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Preliminary Core Design Study of Small Supercritical Fast Reactor with Single-Pass Cooling

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Abstract: A supercritical water-cooled reactor (SCWR) adopts a once-through direct cycle, which is compatible with a small modular reactor class (SMR) plant system. The core is cooled by supercritical light water, which does not exhibit phase change, but undergoes large temperature and density changes. A super fast reactor (Super FR) is a fast reactor type concept of SCWR. Unlike other SCWR core concepts, it adopts the single coolant pass flow scheme, in which the coolant passes the core only once from the bottom to the top without any reverse flows or preheating stages. In the meantime, reducing the core size tends to increase the core power peaking and reduce criticality. Therefore, the key issues with the small Super FR core design is reducing the core power peaking and achieving high average core outlet temperature with the single coolant pass scheme. This study aims to highlight the design issues through conceptual core designs of SMR class Super FR. To evaluate the core characteristics, three-dimensional coupled core calculations are carried out. The proposed design with small fuel assemblies, which are equivalent to those of boiling water reactors, attains a high core average outlet temperature of about 500 °C, which is compatible to that of typical large SCWR core design.

Keywords: supercritical water-cooled reactor (SCWR); Super FR; small modular reactor (SMR); conceptual core design; coupled neutronics; thermal-hydraulics core calculations

1. Introduction

A supercritical water-cooled reactor (SCWR) is one of the Generation IV reactor concepts, which is based on the current light water reactor (LWR) and supercritical pressure fossil fired power plant technologies. Thus, SCWR can be developed incrementally step-by-step from current water-cooled reactors. As the coolant does not exhibit phase change, the simple and compact once-through direct cycle plant system is adopted, in which the entire reactor coolant is directly fed to the turbine. More about the advantages and challenges in SCWR development may be found in the Generation IV International Forum portal and various papers given in the portal as references. So far, most of the research and development effort has been focused on development of large-scale plant systems with thermal power ranging from about 2300 to 4039 MWth [1].

In recent years, small modular reactors (SMRs), with electric power of 300 MWe or less, have gained large interest due to various driving forces such as, but not limited to: offering better economic affordability; replacing the ageing fossil fuel-fired power plants; options for remote regions without established electricity grid infrastructures and so on [2]. However, studies on SMR class SCWR are limited. One of the studies on SMR-scale SCWR is the Canadian pressure tube type reactor concept, SUPERSAFE, which is aimed at 300 MWe (670 MWth). It adopts reactor grade PuO_2 and ThO_2 fuel, cooled by supercritical light water and moderated by D_2O [3]. According to the study, SUPERSAFE can remove decay heat with natural circulation due to the large coolant density change, which naturally induces coolant flow in the core. However, the study is limited to some preliminary evaluation of fuel channels without core-wise evaluations [4,5].

In the view of the authors, there is lack of conceptual studies to reveal expected characteristics of a reactor pressure vessel (RPV) type SCWR for SMR class. In particular, conceptual core design studies are needed to reveal the impact of reducing the core size with respect to the basic core characteristics, such as the core power peaking, the average core outlet temperature, the operation cycle length (reactivity), and the coolant density (or temperature) reactivity characteristics. In general, reducing the core size tends to increase the core power peaking. Since SCWR is cooled with a single-phase flow, increase in the core power peaking may influence uniform core cooling and may significantly reduce the attainable average core outlet temperature. Raising the average core outlet temperature is one of the main design issues of SCWR concept as it has large impact on the plant thermal efficiency [6,7]. In the meantime, reduction in core size reduces the core criticality and impact on the required fuel enrichment to attain the target operation cycle length (or discharge burnup), which may impact on the void reactivity characteristics.

