Sensitivity and Uncertainty Analysis in Best-Estimate Modeling for PGSFR Under ULOF Transient

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Abstract. In this research, uncertainty analyses for multiple safety parameters were performed for Unprotected Loss of Flow (ULOF) for the Prototype Gen-IV Sodium-cooled Fast Reactor (PGSFR) by using the PArallel Computing Platform IntegRated for Uncertainty and Sensitivity analysis (PAPIRUS). The objective of the global uncertainty analysis is to evaluate all safety parameters of the system in the combined phase space formed by the parameters and dependent variables. The uncertainty propagation was performed by mapping the uncertainty bands of the model parameters through the MARS-LMR to determine the distributions for the fuel centerline, cladding, and coolant temperatures. The Best Estimate Plus Uncertainty (BEPU) analysis adopted for uncertainty quantification of the code predictions has been performed through a statistical approach where the Figure of Merit (FOM) is evaluated multiple times by using several combinations of parameters that are randomly generated according to their distributions. The statistical approach of uncertainty quantification is known to be very powerful for estimating response distributions, but sometimes inapplicable owing to demanding calculation requirements. In this research, Wilks' formula was used to estimate the 95% probability value of the FOM from a limited number of code calculations. This paper also introduces the application of data assimilation in best-estimate modeling to improve the prediction of the reactor system performance by refining various sources of uncertainties through model calibration technique. An inverse problem was formulated based upon Bayes theorem and solved to estimate the posteriori distributions of parameters.

Key Words: PGSFR, Uncertainty Analysis, PAPIRUS, ULOF

1. Introduction

A Prototype Gen-IV Sodium-cooled Fast Reactor (PGSFR) is a 150-MWe pool-type fast reactor designed using U-TRU-Zr metal fuel. There are several Design Extension Condition (DEC) events of PGSFR, such as unprotected transient over power (UTOP), unprotected loss of flow (ULOF), unprotected loss of heat sink (ULOHS), large partial subassembly blockage, large steam generator tube rupture (SGTR), large sodium leak and station black out (SBO). In this research, the ULOF accident was selected as the target scenario for the best estimate uncertainty analysis. ULOF is caused by the loss of core cooling capability owing to pumping failure of the primary pump and no leaking coolant unlike pressurized water-cooled reactor (PWR).

The quality of the prediction will impact reactor safety of the PGSFR through the introduction of the safety margins on the reactor design to ensure a proper operation. The best estimate plus uncertainty (BEPU) analysis [1], [2] adopted for uncertainty quantification of the code predictions has been performed through a statistical approach where the figure of merit (FOM) is evaluated multiple times by using several combinations of parameters that are randomly generated according to their distributions. The statistical approach of uncertainty quantification is powerful methodology for estimating response distributions, but inapplicable if the calculations are too demanding. In order to avoid the demanding calculation requirements, Wilks' formula [3] was used when estimating the 95% probability value of the FOM.

2. Methodology

The objective of the global uncertainty analysis is to evaluate all the safety parameters of the system in the combined phase space formed by the parameters and dependent variables. The methods for uncertainty analysis are based on statistical or deterministic procedures. Deterministic methods are used for linear systems while statistical methods are used for nonlinear systems. The deterministic approach of the uncertainty propagation utilizes the Taylor series expansion of the response around the nominal parameter values. The various moments of the random variables can be obtained by integrating the Taylor series expansion of the random variables over the unknown joint probability distribution for the parameters. The statistical approach of the uncertainty propagation is based on sampling variables from the possible values of the parameters. More specifically, sampling-based uncertainty propagation involves the following steps:

1. Determine important or the most influential parameters. Define the subjective dist ributions for characterizing the uncertain parameters.
2. Use the distributions to generate multiple samples.
3. Use each parameter sample to perform model calculations that then generate resp onse distributions
4. Perform an uncertainty analysis based on the response distributions obtained in S tep 3

The model identification and ranking table (MIRT) was developed based on the Phenomena Identification and Ranking Table (PIRT) developed for the SFR reactor design at Korea Atomic Energy Research Institute (KAERI). Developing the MIRT is the process of constructing the models used to calculate the safety parameters, e.g., coolant temperature, for certain phenomena. In the ULOF, there are fifteen models for the reactor core, with four models for the PHTS and IHTS, respectively. The uncertainty range of each parameter indicates 2σ deviation and are determined based on the literature and expert opinions. Uncertainties were quantified for the FOM for ULOF scenario using the parameter sample distributions obtained from Table 1. Brief introduction to the models are as follows.

1. Fuel thermal conductivity

To check the behavior of the nuclear fuel, the temperature of the fuel must be predicted. The important parameter related to the behavior of nuclear fuel is the fuel thermal conductivity. The uncertainty range of this parameter is $\pm$ 0.29 W/m·K with a normal distribution based on the SCDAP/RELAP5/MOD3.3 code manual (NUREG/CR-6150, vol. 4, Rev.2) [4].

