Whole Core Thermal-Hydraulic Design of a Sodium Cooled Fast Reactor Considering the Gamma Energy Transport

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1. Introduction

Since a fuel cladding failure is the most important parameter in a core thermal-hydraulic design, the conceptual design stage only involves fuel assemblies. However, although non-fuel assemblies such as control rod, reflector, and B4C generate a relatively smaller thermal power compared to fuel assemblies, they also require independent flow allocation to properly cool down each assembly. The thermal power in non-fuel assemblies is produced from both neutron and gamma energy, and thus the core thermal-hydraulic design including non-fuel assemblies should consider an energy redistribution by the gamma energy transport. To design non-fuel assemblies, the design-limiting parameters should be determined considering the thermal failure modes. While fuel assemblies set a limiting factor with cladding creep temperature to prevent a fission product ejection from the fuel rods, non-fuel assemblies restrict their outlet temperature to minimize thermally induced stress on the upper internal structure (UIS).

This work employs a heat generation distribution reflecting both neutron and gamma transport. The whole core thermal-hydraulic design including fuel and non-fuel assemblies is then conducted using the SLTHEN (Steady-State LMR Thermal-Hydraulic Analysis Code Based on ENERGY Model) code [1]. The other procedures follow from the previous conceptual design [2].

2. Core Heat Generation

During a nuclear reaction in a sodium-cooled fast reactor, the major heat sources are neutron and gamma rays. In the previous conceptual design, the neutron transport is only evaluated and the gamma transport is neglected. This means that a gamma ray is assumed to be simultaneously generated and absorbed in the same position during the nuclear reaction. As a result, heat generation in the fuel assemblies is overestimated and reveals conservative flow allocations. On the other hand, the non-fuel assemblies have less thermal power than the real values.

To elucidate the heat generation rate in non-fuel assemblies, a detailed nuclear design including the gamma transport has been recently performed using the MCNP code, and its power distribution result is shown in Fig 1 [3]. To have a conservative thermal design, heat generation in the fuel assemblies comes from the previous conceptual design, and the non-fuel assemblies adopt the thermal power from the MCNP calculation. However, the control rod, reflector, and B4C assemblies are only involved in the present core thermal-hydraulic design. The other non-fuel assemblies such as IVS and radial shields are neglected because their heat generation rates are negligible, even when involving the gamma energy transport.

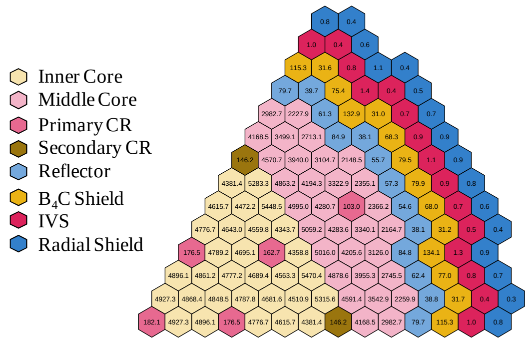

Fig. 1. Thermal power distribution considering both neutron and gamma transport.

3. Thermal-Hydraulic Design

In carrying out the core thermal-hydraulic design, several design criteria need to be met to assure proper performance and safety for the core and upper structure, where the design limits are highly related to temperature distribution in the fuel, cladding, and sodium under various operating conditions. For non-fuel assemblies, the typical thermal design criterion is used to minimize the temperature difference between neighboring assemblies, which is highly related to a thermal striping failure of the upper internal structure. Therefore, each assembly outlet temperature should keep as close to the core average value as possible.

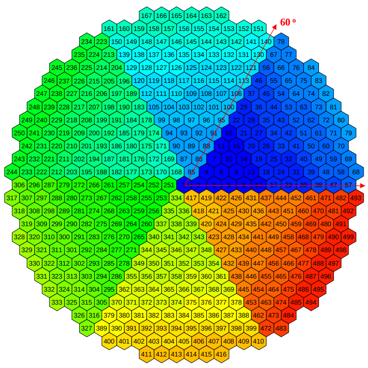

Fig. 2. Assembly index number for the SLTHEN code.