Hence, this study addresses the above issues through conceptual core design studies. The reference large core design is the super fast reactor (Super FR), which is the Japanese RPV type SCWR concept with rated power of 2337 MWth [8]. The core is fueled with reactor grade Pu and is cooled with a single pass coolant scheme, in which the entire coolant flows from the bottom of the core to the core outlet without any reverse flows or preheating stages. The target power for the SMR class core concept to be developed in this study is tentatively determined as about 650 MWth, which corresponds to 300 MWe with assumption of thermal efficiency of 43.8%. The purpose of this study is to reveal the impact of reducing the core size on the above core characteristics and propose a design concept with a design target of: average core outlet temperature of 500 °C with additional design target parameters, which are described in the following chapter.

2. Design Targets and Criteria

2.1. Design Targets

In this study, the following design targets have been tentatively determined by referring to typical design targets of SCWR and also considerations for the expected performance of SMR:

1. Average core outlet temperature ≥ 500 (°C);
2. Average linear heat generation rate (ALHGR) ≥ 15.0 (kW/m);
3. Operation cycle length ≥ 720 (days).

The design target of the average core outlet temperature is important as it directly influences the plant thermal efficiency [6,7]. For a given core inlet temperature of 280 °C at 25 MPa, the target average outlet temperature of 500 °C corresponds to the plant thermal efficiency of about 43.8% [6]. In this study, the target of 500 °C, as also adopted by other SCWR design concepts [1], is tentatively determined. The target of ALHGR is tentatively determined to be 15.0 kW/m or higher by referring to the preceding study [8].

One of the expected roles of SMRs is the electricity (as well as heat) supply in remote regions without having the need to establish large-scale electricity grid infrastructures. For such use, SMRs are expected to operate with long operation cycles without requiring frequent shutdown and maintenance. For this reason, many SMRs aim at long operation cycle length. In this study, the operation cycle length is tentatively determined to be 720 days (2 years) by referring to typical SMR designs [2]. Walk away safety is often discussed for potential merit of SMRs. For attaining such a safety feature, reducing the core pressure loss and decay heat removal with natural circulation is considered to be an advantageous design feature [3]. However, there is large uncertainty regarding evaluation of pressure loss of supercritical water cooling. On the other hand, evaluation of the natural circulation requires rather detailed plant design, which is not readily available for this study. Hence, in this study, establishment of natural circulation is not explicitly considered as the design target. However, the core pressure loss with the currently available correlations is evaluated with intention to reduce to the level; it may be reasonable to expect the possibility of establishing natural circulation by referring to some other studies. For example, the earlier boiling water reactor type design indicates low core pressure loss of 0.04 MPa for attaining natural circulation [9].

2.2. Design Criteria

For the purpose of developing the core design concept, the following design criteria were determined regarding the core coolability, fuel integrity, and inherent safety by referring to the preceding study [8]:

1. Maximum cladding surface temperature (MCST) ≤ 650 (°C);
2. Maximum linear heat generation rate (MLHGR) < 39 (kW/m);
3. Void reactivity coefficient < 0 (%dk/k/%void).

In this study, advanced stainless-steel cladding is assumed as fuel cladding material as in the preceding study. The values of MCST and MLHGR design criteria are tentatively determined for preventing the cladding buckling collapse and pellet cladding mechanical interaction (PCMI) failure during abnormal transients, respectively [10].

3. Analysis Method

The core performance is evaluated by a neutronics- and thermal-hydraulics-coupled three-dimensional core calculations method as adopted in the preceding studies [11], so that the influence of reducing the core size on the core power peaking, coolant temperature and density distributions, and reactivity characteristics can be evaluated. Firstly, the neutronics calculations are carried out with some assumed water density distributions to give core power distributions. Then, the thermal-hydraulics calculations are carried out to update coolant density distributions for the next iteration of the neutronics calculations. These calculations are repeated until converged power distributions and water density distributions are obtained for the given burnup step. At the end of the operation cycle, the fuel shuffling pattern is considered to update the fuel burnup distribution for the next cycle of operation. The coolant flow rate to each fuel assembly is determined so that the MCST design criterion is satisfied in all assemblies throughout the operation cycle. These calculations are repeated until the core reaches the equilibrium state.

