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Fuel, Core Design and Subchannel Analysis of a Superfast Reactor

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A compact supercritical water-cooled fast reactor (superfast reactor) core with a power of 700 MWe is designed by using a three-dimensional neutronics thermal-hydraulic coupled method. The core consists of 126 seed assemblies and 73 blanket assemblies. In the seed assemblies, 251 fuel rods, consisting of MOX pellets, stainless steel (SUS304) cladding, and fission gas plenum are arranged into a tight triangle lattice along with 19 guide tubes for control rods and instrumentation. A zirconium hydride (ZrH) layer is employed in the blanket assemblies to reduce void reactivity. The results of the coupling three-dimensional neutronics and thermal hydraulic calculations show that this core has a high power density of 158.8 W/cm^3 with a maximum linear heat generation rate (MLHGR) less than 39 kW/m, that an average coolant outlet temperature of 500°C is achieved with a maximum cladding surface temperature (MCST) less than 650°C, and that void reactivity coefficients are negative throughout the cycle. Since the thermal-hydraulic part of the core design is based on single-channel analyses, subchannel analyses are also performed on all the seed assemblies to clarify the influence of cross-flow.

KEYWORDS: supercritical pressure, fast reactor, fuel design, core design, subchannel analysis

I. Introduction

The concept of a supercritical water-cooled reactor (SCWR) has been well developed at the University of Tokyo since 1989.[1)] At present, research activities on the SCWR are ongoing worldwide. The SCWR concept was selected as one of the candidates of the generation IV reactor concepts. The SCWR is characterized by

(1) high enthalpy rise and low flow rate in the core, which is very effective for reducing the main pump power,
(2) once-through direct cycle, which makes the reactor system compact and simple,
(3) high main steam enthalpy, which makes the turbine system compact and the thermal efficiency high,
(4) utilizations of the current well-developed technologies of fossil-fired power plants (FPPs) and light water reactors (LWRs),
(5) single-phase coolant without boiling transition or dry-out phenomenon, and
(6) being available for both thermal and fast neutron spectrum cores with the same plant system.

The fast spectrum version of the SCWR not only inherits all the advantages above but also has some additional advantages, *i.e.*, no need of neutron moderation, high power density, and compact core design compared with a thermal spectrum SCWR. The advanced fuel cycle of the fast spectrum SCWR, which can effectively utilize the spent fuel of LWRs in the form of MOX fuel, also reduces the risk of Pu proliferation. Moreover, it has potential for transmutation of minor actinides (MAs) and long-lived fission products (LLFPs), or conversion of fertile fuel.

In the past studies of the SCWR, considerable effort has been given to its thermal spectrum version, called the SCLWR or the super-LWR.[2–4)] Fuel and core design with a three-dimensional neutronics thermal-hydraulic coupled method has been well conducted recently. However, past studies on the core design of the fast spectrum SCWR were mainly based on a two-dimensional R-Z approximation[5,6)] or Monte Carlo method.[7)] Recently, Yoo *et al.*[8)] have developed a three-dimensional neutronics thermal-hydraulic coupled method for the core design of the fast spectrum SCWR. Based on this method, a 1,000 MWe reactor was designed with MOX fuel and stainless steel cladding. A zirconium hydride (ZrH) layer was employed to reduce void reactivity.

A research program, named "Research and Development of the Superfast Reactor," was entrusted by the Ministry of Education, Culture, Sports, Science and Technology (MEXT) in December 2005 as one of the research programs of Japanese Nuclear Energy Research Initiative (NERI).[9)] The superfast reactor is a fast spectrum SCWR developed in this particular program. This program consists of a very wide range of topics including fuel and core design, control

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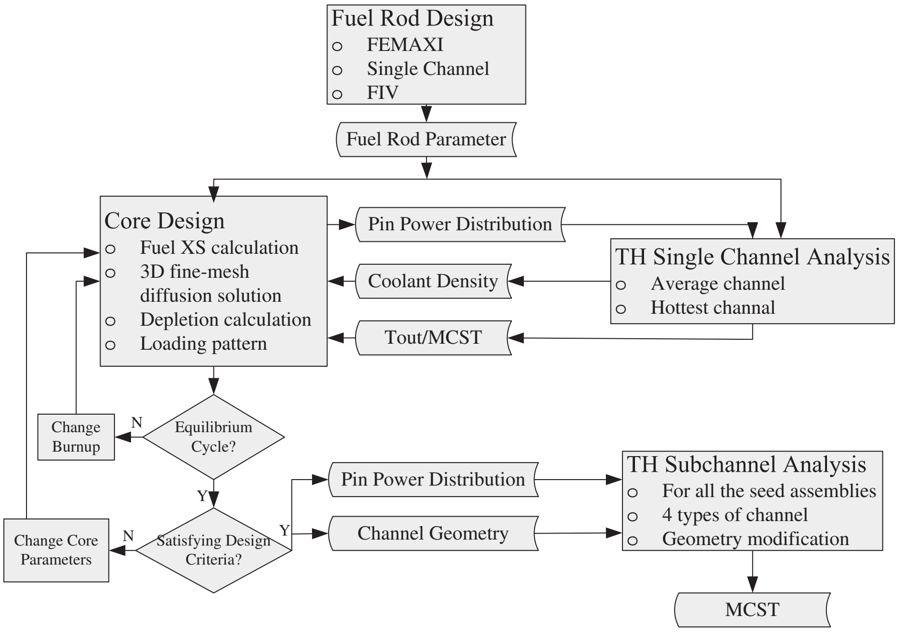

Fig. 1 Overall design procedure for the superfast reactor core design

system design, safety considerations and analyses, thermal-hydraulic analyses and experiments, material development, and backend fuel management. The aim of this study is to design a middle-scale commercial superfast reactor having an electric capacity of 700 MWe as the reference core for the whole program by using a three-dimensional neutronics thermal-hydraulic coupled method. Thermal-hydraulic calculation in the core design is based on single-channel analyses. Therefore, subchannel analyses are also performed for all the seed assemblies to precisely evaluate the maximum cladding surface temperature (MCST) by considering cross-flow. The data of core design and subchannel analyses will be used for control system design, safety analyses, computational fluid dynamics (CFD) analyses, and backend fuel cycle risk evaluation. For pursuing economic competitiveness, this core should be designed with a high power density while satisfying the design criteria.

