Uncertainty Assessment of the Core Thermal-Hydraulic Analysis Using the Monte Carlo Method

Sun Rock Choi *, Jaewoon Yoo, Dae-Hyun Hwang, Sang-Ji Kim
Korea Atomic Energy Research Institute, 1045 Daedeok-daero, Yuseong-gu, Daejeon 305-503, Republic of Korea
**Corresponding author: choisr@kaeri.re.kr*

1. Introduction

In the core thermal-hydraulic design of a sodium cooled fast reactor, the uncertainty factor analysis is a critical issue in order to assure safe and reliable operation. The deviations from the nominal values need to be quantitatively considered by statistical thermal design methods. The hot channel factors (HCF) were employed to evaluate the uncertainty in the early design such as the CRBRP [1]. The improved thermal design procedure (ISTP) calculates the overall uncertainty based on the Root Sum Square technique and sensitivity analyses of each design parameters [2]. Another way to consider the uncertainties is to use the Monte Carlo method (MCM) [3]. In this method, all the input uncertainties are randomly sampled according to their probability density functions and the resulting distribution for the output quantity is analyzed. It is able to directly estimate the uncertainty effects and propagation characteristics for the present thermal-hydraulic model. However, it requires a huge computation time to get a reliable result because the accuracy is dependent on the sampling size.

In this paper, the analysis of uncertainty factors using the Monte Carlo method is described. As a benchmark model, the ORNL 19 pin test is employed to validate the current uncertainty analysis method [4]. The thermal-hydraulic calculation is conducted using the MATRA-LMR program which was developed at KAERI based on the subchannel approach. The results are compared with those of the hot channel factors and the improved thermal design procedure.

2. Methods and Results

2.1 Thermal-Hydraulic Analysis

The MATRA-LMR has been developed for the thermal-hydraulic analysis of the sodium cooled fast reactor where design limits are highly related to temperature distribution in fuel, cladding and sodium under various operating conditions [5]. For example, the maximum temperatures of fuel pins should be lower than their melting point. Assuming the uniform radial heat generation in an assembly, the fuel pin with the maximum temperature is located in the central position of the assembly. Therefore the detailed thermal analysis is conducted for the central fuel pin. Figure 1 show the axial power and temperature profiles of the central fuel for the ORNL 19 pin tests. It is clearly shown that the highest coolant temperature is found at the exact end of the heat generation profile. On the other hand, the highest fuel centerline is slightly above the fuel midplane because the temperature at the fuel centerline is proportional to both coolant temperature and heat generation rate.

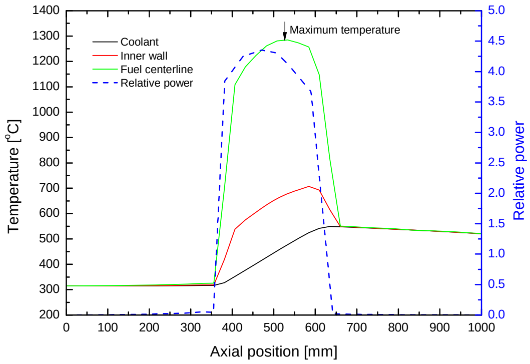

Fig. 1. Temperature and heat generation distribution along the axial location at the central fuel

2.2 Monte Carlo Method

The statistical thermal design procedures are categorized into two types. One method uses the statistical factors and their combination. The HCF and ISTP are included in this method which only considers the standard deviation and size of input parameters. The other method employs the random number generation based on the probability density functions (PDF) which combine into the thermal-hydraulic model. This method has its value when linearization of the model provides an inadequate result, or the PDFs depart from a Gaussian distribution or a t-distribution due to significant asymmetry. The PDFs for the input quantities propagate through the thermal-hydraulic model F(**x**) to offer the overall PDF for the output quantity as shown in fig. 2.

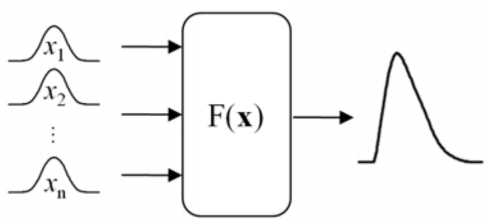

Fig. 2. Schematic diagram of the PDF propagation to evaluate the uncertainty for the output quantity

2.3 Uncertainty Factors

The thermal-hydraulic design involves various engineering uncertainties, often arisen from fabrication, measurement, instrumentation and calculation. The uncertainty factors in this paper are based on the values (3σ) for Integral Fast Reactor [6].

