Research and Development of Supercritical-pressure light water cooled reactors, Super LWR and Super FR

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Outline

1. Introduction
2. Fuel and core design
3. Safety
4. Fast reactor
5. R&D
Change of density and specific heat of water with temperature at supercritical pressure (25 MPa)
Super LWR

- Super LWR: Supercritical-pressure light water cooled and moderated reactor developed at Univ. of Tokyo
- Once-through direct cycle thermal reactor

- Pressure: 25 MPa
- Inlet: 280°C
- Outlet (average): 500°C
- Flow rate: 1/8 of BWR
Evolution of boilers

Circular Boiler

Water tube boiler

Once-through boiler

Super LWR, Super FR (SCWR)
Supercritical fossil-fired power plants

Once-through boilers
Number of units are larger than that of LWRs.
Proven technologies; turbines, pumps, piping etc.
USA; developed in 1950’s, Largest unit is 1300MWe.
Japan; deployed in 1960’s and constantly improved.
Many plants in Russia and Europe.

Compact SC turbine (700MWe, 31.0MPa, 566°C)
Features of Super LWR/Super FR

• Compact & simple plant systems; Capital cost reduction
  – No steam/water separation and no SGs: Coolant enthalpy inside CV is small.
  – High specific enthalpy & low flow rate: Compact components

• High temperature & thermal efficiency (500°C, ~44%)

• Utilize LWR and Supercritical FPP technologies:
  - Temperatures of major components below the experience
  - Same plant system between thermal and fast reactor

\[
\text{BWR} \quad \text{PWR} \quad \text{Supercritical FPP (once-through boiler)}
\]

\[\text{Super LWR/Super FR}\]
Fuel and core design
Core design criteria

Thermal design criteria

- Maximum linear heat generation rate (MLHGR) at rated power $\leq 39\text{ kW/m}$
- Maximum cladding surface temperature at rated power $\leq 650\text{C}$ for Stainless Steel cladding
- Moderator temperature in water rods $\leq 384\text{C}$ (pseudo critical temperature at 25MPa)

Neutronic design criteria

- Positive water density reactivity coefficient (negative void reactivity coefficient)
- Core shutdown margin $\geq 1.0\%\Delta K/K$
### Fuel assembly (example)

<table>
<thead>
<tr>
<th>Design requirements</th>
<th>Solution</th>
</tr>
</thead>
<tbody>
<tr>
<td>Low flow rate per unit power (&lt; 1/8 of LWR) due to large $\Delta T$ of once-through system</td>
<td>Narrow gap between fuel rods to keep high mass flux</td>
</tr>
<tr>
<td>Thermal spectrum core</td>
<td>Many/Large water rods</td>
</tr>
<tr>
<td>Moderator temperature below pseudo-critical</td>
<td>Insulation of water rod wall</td>
</tr>
<tr>
<td>Reduction of thermal stress in water rod wall</td>
<td>Uniform fuel rod arrangement</td>
</tr>
<tr>
<td>Uniform moderation</td>
<td></td>
</tr>
</tbody>
</table>
Fuel enrichment

- Fuel enrichment is divided into two regions to prevent top axial power peak
- Average fuel enrichment 6.11wt%

(a) UO$_2$ fuel rod
(b) UO$_2$ + Gd$_2$O$_3$ fuel rod
Coolant flow scheme

Flow directions

<table>
<thead>
<tr>
<th></th>
<th>Coolant</th>
<th>Moderator</th>
</tr>
</thead>
<tbody>
<tr>
<td>Inner FA</td>
<td>Upward</td>
<td>Downward</td>
</tr>
<tr>
<td>Outer FA</td>
<td>Downward</td>
<td>Downward</td>
</tr>
</tbody>
</table>

