Analyses of Core Thermal-Hydraulic Characteristics for the 1,200 MWe SFR Breakeven Reactor

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1. Introduction

The Korea Atomic Energy Research Institute (KAERI) has been developing the Gen-IV sodium cooled fast reactor (SFR) since 2007 under a National Nuclear R&D Program. The goal of the Gen-IV SFR Development Project is to develop the conceptual design of Gen-IV SFR which requires sustainability, economic competitiveness, safety and reliability, proliferation resistance and physical protection.

The core thermal hydraulic design is performed to efficiently extract the core thermal power by distributing the appropriate sodium coolant flow according to the power of each assembly because the 1,200 MWe SFR core is composed of hundreds of ducted assemblies with hundreds of fuel rods.

This paper describes the coolant flow distribution to the assemblies and the coolant/fuel temperature calculations for conceptual design of 1,200 MWe SFR breakeven core.

2. Core configurations

The 1,200 MWe SFR breakeven core meeting the mission of Gen-IV SFR was designed based on the KALIMER-600 breakeven core which was developed under the national long-term nuclear R&D program[1].

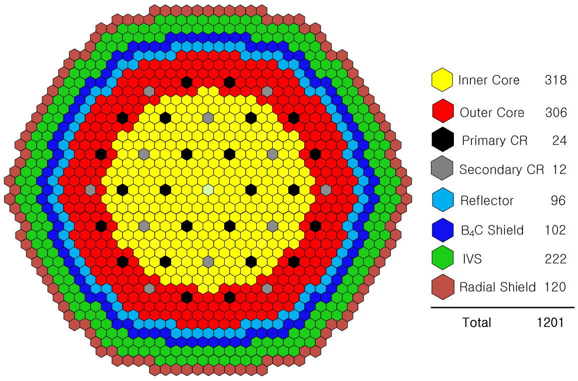
In order to enhance the proliferation resistance, this core eliminated blanket assemblies to prevent the possibility of producing and extracting the weapon-grade plutonium. Also, the core is determined as a homogeneous core of 2 regions with the different fuel enrichment relying on well proven technology in fuel fabrication and fuel performance.

Figure 1 shows the configuration of the 1,200 MWe SFR breakeven core. The core consists of 318 inner core assemblies and 306 middle core assemblies with 271 U-TRU-10%Zr ternary metal fuels.

Table 1 shows the basic design data and operation condition of the 1,200 MWe SFR breakeven core. The length of the fuel element and active core are 355.27 and 80.0cm, respectively. The fuel outer diameter is 8.7 mm and the cladding thickness is 0.595mm.

3. Thermal-hydraulic Design

Thermal-hydraulic design of 1,200 MWe SFR breakeven cores was performed using the SLTHEN code which was based on the SE2-ANL [2]. SE2-ANL is a multi-assembly, steady-state sub-channel analysis code for forced convection and it was successfully applied to thermal hydraulic core design and performance analyses of sodium cooled reactors.



* Another structural assembly instead of control rod was installed at the core centre

Figure 1 Configuration of the 1200 MWe SFR breakeven core.

Table 1 Basic design data and operation condition of 1200 MWe SFR breakeven core

Core Electric Power (MWe)	1200
Core Thermal Power (MWth)	3046.8
Plant Thermal Efficiency (%)	39.4
Coolant Mean inlet Temp. (℃)	390
Coolant Mean outlet Temp. (℃)	545
Fuel Type	U-TRU-10%Zr
Number of Fuel Assembly Types	2
Feed Fuel Enrichment (%)	13.19(IC)/16.83(OC)
Fuel smeared density (%)	75.0
Maximum Core Diameter (cm)	706.0
Duct Wall Thickness (mm)	4.0
Duct Inner Flat to Flat (mm)	172.46
Gap Distance between Ducts (mm)	4.00
Number of Pins per Assembly	271
Fuel Element Length (cm)	355.27
Active Length (cm)	80.0
Pin Outer Diameter (mm)	8.7
Cladding Thickness (mm)	0.595
Pin Pitch (mm)	10.30
Pin P/D ratio	1.184
Wire Wrap Diameter (mm)	1.40
Wire Wrap Pitch (cm)	20.49

The 1,200 MWe SFR breakeven core has 14 flow groups as shown in Figure 2. The total primary loop flow rate including bypass flow is 15,455 kg/s and non-grouped region flow fraction is about 4.57% of total flow rate.

Table 2 shows the flow rate per flow group and the maximum cladding mid-wall temperature with 2σ uncertainty at each flow group. The maximum cladding mid-wall temperature with 2σ uncertainty is estimated to be 650℃. It does not exceed the limit value for the

Mod.HT9 cladding which is expected to be greater than 650℃.

The hot assemblies of the inner and outer core were recorded at (2, 1) and (12, 5), respectively. Table 3 shows the information concerning hot assembly for the inner and outer core.

