-cooled Reactors", GIF Symposium – Paris (France) – 16-17 October 2018

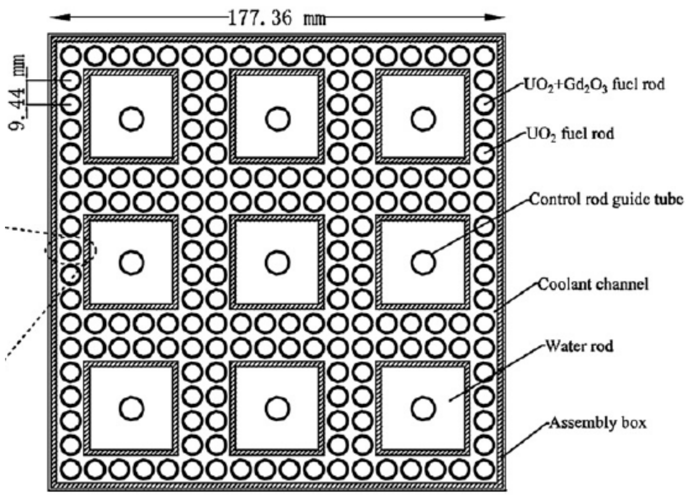
Example of Fuel Assembly (Thermal)

- Low enriched U, advanced SS cladding
- Coolant flow rate distribution with inlet orifice and channel box
- Neutron moderation with water rods
- Reactivity control
  - Burnable poison (Gd_2O_3)
  - Control rods (PWR type／BWR type)

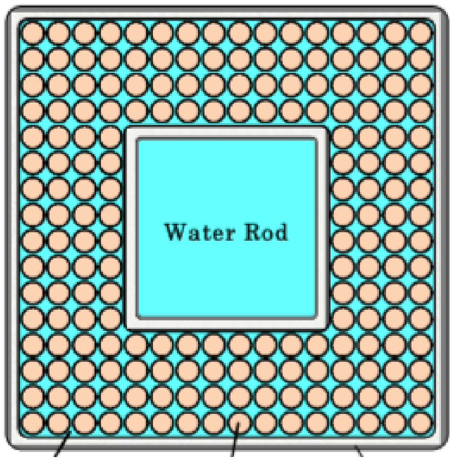




Y. Okano et al., "Design of Water Rod Cores of a Direct Cycle Supercritical-Pressure Light Water Reactor", Ann. Nucl. Energy, Vol. 21, No. 10, pp. 601-611, 1994



C. Zhao et al., "Conceptual design of a supercritical water reactor with double-row-rod assembly", Progress in Nuclear Energy 63 (2013) 86-95, 2013.

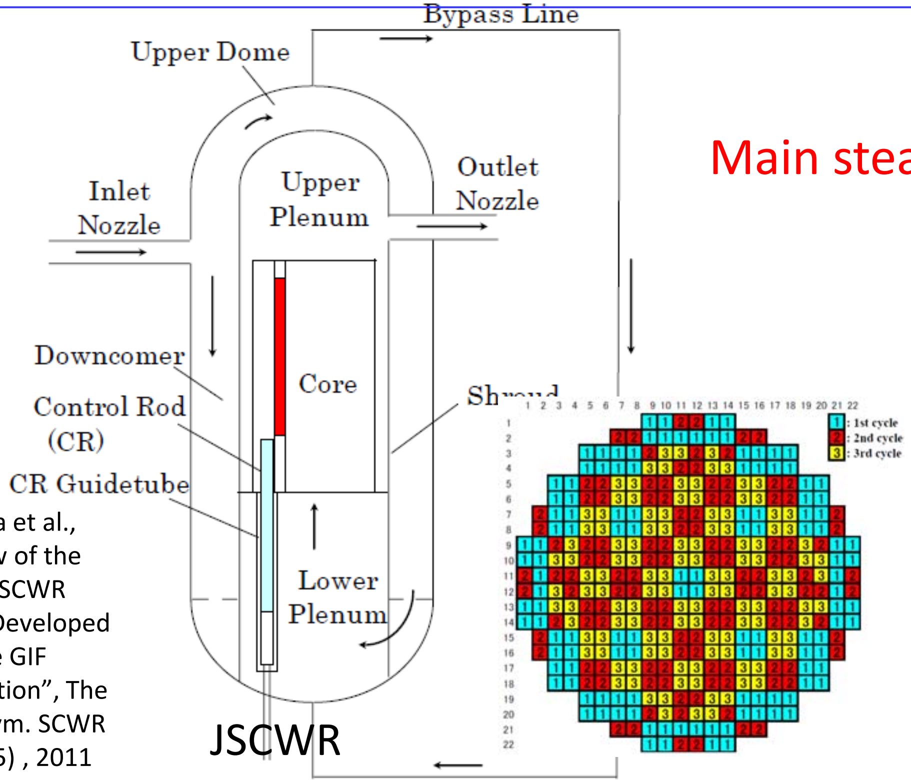


K. Yamada et al., "Overview of the Japanese SCWR Concept Developed Under the GIF Collaboration", The 5th Int. Sym. SCWR (ISSCWR-5) , 2011

Example of RPV and Coolant Flow

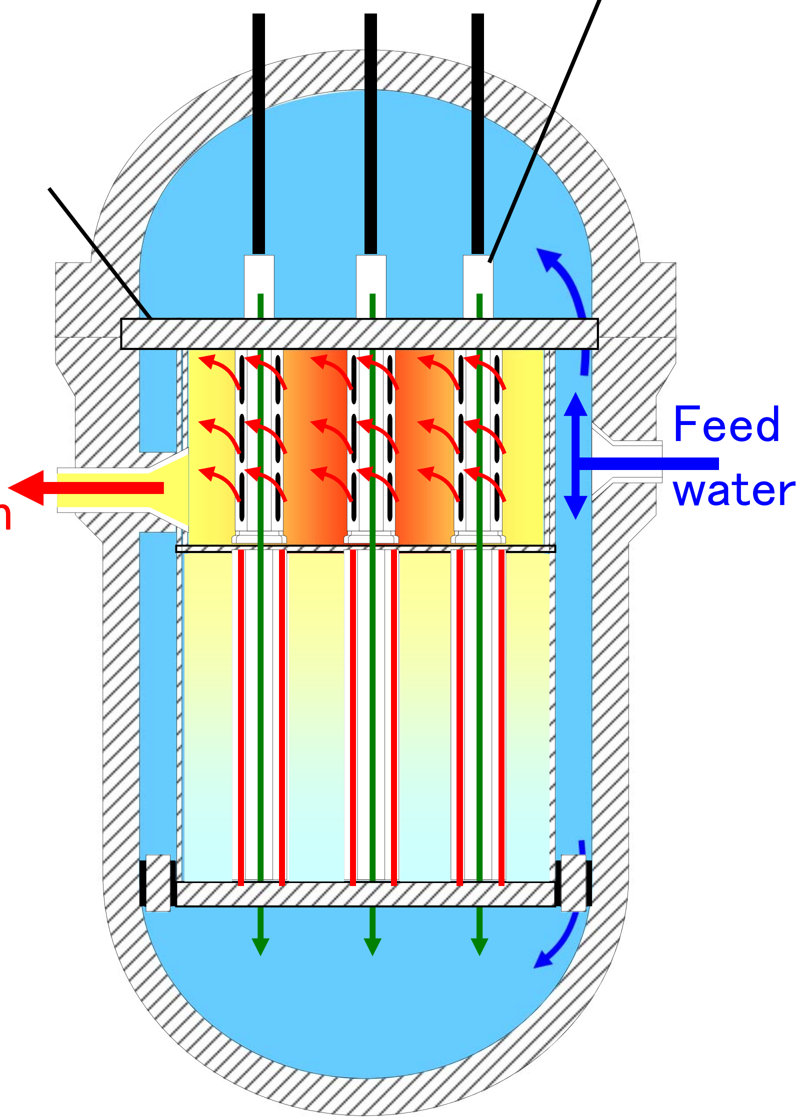
- Large coolant temperature and density changes
- High outlet temperature
- Separating pressure boundary from temperature boundary

Super LWR : ***Single pass*** core design
J.Wu and Y. Oka, "Improved Single pass core design for high temperature Super LWR"、Nuclear Engineering and Design, 267(2014) 100-108



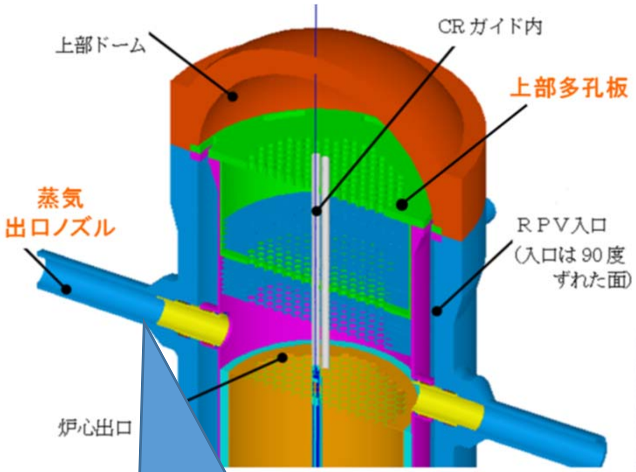
K. Yamada et al., "Overview of the Japanese SCWR Concept Developed Under the GIF Collaboration", The 5th Int. Sym. SCWR (ISSCWR-5) , 2011