A. Flow rate through an assembly: The uncertainty in flow rate is found to be 7%, generally arisen from flow maldistribution in the inlet plenum, orifice uncertainties and loop flow imbalance.

B. Heat transfer coefficient: The uncertainty in the heat transfer coefficient between fuel rods and coolant is determined by experimental data and an empirical correlation. In this work, uncertainty of 16% is used.

C. Cladding thickness: The fabrication tolerance is ±0.0005 in. from which an uncertainty of 3% is expected.

D. Cladding thermal conductivity: The uncertainty in the thermal conductivity of cladding is due to experimental errors and irradiation effect. HT-9 has an uncertainty of about 7%.

E. Fuel thermal conductivity: The fuel conductivity is highly dependent on fuel density and content. In addition, no data are available for irradiated fuel. Therefore, the uncertainty in the thermal conductivity is expected to be 25%.

F. Fuel diameter: The metallic fuel diameter has a tolerance of ±0.003 in., which results in an uncertainty of 2%. Since linear power is proportional to the square of a fuel diameter, the actual uncertainty is 4% for the fuel heat generation rate.

G. Fissile fuel concentration: The uncertainty in fissile fuel concentration is 0.5 wt. % considering the fabrication tolerance. However an uncertainty of 1% is used conservatively.

2.4 Results and Discussions

The statistical uncertainty is evaluated by incorporating the Monte Carlo sampling into the thermal-hydraulic analysis code MATRA-LMR. About 30,000 calculations are conducted to eliminate the sampling size effect and verify the present uncertainty assessing method. Figure 3 represents the overall distribution of the maximum fuel centerline temperature with a comparison of the improved thermal design procedure. The average and standard deviation is determined to be 1008 and 38.87 ℃ respectively which are slightly larger than those of the other methods as shown in Table I. Figure 3 also exhibits the endpoint of at least 95% probability indicated by vertical lines, reflecting the asymmetry propagation of the input uncertainties in the Monte Carlo method.

The calculated results of the hot channel factor are close to those of the improved thermal design procedure owing to their similar statistical analyses. In the present analysis, it is clear that the dominant input parameter is the fuel thermal conductivity considering its uncertainty value and temperature increase.

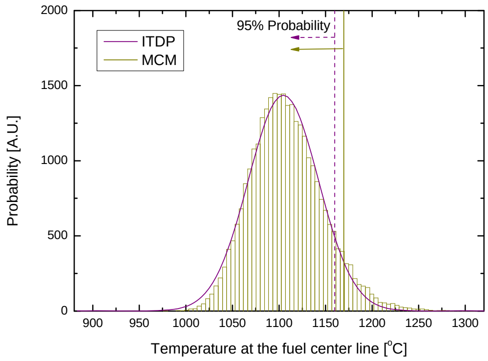

Fig. 3. Distribution histogram of the fuel centerline temperature.

Table I: Calculation Results

	Average (℃)	Standard deviation (℃)
HCF	1104.759	37.66
ISTP	1104.759	37.65
MCM	1108.265	38.87

3. Conclusions

The statistical thermal design procedure for a sodium fast reactor is developed based on the Monte Carlo method combined with subchannel analysis code. The calculated uncertainty for the ORNL 19 pin tests is slightly larger than those of the other methods and also exhibits the asymmetrical characteristics.

REFERENCES

[1] Mario D. Carelli and Aaron J. Friendland, Hot Channel Factor for Rod Temperature Calculations in LMFBR Assemblies, Nuclear Engineering and Design, Vol. 62, p. 155, 1980.

[2] H. Chelemer, L.H. Boman, D.R. Sharp, Improved Thermal Design Procedure, Westinghouse, WCAP 8568, 1975.

[3] JCGM 101:2008, Evaluation of measurement data — Supplement 1 to the Guide to the expression of uncertainty in measurement — Propagation of distributions using a Monte Carlo method.

[4] M.H. Fontana, R.E. MacPherson, P.A. Gnadt and L.F. Parsly, Temperature distribution in the duct wall and at the exit of a 19-rod simulated LMFBR fuel assembly (FFM Bundle 2A), Nuclear Technology, Vol. 24, p. 176, 1974.

[5] W. S. Kim, Y. G. Kim, Y. J. Kim, A Subchannel Analysis Code MATRA-LMR for Wire Wrapped LMR Subassembly, Annals of Nuclear Energy," Vol. 29, p. 303, 2002.

[6] J. Y. Ku, L. K. Chang, D. Mohr, Analysis of IFR Driver Fuel Hot Channel Factors, the Fourth International Topical Meeting on Nuclear Thermal Hydraulics, Operation and Safety, 1994, Taipei, Taiwan.