The overall design procedure consists of preliminary fuel rod design, fuel assembly (seed and blanket) design, neutronics core design coupled with single-channel analyses, and thermal hydraulic fuel assembly design by subchannel analyses.

II. Design Goals and Criteria

In order to achieve high economical competitiveness for the superfast reactor, the following are established as the design goals:

(1) 700 MWe class middle-scale commercial reactor,
(2) core average thermal power density over 100 W/cm^3,
(3) core average outlet temperature over 500°C,
(4) core average linear heat generation rate around 17 kW/m, and
(5) average fuel assembly discharge burnup around 70 MWd/kgHM.

Among them, the high power density is the key issue because it is beneficial for reducing the size of the reactor pressure vessel (RPV) and containment vessel (CV), which is essential for achieving the economical property.

The following principles of thermal design criteria are considered to ensure fuel and cladding integrity:

(1) Maximum linear heat generation rate (MLHGR) should be less than 39 kW/m.
(2) MCST should be less than 650°C.
(3) Buckling collapse and creep collapse of the fuel cladding, caused by the high coolant pressure, should be avoided.
(4) Flow-induced vibration (FIV) should satisfy the design criterion of constructed liquid-metal fast breeder reactors (LMFBRs).[10)]

$$\left(\frac{1}{2}\rho V^2\right)^{1.16} \mathrm{Re}^{0.25} < 2.0 \times 10^6 \quad \text{or}$$
$$\frac{1}{2}\rho V^2 < 0.02\,\mathrm{MPa} \tag{1}$$

(5) Positive coolant density reactivity (negative void reactivity) should be achieved to ensure an inherent safety requirement for a water-cooled reactor core.

III. Design Methods

The design methods are very similar to those of the previous study.[8)] The overall procedures of the fuel and core design are depicted in **Fig. 1**.

The core design is carried out by coupling three-dimensional neutronics and thermal hydraulic calculations. The neutronics calculation is based on three-dimensional fine-mesh multigroup neutron diffusion solution. As the superfast reactor is designed with a closed loop for each assembly such that there is no cross-flow between assemblies, single-channel analysis can be employed to perform the thermal hydraulic analysis in the phase of core design.

Single-channel analysis has been widely used for thermal hydraulic analyses and proved to be conservative in current LWR fuel assemblies with large fuel rod gap clearance. However, it is not well known if such conservatism can be kept in the superfast reactor operating at supercritical pressure with small fuel rod gap clearance. To consider the cross-flow between channels inside one assembly, subchannel analysis based on a control volume approach is used to evaluate the MCST and verify the results of single-channel analysis.

1. Nuclear Design Method

The SRAC system developed by Japan Atomic Energy Agency (JAEA) has been used as the neutronics solver.[11] It is a multipurpose code system applicable to neutronics analyses of a variety of reactor types. The PIJ code of SRAC based on an integral neutron transport method called the collision probability method (CPM) is used to perform the cell transport calculation from the 61-fast and 46-thermal energy groups of the JENDL-3.3 nuclear data library and generate the cell homogenized cross sections. The SRAC system also includes two auxiliary codes: ASMBURN and COREBN. The ASMBURN code, which works by the combination of flux calculation by the CPM and burnup calculation by interpolation of macroscopic cross sections from cell calculation, is used to perform the assembly burnup calculations. The COREBN code based on the multidimensional diffusion code CITATION is used for three-dimensional core burnup calculation.

2. Thermal Hydraulic Calculation

Thermal hydraulic calculation is based on single-channel analyses for multicoolant channels. A fuel assembly is expressed as two representative coolant channels; one is the peak fuel rod channel and the other is the average fuel rod channel. Both peak and average coolant channels have the same thermal hydraulic parameters but different linear heat generation rate distributions. The peak fuel rod is defined as a fuel rod having the peak linear heat generation rate within an assembly, which is used to evaluate the peak cladding surface temperature (PCST) within an assembly. The average coolant channel shows the average linear heat generation rate over a fuel assembly, which is used for evaluating the average coolant density distribution along the fuel channel to be utilized in thermal-hydraulic coupled calculation.

Mass conservation, energy conservation and state equations are solved in the single-channel analysis code.[8]

3. Subchannel Analysis

A subchannel analysis code for supercritical pressure, which was developed by the University of Tokyo[12,13] and verified against the PRANDTL experiment, has been applied to the thermal-hydraulic fuel assembly design and used to evaluate the MCST of the superfast reactor. The code is based on a control volume approach. Each subchannel is treated as a control volume that communicates with an adjacent control volume through the gaps between the fuel rods. The subchannel analysis method consists of four governing equations at a steady state.

Mass continuity equation

$$\frac{\partial}{\partial z}(\rho_i u_i A_i) + \sum_j (\rho' v_{ij}) S_{ij} = 0 \tag{2}$$

Axial momentum conservation equation

$$\begin{aligned}\frac{\partial}{\partial z}(\rho_i u_i^2 A_i) + \sum_j (\rho' u' v_{ij}) S_{ij} &= -A_i \frac{\partial P_i}{\partial z} \\ &- \frac{1}{2}\left(\frac{f}{D_h} + \frac{k}{\Delta z}\right)(\rho_i u_i^2) A_i - A_i \rho_i g \cos\theta \\ &- \sum_j C_t w'_{ij}(u_i - u_j)\end{aligned} \tag{3}$$

Transverse momentum conservation equation

$$\begin{aligned}&\frac{\partial}{\partial z}(\rho_i u' v_{ij} S_{ij}) + C_s \sum_k \frac{\rho' v_k^2}{l_{ij}} \cos\beta_k S_{ij} \\ &= \frac{P_i - P_j}{l_{ij}} S_{ij} - \frac{1}{2} K_g \frac{\rho' v_{ij}^2}{l_{ij}} S_{ij} - \rho_i g \sin\theta \cos\gamma S_{ij}\end{aligned} \tag{4}$$

Energy conservation equation

$$\begin{aligned}&\frac{\partial}{\partial z}(\rho_i u_i h_i A_i) + \sum_j (\rho' h' v_{ij}) S_{ij} = \sum_i q' p_h \Delta z \\ &+ \frac{\partial}{\partial z}\left(A_i k \frac{\partial T}{\partial z}\right) - \sum_j Ck \frac{T_i - T_j}{l_{ij}} - \sum_j w'_{ij}(h_i - h_j)\end{aligned} \tag{5}$$

IV. Preliminary Fuel Rod Design

The purpose of preliminary fuel rod design is to determine the fuel rod diameter, the pitch-to-diameter ratio (P/D), the thickness of the fuel cladding, and the core active height by considering its thermo-hydrodynamic and mechanical characteristics.