JSCWR

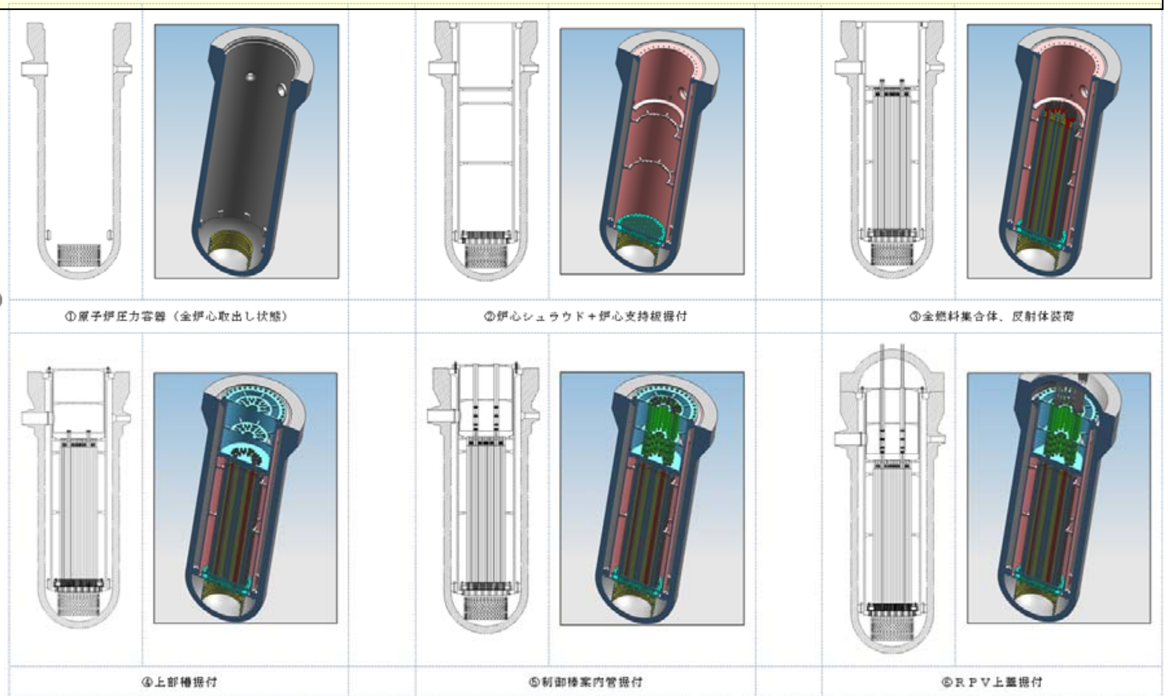


In-core Structure Designs

◆Structures to accommodate high temperature, maintenance, fuel replacement



Sleeve structure to reduce thermal stress



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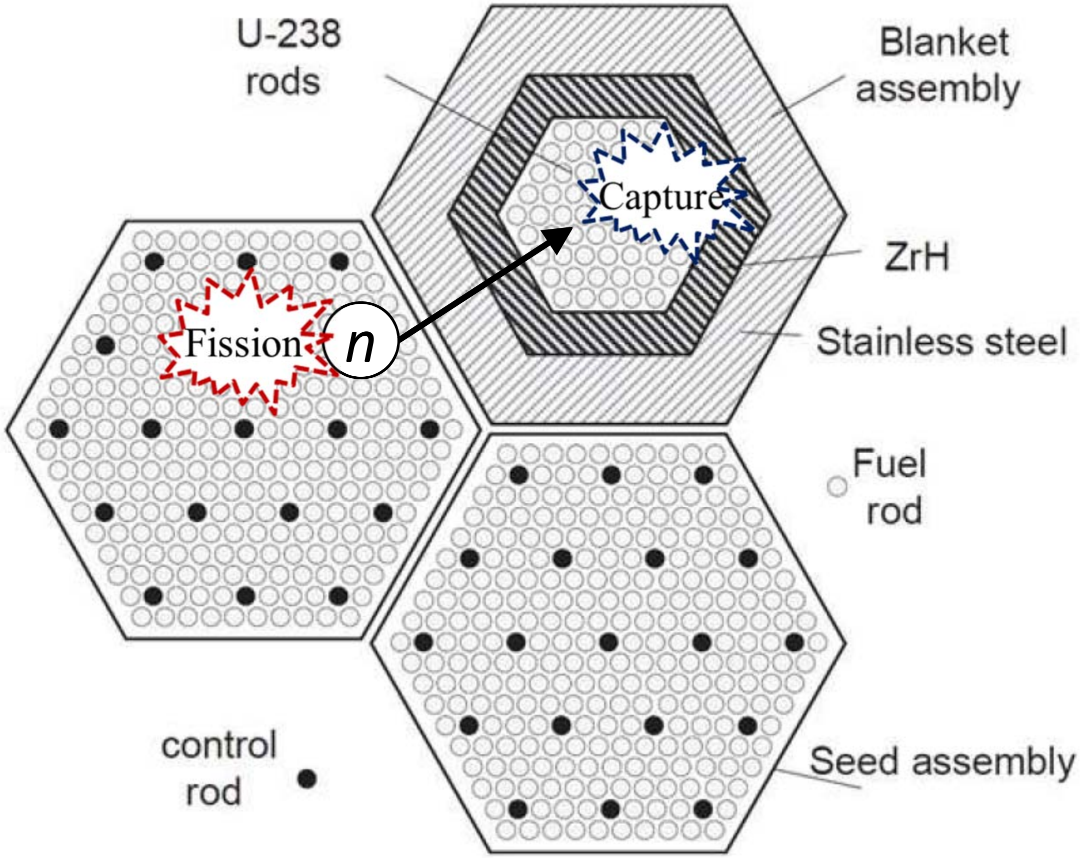
革新技術創出発展型　平成22年度採択課題
「軽水冷却スーパー高速炉に関する研究開発」

Example of Fuel Assembly (Fast)

- Tight hexagonal fuel lattice without water rods
- Heterogeneous core configurations with seed (MOX) and blanket (depleted U)
- Cluster-type control rods
- Use of solid moderator（ZrH）to attain negative void reactivity characteristics without reducing the core height (neutron leakage)



Transmutation of Long Lived Fission Products（LLFP）and Minor Actinides（MA）

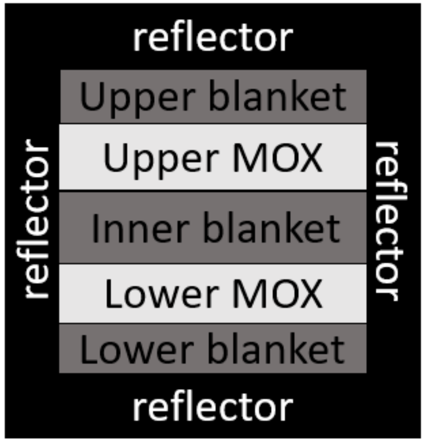
Lu et al., “Study on the LLFPs transmutation in a super-critical water-cooled fast reactor”, Nuclear Engineering and Design 241 (2011) 395–401.

Cao et al., “Research and Development of a Super Fast Reactor (4) Transmutation Analyses of Minor Actinides and Transuranium Elements” Proc. 16PBNC, 2008.

L. Cao et al., “Fuel, Core Design and Subchannel Analysis of a Superfast Reactor”, Journal of NUCLEAR SCIENCE and TECHNOLOGY, Vol. 45, No. 2, p. 138–148 (2008)

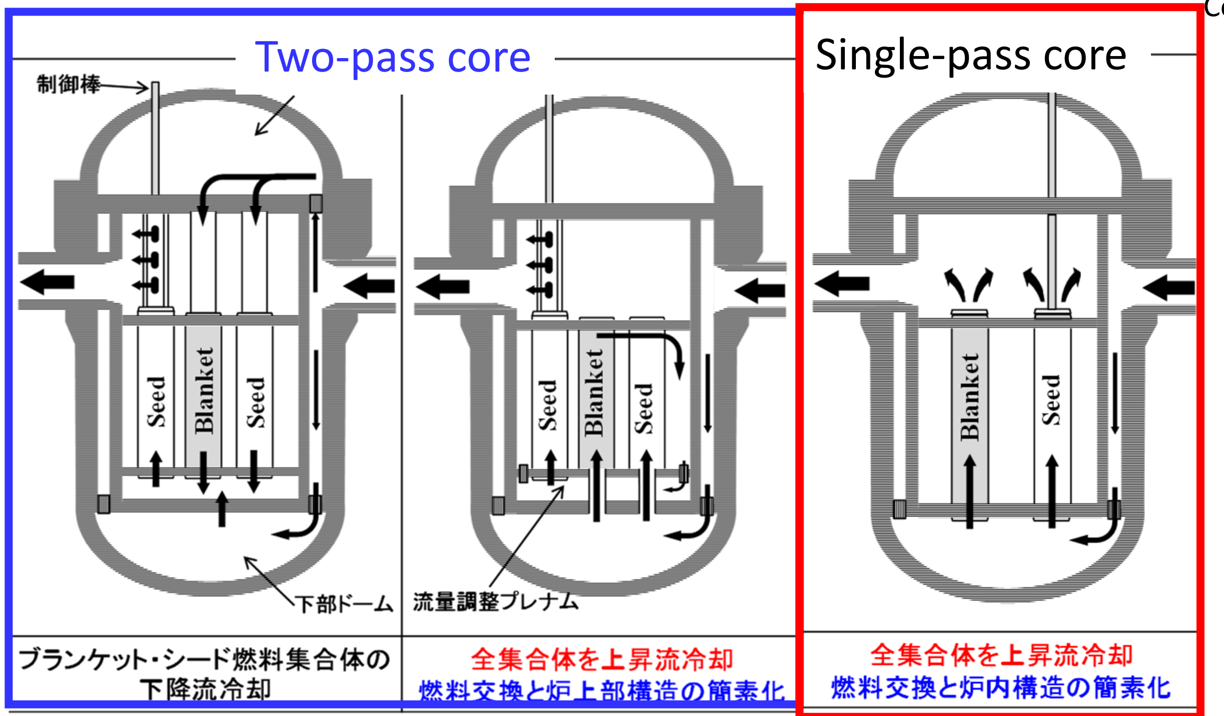
Example of Fast Reactor Design

- From high conversion type to breeding type
- Negative void reactivity characteristics is necessary, because of high-pressure system
- Seed / blanket heterogeneous core configurations
- Buildup of fissile Pu in blanket leads to “power swing” → efficient core cooling is the design issue.

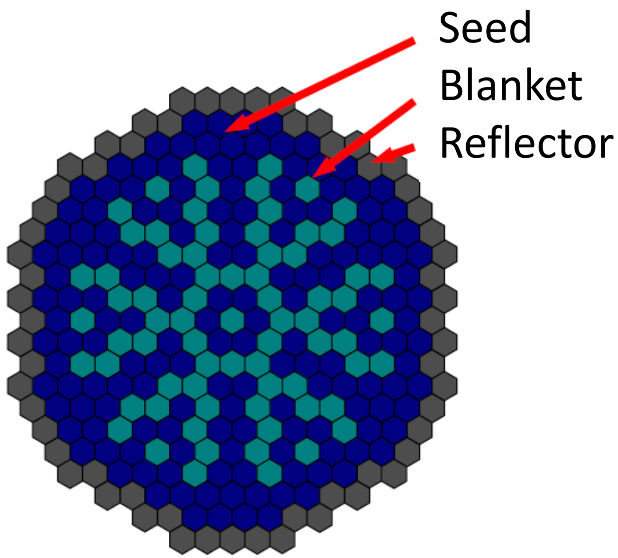


Axially heterogeneous core

S. Noda et al., “Core Design Study of Super FBR with Multi-Axial Fuel Shuffling and Different Coolant Density”, Proc. ICONE26, July 22-26, 2018,



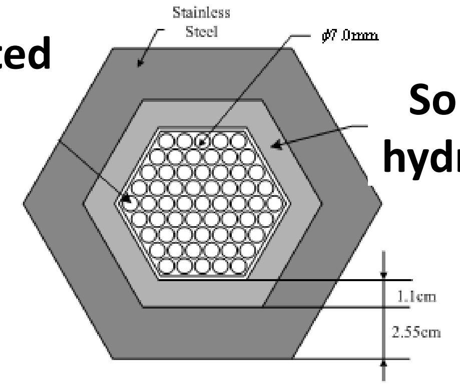
Q. Liu and Y. Oka, “Single pass core design for a Super Fast Reactor”, Annals of Nuclear Energy 80 (2015) 451–459



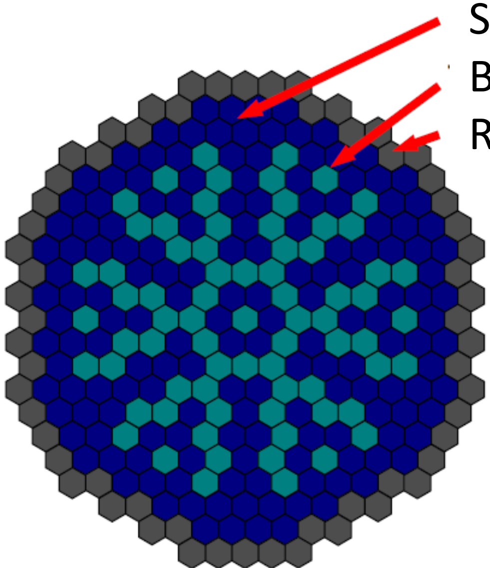
Radially heterogeneous core

Basic Reactor Physics Data from Experiments

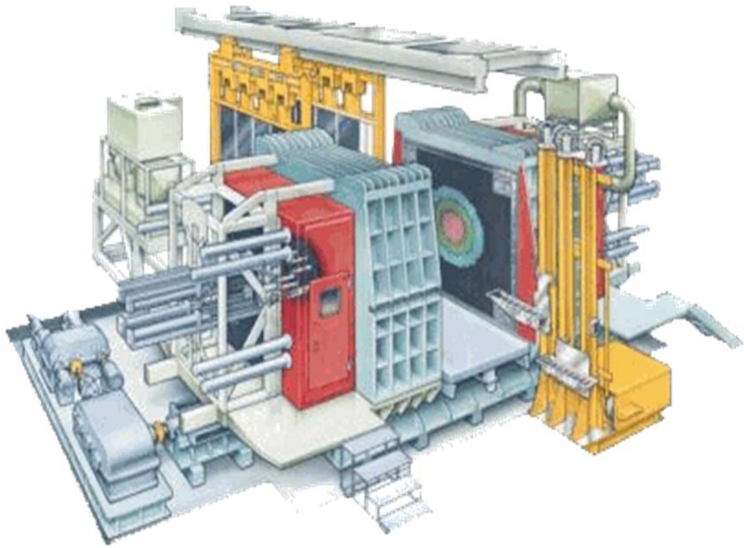
- Basic reactor physics data of solid moderator (ZrH) obtained from Fast Critical Assembly (FCA) of Japan Atomic Energy Agency (JAEA)



