Development of Supercritical-water Cooled Power Reactor
Conducted by a Japanese Joint Team

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The supercritical-water cooled power reactor, which operates above the thermodynamic critical point of water (374°C, 22.1 MPa), gains advantage over the current generation light water reactors (LWRs) on three points: high thermal efficiency exceeding 40%, compact and simple major components due to high-enthalpy single phase working fluid, and small R&D cost potential resulting from utilization of matured LWR technologies as well as matured supercritical-pressure fossil power technologies. To realize these advantages, a Japanese joint team consisting of Univ. of Tokyo, Kyushu Univ., Hokkaido Univ., Hitachi, Ltd, and Toshiba Co. launched a four-year development project of SCPR funded by the Institute of Applied Energy, Ministry of Economy, Trade and Industry, Japan in the fiscal year 2000. The objective of this project is to provide technical information essential to SCPR technology demonstration through concentrating three sub-themes: plant conceptual design, thermohydraulics, and material & water chemistry. This paper overviews the project and reports the progress up to fiscal year 2002.

KEYWORDS: Supercritical water cooled reactor, SCPR, development project

I. Introduction

The Supercritical-water Cooled Power Reactor (SCPR) operate above the thermodynamic critical point of water (374°C, 22.1 MPa). The use of supercritical pressure (SCP) water as reactor core coolant potentially brings advantages over the current generation light water reactors (LWRs).

- The estimated thermal efficiency is exceeding 40% due to high-pressure (HP), high-temperature (HT) steam at the turbine inlet.
- Heat exchangers (HXs) and turbines are compact due to high-enthalpy coolant. Steam separation systems, re-circulation systems or steam generators are unnecessary because no phase change occurs in supercritical regime.
- The R&D cost and duration are potentially minimized because SCPR technology is based on matured LWR as well as SCP fossil power technologies.

To make these advantages viable, University of Tokyo, Kyushu University, Hokkaido University, Hitachi, Ltd, and Toshiba Co. organize a joint team under the frame of Innovative and Viable Nuclear Energy Technology Development Project funded by Ministry of Economy, Trade and Industry (METI), Japan in fiscal year (FY) 2000.

This SCPR development project is positioned in the viability step followed by the demonstration and the prototype steps of SCPR deployment scheme (Fig.1). The viable step follows the basic step which has been conducted mainly by Univ. of Tokyo since late 1980s. The demonstration step are costly and time consuming because they need to simulate the SCPR conditions as closely as possible including neutron irradiation, high-temperature SCP water and geometric sizes. To minimize the demonstration step, the viable step plays an important role.

Major R&D areas suggested from the present project are essential to achieve an early, rational SCPR deployment. Those R&D areas are:

- HP, HT structure development including components as well as elements/parts unique to SCPR
- Understanding supercritical water (SCW) chemistry and relevant system development such as chemical control system and purification system
- Extended material R&D and reactor irradiation tests of materials under HT, SCP condition
- Large thermohydraulic (T/H) tests simulating the SCPR core under HT, SCP water condition

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Fig. 1 The joint team is concentrating three sub-themes: plant conceptual design, thermo-hydraulics, and material & water chemistry. The active duration of this project is four years; the major R&D items as well as schedules are shown in Fig. 1. The major targets of each sub-theme are as follows:

- **Plant conceptual design;** Optimization as well as rationalization as a nuclear system is being performed through design study of key components/systems essential to the SCPR concept. The target is large reduction in amount of volume of such components/systems compared with the conventional LWRs while achieving high thermal efficiency without sacrificing the level of safety.

- **Thermo-hydraulics;** Heat transfer characteristics of supercritical water as a coolant of the SCPR are examined experimentally and analytically focusing on minimizing the uncertainty. The experiments are being performed using substitute fluid (HCFC22) for water in a fossil boiler test facility at Kyushu Univ., Japan. The experimental results are being incorporated in LWR analytical tools together with an extended steam/HCFCC22 table.