1. Thermo-hydrodynamic Design

Two limiting conditions are taken into account in thermo-hydrodynamic design. One is flow-induced vibration (FIV) consideration and the other is the coolant outlet temperature.

A large core height requires a high flow rate so as to limit the MCST, which results in a large vibration. A previous experimental study[14] shows that the FIV is related to the material and many fuel pin parameters; however, it most strongly depends on the flow dynamic pressure and Reynolds number. To roughly evaluate the FIV, we combine and simplify the past experimental correlations as

$$\left(\frac{\delta}{D}\right) \propto \left(\frac{1}{2}\rho V^2\right)^{1.16} Re^{0.25} \quad \text{or} \quad \left(\frac{\delta}{D}\right) \propto \frac{1}{2}\rho V^2. \tag{6}$$

As the materials of the fuel pin in this study are the same as those of the past LMFBR, the same degree of rod displacement as that of the past LMFBR can be taken as the design criterion. Therefore, the criterion for the FIV in the past LMFBR design given in Eq. (1) is adopted in this study. This criterion limits the upper boundary of the core height. The results of the single-channel analyses are shown in **Fig. 2**. The real lines in the figure denote the upper bounda-

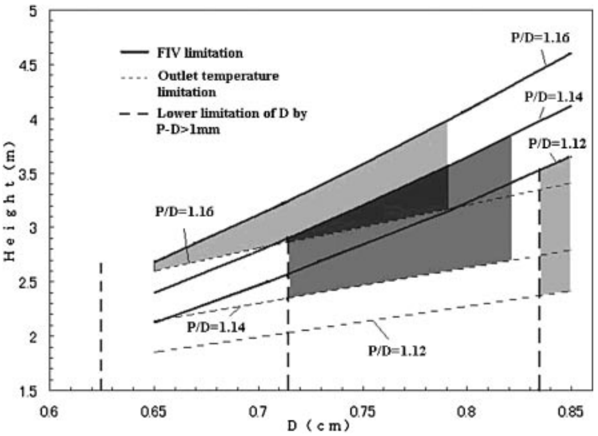

Fig. 2 Available ranges of diameter, height, and P/D

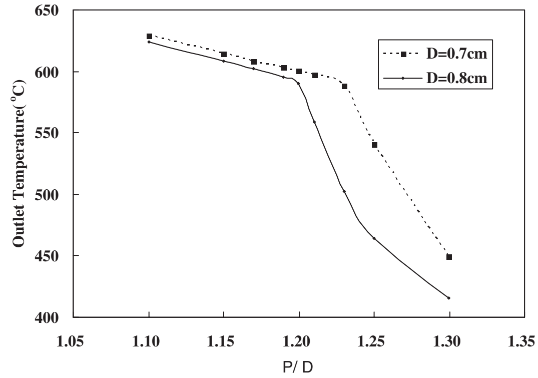

Fig. 3 Sensitivity of P/D to outlet temperature (MCST is kept at 650°C.)

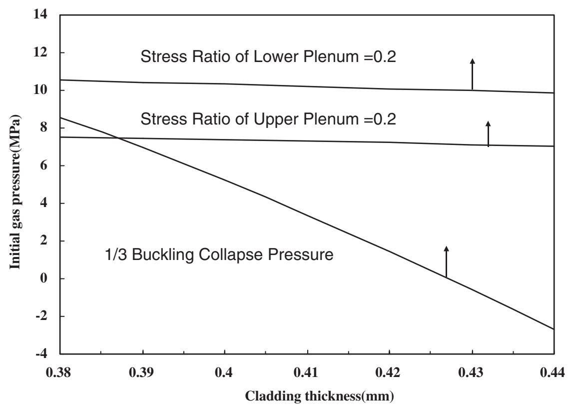

Fig. 4 Relationship between initial gas pressure and cladding thickness

ries of the core height with different P/D values limited by the FIV.

On the other hand, a small core height requires a low flow rate so as to keep the coolant outlet temperature. However, the low flow rate gives a low heat transfer coefficient; thus, the coolant outlet temperature needs to be low so as to limit the MCST. The past experiences[8)] indicate that if we keep the coolant outlet temperature of the peak fuel rod channel higher than 610°C, the core average outlet temperature will be over 500°C. This criterion will make the lower boundary of the core height. The dotted lines in Fig. 2 denote the lower boundaries of the core height with different P/D values.

It has not yet been known how small we can take the fuel rod gap from the viewpoint of fabrication, burnup bending, cladding swelling, heat transfer, *etc.* In this study, the fuel rod gap is set to be over 1 mm by referring to currently developed high-conversion LWRs with tight lattice bundles.[15)]

$$P - D > 1\,\mathrm{mm} \tag{7}$$

This requirement gives the lower boundaries of D corresponding to different P/D values; for example, as shown in Fig. 2, D should not be less than 0.833, 0.714, and 0.625 cm for P/D values of 1.12, 1.14, and 1.16, respectively.