To keep high average coolant outlet temperature

Kamei, et al., ICAPP'05, Paper 5527
3-D N-T Coupled Core Calculation

- T-H calculation based on single channel model
- Neutronic calculation; SRAC

Core consists of homogenized fuel elements

Single channel T-H model

- Coolant
- Water rod wall
- Pellet
- Cladding
- Moderator

Homogenized Fuel element

1/4 core

Fuel assembly

Single channel T-H analyses
Fuel load and reload pattern

- 120 FAs of 1\textsuperscript{st}, 2\textsuperscript{nd} and 3\textsuperscript{rd} cycle fuels and one 4\textsuperscript{th} cycle FA
- 3rd cycle FAs which have lowest reactivity are loaded at the peripheral region of the core to reduce the neutron leakage.

The low leakage core with high outlet temperature is made possible by downward flow cooling in peripheral FAs.

(a) 1\textsuperscript{st} \rightarrow 2\textsuperscript{nd} cycle
(b) 2\textsuperscript{nd} \rightarrow 3\textsuperscript{rd} cycle
(c) 3\textsuperscript{rd} \rightarrow 4\textsuperscript{th} cycle

\(\frac{1}{4}\) symmetric core
Coolant flow rate distribution

- Flow rate to each FA is adjusted by an inlet orifice
- 48 out of 121 FAs are cooled with descending flow

Relative coolant flow distribution (1/4 core)
Control rod patterns

- X : withdrawn rate (X/40)  Blank box : complete withdrawal (X=40)
- At the EOC, some CRs are slightly inserted to prevent a high axial power peak near the top of the core

<table>
<thead>
<tr>
<th>0.0GWd/t</th>
<th>0.22GWd/t</th>
<th>1.1GWd/t</th>
<th>2.2GWd/t</th>
<th>3.3GWd/t</th>
</tr>
</thead>
<tbody>
<tr>
<td>12</td>
<td>16</td>
<td>24</td>
<td>24</td>
<td>28</td>
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<td>32</td>
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<td>3216</td>
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</tbody>
</table>

<table>
<thead>
<tr>
<th>4.4GWd/t</th>
<th>5.5GWd/t</th>
<th>6.6GWd/t</th>
<th>7.7GWd/t</th>
<th>8.8GWd/t</th>
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</tr>
<tr>
<td>9.9GWd/t</td>
<td>11.0GWd/t</td>
<td>12.1GWd/t</td>
<td>13.2GWd/t</td>
<td>14.3GWd/t</td>
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<td>36</td>
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</tbody>
</table>
MLHGR and MCST

MLHGR and MCST are kept below 39 kW/m and 650°C throughout a cycle. Thermal design criteria are satisfied.

(a) MLHGR

(b) MCST
Water density reactivity coefficient and Shutdown margin

- Positive water density reactivity coefficient (Negative void reactivity coefficient)

- Shutdown margin is 1.27 %dk/k
  - One rod stuck
  - Cold and clean core

Neutronic design criteria are satisfied
### Super LWR characteristics summary

<table>
<thead>
<tr>
<th>Core</th>
<th>Super LWR</th>
</tr>
</thead>
<tbody>
<tr>
<td>Core pressure [MPa]</td>
<td>25</td>
</tr>
<tr>
<td>Core thermal/electrical power [MW]</td>
<td>2744/1200</td>
</tr>
<tr>
<td>Coolant inlet/outlet temperature [C]</td>
<td>280/500</td>
</tr>
<tr>
<td>Thermal efficiency [%]</td>
<td>43.8</td>
</tr>
<tr>
<td>Core flow rate [kg/s]</td>
<td>1418</td>
</tr>
<tr>
<td>Number of all FA/FA with descending flow cooling</td>
<td>121/48</td>
</tr>
<tr>
<td>Fuel enrichment bottom/top/average [wt%]</td>
<td>6.2/5.9/6.11</td>
</tr>
<tr>
<td>Active height/equivalent diameter [m]</td>
<td>4.2/3.73</td>
</tr>
<tr>
<td>FA average discharged burnup [GWd/t]</td>
<td>45</td>
</tr>
<tr>
<td>MLHGR/ALHGR [kW/m]</td>
<td>38.9/18.0</td>
</tr>
<tr>
<td>Average power density [kW/l]</td>
<td>59.9</td>
</tr>
<tr>
<td>Fuel rod diameter/Cladding thickness (material) [mm]</td>
<td>10.2/0.63 (Stainless Steel)</td>
</tr>
<tr>
<td>Thermal insulation thickness (material) [mm]</td>
<td>2.0 (ZrO₂)</td>
</tr>
</tbody>
</table>
Sub-channel analysis coupled with 3D core calculation
Reconstruction of pin power distributions