- **Material & water chemistry;** Promising material candidates for fuel cladding and internals of the SCPR are being chosen from materials commercially available mainly through mechanical test, corrosion test, and simulated irradiation test under the SCPR condition considering water chemistry. In particular, stress corrosion cracking (SCC) susceptibility of the materials is being investigated as well as general corrosion and swelling characteristics.

## II. Progress Report

Major progress as of FY 2002 is reported in this paper.

### 1. Plant Conceptual Design

We first reviewed an SCPR concept (SCLWR-H) proposed by Univ. of Tokyo\textsuperscript{234} to identify the areas that need further optimization. The whole plant image is depicted in Fig. 2. At present, we revise the concept from nuclear manufacturer's point of view.

The core is cooled by SCP water at 25 MPa, which was chosen from the matured SCP fossil power technologies. Along the SCLWR-H (1000 MWe) core, coolant is heated from 280 to 508°C at 25 MPa. The core flow rate is 1.2 t/h.

The reactor core consists of relatively large 96 square fuel assemblies (about 300×300 mm) as shown in Fig. 3. The fuel assembly consists of 301 fuel rods and 36 square water rods. The fuel rod mainly contains UO\textsubscript{2} like LWRs in nickel alloy cladings\textsuperscript{1}. The water rods compensate small moderation resulted from the large change in coolant (water) density along the core as well as to guide the control rod insertion into the core.

In the water rods, cold coolant descends from the RPV upper region through the control rod guide tubes located in the hot plenum. Along the coolant channels between the fuel elements, the coolant’s temperature increases and at 280°C, it enters into the hot plenum from the bottom of the RPV. The hot coolant at 508°C is delivered to the hot plenum with the pressurizer working as a closed system.

Fig. 1 An SCPR deployment scheme

\textsuperscript{1} The cladding material is subject to change in the progress of this R&D.
rods or between the fuel rod and the water rod in the channel box, coolant ascends from the core bottom to the hot plenum. This arrangement facilitates neutron moderation by cold water in the water rods and core cooling by water along the coolant channels.

According to our preliminary evaluation, U235 enrichment for the equilibrium core exceeds 6% to achieve discharge burnup similar to current LWR fuels. This relatively high enrichment is mainly resulted by relatively high neutron capture of the structure materials (non-Zircaloy) including the fuel cladding and the channel boxes. To ensure adequate shut down margin during the entire operation cycle, gadolinia (burnable poison) is inserted in the fuel.

The reactor is shut down by inserting cluster type control rods into all fuel assemblies from the top of core. We investigated the requisites for the control rods to achieve cold shutdown. The result shows that the radius of the control rods should be greater than 0.7 cm and that 16 or more control rods are necessary per fuel assembly.

The reactor pressure vessel (RPV) is simple and similar to that of PWRs (Fig.4). There is no active primary component used inside the RPV. High-pressure cold coolant is fed by the feedwater pumps like BWRs. No coolant circulation nor re-circulation pump is used. No steam separation systems are necessary.

We roughly designed such a simple RPV for SCPR under the design principle: maximum utilization of existing technologies as well as existing design standards. Major measures to satisfy the design principle are following:

- Minimization of the RPV radius
- Cooling the pressure boundaries (RPV wall, main steam lines, vessel flange, etc.)
- Protection of the pressure boundaries from radiation from the core
- Avoiding influence from rapid, large temperature change of the coolant in case of transients and accidents

The result suggests that the inner diameter is about 4.3 m; the total height is 15 m; the thickness of the shell is about 0.39 m. Base material of the RPV is chosen from the existing materials: SFVQ1A (ASTM A508) with stainless steel lining. The inner surface of the RPV is cooled by feedwater (280°C); The cold feedwater path serves as radiation shield. The hot plenum is isolated from the RPV wall; The hot plenum wall is thin and covered with thermal insulator. The control rod guide tubes (CRGTs) are also covered with thermal insulator along the hot plenum to avoid heat-up of the moderator (coolant) in them. The CRGTs need special seal mechanism to avoid coolant leakage from the fuel channels as well as from the top boundary of the hot plenum.