The shadow areas in Fig. 2 denote the available ranges corresponding to different P/D values. The core average power density tends to increase as the fuel rod diameter decreases. Thus, although there is no particular upper limitation on the fuel rod diameter, we must choose a small value to achieve a high power density. On the other hand, the single-channel analyses show that the coolant outlet temperature is low with a large P/D when the MCST is kept at 650°C. This is because the large P/D gives a small mass flux and a low heat transfer coefficient. The sensitivity of the P/D to the coolant outlet temperature is shown in **Fig. 3** with the fuel rod diameters of 0.7 and 0.8 cm. Furthermore, tight fuel pin arrangement is also essential for hardening the neutron spectrum for the fast reactor.

Therefore, the P/D is chosen to be 1.16 by considering the balance of power density and outlet temperature. For conservatism, the design point of D and H is determined to be the middle point of the available P/D range of 1.16, which means that the fuel outer diameter is 7.0 mm and the active core height is 3.0 m.

2. Mechanical Design

The main concern regarding cladding thickness is the mechanical consideration including buckling collapse pressure and creep collapse. The buckling collapse pressure is calculated using[16)]

$$P_{collapse} = 2.2E\left(\frac{t}{D-t}\right). \tag{8}$$

So as to avoid creep collapse, the compress to yield stress ratio (σ_C/σ_Y) is limited to 0.2. The compressive stress arising from the pressure difference is calculated using

$$\sigma_C = (P_o - P_i)\frac{\overline{R}}{t}. \tag{9}$$

The calculated results for the relationship between the cladding thickness and the initial gas pressure are given in **Fig. 4**, from which we can see that the collapse pressure is sensitive to the cladding thickness, while the stress ratio is not so sensitive. Thus, if we set the initial gas pressure to be 7.5 MPa, the minimum cladding thickness will be around 0.39 mm. However, the superfast reactor will be operated under high pressure and temperature conditions, and the study on the corrosion of the cladding material under such condi-

Table 1 Preliminary fuel rod design results

Fuel material	MOX
Fuel density	95%TD
Fuel rod outer diameter [mm]	7.0
P/D	1.16
Cladding material	SUS304
Cladding thickness [mm]	0.43
Active core height [cm]	300
Average linear heat generation rate [kW/m]	17
Initial gas plenum pressure [MPa]	7.5

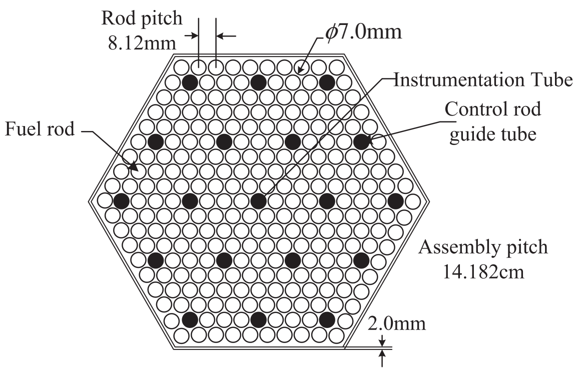

Fig. 5 Seed fuel assembly design

tions has not been well conducted yet. For conservatism, we assume that 10% of the cladding thickness is damaged by corrosion in the whole life. Thus, the final cladding thickness is determined to be 0.43 mm.

3. Fuel Rod Design Parameters

By the considerations above, the preliminary fuel rod design parameters have been determined. They are summarized in **Table 1**.

V. Fuel Assembly Design

Like most of liquid-metal-cooled fast reactors, the super-fast reactor consists of seed and blanket assemblies. The seed assemblies are used to generate thermal power and fast neutrons. Since the purpose of this study is to design a high-power-density core, not a breeder or conversion core, the blanket assemblies are used to achieve negative void reactivity. This is explained in Chap. 5.3.

1. Seed Assembly Design

A seed assembly is designed with 252 MOX fuel rods and 18 control rod guide tubes and 1 instrumentation tube in hexagonal arrangement, as shown in **Fig. 5**. The seed assembly has a very similar configuration to those of typical liquid metal fast reactors[17] except for the tighter fuel rod arrangements. Each fuel assembly is surrounded by a 2-mm-thick wrapper tube that separates individual flow rates, such as those of BWRs and LMFBRs. The individual flow rate in a fuel assembly is controlled by an orifice attached to the assembly inlet or the lower core plate. The assemblies are arranged in the core with a gap of 2 mm. The assembly pitch

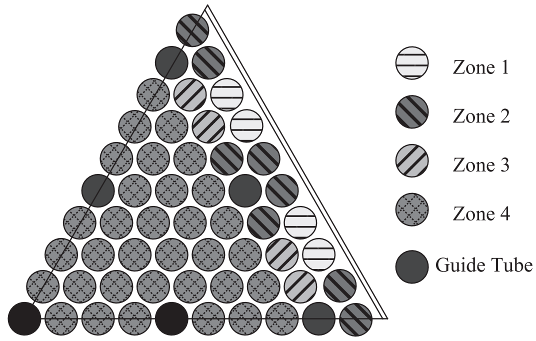

Fig. 6 Radial Pu enrichment zoning

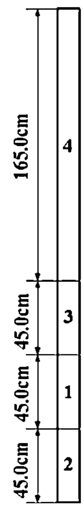

Fig. 7 Axial Pu enrichment zoning

Table 2 Fissile Pu enrichment zoning (wt%)

	Radial zone 1	Radial zone 2	Radial zone 3	Radial zone 4
Axial zone 1	17.340	18.496	23.120	26.010
Axial zone 2	17.918	19.074	23.698	26.588
Axial zone 3	18.496	19.652	24.276	27.166
Axial zone 4	19.074	20.23	24.854	27.174

(center-to-center distance between adjacent assemblies) is 14.2 cm. The clearance between the wrapper tube and the peripheral fuel rods is 1.12 mm, which is the same as the fuel rod gap clearance.

As a ZrH layer is employed in a blanket assembly in order to reduce void reactivity (see Chap. 5.3), it makes the adjacent seed assembly to have a high power peaking in the peripheral fuel rods. In order to flatten the power distribution in the seed assembly and thus limit the MCST, the radial Pu enrichment zoning shown in **Fig. 6** is employed. The axial Pu enrichment zoning shown in **Fig. 7** is also employed to reduce the power peaking along the axial direction. The Pu enrichment of each zone is given in **Table 2**.

