Monte Carlo Simulation on the Preliminary Conceptual Design for a Prototype Experimental Sodium-cooled Fast Reactor

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1. Introduction

Management of spent nuclear fuel has been a technical issue worldwide for the last several decades. Intensive R&D activities have been done for the Generation IV fast reactors with viable goals; such as improvements in safety and economics, fuel cycle sustainability with proliferation resistance. The prototype reactor design and construction is planned with the sodium-cooled fast reactor (SFR) type at our country for an option of waste incineration.

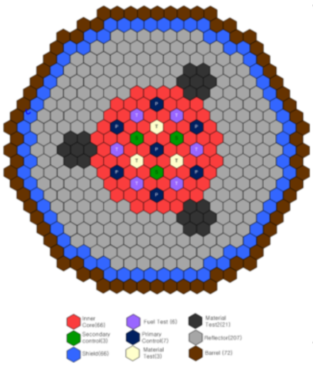
At Kyung Hee university, the early stages of conceptual design for Multipurpose Experimental Sodium-cooled Fast Reactor (MESOF) was done in the reactor size of prototype. In that study, an operational validation was checked under various experimental conditions for fuels and materials. A code system of TRANSX, DANTSYS, and REBUS-3 was used in the conceptual reactor core design [1].

In this work, a preliminary conceptual design for MESOF was compared with MCNPX code results for the design verification. Reliable limit and error range of this code system was also searched for.

2. Benchmark Problems and Modeling

2.1 Configuration of the Reference Model

A preliminary conceptual design for MESOF was done based on ABTR core model. [1] Unlike ABTR with TRU fuel loading, MESOF was loaded with LEU fuel. In MESOF, assembly pitch and number of pins were increased for neutron economy. The pin height was also increased to 87 cm and number for reflector assemblies of 207 was much larger than that of ABTR.



Reactor power, MWth	300
Cycle length, effective full power days	120
Number of driver assemblies	72
Fuel form	U-10Zr
U enrichment, %	19.5
Initial core loading (heavy metal /Fissile), MT	6.16 / 1.20
Power density (average/peak), kW/l	212.23 / 414.87
Burnup reactivity swing (%ΔK)	0.88
Power peaking factors (BOEC/EOEC)	1.95475/ 2.02298
Peak linear power, kW/m	34.67
Linear power limit, kW/m	44.0
Core average flux, 10^{15} n/cm^2-s	1.69
Material Test Assembly flux, 10^{15} n/cm^2-s	1.94
Fast flux fraction	0.69

Fig. 1. Radial core layout (left), Performance Characteristics & Kinetic Parameters (right) of MESOF Design-S

Another different feature is an installation of autonomous loops (as independent fuel test loops) placed in the outer reflector zone. Separate and independent irradiation loops are required for the irradiation experiment of fuels under different coolant or at different operation conditions. In a reference model, all fuel test assemblies was loaded with TRU fuel(Spent fuel), whereas reflector and material test assemblies are loaded with HT-9. In an autonomous loop, duct is filled with Na coolant without fuel.

2.2 Benchmark Problems

The configuration and feature of MESOF was modeled by using MCNPX 2.6.0 code. In a previous design stage, REBUS-3 has only a nodal diffusion module, DIF-3D. It was known that this module showed relatively large differences in k-effective compared with transport theory code module, VARIANT. A benchmark study showed results of core model of ABTR from DIF-3D, ERANOS, VARIANT and MCNP. MCNP or transport code model showed higher k-eff by about 2-3% compared with diffusion models [2].

In this study, MCNPX code is used for the validation for depletion effects. The following figure 2 represents a part of MCNPX model for MESOF core.

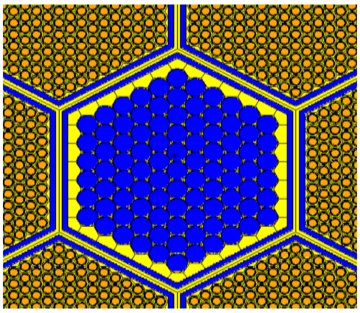


Fig. 2. Part of MCNPX model for a MESOF

3. Comparison of REBUS with MCNPX

3.1 K-effective Calculation

K-effective value was compared in order to verify MESOF modeling. Comparison was done for three kinds of similar core models which are different in the existence of autonomous loops and layout of FTAs and MTAs.

Table 1. Comparison of K-effective

		K-effective (BOC)	Difference (%)
Design-R	REBUS-3	1.01139	2.287%
	MCNPX	1.03426	
Design-M	REBUS-3	1.00610	2.023%
	MCNPX	1.02633	
Design-S	REBUS-3	1.00385	2.003%)
	MCNPX	1.02388	

Diffusion model in REBUS showed big differences in k-eff by about 2%. DIF-3D underestimated consistently compared with MCNPX. This error seems to come from over-estimation of neutron leakage at the core boundary in this small size core of MESOF.