The wet type reinforced concrete containment vessel (RCCV) adopted in SCLWR-H. The RCCV volume is...
expected smaller than ABWR’s because of following two reasons. The total coolant enthalpy in SCLWR-H RPV is smaller than that in ABWR RPV so that the wet well volume in the RCCV would be smaller enough to condensate steam discharged from the RPV during a loss of coolant accident (LOCA). SCLWR-H adopts top entry type control rod drives (CRDs) like PWRs, which results in volume reduction of lower dry well for CRD maintenance.

The steam cycle of SCPR employs a two-stage re-heating and eight-stage re-generative system (Fig.5), whose thermal efficiency was preliminarily estimated at reaching 42%. The volumetric capacity of the turbines as well as the feedwater heaters used is much smaller than those used in conventional LWR plants because of the small volumetric flow rate per electricity production resulted from high enthalpy/pressure of SCP coolant.

![A conceptual steam cycle](image)

**Fig.5** A conceptual steam cycle

The turbine has one dual-exhaust high-pressure (HP) section, one dual-exhaust intermediate-pressure (IP) section and two dual-exhaust low-pressure (LP) sections. The low-pressure turbine is a four flow, tandem compound, 1500 rpm (for 50 Hz) machine with 52 inches last stage blades. The cycle uses moisture separator reheaters (MSH) with two-stage reheat. This cycle produce about 950 MWe with a thermal reactor output of 2273 MW.

Steam bled from the HP, IP and LP sections is conveyed to high (HP-FWH) and low-pressure feedwater heaters (LP-FWH), respectively. The feedwater heaters provide a final feedwater temperature of 280°C at the rated condition.

The condensate collected in the condensers is pumped by low-pressure condensate pumps (LP-CP) and passes an air ejector, a grand steam condenser and a set of demineralizer and filters. Being pumped by high-pressure condensate pumps (HP-CP), the condensate passes the LP-FWHs including deaerator. Then the condensate up to supercritical pressure of about 27 MPa and passes HP-FWHs to be heated.

It should be noted that all these steam cycle technologies are matured and that little R&D is necessary.

2. Thermohydraulics

Although thermohydraulics of SCP water has been extensively studied for fossil boilers, thermohydraulic conditions of SCPR core are different from those of fossil boilers in the following three areas (Fig.6).

- Small hydraulic diameter vs. large diameter
- Rod outer surface cooling vs. tube inner surface
- Rod bundle with spacer/grid vs. single tube with heat transfer enhancer

Besides, "heat transfer deterioration (abnormal heat transfer)", which is the unique phenomena observed under very limited flow conditions, may cause abnormal, but milder heat-up on the rod surface unlike dry-out at subcritical pressure.

To minimize the uncertainty brought from those differences including heat transfer deterioration, we are carrying out experimental study as well as analytical study. The objectives of the study are to find heat transfer correlations presenting good agreement with the HCFC22 experimental results, to develop correlations predicting heat transfer of SCP water, and to incorporate the relations into thermohydraulic design codes for SCPR development. A series of tests are being performed with three types of test section: single tube, single rod and bundle mounted on a SCP HCFC22 loop (Fig.7) at Kyushu University, Japan.

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2 Because of relatively large steam flow rate compared with existing SCP fossil plants, high-speed (3000 rpm for 50 Hz) turbines are not currently available.
This HCFC22 loop has been used to develop boiler tubes for SCP fossil plants. HCFC22 is often used as a substitute for water because its pseudo-critical pressure (5 MPa) is far less than that of water (22 MPa).

The range of HCFC22 test conditions is determined to simulate SCLWR-H thermohydraulic conditions such as system pressure, mass flux and heat flux (Table 1). The system pressure is 110% of the critical pressure of HCFC22 to create an SCP condition equivalent to the SCLWR-H condition. The range of mass flux is chosen so that Reynolds number (flow character) under the test conditions is equivalent to the SCLWR-H condition including flow transient condition. The range of heat flux is chosen to simulate up to the rated power condition of SCLWR-H as well as heat transfer deterioration caused by flow transient events.