Blanket fuel assembly



Super FR core configuration



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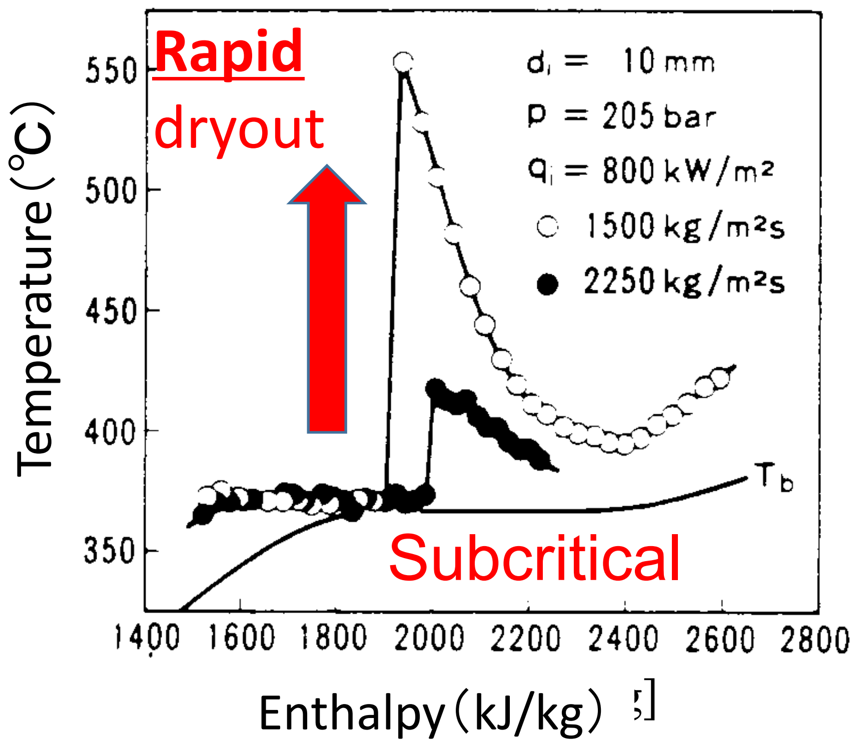
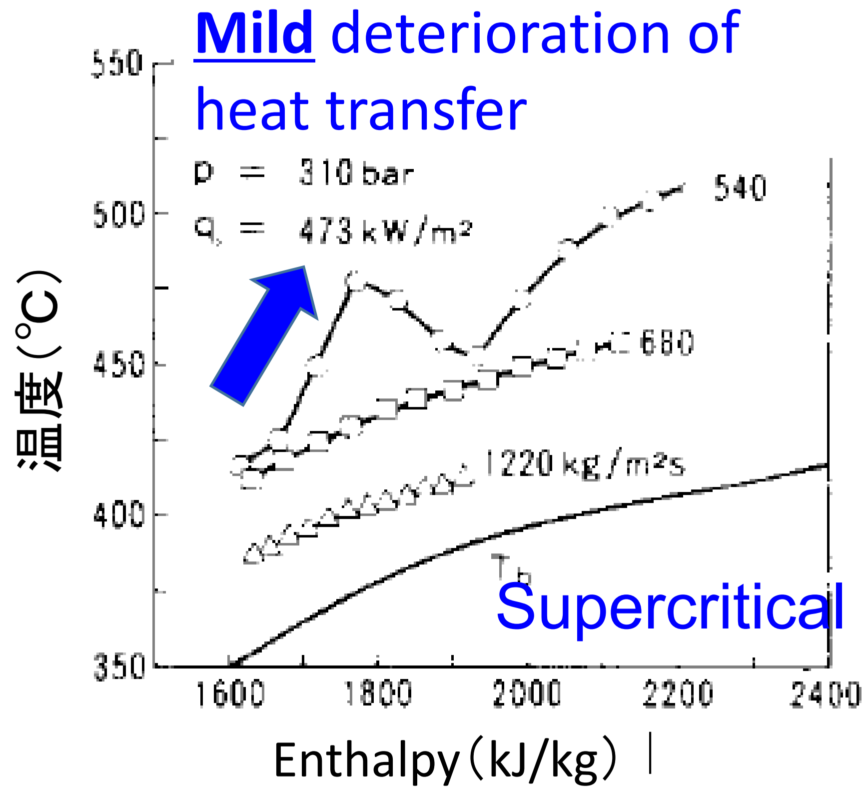
Safety Feature for Ensuring Core Cooling