3.1. Neutronic Calculations

The SRAC code system [12] and JENDL-3.3 nuclear data library [13] developed by Japan Atomic Energy Agency (JAEA, Ibaraki, Japan) is used for neutronics calculations. The SRAC code system consists of three calculation modules: SRAC, ASMBURN, and COREBN. In SRAC and ASMBURN, cell and assembly burnup calculations including the branch-off calculations with various water densities are carried out with collision probability method. With these calculations, macroscopic cross sections are tabulated as the function of water density and fuel burn-up for different burnup histories. At this time, the fuel temperature is fixed at 1200 °C. In reality, the fuel reactivity also depends on the fuel temperature. However, in this study, constant fuel temperature is assumed as its influence is expected to be relatively small compared with other reactivity changes such as the coolant density feedback. Meanwhile, the heterogenous form factor (HFF) is evaluated to calculate pin-wise power distributions in fuel assemblies. In COREBN, three-dimensional core neutronics calculations are carried out with interpolation of macroscopic cross sections with water density and fuel burnup to give homogenized core power distributions at various burnup steps. Combining these results with the evaluated HFF, pin-wise power distribution of the core is evaluated and used for the thermal-hydraulics calculations.

3.2. Thermal-Hydraulics Calculations

The SPROD code developed by University of Tokyo (Tokyo, Japan) is used for thermal-hydraulics calculation [11]. It is based on a single-channel model to calculate heat transfer from the fuel rod to the cladding and coolant and pressure loss. For each fuel assembly, the average power channel and the hot channel are evaluated. The Watts' correlation [1,14] is adopted for evaluation of the heat transfer with supercritical water cooling as also adopted by some other design studies. [1]. For evaluation of pressure loss, the friction pressure loss, gravity pressure loss and acceleration pressure loss are considered [15].

4. Core Design

4.1. Neutronic Calculations

In order to investigate the impact of reducing the core size on the core performance, a set of fuel design parameters have been determined as the common design parameters for the hexagonal fuel assembly as summarized in Table 1. In this study, the cladding material is assumed to be advanced stainless steel by referring to the preceding study [8] with main concerns on evaluating neutronics performance of the core. Development of the cladding material candidates, including stainless steels, are currently being carried out with various issues such as irradiation behavior and corrosion characteristics and so on [16]. With the fuel assemblies described below, cores with different sizes are configured as shown in Figure 1. For the sake of simplicity in the comparisons, the cores are assumed to be configured with the fuel assemblies, which are loaded in the core up to three cycles of operation with out-in fuel loading patterns. Table 2 summarizes the corresponding core design specifications. The average fuel discharge batch number, the average number of operation cycles for which a fuel assembly stays in the core before being discharged, is kept about the same among the different cores.

Table 1. Design parameters of the fuel rod and the hexagonal fuel assembly.

Fuel rod outer diameter (mm)/Fuel rod pitch (mm)	10.0/11.0
Cladding material/thickness (mm)	advanced stainless steel/0.60
Number of fuel rods per assembly	469
Average fuel temperature (°C)	1200
Assembly pitch (mm)	246.6
Channel box thickness (mm)/Gap between assemblies (mm)	1.0/2.0
Heated height (m)	2.0

Table 2. Core design specifications for the different core sizes.

ALHGR (kW/m)/Operation Cycle Length (Days)	**15.0/720**			
	Core 1	**Core 2**	**Core 3**	**Core 4**
Number of assemblies	127	91	61	37
Core equivalent diameter (cm)	292	247	202	158
Thermal power (MWth)	1800	1280	860	520
Average fuel discharge batch number	2.1	2.3	2.0	2.1
Average fuel discharge burnup (GWd/ton)	42.4	46.4	41.0	43.1
Void reactivity coefficient BOEC/EOEC (beginning/end of equilibrium cycle) (%dk/k/%void)	0.008/ −0.065	0.006/ −0.068	−0.002/ −0.083	0.014/ −0.041
Pressure loss (MPa)	0.030	0.033	0.036	0.045