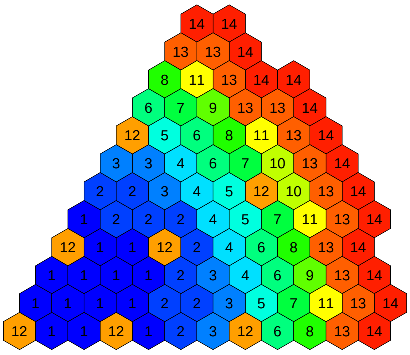

Fig. 3. Flow grouping diagram for the whole core analysis.

Table I: Flow distribution results

Flow Group No.	Assembly Type	Assembly Flow (kg/s)	Group Flow (kg/s)	Fraction (%)
1	IC	26.52	1750.06	20.92
2	IC	24.51	1323.52	15.82
3	IC	22.75	682.53	8.16
4	OC	26.91	807.16	9.65
5	OC	24.68	592.26	7.08
6	OC	22.62	678.62	8.11
7	OC	19.05	457.11	5.46
8	OC	17.15	308.69	3.69
9	OC	15.38	184.58	2.21
10	OC	13.82	165.80	1.98
11	OC	13	312	3.73
12	CR	1.44	36.08	0.43
13	Ref	0.53	38.48	0.46
14	B4C	0.57	44.18	0.53
Etc.	Inter assembly + IVS+ Radial shield etc.		985.05	11.77
Total			8366.1	100

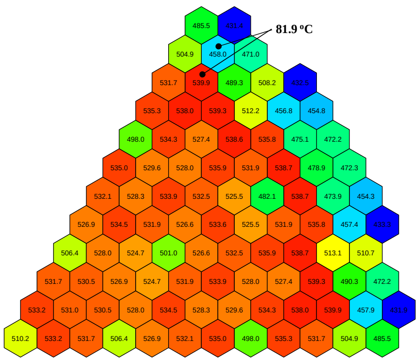

Fig. 4. Outlet temperature distribution.

In addition to the conceptual design with fuel assemblies, the whole core thermal-hydraulic design including non-fuel assemblies is performed. Considering the hexagonal repetition with a period of 60^o, the design of only a 1/6 core part was provided with a faster calculation time. Since fuel assemblies generate a large amount of thermal energy, they also need a massive coolant flow. This means that the heat transfer in a fuel assembly is purely convective, and the assembly boundary conditions rarely affect the temperature distribution. However, the non-fuel assemblies have much a smaller coolant flow, and the heat transfer between neighboring assemblies is highly important. Therefore, the present core thermal-hydraulic design utilizes the whole core analysis as shown in Fig. 2. The flow grouping with a 1/6 core and repetitive application to the whole core are performed based on the assembly power rate, as shown in Fig 3. The non-fuel assemblies are made using a single group dependent on their type.

The representative core flow grouping results for the whole core analysis are detailed in Table I. 14 flow groups in total were specified for the present core design, as shown in this table, where the inner core fuel assemblies utilize 3 flow groups, and the outer core fuel assemblies utilize 8 flow groups. Most of the flow is distributed to the fuel assemblies owing to the huge portion of thermal power. The control rod, reflector, and B4C assemblies have flow allocations of 0.43, 0.46 and 0.53%, respectively, which are significantly smaller than those of the fuel assembly but cannot be neglected. The remaining flow rates for inter-assembly, IVS, radial shield, etc. were about 11.77%. The entire core region was kept below the limiting temperatures. The outlet temperature distribution of the whole core is displayed in Fig. 4, and indicates the neighboring assemblies having the most temperature difference. The maximum temperature difference exiting in adjacent assemblies was 81.9^{o}C, which assure the structural integrity of the upper internal structures.

4. Conclusions

A whole core thermal-hydraulic analysis including fuel, control rod, reflector and B4C assemblies was performed based on the neutron and gamma energy transport. The design limiting criteria for fuel and non-fuel assemblies are the cladding mid-wall temperature and assembly outlet temperature, respectively. The results show that the present design provides enough thermal margins considering the temperature distribution and remaining flow quantity.

ACKNOWLEDGEMENT

This study was performed under the nuclear R&D program sponsored by the Ministry of Education, Science and Technology of Korea.

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