Criterion to ensure sufficient cooling

Heat flux

Direct evaluation of clad surface temperature

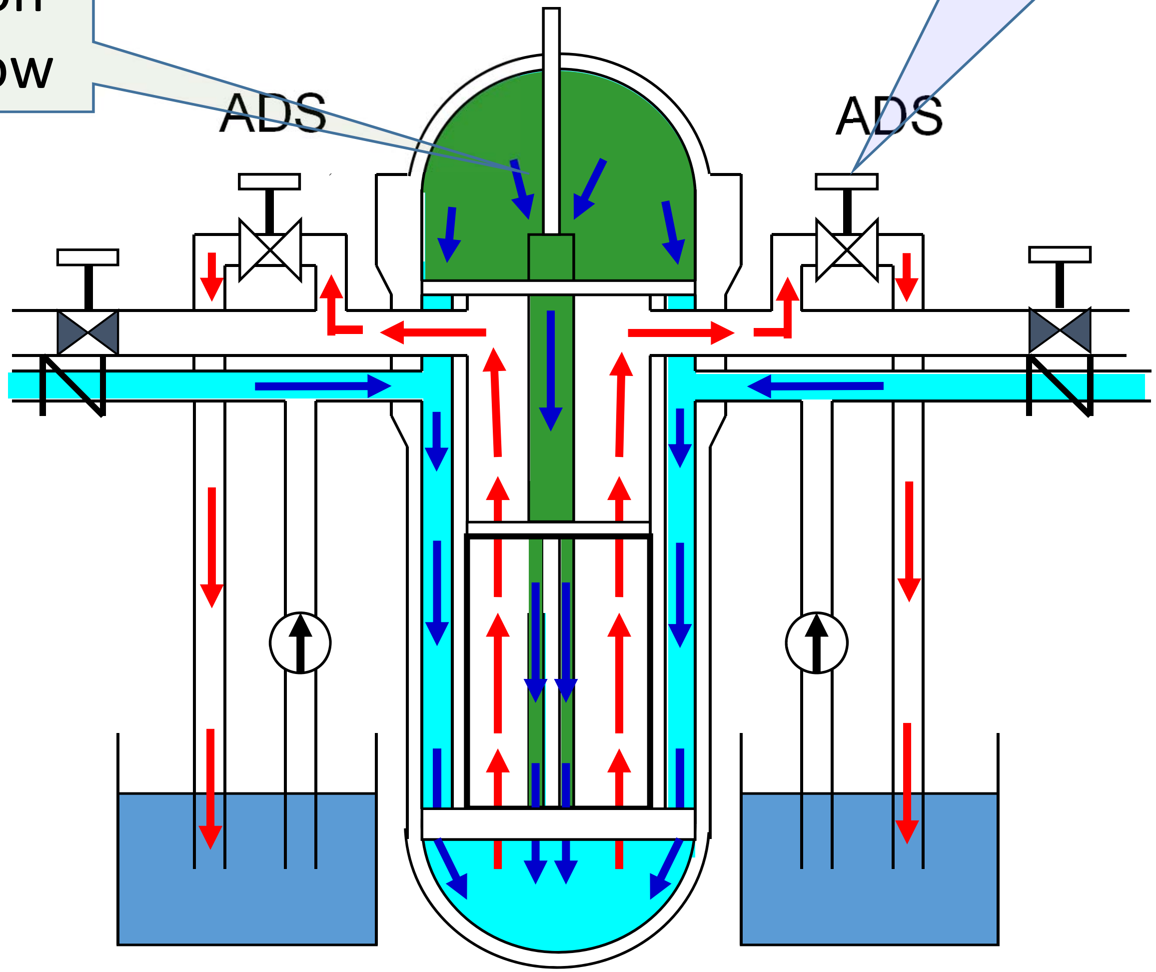
High temperature core design > 500 C

Safety Feature of Once-through Direct Cycle

Simple principle:
Keep inlet and outlet open

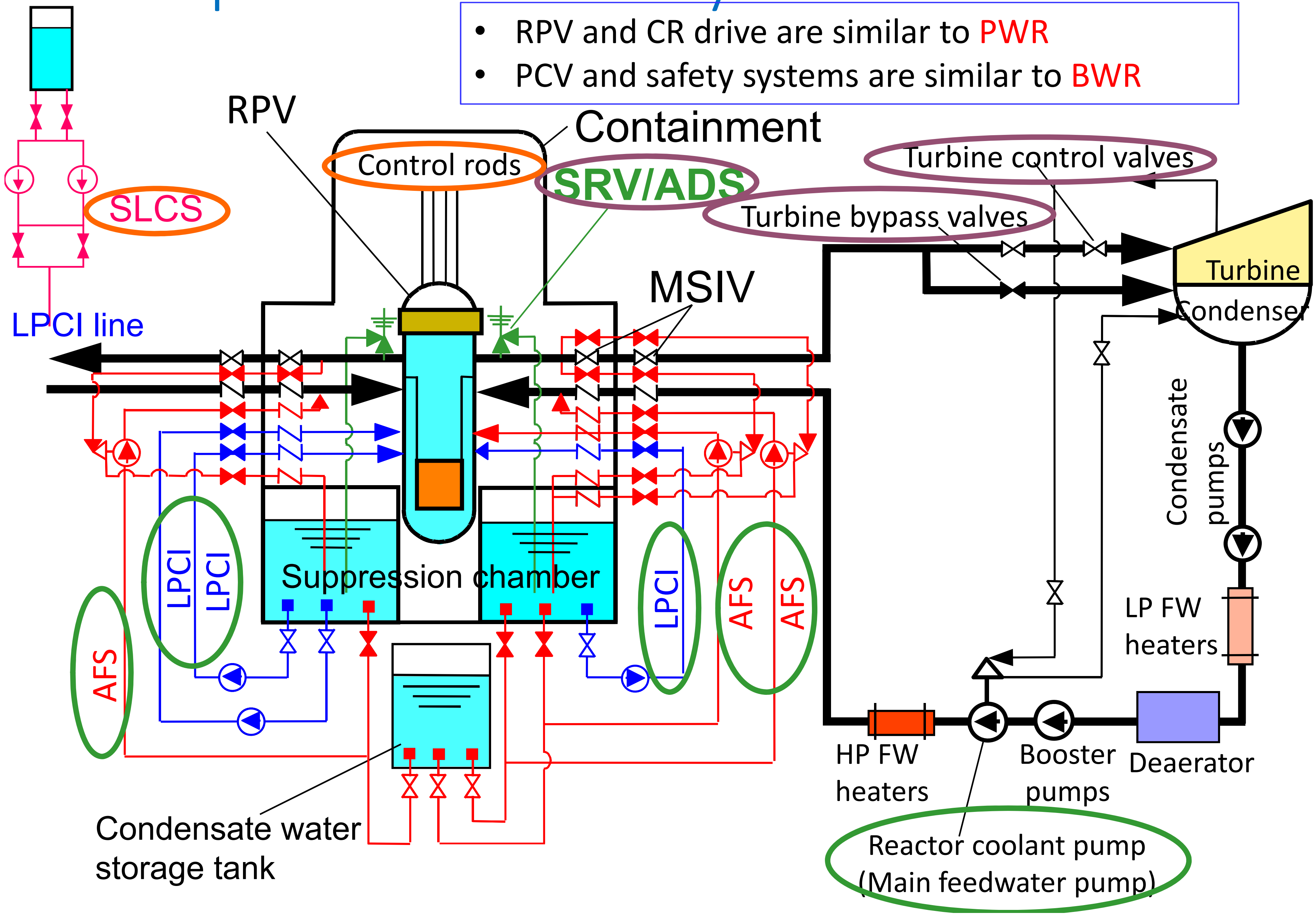
Keep outlet open→ core cooling

Depressurization induces core flow



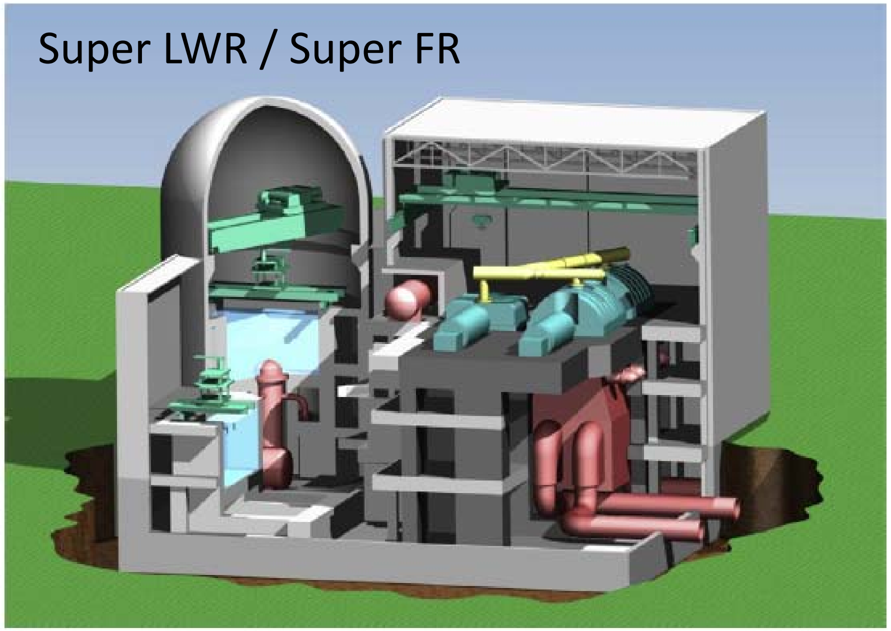
Example of Plant Safety and Control

- RPV and CR drive are similar to PWR
- PCV and safety systems are similar to BWR

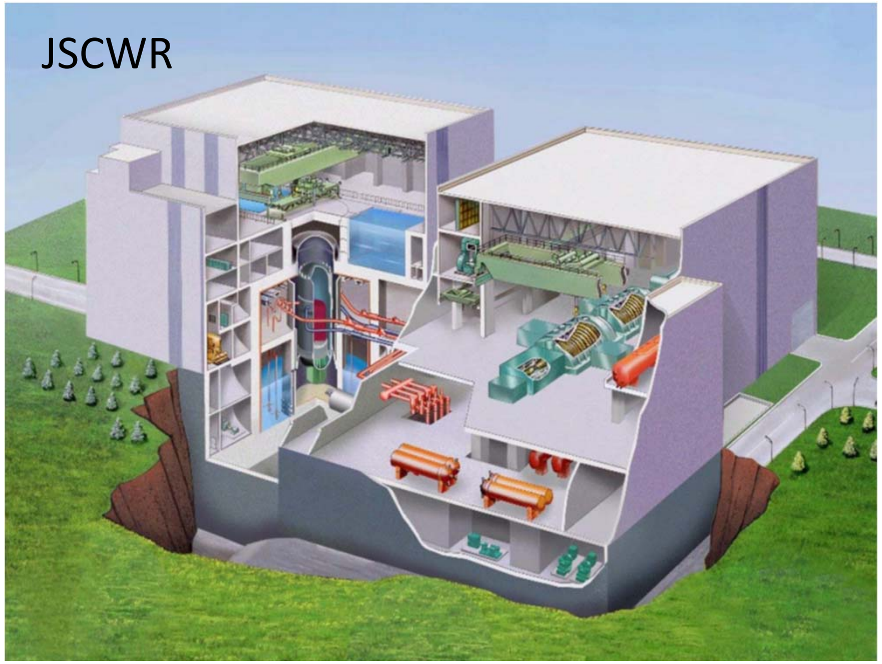


Examples of Plant Design

- RPV is similar to PWR
- Control rod drive mechanisms
  - Similar to PWR（Super LWR / Super FR）
  - Similar to BWR（JSCWR）
- Wet well based containment design, similar to ABWR
- Passive safety

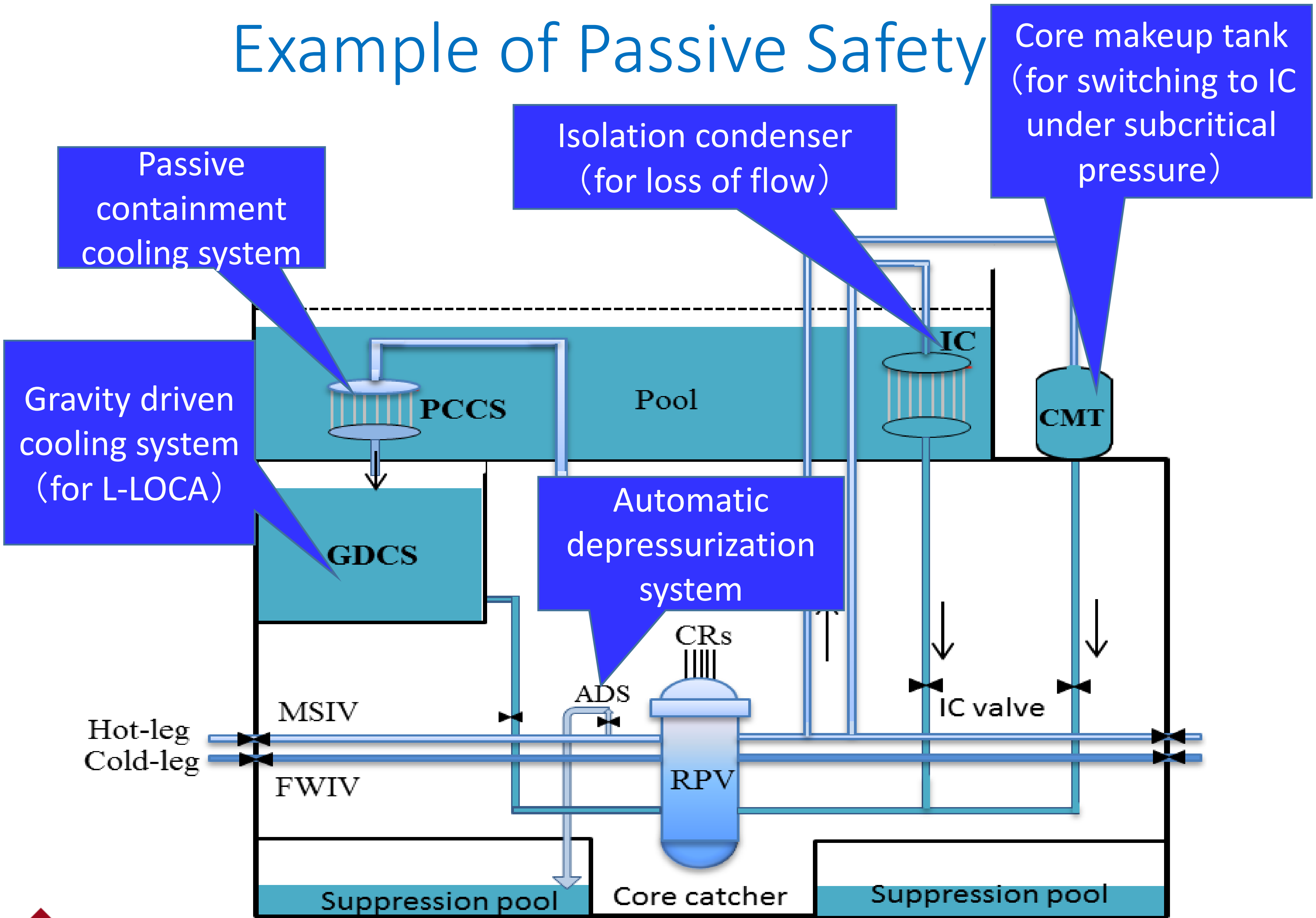


Y. Oka, H. Mori, "Supercritical-Pressure Light Water Cooled Reactors", ISBN 978-4-431-55024-2, Springer Tokyo Heidelberg New York Dordrecht London, 2014.



K. Yamada et al., "Overview of the Japanese SCWR Concept Developed Under the GIF Collaboration", The 5th Int. Sym. SCWR (ISSCWR-5) , 2011

Example of Passive Safety



Sutanto, Y. Oka, "Passive safety system of a super fast reactor", Nuclear Engineering and Design 289 (2015) 117–125.

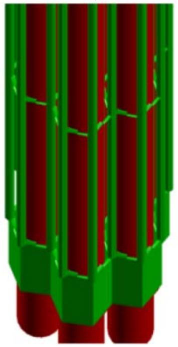
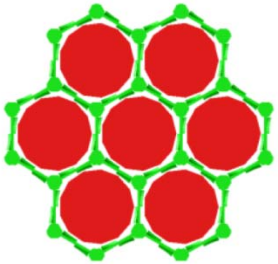
Core Thermal-hydraulics

Examples of R&D

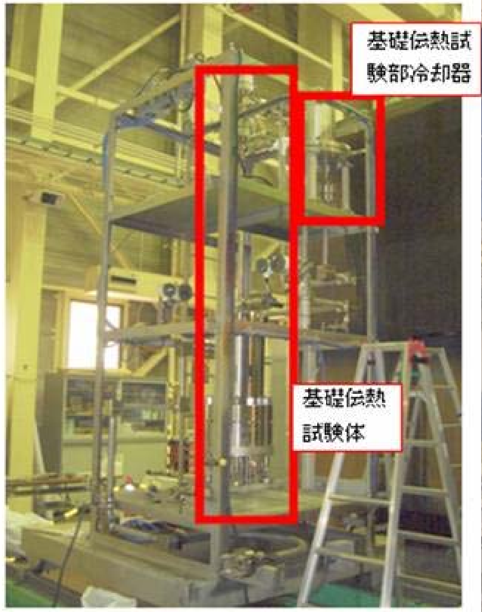
- Improving accuracy of the clad surface temperature evaluation
  - **Cross-flow** between adjacent subchannels
  - Effects of spacers on **turbulence**, **pressure loss**, and **heat transfer**
  - **Heat transfer deterioration** (supercritical pressure)
  - **Critical heat flux** (subcritical pressure)
- Abnormal transients and accidents
  - **Condensation** of supercritical water in suppression pool
  - **Critical flow** of supercritical water from break