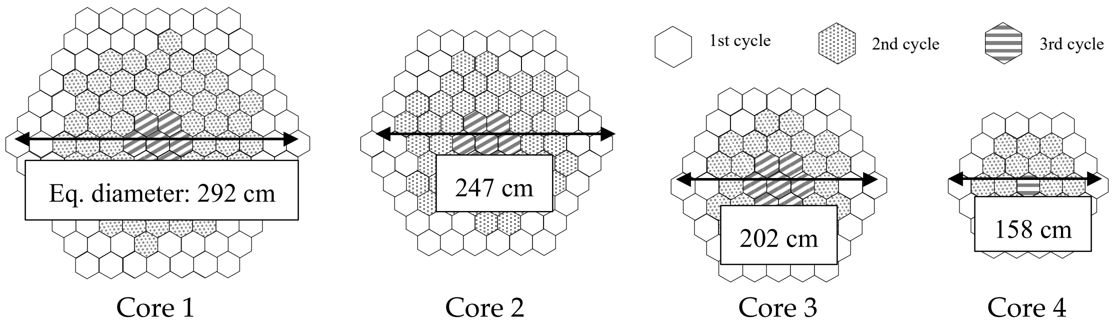

Figure 1. Core configurations.

Figure 2 shows the average core outlet temperature and MLHGR for the four cores. The results indicate that as the core size is reduced with less fuel assemblies, the MLHGR is increased. Figure 3 compares the core radial power distributions of Core 1 and Core 4 at BOEC. The radial and the local power peaking is higher for the smaller Core 4 compared with that of Core 1 and there is such consistent tendency among the four cores. These results indicate that for the given ALHGR, the MLHGR is increased from Core 1 to Core 4 as the core size is reduced due to larger radial and local power peaking. In addition, increasing power peaking results increasing flow rate. Therefore, pressure loss increases by reducing the core size.

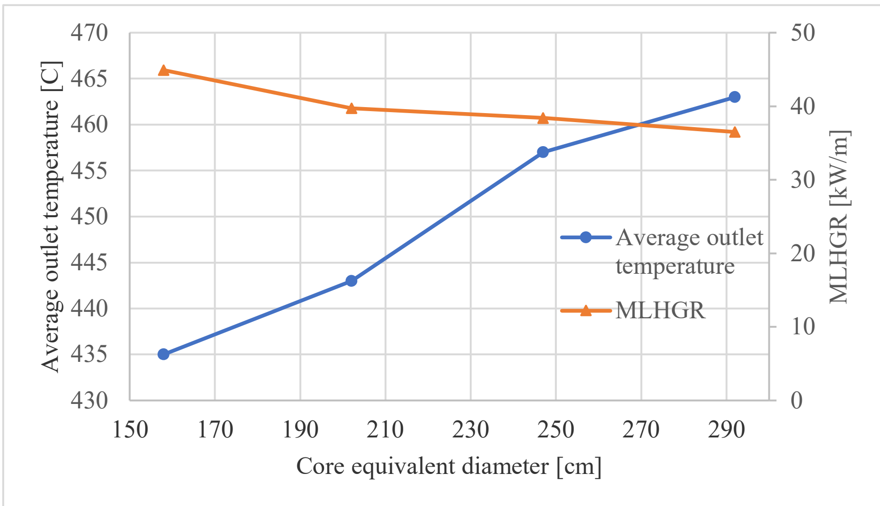

Figure 2. Dependence of maximum linear heat generation rate (MLHGR) and average outlet temperature on the core size.