2. Principle for Achieving Negative Void Reactivity

Generally, large-scale fast spectrum cores tend to have positive void reactivity because, under void conditions, the

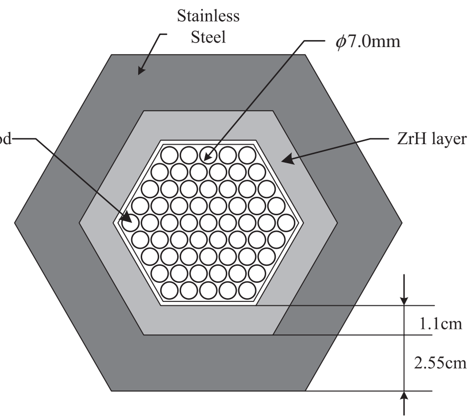

Fig. 8 Blanket assembly design

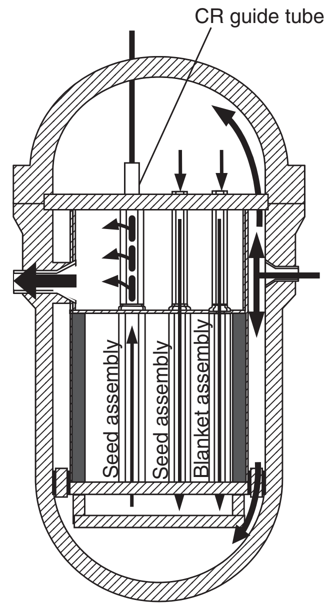

Fig. 9 In-vessel flow pattern

effects of increasing fast fission and decreasing neutron absorption by the coolant are possibly stronger than the effect of increasing neutron leakage in the core peripheral region. So as to achieve negative void reactivity, for example, flat cores are selected in the design of high-conversion BWRs.[15) However, the core height cannot be too small for the superfast reactor, having a much smaller mass flux than BWRs, due to the limitation of the coolant outlet temperature (see Chap. 4.1). Also, a flat core has a larger diameter than a non-flat core with the same thermal power, which results in a thicker RPV. Therefore, the flat core is not used for the superfast reactor. We use a unique method for negative void reactivity, which is explained in Chap. 5.3.

3. Blanket Assembly Design

The principal role of blanket fuel in fast reactor design is to convert fertile nuclide to a fissile material. In addition, the blanket fuel also plays the role of a neutron absorber when the coolant is voided. As shown in **Fig. 8**, the ZrH layer is employed in the blanket assembly to reduce void reactivity. The fast neutrons coming from seed fuel regions are slowed down through the ZrH layer and then absorbed in the blanket fuel. A neutron absorber is essential for reducing the coolant void reactivity cooperating with the heterogeneous ZrH layer.

A blanket fuel rod has the same dimensions as the seed fuel rod. Depleted UO_2 fuel discharged from LWRs with 0.2 wt% ^{235}U content is used. The blanket fuel assembly is also enveloped with a wrapper tube for flow separation. The ZrH layer is placed in the wrapper tube. The thickness and position of the ZrH layer are optimized to achieve the most negative void reactivity.

The past study[8)] showed that the existence of a ZrH layer tends to increase the power peaking of its neighboring seed assemblies dramatically. As a remedy, a stainless steel duct tube is used to substitute the peripheral fuel rods in the blanket assembly.

VI. Core Design

1. Coolant Flow Scheme

The power fraction of the blanket assemblies changes largely with the burnup. Since the flow rate fraction for the blanket, not changing with the burnup, must be so large to keep the MCST of the blanket below the criterion throughout the cycle, the coolant outlet temperature of the blanket assemblies is lower than that of the seed assemblies, which leads to degradation of the core average outlet temperature. This situation is not only for the blanket because some of the seed assemblies, in the core peripheral and innermost regions, have a very steep power gradient and cannot easily be controlled by radial enrichment zoning. In order to avoid degradation of the average core outlet temperature resulting from the blanket power swing and the large power gradient in some of the seed assemblies, downward flow in these assemblies is employed. The coolant flow scheme is shown in **Fig. 9**. Half of the feedwater is first derived from the top dome and flows downward through the blanket or seed assemblies and then mixed with the flow coming from the downcomer at the lower plenum. The flow pattern of each assembly is shown in **Fig. 10**.

Besides improving the coolant outlet temperature, this flow scheme offers additional advantages in aspects of mechanical integrity of the pressure vessel, neutron shielding, and reactor safety. The temperature of the coolant facing the pressure vessel is kept low as inlet temperature so that the coolant density is always high enough; the coolant acts as a good neutron shield for the pressure vessel. The large portion of the coolant accumulating in the top dome serves as an internal accumulator and delays blow down time during LOCA.[18)]

2. Core Arrangement and Loading Pattern

As core arrangement, *i.e.*, the location of the blanket and seed assemblies, is very sensitive to void reactivity, several types of core arrangement have been tested to compare the void reactivities at the beginning of the equilibrium cycle (BOEC) and the end of the equilibrium cycle (EOEC).