![Diagram](image)

**Table 1** Major thermohydraulic condition

<table>
<thead>
<tr>
<th>Item</th>
<th>Unit</th>
<th>Experiment</th>
<th>SCLWR-H</th>
</tr>
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<tbody>
<tr>
<td>Fluid</td>
<td>HCFC22</td>
<td>Water</td>
<td></td>
</tr>
<tr>
<td>$P_c$</td>
<td>MPa</td>
<td>5</td>
<td>22</td>
</tr>
<tr>
<td>$T_c$</td>
<td>K</td>
<td>369</td>
<td>647</td>
</tr>
<tr>
<td>System pressure</td>
<td>MPa</td>
<td>5.5</td>
<td>25</td>
</tr>
<tr>
<td>Mass flux</td>
<td>kg/m²/s</td>
<td>200-2000</td>
<td>1200 (rated)</td>
</tr>
<tr>
<td>Heat flux</td>
<td>kW/m²</td>
<td>10-190</td>
<td>620 (rated)</td>
</tr>
<tr>
<td>Geometry</td>
<td>Tube/rod/bundle</td>
<td>Rod bundle</td>
<td></td>
</tr>
<tr>
<td>Heated length</td>
<td>m</td>
<td>2</td>
<td>3.7</td>
</tr>
<tr>
<td>Hydraulic diameter</td>
<td>mm</td>
<td>4.4</td>
<td>4.4</td>
</tr>
<tr>
<td>Non-dimensional</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>$Re$</td>
<td>$\times 10^4$</td>
<td>2.5 (at inlet)</td>
<td>5 (rated, at inlet)</td>
</tr>
<tr>
<td>$Pr$</td>
<td></td>
<td>2.3 (at inlet)</td>
<td>0.8 (at inlet)</td>
</tr>
</tbody>
</table>

As of FY2002, the small hydraulic diameter tests using the single tube test section have been completed to identify difference in heat transfer characteristics between small (4.4 mm) and large hydraulic diameter (9 mm or larger).

Heat transfer coefficient (HTC) of the small hydraulic flow as well as larger hydraulic flows increases with bulk fluid enthalpy and reaches the maximum at near pseudo-critical enthalpy ($h_{pc}$) (Fig.8). HTC increases with decrease of hydraulic diameter. Most HTCs at below $h_{pc}$ are generally predicted by Dittus-Boelter's equation. Those facts suggest that heat transfer characteristics for the small hydraulic tube flow is not unique compared with those for the larger tube flow.

From a lot of heat transfer correlations (Nu numbers) including Dittus-Boelter's equation for single phase fluids as well as SCP fluids, we are seeking for the best correlation that can sufficiently predict the HCFC22 test results over a wide range of thermohydraulic conditions. Judging from the comparison between test and analytical results as of FY2002, Watts' correlation shows the best predictability among them for normal heat transfer regime, mass flux from 200 to 2000 (kg/m²/s), heat flux from 2 to 190 kW/m² (Fig.9). It should be noted, however, that HTC in the abnormal heat transfer regime or low mass-flux and high heat-flux area is different from those and that they are currently under investigation.
3. Material & Water Chemistry

Zirconium alloys (Zircaloy), which are widely used for LWR fuel claddings, are difficult to use for SCPR mainly because of their poor mechanical as well as corrosion properties under high-temperature water condition (more than 400°C). To principally find Zircaloy alternatives, a material screening is performed through simulated irradiation tests, corrosion tests and mechanical tests. Furthermore, stress corrosion cracking (SCC) susceptibility is being investigated for potential structure materials being used in SCPR. Influences of water chemistry on such material properties are also examined to find preferable water conditions as well as to develop rational water chemistry controlling methods.

The process of material screening consists of three steps: candidate selection from commercially available materials, material test and material application (Fig.10). The goal of this process is to find promising materials worth further examining under neutron radiation field, which is planned after this development project.

In the first step, candidates were chosen from materials used in three industrial fields: fossil power, waste processing and nuclear energy. The major candidate materials chosen are listed in Table 2.