Core power distributions
(3-D core calculations)

Homogenized FA

Coupled subchannel analyses

Pin power distribution
$f$(burnup history, density, CR insertion)

Reconstructed pin power distribution

Height [m]

Normalized power

[Graph and diagrams depicting power distribution]
Statistical thermal design

• Taking uncertainties into evaluation of peak cladding temperature
Monte Carlo statistical procedure

Engineering uncertainty is evaluated as: \( k\sigma_T \)
\( k = 1.645 \) is to ensure 95/95 limit.

\[
\sigma_T^2 = \sigma_{PF}^2 + \sigma_C^2
\]
Peak Cladding Surface Temperature

Failure limit

Limit for design transients

Max. peak steady state condition

Nominal peak steady state condition

Nominal peak steady state condition (Homogenized FA)

Nominal steady state core average condition

Peak cladding surface temperature

Criterion: ? °C

740°C

(ΔT_4 = ? °C)

708°C

(ΔT_3 = 32°C)

650°C

(ΔT_2 = 58°C)

3-D core calculations

(ΔT_1 = 150°C)

Ave. outlet: 500°C

Plant safety analyses

Statistical thermal design

Subchannel analyses

ΔT_1 = 150°C

ΔT_2 = 58°C

ΔT_3 = 32°C

ΔT_4 = ? °C

708°C

740°C

Ave. outlet: 500°C

Limit for design transients
Safety
Depressurization induces core coolant flow of the once-through cycle reactor

Once-through system ⇒ Coolant flow induced in the core
Large water inventory of Top dome ⇒ In-vessel accumulator
Negative void reactivity ⇒ Power decreasing
Safety principle of Super LWR

- Keeping coolant inventory is not suitable due to no water level and large density change.

Safety principle is keeping **core coolant flow rate**.

<table>
<thead>
<tr>
<th></th>
<th>BWR</th>
<th>PWR</th>
<th>Super LWR</th>
</tr>
</thead>
<tbody>
<tr>
<td>Requirement</td>
<td>RPV inventory</td>
<td>PCS inventory</td>
<td>Core flow rate</td>
</tr>
<tr>
<td>Monitoring</td>
<td>RPV water</td>
<td>Pressurizer</td>
<td>Main coolant flow rate,</td>
</tr>
</tbody>
</table>
Plant and safety system
### Abnormal levels and actuations

<table>
<thead>
<tr>
<th>Condition</th>
<th>Description</th>
<th>Levels</th>
<th>Actions</th>
</tr>
</thead>
<tbody>
<tr>
<td>Flow rate low</td>
<td>⇔ Coolant flow from cold-leg</td>
<td>Level 1 (90%)*</td>
<td>Reactor scram</td>
</tr>
<tr>
<td></td>
<td></td>
<td>Level 2 (20%)*</td>
<td>AFS</td>
</tr>
<tr>
<td></td>
<td></td>
<td>Level 3 (6%)*</td>
<td>ADS/LPCI</td>
</tr>
<tr>
<td>Pressure high</td>
<td>⇔ Coolant outlet at hot-leg</td>
<td>Level 1 (26.0 MPa)</td>
<td>Reactor scram</td>
</tr>
<tr>
<td></td>
<td></td>
<td>Level 2 (26.2 MPa)</td>
<td>SRV</td>
</tr>
<tr>
<td>Pressure low</td>
<td>⇔ Valve opening, LOCA</td>
<td>Level 1 (24.0 MPa)</td>
<td>Reactor scram</td>
</tr>
<tr>
<td></td>
<td></td>
<td>Level 2 (23.5 MPa)</td>
<td>ADS/LPCI</td>
</tr>
</tbody>
</table>