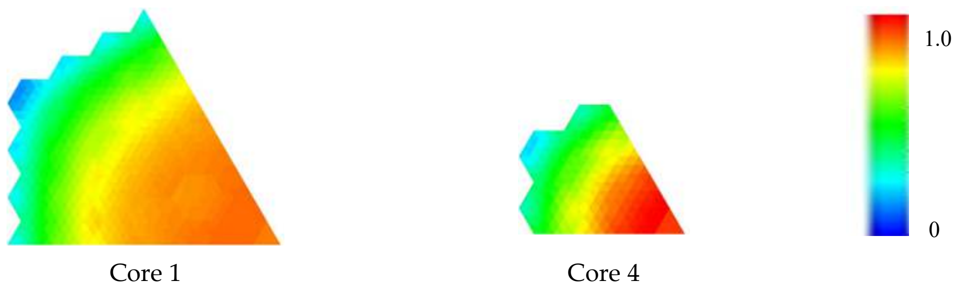

Figure 3. Normalized radial power distribution of the 1/6 core (BOEC).

Figure 2 also indicates that as the core size is reduced, the average outlet temperature is decreased. With consideration of Figure 3, it can be understood that the higher local power peaking with the smaller core leads to larger mismatch between the coolant flow rate and the fuel rod power and such power to flow mismatch is responsible for the decrease in the average outlet temperature. The smaller core also has larger fraction of the fuel assemblies loaded in the core periphery, which is another reason for the decrease in the average outlet temperature.

4.2. Increasing Average Outlet Temperature by Reducing Fuel Assembly Size

In order to increase the average outlet temperature, the power flow mismatch has to be reduced for each fuel assembly. For this purpose, the assembly size is reduced so that the local power peaking can be reduced and a fraction of the assemblies loaded in the core periphery is reduced. Thus, the assembly size and pitch are reduced, but the minimum assembly size is tentatively determined as the equivalent size of the boiling water reactor (BWR) fuel assembly. For each assembly size, the core configuration and fuel loading patterns are considered in a similar manner as described in Figure 1. The design specifications are summarized in Table 3.

Table 3. Core design specifications with different fuel assembly sizes.

	Core 4-1	Core 4-2	Core 4-3	Core 4-4
Assembly pitch (cm)	24.66	20.82	18.89	15.04
Number of fuel rods per assembly	469	331	271	169
Number of fuel assemblies	37	61	91	127
Core equivalent diameter (cm)	158	171	189	178
Core thermal power (MWth)	510	670	740	650
Average fuel discharge batch number	2.1	2.0	2.3	2.1
Average fuel discharge burnup (GWd/t)	43.1	39.4	41.8	44.0

With the above described size reduction of the fuel assembly, the local power peaking is reduced and the average outlet temperature is increased. The dependence of the average outlet temperature and the local power peaking on the assembly size are shown in Figure 4.

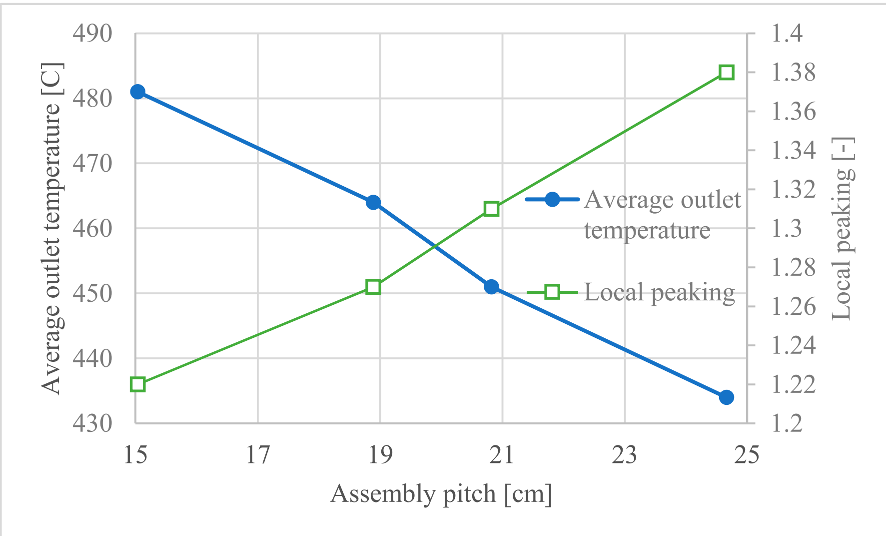

Figure 4. Dependence of the average outlet temperature and the BOEC local peaking on the assembly size.