Database and Correlation for Designing

- Simulated experiment with freon gas （HCFC22、HCFC123）（Kyushu U.）
- Supercritical water loop（JAEA）
- CFD development（JAEA/ACE-3D）

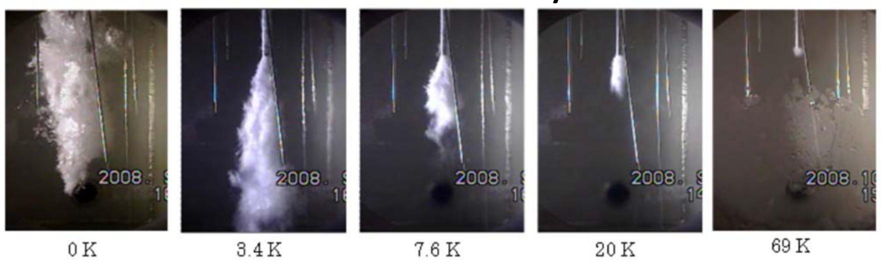




Design and analysis of grid spacers（Waseda U.）



Supercritical water loop （JAEA）

Turbulent mixing cross-flow test （Kyushu U.）



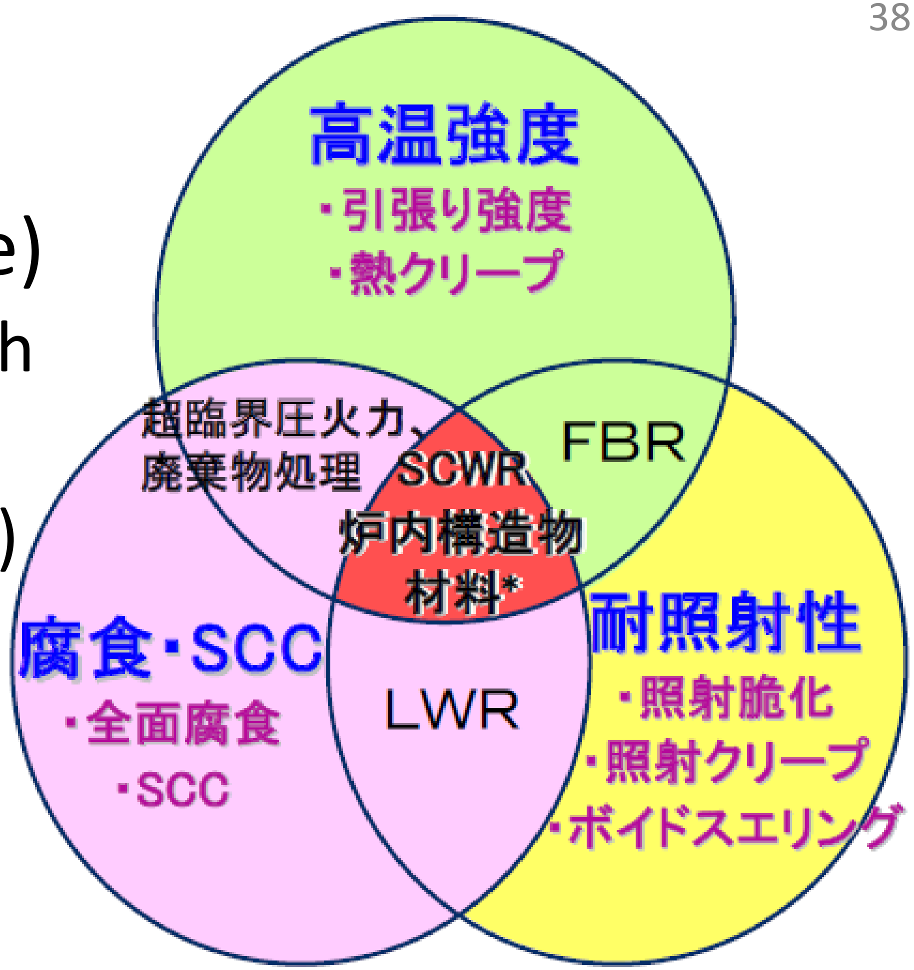
Condensation of supercritical fluid （Kyushu U.）

Advances in reducing uncertainties with thermal-hydraulics

Materials and Corrosions

Examples of R&D

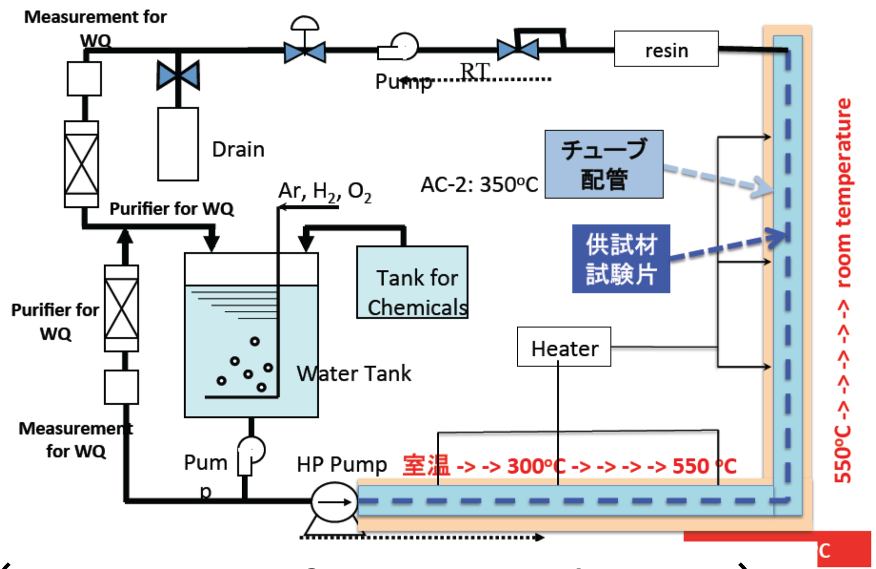
- Required properties (core)
  - High temperature strength
  - Irradiation resistance
  - Corrosion resistance (SCC)
- Candidates
  - Advanced stainless-steel
  - Ni based alloys
  - etc



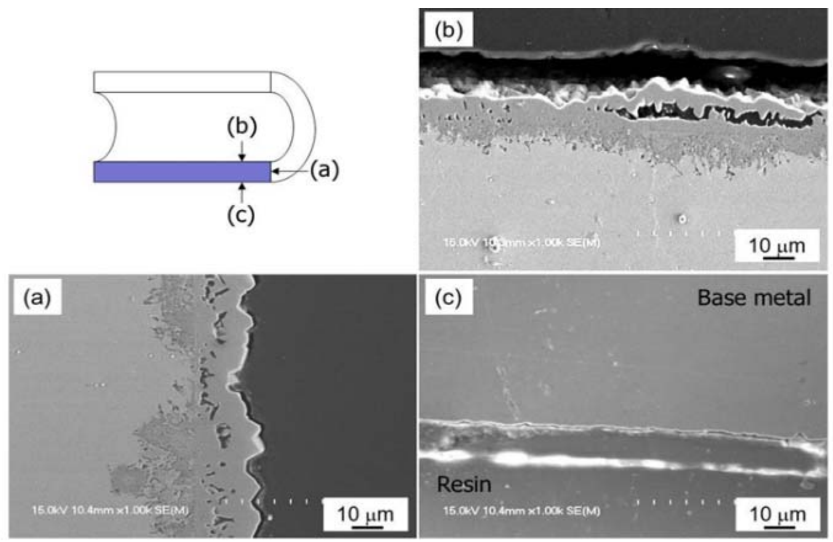
- Other issues may include, but not limited to:
  - Optimization of clad manufacturing, long term data accumulation, corrosion with irradiation, integrated fuel performance test

Examples of Database for Material Performances

- 15Cr-20Ni-advanced austenitic SS cladding（JAEA）
  - Corrosion resistance improvement of SFR cladding (PNC1520)
- Oxidation in supercritical water（Tohoku U.）
  - Oxide layer growth data
- Corrosion, elution, precipitation test and analyses（mass transfer test, Tokyo U.）
- Thermal insulator development



（Mass transfer test, Tokyo U.）



Post test SEM of oxidation test
（Tohoku U.）

Domestic and International Activities

References :

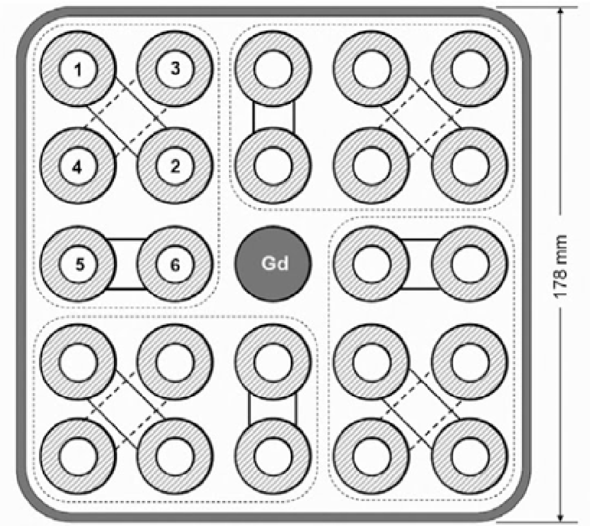
Heat Transfer Behaviour and Thermohydraulics Code Testing for Supercritical Water Cooled Reactors (SCWRs), **IAEA-TECDOC-1746**

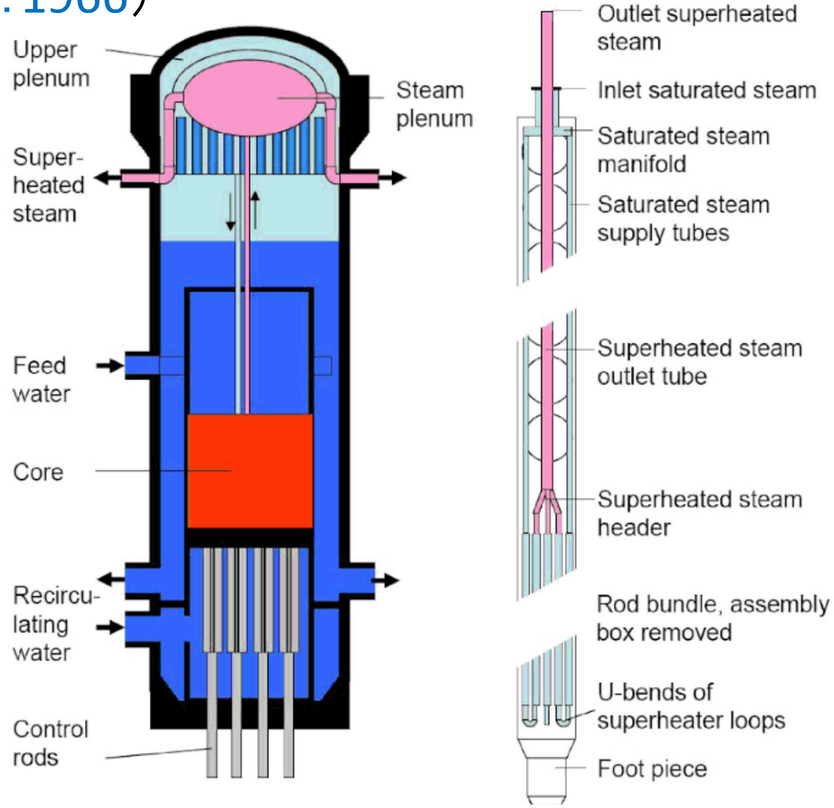
L.K.H. Leung et al., “An Update on the Development Status of the Super-Critical Water-cooled Reactors”, GIF Symposium – Paris (France) – 16-17 October 2018

Early Days

- HW mod・Supercrt. LW cool・**Indirect cycle**（design：Westinghouse, 1957）
- HW mod ・ Supercrt. LW cool・ **Direct cycle** （design：General Electric, 1959）
- Carbon mod・ Supercrt. LW cool・**P tube Direct**（design：Westinghouse, 1962）
- PWR type・primary at Supercrt.（Proposed：1966）

BWR type・superheater・AEG Co.・HDR（Heissdampfreaktor）、Critical 1969、100MWt、main steam 457℃／7.3MPa





Schulenberg, T., Starflinger, J., 2012. High Performance Light Water Reactor, Design and Analyses. KIT Scientific Publishing, Karlsruhe.