*100% corresponds rated flow rate*
Safety system design

Capacity:
AFS          TD 3 units: 50kg/s/unit (4%)* at 25MPa
LPCI/RHR     MD 3 units: 300kg/s/unit (25%)* at 1MPa
SRV/ADS      8 units: 240kg/s/unit (20%)* at 25MPa

Configuration:

```
AFS            AFS
LPCI   LPCI
AFS            AFS
LPCI   LPCI
AFS            AFS
LPCI   LPCI
```

*100% corresponds to rated flow rate
Water rods mitigate loss-of-flow events.

Under loss-of-flow condition:
Heat conduction to water rods increases. → “Heat sink” effect
Water rods supply their inventory to fuel channels due to thermal expansion. → “Water source” effect
Alternative action is not necessary under ATWS conditions (Super LWR)

Analysis results for ATWS events without an alternative action

Loss of offsite power  Loss of turbine load without bypass  Uncontrolled CR withdrawal at normal operation
Good inherent safety characteristics of Super LWR

Why ATWS is mild?

1. Small power increase by valve closure.
   - flow stagnation mitigates density increase
   - no void collapse

2. Power decreases with core flow rate due to density feedback.

Good ATWS behavior without alternative action inserting negative reactivity
Summary of safety analysis results

### Transients

1. Partial loss of reactor coolant flow
2. Loss of offsite power
3. Loss of turbine load
4. Isolation of main steam line
5. Pressure control system failure
6. Loss of feedwater heating
7. Inadvertent startup of AFS
8. Reactor coolant flow control system failure
9. Uncontrolled CR withdrawal at normal operation
10. Uncontrolled CR withdrawal at startup

### Accidents

1. Total loss of reactor coolant flow
2. Reactor coolant pump seizure
3. CR ejection at full power
4. CR ejection at hot standby
5. Large LOCA
6. Small LOCA
ΔMSCT for abnormal events

Failure limit for accident

Criterion for accident

Failure limit for transient

Criterion for transients

Maximum peak steady state condition

Nominal steady state core average condition

Margin

3-D core design
Subchannel analysis
Statistical thermal design

380°C
Small LOCA
330°C
Large LOCA

ATWS

520°C

220°C
Loss-of-flow
250°C

110°C
Transient

60°C

850°C

240°C

740°C

Ave. outlet: 500°C

1260°C
Summary of safety characteristics of Super LWR

- Core cooling by depressurization
- Top dome and water rods serve as an “in-vessel accumulator”
- Loss of flow mitigated by water rods
- Short period of high cladding temperature at transients
- Mild behavior at transients, accidents and ATWS
- Simple safety principle (keeping flow rate) due to once-through cooling cycle
Super fast reactor

Tight fuel lattice

Supercritical-pressure light water cooled fast reactor

Same plant system as Super LWR

Plant system of Super LWR and Super FR
Advantages of Super Fast Reactor

Low reactor coolant flow rate due to high enthalpy rise
High head pumps of the once-through direct cycle plant
- Compatible with tight fuel lattice core of Super FR, a light water cooled fast reactor
- No pumping power increase and instability problems of high conversion LWR

Same plant system as Super LWR, the thermal reactor
Fast reactors have higher power densities than thermal reactors due to no moderator necessary.
- Making capital cost of Super FR lower than LWRs
  (Capital cost; Super FR< Super LWR< LWRs)
R&D of Super Fast Reactor