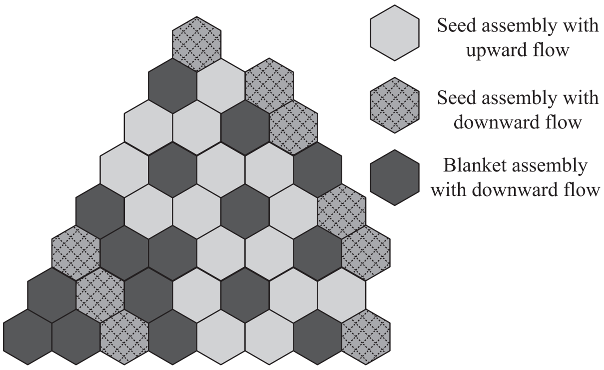

Fig. 10 Radial core flow pattern

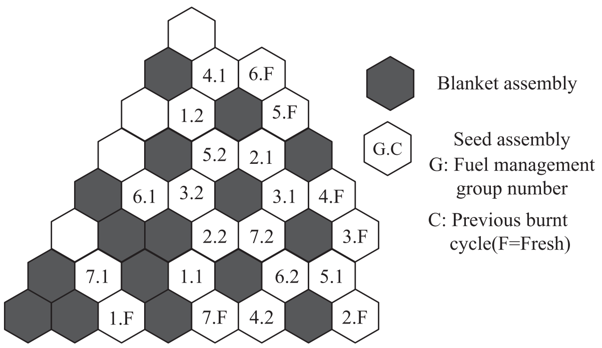

Fig. 11 Core arrangement and loading pattern of equilibrium cycle

Optimization of the loading pattern of the equilibrium cycle is essential for flattening the radial power distribution over the cycle and balancing the assembly discharge burnup. A loading pattern, which is optimized manually, is used in the current design. A typical out-in fuel management scheme is used with three batch cycles and 380EFPD cycle length, where most of the fresh fuel assemblies are loaded in the core peripheral regions. The core arrangement and its loading pattern of the equilibrium cycle are shown in **Fig. 11**, where the fuel management group numbers in the seed assemblies stand for the shuffling scheme. For example, assembly 1.F is moved to the position of assembly 1.1 at the end of its first cycle and to the position of assembly 1.2 at the end of its second cycle.

3. Characteristics of Core Design

Some preliminary parameters of the designed superfast reactor are summarized in **Table 3**. We can see that the average coolant outlet temperature is higher than 500°C, that the average power density is much higher than 100 W/cm^3, and that the flow rate is only 850.0 kg/s, which is only about 1/10 of that of a BWR or PWR with the same electric power. The calculated results for void reactivity are negative at both BOEC and EOEC.

The calculated results for the coolant outlet temperature, hottest cladding surface temperature, local power peaking factor, and coolant flow flux for each seed assembly are given in **Figs. 12–15**, where "Max." and "Min." mean the maximum and minimum values, respectively, over the whole fuel cycle. The radial and axial power distributions at BOEC, MOEC, and EOEC are plotted in **Figs. 16** and **17**, respectively. The maximum linear heat rate along burnup is given in **Fig. 18**.

From Fig. 12, we can see that the coolant outlet temperature is relatively uniform along with burnup and that the largest coolant outlet temperature difference over the cycle is less than 20°C, which happens in the assembly surrounded

Table 3 Preliminary parameters of the designed superfast reactor

Parameter	
Core thermal power [MWt]	1,650
Core height [cm]	300
Equivalent diameter [cm]	210
Number of seed assemblies	126
Number of blanket assemblies	73
Number of seed assemblies with downward flow	42
Average fissile Pu enrichment [wt%]	24.87
Fissile Pu inventory [t]	6.571
Heavy metal inventory [t]	26.42