<table>
<thead>
<tr>
<th>Table 2</th>
<th>Materials chosen from the existing industries</th>
</tr>
</thead>
<tbody>
<tr>
<td>Stainless steel</td>
<td>Austenitic</td>
</tr>
<tr>
<td></td>
<td>Ferritic</td>
</tr>
<tr>
<td>Nickel alloy</td>
<td>Alloy690</td>
</tr>
<tr>
<td></td>
<td>Alloy825</td>
</tr>
<tr>
<td></td>
<td>Alloy600</td>
</tr>
<tr>
<td>Titanium alloy</td>
<td>Ti-3Al-2.5V</td>
</tr>
<tr>
<td></td>
<td>Ti-15V-3Al-3Sn-3Cr</td>
</tr>
</tbody>
</table>

In the second step, simulated irradiation test, corrosion test and mechanical test are being performed for the candidate materials to obtain data under the condition partly simulating the SCPR. Irradiation tests are performed with high-energy electron beam at the temperature range expected for the SCPR condition. General corrosion tests and SCC tests are performed considering SCPR water chemistry. Mechanical or tensile tests are performed at the
temperature range estimated in SCPR to evaluate high-temperature mechanical properties.

In the third step, the test results are fed back to improve material performance by trace element additive, fine grainning and so on. The improved materials are being tested again. Finally, the promising material candidates will be proposed for further investigation.

Simulated irradiation tests up to five displacement per atom (dpa) are performed at 290, 450 and 550°C with electron beam of 1 MeV at Hokkaido University. After the irradiation, void formation is observed with using a high-energy transmission electron microscope (TEM) to evaluate swelling performance of the materials chosen in STEP 1. In general, some void formations are observed on the stainless steels; on the nickel base alloys, no void formation was observed. Those swelling behaviors are summarized on Fig. 11. The other materials are being investigated.

![Fig. 11 Swelling performance (5dpa, 5 MeV electron)](image)

Corrosion and SCC tests are being performed in the test section of an SCP water loop (Fig. 12)\(^1\). The test section and associated loop were assembled to simulate the SCPR condition up to 30 MPa, 600°C. Water chemistry condition is set up for the corrosion tests using the similar way adopted in LWRs: purification and oxygen concentration control.

General corrosion and SCC susceptibility are measured as follows. Oxide film on the specimens is analyzed in terms of thickness, morphology, chemical composition and chemical form to understand corrosion behavior under SCP condition. SCC susceptibility is being examined by two means: double U-bend tests and slow strain rate tests (SSRT). The double U-bend specimens are cut along the longitudinal centerline and crack depth is measured in the cross section after SCP water exposure. The test specimens for each test are depicted on Fig. 12.

During SSRT, stress and strain are measured continuously for SSRT specimens. After SCP water exposure, fracture surface of the specimens is inspected with a scanning electron microscope (SEM) to identify fracture mode. SCC area ratio on the fracture surface is measured to evaluate relative SCC growth rate or inter granular SCC (IGSCC).

IGSCC ratio of two kind of stainless steel was first evaluated from the observation after the SSRT. The stainless steel specimens were exposed heat treatment (620°C, 24 hours) before the test. The strain rate was set at \(4 \times 10^{-7} /\text{s}\), the temperature range: 290-550°C and the pressure: 25 MPa in high purity water with 8 ppm dissolved oxygen (DO). The IGSCC area ratio in the fracture surface was measured to evaluate relative SCC susceptibility.

![Fig. 12 Corrosion and SCC test loop, test specimens](image)
The result shows that the IGSCC ratio of the sensitized SUS304 decreases with increasing temperature and bottoms out at zero at 400°C; the IGSCC ratio of the thermal-treated SUS316L has bottomed at zero from low to high temperature (Fig. 13). The mechanism of this IGSCC behavior is under investigation. It seems that density of water as well as inorganic solubility influences IGSCC as shown in Fig. 13 because both rapidly decrease at around pseudo-critical temperature as the IGSCC ratio dose. The other materials are being studied.

Mechanical tests at 550°C in air with strain rate of 5x10⁻³/s are conducted to evaluate the mechanical integrity including 0.2% yield stress, tensile stress, total elongation. Overall, nickel base alloys possess higher strength than stainless steels. Among stainless steels, austenitic steels are generally better than ferritic steels in the mechanical properties. Some titanium base alloys possess high strength; the others do not.