Domestic R&D

- Conceptual development（Tokyo U.1989～2010／Waseda U. 2010～）
- JSPS projects（Tokyo U.、1998-2002）
- METI projects
  - Feasibility study：Toshiba, Hitachi, Tokyo U., Kyushu U., Hokkaido U.（2000-04）
  - Materials：Toshiba, Hitachi-GE, Hitachi, Tohoku U., Tokyo U., JAEA（2004-08）
  - GIF Collaboration Phase 1: IAE, Toshiba, Tokyo U., Kyushu U., JAEA, Hitachi-GE, Hitachi（Phase 1: 2008-2010）
- MEXT Projects
  - Tokyo U., JAEA, CRIEPI, Hitachi, Toshiba （2002-06）
  - Tokyo U., JAEA, TEPCO, Kyushu U.（2005-09）
  - Tokyo U., TEPSYS, Kyushu U., Tohoku U., JAEA, AIST（2010-13）
  - Kaiyo U., Tokyo U., Waseda U. （2015-18）
- JP-US Joint Project（Tohoku U., Toshiba, Hitachi, Tokyo U., 2004-06)

International Activities

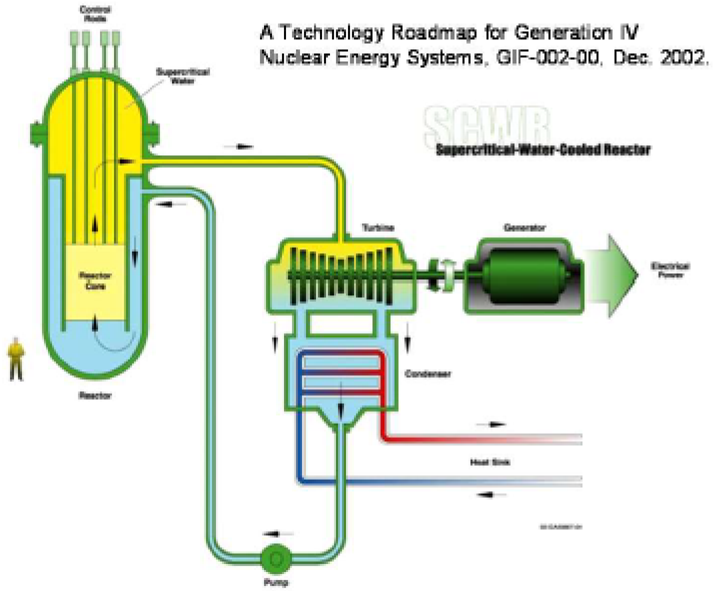
- Generation IV International Forum
  - Technical secretariat: OECD/NEA
  - Canada, Euratom, Japan, Russia, China
- IAEA
  - Coordinated Research Project（2008-2012, 2014-2018）
    - Thermal-hydraulics experiments and data
    - Benchmark analyses
  - Technical meetings（5）：thermal-hydraulics, materials, corrosions, water chemistry
  - Training courses（4）
  - Canada, China, Germany, Hungary, India, Italia, Russia, UK, Ukraine, USA
- International symposium（ISSCWR）
  - 1^{st} symposium：Tokyo U.（2000）
  - Japan, China, Germany, Canada, Finland
  - 10^{th}： Prague, Czech Republic, Week 15-19 March 2021 (tentative)

Generation IV International Forum

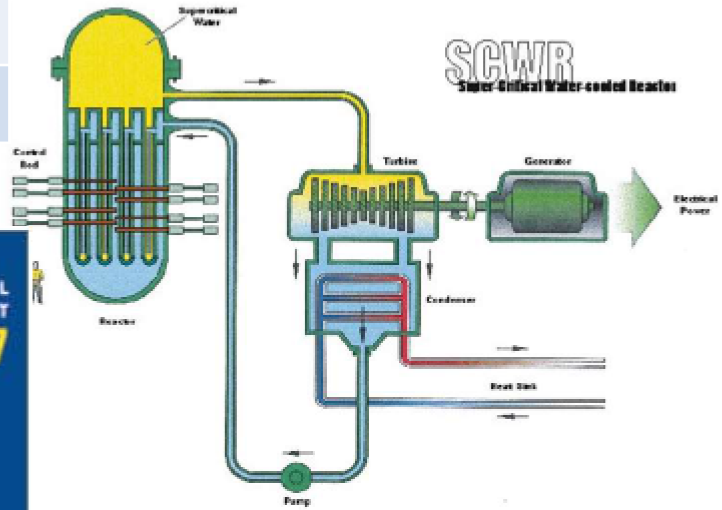
- SCWR selected in 2002
- System Steering Committee

（2019年12月現在）

参加国	署名機関
Canada	AECL (with CNL)
Euratom	JRC
Japan	METI / ANRE
China	中国国家原子能機構(CAEA) 中国科学技術部(MOST)
Russia	ROSATOM



A Technology Roadmap for Generation IV Nuclear Energy Systems, GIF-002-00, Dec. 2002.



Project management

- Material & Chemistry
- Thermal-hydraulics & Safety
- (System Integration and Assessment)

Thermal-hydraulics
Material Corrosions

Canada, Finland, Netherlands, Spain, Japan

Corrosion Science 106 (2016) 147–156

Contents lists available at ScienceDirect

Corrosion Science

journal homepage: www.elsevier.com/locate/corsc

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The reproducibility of corrosion testing in supercritical water—Results of an international interlaboratory comparison exercise

CrossMark

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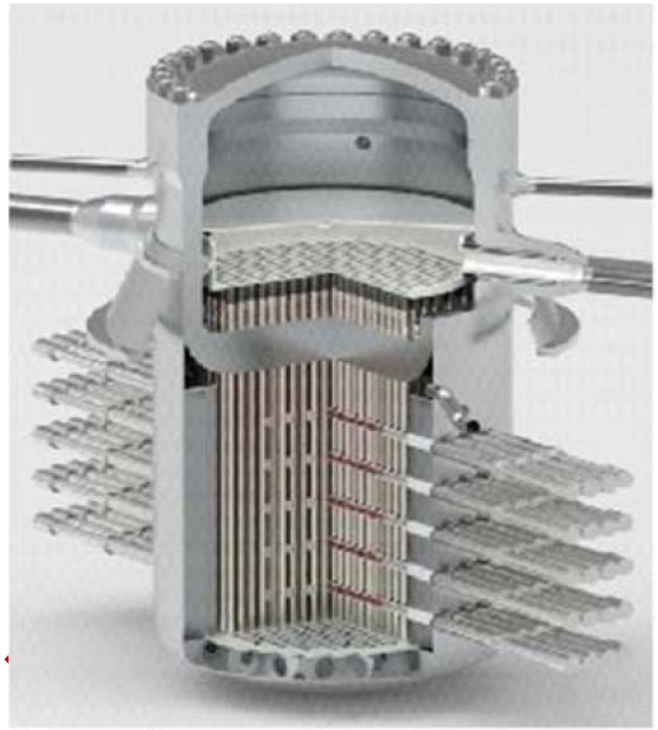
Netherlands, Italy, Hungary, Canada, Czech Republic, Germany, Sweden, Poland, Japan

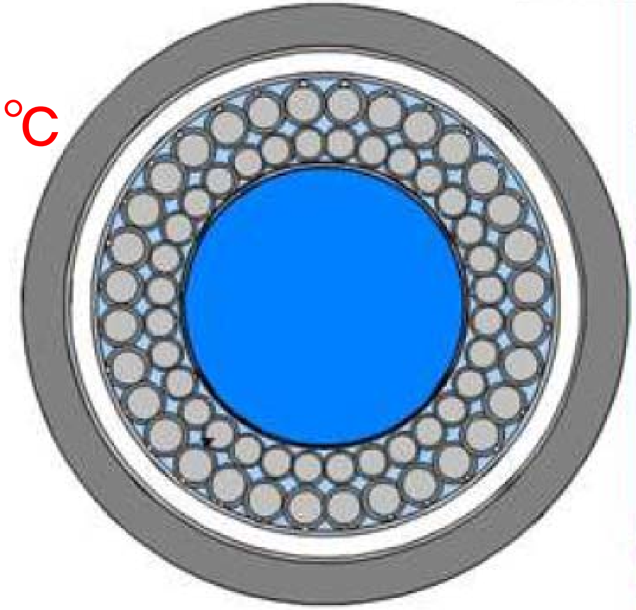
A Blind, Numerical Benchmark Study on Supercritical Water Heat Transfer Experiments in a 7-Rod Bundle

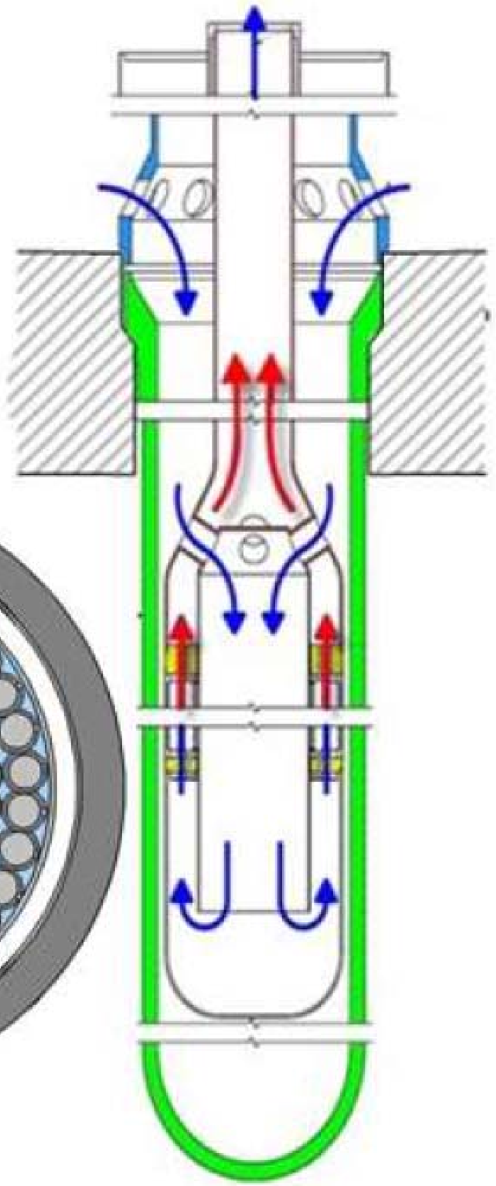
Heat transfer in supercritical water reactors (SCWRs) shows a complex behavior, especially when the temperatures of the water are near the pseudocritical value. For example, a significant deterioration of heat transfer may occur, resulting in unacceptably high cladding temperatures. The underlying physics and thermodynamics behind this behavior are not well understood yet. To assist the worldwide development in SCWRs, it is therefore of paramount importance to assess the limits and capabilities of currently available models, despite the fact that most of these models were not meant to describe supercritical heat transfer (SCHT). For this reason, the Gen-IV International Forum initiated the present blind, numerical benchmark, primarily aiming to show the predictive ability of currently available models when applied to a real-life application with flow conditions that resemble those of an SCWR. This paper describes the outcomes of ten independent numerical investigations and their comparison with wall temperatures measured at different positions in a 7-rod bundle with spacer grids in a supercritical water test facility at JAEA. The wall temperatures were not known beforehand to guarantee the blindness of the study. A number of models have been used, ranging from a one-dimensional (1-D) analytical approach with heat transfer correlations to a RANS simulation with the SST turbulence model on a mesh consisting of 62 million cells. None of the numerical simulations accurately predicted the wall temperature for the test case in which deterioration of heat transfer occurred. Furthermore, the predictive capabilities of the subchannel analysis were found to be comparable to those of more laborious approaches. It has been concluded that predictions of SCHT in rod bundles with the help of currently available numerical tools and models should be treated with caution. [DOI: 10.1115/1.4031949]