University of Tokyo, JAEA, Kyusyu Univ. and TEPCO entrusted by MEXT as one of the Japanese NERI, 5 years, Dec. 2005-March 2010

Leader: Y. Oka (University of Tokyo)
Fuel and Core (example)

- MOX fuel with SS cladding (Fuel rod analysis)
- Core design: 3-D N-TH coupled core burn-up calculation, subchannel analysis

ZrH₂ layer (for coolant void reactivity reduction)
Core Structure and Plant Control and Safety

Core characteristics

<table>
<thead>
<tr>
<th>(700 MWe)</th>
<th>Core 1</th>
<th>Core 2</th>
</tr>
</thead>
<tbody>
<tr>
<td>Fuel</td>
<td></td>
<td></td>
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<tr>
<td>Fuel (Seed/Blanket)</td>
<td>MOX/dep.UO₂</td>
<td></td>
</tr>
<tr>
<td>Fuel pellet density</td>
<td>95%TD</td>
<td></td>
</tr>
<tr>
<td>Rod OD [mm]</td>
<td>7.0</td>
<td>5.5</td>
</tr>
<tr>
<td>Pitch/OD</td>
<td>1.16</td>
<td>1.19</td>
</tr>
<tr>
<td>Cladding Material</td>
<td>SUS304</td>
<td></td>
</tr>
<tr>
<td>Thickness [mm]</td>
<td>0.43</td>
<td>0.4</td>
</tr>
<tr>
<td>Effective heating length [cm]</td>
<td>300</td>
<td>200</td>
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<tr>
<td>Core</td>
<td></td>
<td></td>
</tr>
<tr>
<td>No. of seed fuel assemblies</td>
<td>126</td>
<td>162</td>
</tr>
<tr>
<td>No. of blanket fuel assemblies</td>
<td>73</td>
<td></td>
</tr>
<tr>
<td>Pitch of FA</td>
<td>14.2</td>
<td>11.6</td>
</tr>
</tbody>
</table>

RPV and the coolant flow
Thermal hydraulic experiments

Kyusyu University; HCFC22 (Freon) JAEA Naka-lab; Supercritical Water

(1) single tube and 7-rod bundle
(2) critical heat flux near critical pressure
(3) critical flow and condensation
Experimental results; HCFC22 (Freon)

Wall temperature and heat transfer coefficient of 7-rod bundle test

Maximum wall temperature at critical heat flux
Need for Developing High Creep Strength Clad

- Max. stress on clad at peak T (700-750°C): 70-100MPa
  - Exceed creep strength of SS for LWR (SUS316L)
  - Advanced SS for LMFBR (PNC1520) almost satisfies the requirement but SCC susceptibility, corrosion and neutron absorption properties need to be improved

- High creep strength clad needs to be developed for Super FR

**Fuel rod analysis results (Super LWR)**

**Creep rupture strength of advanced SS**
Developed Good Thermal Insulator
Yttria stabilized zirconia (YSZ)

- Large $\Delta T$ (~250°C)
- Thermal insulator is required for:
  - reduction of thermal stress
  - maintaining coolant temperature

Graph showing differences in thermal stress with and without insulation.

Thermal stress on the wall:

- $\sigma < (1/2 \times Su)$
- $(1/2 \times Su) < \sigma < Su$
- $\sigma > Su$

$Su$: tensile strength

~1/20 of Zirconia
Elution of structural material in SC water

Elution decreases with temperature (at 25 MPa)

<table>
<thead>
<tr>
<th>Temperature</th>
<th>Absolute value (g/m²)</th>
<th>Relative value (Normalized at 300 °C)</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Deaerated</td>
<td>200 ppb O₂</td>
</tr>
<tr>
<td>300 °C</td>
<td>0.203</td>
<td>0.102</td>
</tr>
<tr>
<td>400 °C</td>
<td>0.0098</td>
<td>0.0085</td>
</tr>
<tr>
<td>450 °C</td>
<td>0.0045</td>
<td>0.0045</td>
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<tr>
<td>550 °C</td>
<td>&lt; 0.002</td>
<td>0.0062</td>
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</table>