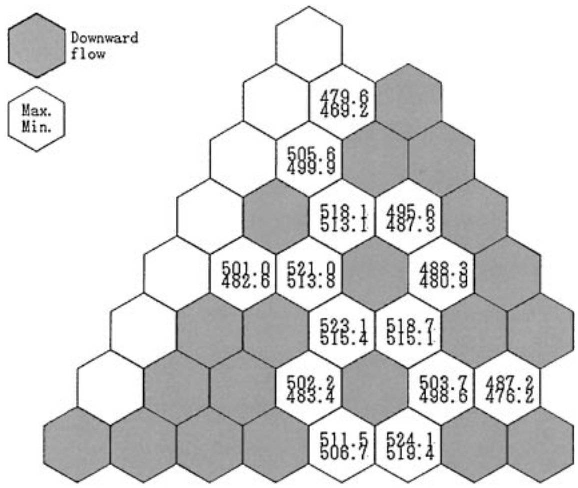

Fig. 12 Coolant outlet temperature

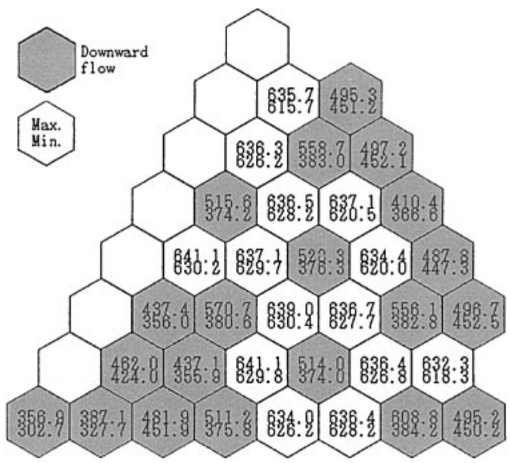

Fig. 13 Hottest cladding surface temperature

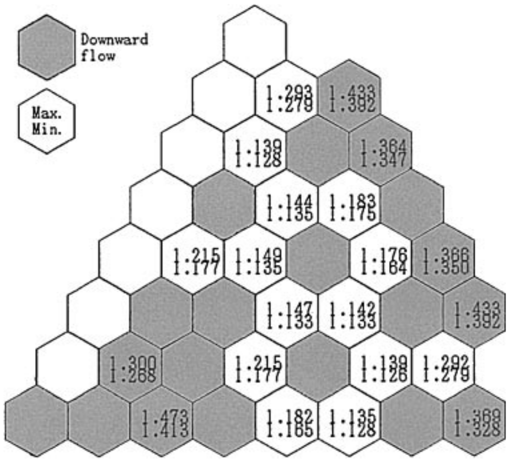

Fig. 14 Local power peaking

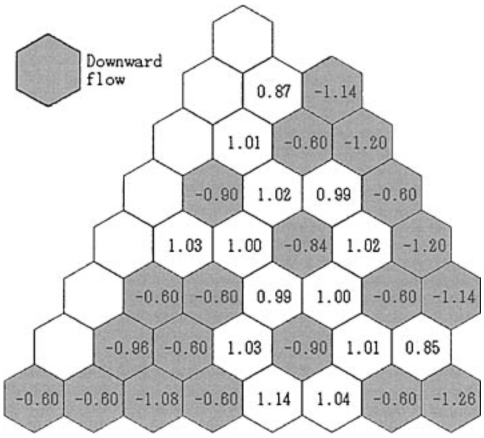

Fig. 15 Relative mass flux distribution

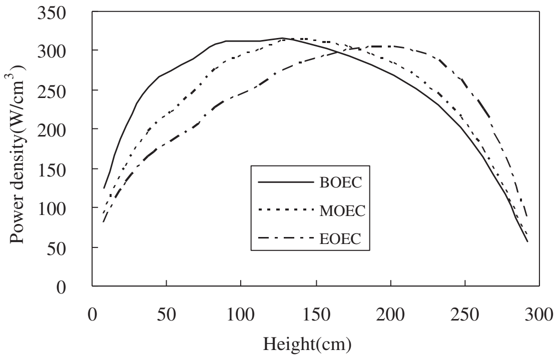

Fig. 17 Core average axial power distribution

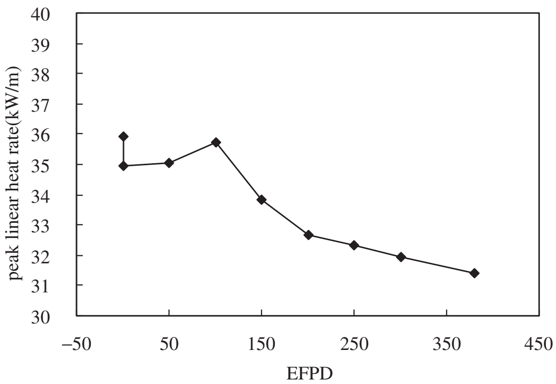

Fig. 18 Peaking linear heat rate

by four blanket assemblies. This shows that the reactivity swing of the seed assemblies along with burnup is small and can easily be controlled. The coolant outlet temperature of different assemblies directly corresponds to the local power peaking observed in Fig. 14, *i.e.*, small power peaking yields high outlet temperature.

The hottest cladding surface temperature distribution given in Fig. 13 shows that the MCST over the cycle is 641.1°C. The cladding surface temperatures of the assemblies with downward flow are lower than 500°C, which gives a large margin for the uncertainty of single-channel analysis for downward flow. Figure 15 gives the relative flow mass flux distribution, which is kept constant over the cycle. The values depend strongly on the power distribution of the assemblies.

From Fig. 16, we can see that the radial power distribution does not markedly change with burnup, but the power share of the blanket increases. The axial power distribution, which is given in Fig. 17, shifts from the bottom peak to the top peak with burnup. Figure 18 shows that the peaking linear heat generation rate decreases with burnup except a local increase at 100EFPD because of the xenon oscillation. The MLHGR is 35.9 kW/m, which is well below the design criterion and happens at BOEC.

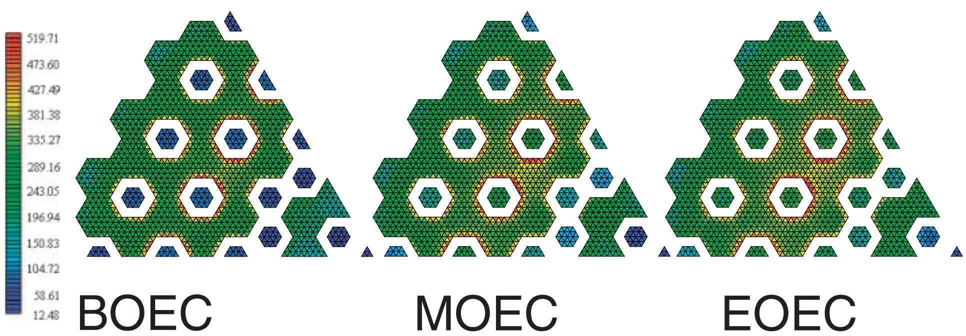

Fig. 16 Radial power distribution

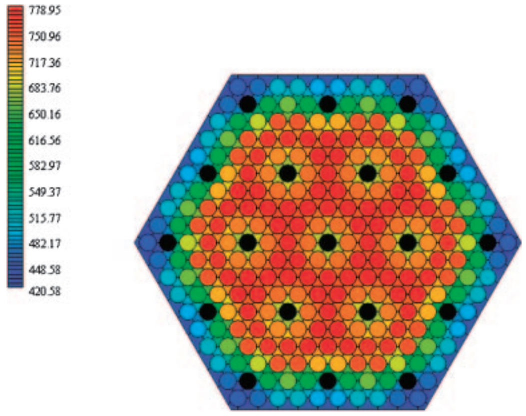

Fig. 19 Temperature distribution with uniform power distribution

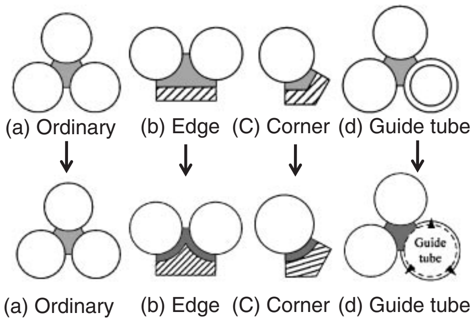

Fig. 20 Channel shape in seed assembly

VII. Subchannel Analyses

Subchannel analyses have been performed on all of the seed assemblies in the above designed core. The results show that the MCST is 706.3°C and that the maximum difference in MCST between single-channel and subchannel analyses is as high as 101.5°C. This is mainly due to the strong subchannel heterogeneity of different types of channel. **Figure 19** gives the temperature distribution of the subchannel analyses results of the originally designed assembly with a uniform power distribution. We can find that the peripheral fuel pins are strongly overcooled and that the fuel pins around the CR guide tubes are slightly overcooled.

There are four types of channel in one assembly, as shown in **Fig. 20**. The degree of subchannel heterogeneity can be represented by the ratio of heated perimeter to flow area, H/A. The H/A values of the four types of channel are given in **Table 4**. In order to balance the H/A value and consequently reduce the degree of subchannel heterogeneity, we propose two modifications for the channel geometry. One is to add protrusions in the wrapper tube and the other is to enlarge the diameter of the CR guide tubes. After these modifications, the hydraulic parameters are changed to be those given in Table 4.