Because water chemistry influences corrosion, SCC as well as corrosion product (CP) transport in the coolant, understanding of SCP water chemistry is important. It may be indispensable to develop SCP water purification systems to reduce radioactive exposure from the coolant containing radioactive CPs generated in the core. Although less is understood for SCP water chemistry under radiation field, the steam cycle of SCP employs a two-stage re-heating and eight-stage re-regenerative system. The thermal efficiency was preliminarily estimated at reaching 42%.

To minimize the uncertainty in heat transfer characteristics of SCP fluid, we are carrying out experimental/analytical thermohydraulic study using SCP HCFC22. As of FY2002, the small hydraulic diameter tests simulating SCP coolant channel have been completed and analyzed by using different heat transfer correlations proposed in the past studies. The heat transfer characteristics for the small hydraulic tube flow is not unique compared with those for the larger tube flow. So far Watts' correlation shows the best predictability for normal heat transfer regime among the heat transfer correlations for wide range of mass flux from 400 to 2000 kg/m²/s.

III. Summary

Operating above the thermodynamic critical point of water, the SCPR surpasses the current generation LWRs in thermal efficiency as well as in compactness/simplicity of design. Little R&D is necessary because the SCPR technologies are based on the matured LWR and SCP fossil power technologies. To make these advantages viable, a Japanese joint team started a development project of SCPR with a national fund in the fiscal year 2000.

Major R&D areas succeeding the present project were discussed. High-temperature, high-pressure structure development, understanding SCP water chemistry and relevant system development, extended material R&D and reactor irradiation tests, and large thermohydraulic tests simulating the SCPR core are important to achieve an early, rational SCPR deployment. The authors hope that the Generation-IV activity accelerates those R&D.

Plant conceptual design studies were carried out on the SCPR core, RPV and the BOP based on the existing concept proposed by Univ. of Tokyo. We investigated the requisites for the control rods to achieve cold shutdown. We roughly designed a simple RPV for SCPR under the design principle: maximum utilization of existing technologies as well as existing design standards. The steam cycle of SCPR employs a two-stage re-heating and eight-stage re-regenerative system. The thermal efficiency was preliminarily estimated at reaching 42%.

Fig. 13 IGSCC ratio of the sensitized SUS304 and thermal-treated SUS316L
To principally find Zircaloy alternatives, a material screening is being performed. Some promising materials will be chosen from electron irradiation tests, SCC tests and mechanical tests at high temperatures. High energy electron irradiation tests up to 5 dpa were performed at 290, 450 and 550°C. Nickel alloys generally show better swelling behavior than stainless steels. IGSCC ratio of the sensitized SUS304 decreases with increasing temperature and bottoms out at zero at 400°C. That of thermal-treated SUS316L has been zero from low to high temperature. Mechanical tests at 550°C in air showed that nickel base alloys possess higher strength than stainless steels. Some titanium base alloys possess high strength; the others do not.

Nomenclature

<table>
<thead>
<tr>
<th>Symbol</th>
<th>Description</th>
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<tbody>
<tr>
<td>Con</td>
<td>Electric conductivity</td>
</tr>
<tr>
<td>DO</td>
<td>Dissolved oxygen concentration</td>
</tr>
<tr>
<td>G</td>
<td>Mass flux</td>
</tr>
<tr>
<td>h_b</td>
<td>Bulk fluid enthalpy</td>
</tr>
<tr>
<td>P</td>
<td>Pressure</td>
</tr>
<tr>
<td>P_pC</td>
<td>Pseudo-critical pressure</td>
</tr>
<tr>
<td>q</td>
<td>Heat flux</td>
</tr>
<tr>
<td>HCFC22</td>
<td>Hydrochlorofluorocarbon -22 (CHClF₂)</td>
</tr>
<tr>
<td>SS</td>
<td>Stainless steel</td>
</tr>
<tr>
<td>T_pC</td>
<td>Pseudo-critical temperature</td>
</tr>
<tr>
<td>α</td>
<td>Heat transfer coefficient</td>
</tr>
<tr>
<td>ε</td>
<td>Strain rate</td>
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</table>

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