Elution depends on O₂

Experimental devices
SCWR R&D in the world

- Japan: University of Tokyo; Super LWR concept (since 1989), Super FR R&D (2005-2010). Toshiba; SCPR R&D, Consortium for GIF R&D

- China; Shanghai JTU (8 organizations) SCWR R&D (2007-2012), CGNPC announced the plan of constructing an experimental SCWR from 2016.

- EU; HPLWR phase 1 (FZK, 2000-2), phase 2 (FZK, 10 organizations of 8 countries 2006-9), planning of phase 3

- Canada: pressure tube type SCWR R&D: NSERC/NRCan/AECL-Universities program

- Korea: thermal hydraulics (KEARI)

- Russia: SC thermal hydraulic loops of IPPE, WS at NIKIET in 2008

- USA: TH and materials at Univ. Wisconsin and Univ. Michigan (finished)

- GIF SCWR OECD/NEA (Canada, EU, Japan and other countries) phase 2

- IAEA: CRP of supercritical thermal hydraulics

SCR symposiums; 1st and 2nd at University of Tokyo in 2000 and 2003, 3rd at Shanghai JTU in 2007 and 4th in Heidelberg in 2009
Thank you
Control and start up
Plant control system
Sliding Pressure Startup System

Nuclear heating starts at subcritical pressure.

Water separator is installed on a bypass line.
Calculation Model for Sliding Pressure Startup
Sliding Pressure Startup Procedure

1. Start of Nuclear Heating
2. Turbine Startup
3. Pressurization to 25 MPa
4. Line Switching
5. Temperature Raising
6. Power Raising
Pressurization phase

- Sliding pressure startup system (nuclear heating starts at subcritical pressure)
- Clad temperature increase in pressurization phase is due to BT
- Power / flow region is limited by CHF
- CHF may be increased by grid spacers
Sliding Pressure Startup Curve

(By Thermal-Hydraulic Analysis)

- Reactor power
- Feedwater temperature
- Feedwater flow rate
- Core outlet temperature
- Main steam pressure
- Reactor power

Temperature (°C) / Pressure (bar)

Ratio (%)

Start of feedwater pump
Start of nuclear heating
Turbine startup
Pressurization
Line switching
Temperature raising
Power raising
Linear Stability Analysis (for Supercritical Pressure)
Thermal-Hydraulic Stability (Supercritical pressure)
Coupled Neutronic Thermal-Hydraulic Stability (Supercritical pressure)
Decay ratio increases with power to flow rate ratio.
Stability Analysis during Sliding-Pressure Startup

• Coupled neutronic thermal-hydraulic stability analysis
• Thermal-hydraulic stability analysis
• Thermal-hydraulic analysis
• Sliding pressure startup procedures
Sliding pressure startup curve
(Thermal criteria only)

Sliding pressure startup curve
(Both Thermal and Stability criteria)
Scope of studies and Computer codes

1. Fuel and core
   Single channel thermal hydraulics (SPROD), 3D coupled core neutronic/thermal-hydraulic (SRAC-SPROD), Coupled sub-channel analysis, Statistical thermal design method, Fuel rod behavior (FEMAXI-6), Data base of heat transfer coefficients of supercritical water

2. Plant system; Plant heat balance and thermal efficiency

3. Plant control

4. Safety; Transient and accident analysis at supercritical-and subcritical pressure, ATWS analysis, LOCA analysis (SCRELA)

5. Start-up (sliding-pressure and constant-pressure)

6. Stability (TH and core stabilities at supercritical and subcritical-pressure)

7. Probabilistic safety assessment
Economic potential
Comparison of containments