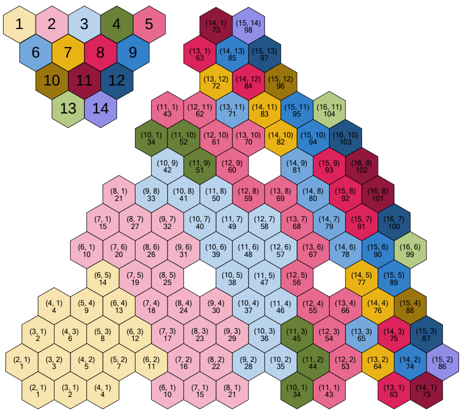


Figure 2 Assembly position and flow group of 1200 MWe SFR breakeven core

Table 2 Flow rate per flow groups and cladding mid-wall temperature of 1,200MWe SFR breakeven core

Flow Group No.	Assy Type	Assy. No.	Assy Flow rate (kg/s)	Group Flow rate (kg/s)	Fraction (%)	Cladding Midwall (2σ)(°C)
1	IC	28.10	78	2191	53.46%	650
2	IC	26.50	108	2862		650
3	IC	24.60	102	2509		650
4	IC	23.30	30	699		649
5	OC	27.10	84	2276	41.97%	650
6	OC	24.40	36	878		649
7	OC	22.90	36	824		648
8	OC	20.60	36	741		649
9	OC	18.70	36	673		649
10	OC	16.00	12	192		649
11	OC	15.00	18	270		648
12	OC	14.10	24	338		649
13	OC	12.80	12	153		648
14	OC	11.60	12	139		649

Table 3 Hot assembly position, assembly flow rate, and power.

Assembly Type	Assembly Position	Flow Group No.	Flow rate (kg/s)	Power (MWth)
IC	(2,1)	1	28.10	6.29
OC	(12,5)	5	27.10	5.55

4. Conclusion

The Core thermal-hydraulic characteristic analyses for the 1,200 MWe SFR breakeven core were performed. The core has 14 flow groups for the fuel assemblies, and the flow rate per assembly was assigned from 11.60 kg/s up to 28.10 kg/s.

The maximum cladding mid-wall temperature with 2σ uncertainty of IC/OC is estimated to be 650/650℃, respectively. The estimated bundle pressure drop is 0.15MPa with 20% uncertainty.

The results show that the conceptual design for 1,200 MWe SFR breakeven core satisfy the design requirement such as maximum cladding temperature and pressure drop.

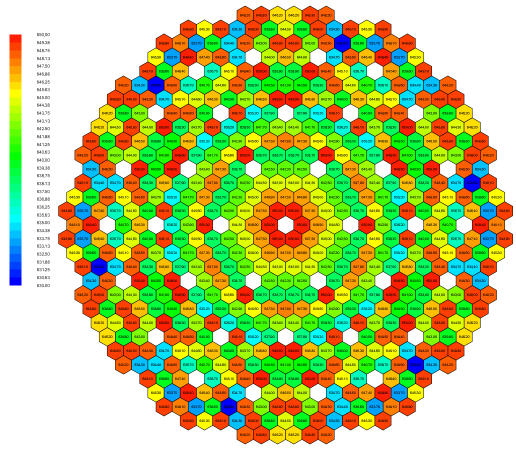


Figure 3 Maximum cladding mid-wall temperature with 2σ uncertainty distribution.

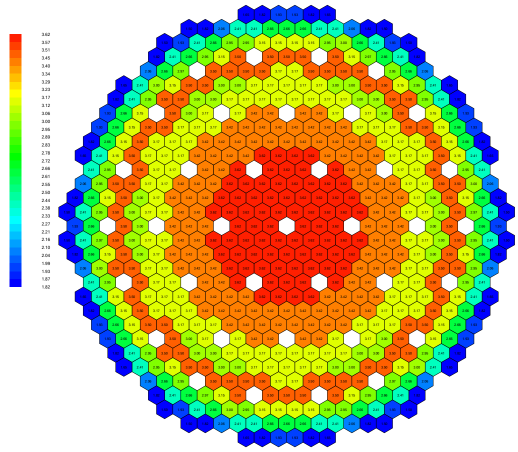


Figure 4 Core outlet coolant mean velocity distribution

ACKNOWLEDGEMENT

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REFERENCES

[1] D. H. Hahn et al., KALIMER-600 Conceptual Design Report, KAERI/TR-3381/2007.
[2] Yang W. S. and Yacout A. M. , Assessment of the SE2-ANL Code Using EBR-II Temperature Measurements, Proc. 7th International Meeting on Nuclear Reactor Thermal Hydraulics, NUREG/CP-0142, p. 2934, Saratoga Springs, l0-15 September, 1995.