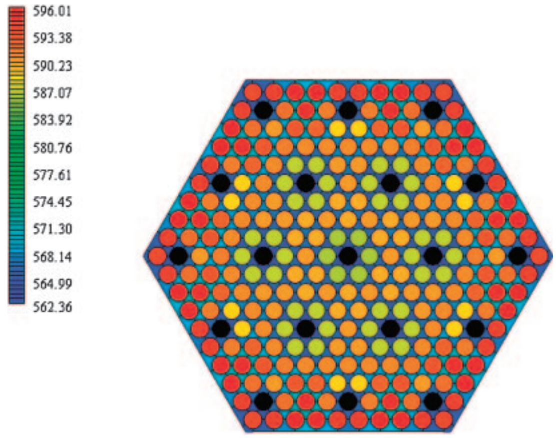

Fig. 21 Temperature distribution of modified geometry

The hottest cladding surface temperature of each fuel rod and the coolant outlet temperature of each subchannel with a uniform power distribution after modification are given in **Fig. 21**. Compared with Fig. 19, the temperature distribution is much more uniform. The calculated results for the hottest cladding surface temperatures of the seed assemblies are compared between subchannel and single-channel analyses in **Fig. 22**.

The MCST is 639.8°C. The maximum difference in MCST between single-channel and subchannel analyses is 76.1°C, which is obtained at the peripheral assembly with a relatively low temperature. This shows that subchannel heterogeneity has a strong influence on the precision of single-channel analysis, especially for the assembly with downward flow. However, it can be well controlled by making some small changes on the geometry of the channels.

VIII. Final Design Results

Table 5 summarizes the main parameters and calculated results of the final core design.

This core is designed with a high power density of 158.8 W/cm^3 by keeping the MLHGR less than 39 kW/m, and an average coolant outlet temperature of 503.7°C is achieved with the MCST less than 650°C. The core equivalent diameter is only 2.1 m, which is much smaller than that

Table 4 Hydraulic parameters of channels

	Channel area [mm^2]	Wetted perimeter [mm]	Heated perimeter [mm]	H/A^*	Original H/A
Ordinary	9.31	11.00	11.00	1.18	1.18
Edge	11.42	21.12	11.00	0.96	0.60
Corner	4.19	8.38	3.67	0.88	0.62
Guide tube	7.34	11.52	7.33	0.998	0.79

$^*H$ = Heated perimeter A = Channel area

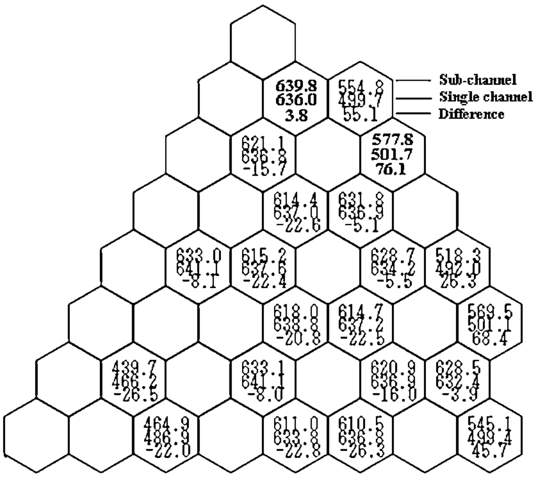

Fig. 22 MCSTs of single-channel and subchannel analyses

Table 5 Summary of main parameters of core design

Parameter	
Core thermal power [MWt]	1,650
Core electrical power [MWe]	723
Core height [cm]	300
Equivalent diameter [cm]	210
Number of seed assemblies	126
Number of blanket assemblies	73
Number of seed assemblies with downward flow	42
Fissile Pu enrichment [wt%]	24.87
Fissile Pu inventory [t]	6.571
Heavy metal inventory [t]	26.42
Coolant outlet temperature [°C]	503.7
Maximum cladding surface temperature [°C]	639.8
Cycle length [EFPD]	380
Average power density [W/cm^3]	158.8
Average linear heat rate [kW/m]	17.3
Maximum linear heat rate [kW/m]	35.9
Flow rate [kg/s]	850.0
Average discharge burnup [MWd/kgHM]	69.3
Coolant void reactivity [%dk/k] BOEC	−1.23
EOEC	−2.07

of a PWR or BWR with the same electric power. This offers great potential for reducing the size of the pressure vessel, consequently reducing the capital cost. The flow rate is 850 kg/s, which is about 1/10 of that of a typical PWR with the same electric power. The void reactivity coefficients are negative throughout the equilibrium cycle.

IX. Conclusions

A middle-scale class supercritical water-cooled fast reactor with a power of 700 MWe, called the superfast reactor, has been designed as part of the research program entrusted by MEXT, Japan. Results obtained by coupling neutronics and thermal hydraulic calculations show that all of the design goals are achieved while satisfying all of the design criteria. Subchannel analyses are also performed on all of the seed assemblies to ensure the MCST under 650°C.

As this work is just the beginning of the whole research, fuel and core designs will be improved. Firstly, local void reactivity, as well as overall void reactivity, should be kept negative for safety. Secondly, the core average power density can be continually increased by optimizing the core parameters. Thirdly, fuel rod behavior should be analyzed in detail.

Nomenclature

A: cross-sectional flow area
C: compensation coefficient for connection between channels
C_t: turbulent Prantle number
D: diameter of the fuel pin
E: Young's modulus
f: frictional coefficient
g: gravitational acceleration
h: specific enthalpy
K_g: loss coefficient of grid spacer
k: heat conductivity
l: mixing length
P: coolant pressure
p_h: heated perimeter
P_o: external pressure
P_i: inner pressure
$\bar{R}$: mean radius of fuel cladding
Re: Reynolds number
S: fuel rod gap clearance
T: coolant temperature
t: cladding thickness
u: axial coolant velocity
v: transverse coolant velocity
w: transverse mixing mass flux
ρ: coolant density
σ_c: compressive stress
σ_Y: yield strength

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