Canada

- Atomic Energy of Canada Limited（AECL）
- Pressure Tube type（LW cool・HW moderated）
- Vertical arrangement with fuel batch replacement
- Downward flow core cooling
- Thermal reactor
- Th-Pu fuel
- Main steam temperature：625℃



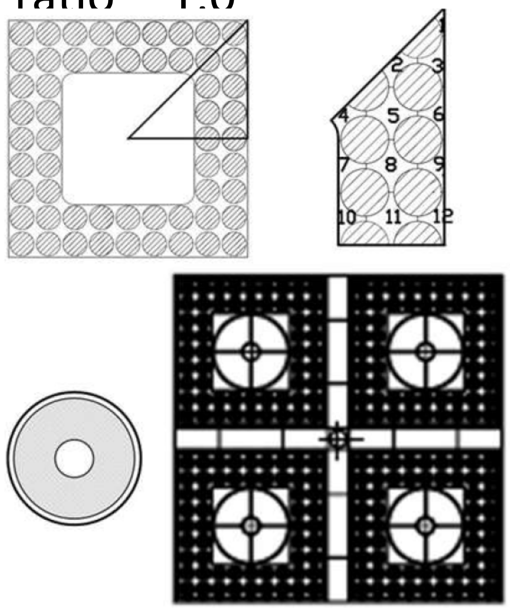


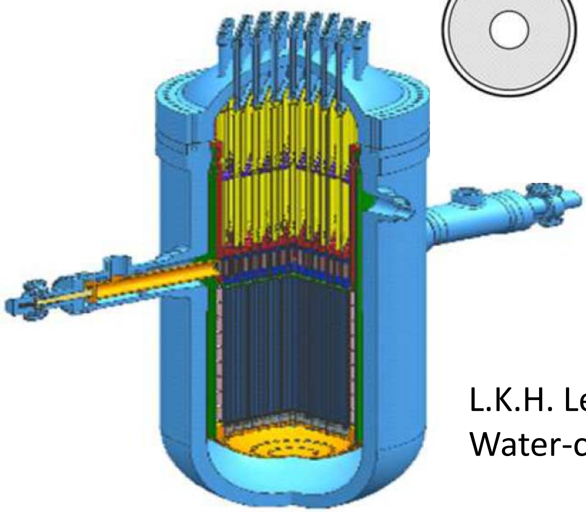


L.K.H. Leung et al., "An Update on the Development Status of the Super-Critical Water-cooled Reactors", GIF Symposium – Paris (France) – 16-17 October 2018

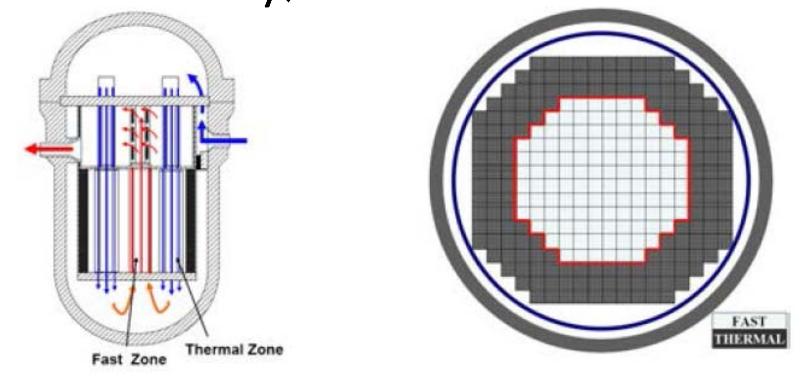
China

- Thermal：CSR-1000（Nuclear Power Institute of China, NPIC）
    - Hollow pellet to reduce fuel temperature
- Thermal-fast MIX：SCWR-M（上海交通大学）
    - Thermal zone：Downward flow cooling（high water density）
    - Fast zone：Upward flow cooling（low water density）
    - Conversion ratio～1.0

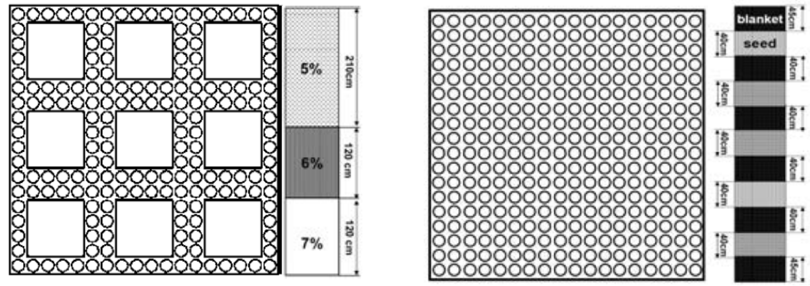




CSR-1000
（Thermal）



(a) Flow paths in the core (b) Fuel assembly arrangement in the core



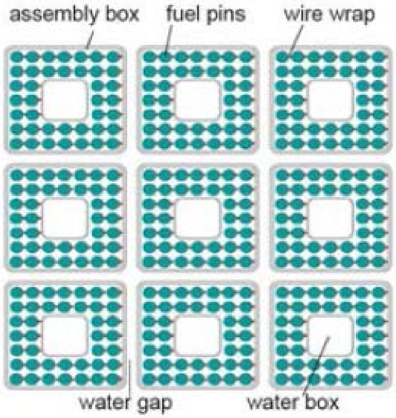
(a) Fuel assembly in the thermal zone (b) Fuel assembly in the fast zone

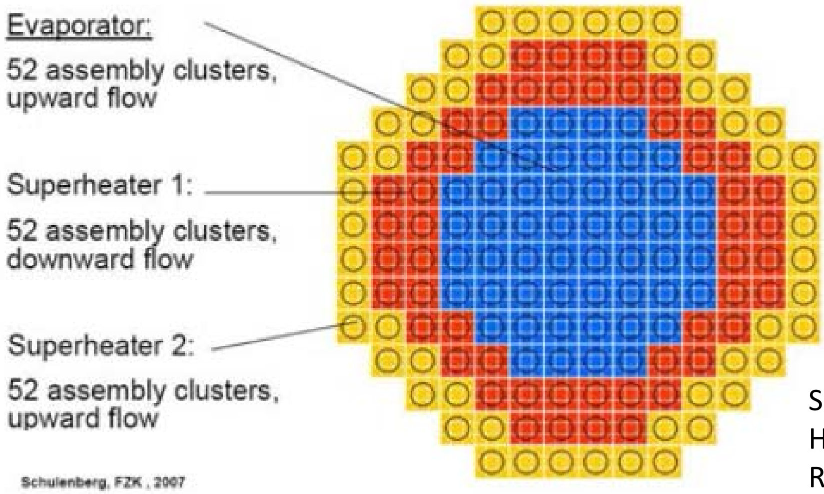
SCWR-M（Thermal-fast-mix）

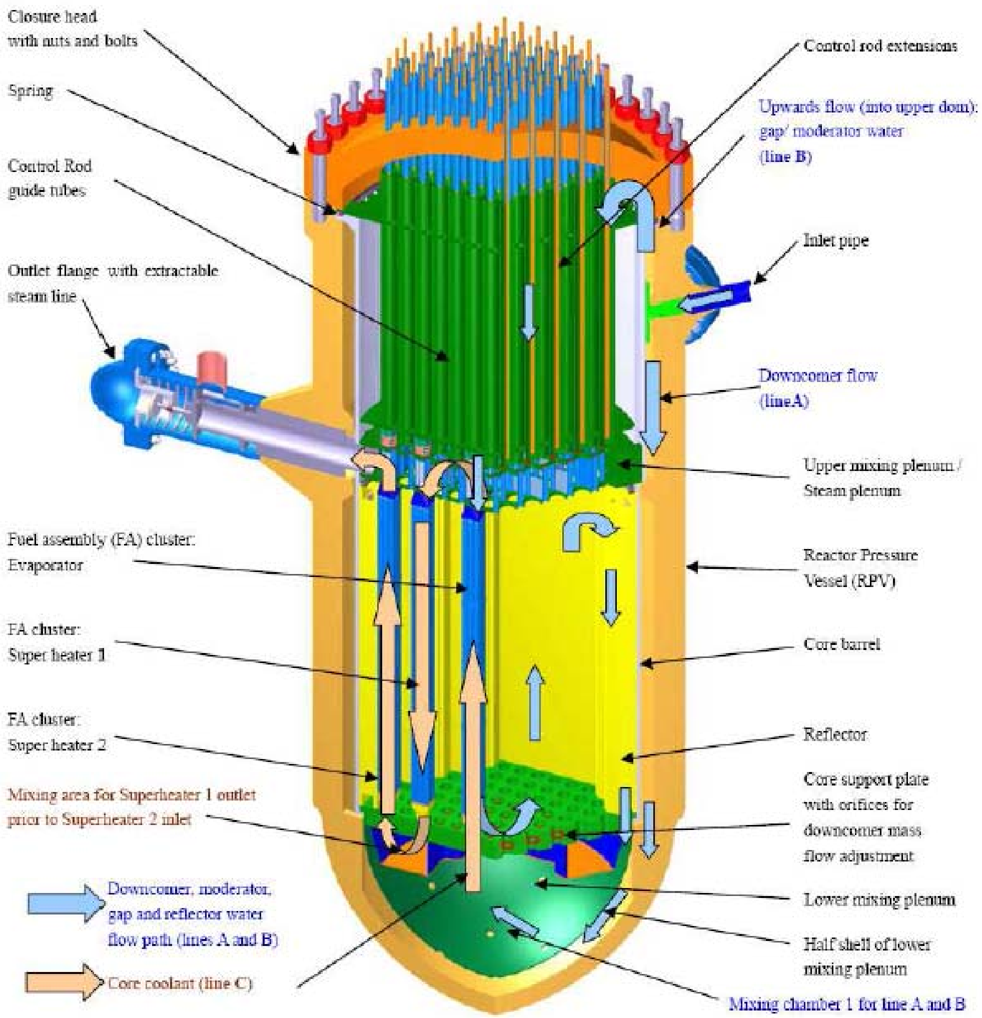
L.K.H. Leung et al., “An Update on the Development Status of the Super-Critical Water-cooled Reactors”, GIF Symposium – Paris (France) – 16-17 October 2018