4.3. Proposed Design for SMR Class Super FR

As discussed above, reducing the core size of Super FR leads to lower average core outlet temperature and higher MLHGR, but the average core outlet temperature can be raised by reducing size of the fuel assemblies. Hence, the following design is proposed as a preliminary core design concept of SMR class Super FR. The fuel assembly size is roughly the same as that of BWR. The average fuel discharge batch number has been increased to 4.2, as shown in Figure 5, to reduce the core radial power peaking, As the result, the average and maximum discharge burnup are 84.8 and 107.1 GWd/t, respectively. Core design with such high discharge burnup is limited to few studies, such as the study on SMR class sodium cooled fast reactor [17]. With consideration of fuel integrity under supercritical water-cooled condition, it may be necessary to reduce the discharge burnup, for example by reducing the average linear heat rate or operation cycle length. Such considerations may be necessary in future research. Pu enrichment zoning is considered within the fuel assemblies to reduce the local power peaking. Table 4 summarizes the proposed design specifications and characteristics.

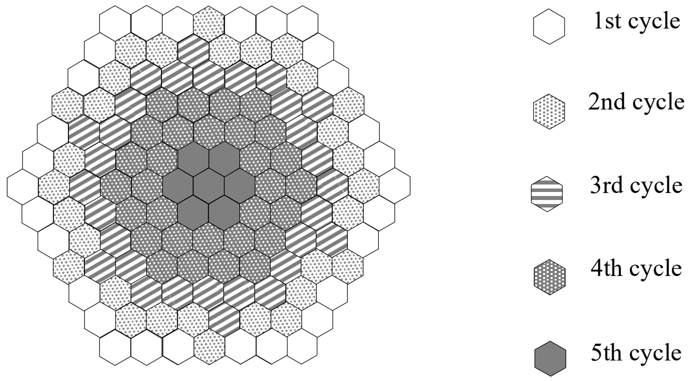

Figure 5. Fuel shuffling pattern of the proposed design.

Table 4. Proposed core design specifications and characteristics.

Thermal power/Electric power (MW)	650/285
Assembly pitch (cm)/Number of assemblies	15.0/127
Cycle length(days)/Batch number	720/4.2
Average Pu enrichment (wt.%)	18.8
Average/Maximaum fuel discharge burnup (GWd/t)	84.8/107.1
Excess reactivity at BOEC (%)	5.0
Fissile Pu surviving ratio	0.97
Void reactivity coefficient BOEC/EOEC (%dk/k/%void)	−0.009/−0.099
ALHGR/MLHGR (kW/m)	15.1/30.6
MCST (°C)	650
Coolant inlet/average outlet temperatures (°C)	280/502
Core pressure loss (MPa)	0.03

5. Conclusions

The main issues of designing an SMR class SCWR core have been discussed to propose SMR class Super FR with a single-pass coolant flow scheme. The issues include, but are not limited to: raising the average core outlet temperature, reducing the power peaking, reducing the core pressure loss, increasing the operation cycle length (reactivity), and attaining negative reactivity feedback with the coolant density (or temperature) changes. Reducing the fuel assembly size and increasing the fuel batch number are effective method to raise the average core outlet temperature and reduce the core power peaking (or MLHGR). By adopting small fuel assemblies, which are equivalent to those of BWR, the proposed design attains an average core outlet temperature of about 500 °C, which is compatible to typical large SCWR design concepts.