3.2 Flux and Power Distribution

The Mesh Tally card of MCNPX was used in order to recognize the whole core flux distribution of MESOF Design-S. By using the F4 Tally, flux values at each node were calculated [3]. Fluxes at three locations; MTA (material test assembly), active core, and separate test loop were compared in the following table. The Fig.3 showed assembly-wise power distribution for the core.

Table 2. Comparison of Flux (MTA, Active Core and Separate Test Loop)

	REBUS-3	MCNPX (Normalized to Core Avg.)
MTA average flux (10^{15}n/cm^2-sec)	2.54	2.40
Active Core average flux (10^{15}n/cm^2-sec)	1.72	1.72
Separate FTL average flux (10^{15}n/cm^2-sec)	0.643	0.670

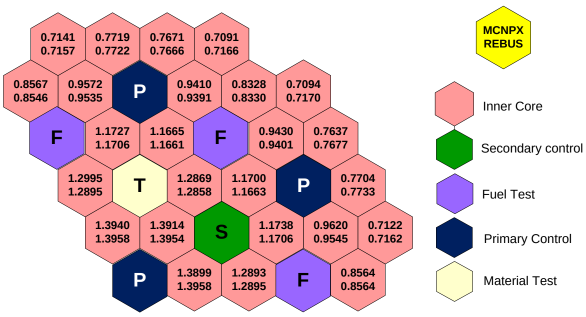


Fig. 3. Normalized Power Distribution for MCNPX & REBUS

The results in Table 2 shows that MCNPX calculated specific fluxes, normalized to core average, are similar to those of REBUS-3. The normalized power distributions of peripheral assemblies are lower than the central assemblies. The difference in power distributions estimated by two codes lies within 1% error range at all locations.

3.3 Depletion Calculation

By using the CINDER90 module, burnup calculation was done in MCNPX. Calculation results were compared with 120 days burnup results from REBUS-3.

Table 3. Comparison of K-effective and Enrichment

		K-effective	Enrichment of U-235
REBUS-3	BOC	1.00385	18.589 w/o
	EOC	0.99482	18.108 w/o
MCNPX	BOC	1.02388	18.585 w/o
	EOC	1.01587	18.083 w/o

Burnup difference was 0.056 GWd/MTU, the k-effective in BOC and EOC showed the same differences with the previous results in Section 3.1. Differences of k-effective were about 2%. Differences in U-235 enrichment at EOC were negligibly small.

3.4 Control Assembly Worth Calculation

Critical control rod position was searched for MESOF Design-A, and the result of calculated control rod worth was compared using MCNPX and REUBS-3 at critical control rod position. 8 positions were simulated with assumption that all the primary control assemblies move together. After that, rod worth was calculated at 7 positions when the central assembly was withdrawn. The assumed delayed neutron fraction was 0.006391 after estimation by MCNP. The results of the calculations were shown in figures 4 and 5.

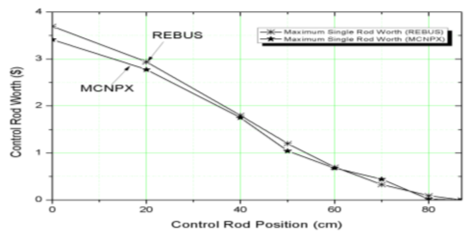


Fig. 4. Comparison of Reactivity of Primary Control System at BOC

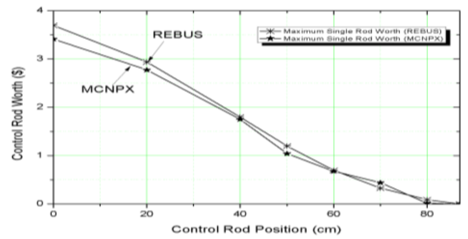


Fig. 5. Comparison of Reactivity of Primary Control System at EOC

REFERENCES

[1] Hee-Hun Lee and Myung-Hyun Kim, " Preliminary Conceptual Design for a Prototype Experimental Sodium-cooled Fast Reactor", Transactions of the Korean Nuclear Society Spring Meeting, Jeju, Korea, May, 2012.
[2] Allen, K., et al., “Benchmark of Advanced Burner Test Reactor model using MCNPX 2.6.0 and ERANOS 2.1”, Progress in Nuclear Energy, 2011.
[3] D.B. Pelowitz, MCNPX^{TM} User's Manual Version 2.6.0, April, 2008.