Europe

- Euratom: HPLWR
- Thermal reactor
- Coolant flow scheme
  - up→down→up



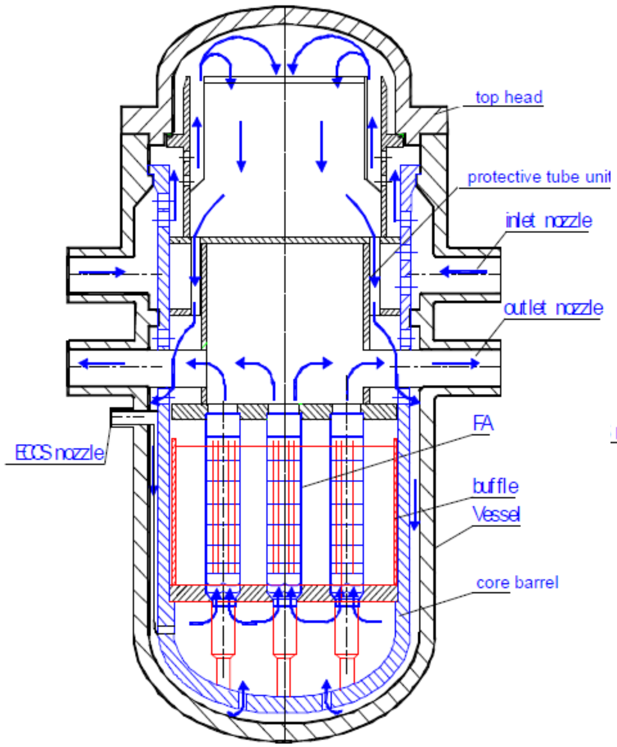




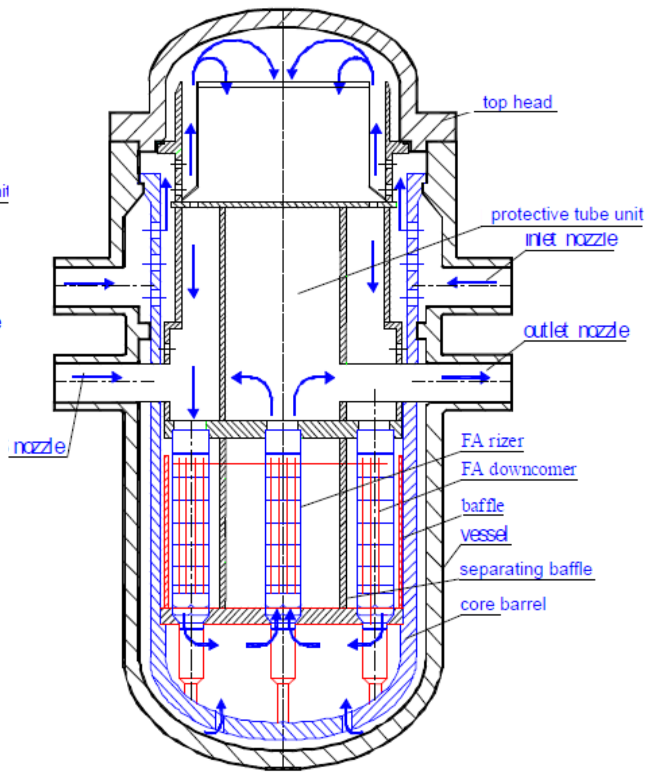
Schulenberg, T. and Leung, L. “Super-critical water-cooled reactors”, Handbook of Generation IV Nuclear
Reactors, Editor: I.L Pioro, Woodhead Publishing Series in Energy: 103, 2016.

Russia

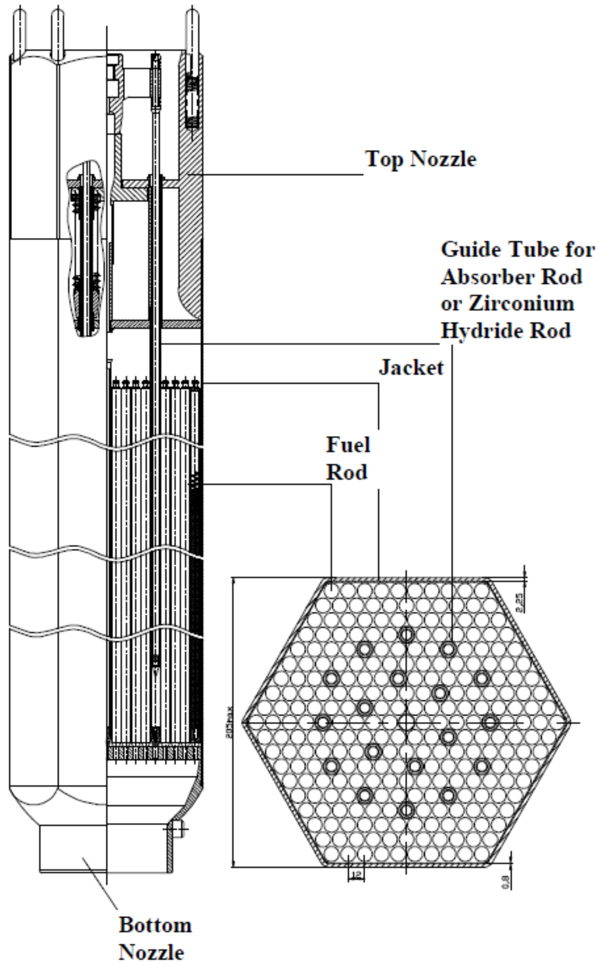
- OKB GIDROPRESS: VVER-SCP
- Fast～resonance（Conversion ratio: 0.9～1）



Upward



Downward／upward

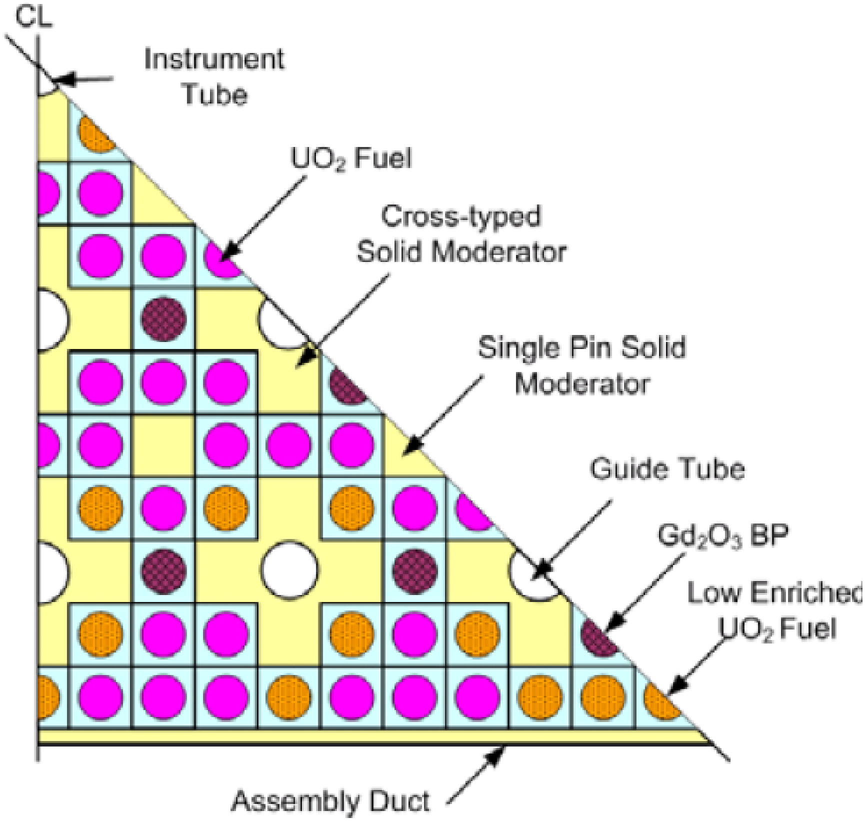


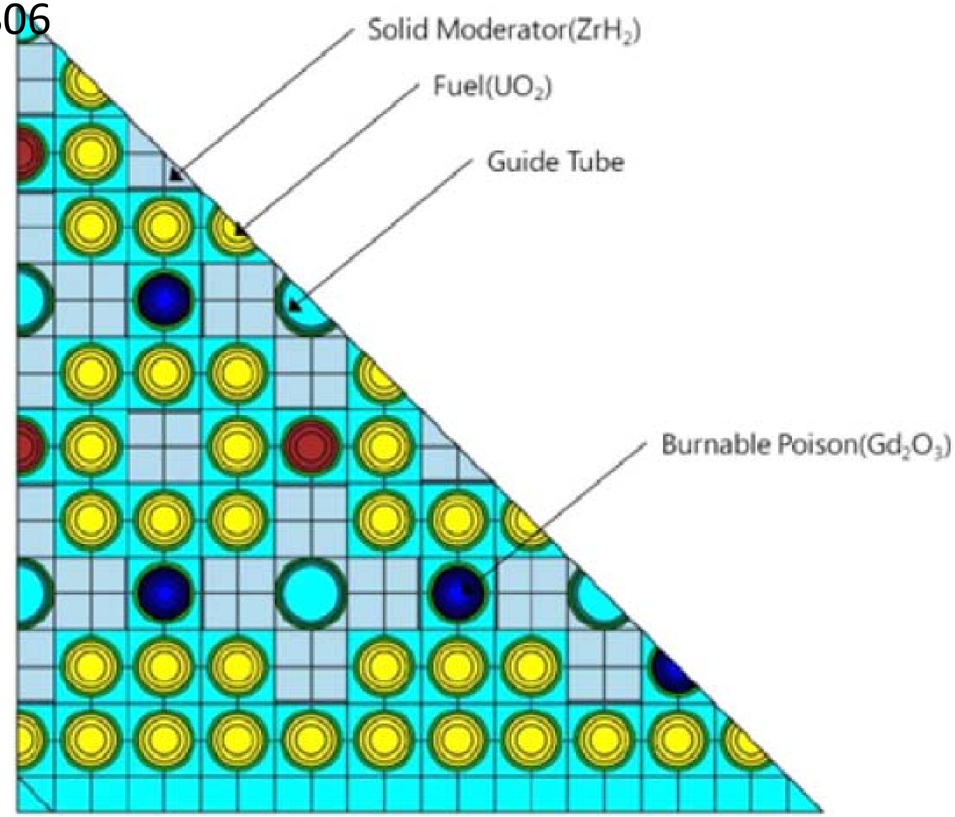
Heat Transfer Behaviour and Thermohydraulics Code Testing for Supercritical Water Cooled Reactors (SCWRs), **IAEA-TECDOC-1746**

Korea

- KAERI: SCWR-SM
- Thermal reactor（1400MWe , 1700MWe）
  - Solid moderator (ZrH_2) to simplify RPV upper dome structure

T. Schulenberg, et. al., “Supercritical Water-Cooled Reactor (SCWR) Development through GIF Collaboration”, IAEA-CN-164-5S06





Plans

- Canada
  - Large ＋ small（300MWe class SuperSafe）designs
  - In-core materials, fuel test, reactor physics test
  - Prototype（under GIF framework）
- China
  - Optimization of CSR-1000
  - In-core fuel test loop design
  - Prototype（CSR-150）design and construction (proposals)
- Euratom
  - Materials, thermal-hydraulics
  - EU･China･Canada joint proposal：materials, corrosion, turbulence heat transfer, mass transfer
  - In-core material irradiations
  - European Small Modular supercritical water Reactor Technology (E-SMART) (proposal)
- Russia
  - Fast reactor
  - Thermal-hydraulics and materials

Future Outlook

- Substantial reduction in risks and uncertainties in R&D
- Reducing capital cost is the must for gaining competitiveness of nuclear power with other power sources
- Issues :
  - Most issues are common in both thermal and fast reactors
  - Improving accuracy of thermal-hydraulics analyses
  - Material behavior and chemistry under irradiation
  - Fuel irradiation test
  - Prototype reactor design and construction

Research at University（人材育成）

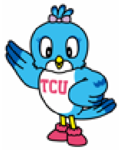
- Understanding interactions between different fields and cross-cutting issues

- SCWR is LWR. Ready to work in industries.

- SCWR is single-phase cooling. Experience from SFR is also useful.

- Creative issues attract bright students.

Thanks for your attention.



東京都市大学・早稲田大学
共同原子力専攻