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References

1. IAEA. *Heat Transfer Behaviour and Thermohydraulics Code Testing for Supercritical Water Cooled Reactors (SCWRs)*; IAEA: Vienna, Austria, 2014; pp. 18–68.
2. IAEA. *Advances in Small Modular Reactor Technology Developments*; IAEA: Vienna, Austria, 2014; pp. 1–130.
3. Yetisir, M.; Pencer, J.; McDonald, M.; Gaudet, M.; Licht, J.; Duffey, R. The Supersafe© Reactor: A Small Modular Pressure Tube SCWR. *AECL Nucl. Rev.* **2012**, *1*, 13–18. [CrossRef]
4. Dimmick, G.; Chatoorgoon, V.; Khartabil, H.; Duffey, R. Natural-convection studies for advanced CANDU reactor concepts. *Nucl. Eng. Des.* **2002**, *215*, 27–38. [CrossRef]
5. Dimmick, G.R.; Khartabil, H.F.; Duffey, R.B.; Chatoorgoon, V. Natural Convection and Circulation Studies for Advanced CANDU Reactor Concepts. In *Single and Two-Phase Natural Circulation*; Genovaviva: Genoa, Italy, 1999; pp. 291–300.
6. Oka, Y.; Koshizuka, S.; Ishiwatari, Y.; Yamaji, A. *Super Light Water Reactors and Super Fast Reactors Supercritical-Pressure Light Water Cooled Reactors*; Springer Science & Business Media: Berlin, Germany, 2010; pp. 1–644.
7. Yamada, K.; Sakurai, S.; Asanuma, Y.; Hamazaki, R.; Ishiwatari, Y.; Kitoh, K. Overview of the Japanese SCWR Concept Developed under the GIF Collaboration. In Proceedings of the The 5th International Symposium on Supercritical Water-Cooled Reactors, Vancouver, BC, Canada, 13–16 March 2011.
8. Liu, Q.; Oka, Y. Single pass core design for a SuperFast Reactor. *Ann. Nucl. Energy* **2015**, *80*, 451–459. [CrossRef]
9. Uchikawa, S.; Okubo, T.; Kugo, T.; Akie, H.; Takeda, R.; Nakano, Y.; Ohunki, A.; Iwamura, T. Conceptual Design of Innovative Water Reactor for Flexible Fuel Cycle (FLWR) and its Recycle Characteristics. *J. Nucl. Sci. Technol.* **2007**, *44*, 277–284. [CrossRef]
10. Yamaji, A.; Oka, Y.; Ishiwatari, Y.; Liu, J.; Suzuki, M. Principle of rationalizing the criteria for abnormal transients of the Super LWR with fuel rod analyses. *Ann. Nucl. Energy* **2006**, *33*, 984–993. [CrossRef]
11. Han, C. *Code System for Core Design of Supercritical Water-Cooled Fast Reactor*; University of Tokyo: Tokyo, Japan, 2010; pp. 1–137.
12. Okumura, K.; Kugo, T.; Kaneko, K.; Tsuchihashi, K. *SRAC2006: A Comprehensive Neutronics Calculation Code System*; JAEA: Ibaraki, Japan, 2007; pp. 1–313.
13. Shibata, K.; Kawano, T.; Nakagawa, T.; Iwamoto, O.; Katakura, J.I.; Fukahori, T.; Chiba, S.; Hasegawa, A.; Murata, T.; Matsunobu, H.; et al. Japanese evaluated nuclear data library version 3 revision-3: JENDL-3.3. *J. Nucl. Sci. Technol.* **2002**, *39*, 1125–1136. [CrossRef]
14. Watts, M.; Chou, C. Mixed Convection Heat Transfer to Supercritical Pressure Water. In Proceedings of the 7th International Heat Transfer Conference, Munich, Germany, 6–10 September 1982.
15. Duderstadt, J.J.; Hamilton, L.J. *Nuclear Reactor Analysis*; John Wiley & Sons Inc.: New York, NY, USA, 1976; Chapter 12.
16. Kano, F.; Tsuchiya, Y.; Saito, N.; Okawa, M. Development of Materials for Supercritical-Water-Cooled Reactor. *Toshiba Rev.* **2004**, *59*, 60–63.
17. Guo, H.; Buiron, L.; Sciora, P.; Kooyman, T. Optimization of reactivity control in a small modular sodium-cooled fast reactor. *Nucl. Eng. Technol.* **2020**, *52*, 1367–1379. [CrossRef]