2. Modified-Schad correlation

A modified-Schad correlation is used as the heat transfer model for the rod bundle for the reactor core within the MARS-LMR code. The value of the Nusselt number (Nu) is a function of the geometry of the fuel bundle and Peclet number (Pe). The uncertainty range of this parameter is $\pm$10% with normal distribution based on the Westinghouse report discussing the heat transfer correlation for analysis of CRBRP assemblies [5].

$$\mathrm{Nu} = [-16.15 + 24.96(P/D) - 8.55(P/D)^2]Pe^{0.3} \quad (1)$$
$$(1.1 \leq P/D \leq 1.5, \qquad 150 \leq Pe \leq 1000)$$
$$\mathrm{Nu} = 4.496[-16.15 + 24.96(P/D) - 8.55(P/D)^2] \ \ (Pe \leq 150) \quad (2)$$

where P/D indicates the pitch-to-diameter.

3. Sodium density reactivity

The change in the reactivity of the density of sodium as a coolant in the reactor core is one of the important parameters related to reactivity feedback. The uncertainty range of this parameter is $\pm$32.6% with uniform distribution. This is based on the BFS-73-1 critical experiment at the BFS-1 facility at IPPE (Institute of Physics and Power Engineering) in Russia for developing PGSFR core design and simulation by using the MC^2-3/DIF3D code [6].

4. GP strain coefficient

The GP supports the core subassemblies and acts as the plenum for sodium to pass through the core assemblies [7]. During the ULOF, the thermal expansion of the materials in the core is also important for determining reactivity feedback. In the MARS-LMR code, this parameter is calculated as a component of the radial expansion reactivity. The uncertainty range of this parameter is $\pm$10% uniform distribution determined following expert opinions.

5. ACLP strain coefficient

The ACLP is also a structure for supporting the core subassemblies above the core. The ACLP strain coefficient is calculated as a component of the radial expansion reactivity with the GP. The uncertainty range of this parameter is $\pm$10% uniform distribution determined following expert opinions.

6. Core radial expansion reactivity coefficient

In the MARS-LMR code, the calculation determining core radial expansion reactivity is provided in Eq. 3 below [8]. The expansion of the materials as well as the core radial expansion reactivity coefficient are considered as follows.

$$\delta\rho^{radial} = \alpha^T\left(\sum_i \xi^i_{ACLP} W^i_{ACLP} + \sum_j \xi^i_{GP} W^i_{GP}\right) \tag{3}$$

where i and j indicate the heat structure node of the ACLP and GP, respectively; ξ and W are the strain and weighting factors, and α^T is the core radial expansion reactivity coefficient. The uncertainty range of this parameter is $\pm$10% uniform distribution obtained based on expert opinions.

7. Fuel density reactivity

The change in fuel density is considered as one of the parameters related to the fuel axial expansion reactivity. In the MARS-LMR code, the table of fuel density according to the temperature can be entered. The uncertainty range of this parameter is $\pm$10% uniform distribution following expert opinions.

8. Cladding strain coefficient

The cladding strain coefficient is also considered with fuel density as a component of the fuel axial expansion reactivity. The uncertainty range of this parameter is $\pm$10% uniform distribution following expert opinions.

9. Fuel axial expansion reactivity coefficient

During the ULOF, the fuel expansion occurs in the axial direction as well as the radial direction. Additionally, the ULOF is related to the change in the total reactivity. The calculation of the fuel axial expansion reactivity is as follows:

$$\delta\rho^{axial} = \sum_j C_j\left(1 - D_j^{t=0}/D_j^{t=t}\right) \tag{4}$$

where *j* indicates the node of the fuel in the axial direction and t is the time, *C* and *D* indicate the reactivity at each node of fuel and fuel density, respectively. The uncertainty of this parameter is $\pm 10\%$ normal distribution having the same reference as the sodium density reactivity [6].

10. Fuel axial expansion reactivity coefficient

Considering the amount of expansion of the CRDL, the thermal expansion coefficient of the CRDL is determined in the MARS-LMR code. The uncertainty range of this parameter is $\pm 10\%$ uniform distribution according to the expert opinions.

11. RV expansion reactivity coefficient

In the CRDL expansion model, the RV expansion is also considered and calculated based on the thermal expansion coefficient of RV. This parameter affects the reactivity of this system returning the positive reactivity feedback because of the extraction of the control rods. The uncertainty range of this parameter is $\pm 10\%$ uniform distribution according to the expert opinions.

12. Control and shutdown rod worth

Control and shutdown rod worth is typically considered as the one of the negative reactivity feedbacks. The uncertainty range of this parameter is $\pm 19.8\%$ normal distribution having the same reference as the sodium density reactivity [6].

13. Doppler reactivity

The Doppler effect is also determined as the one of the negative reactivity feedbacks and related to the inherent safety of the NPPs. The uncertainty of this parameter is $\pm 30\%$ normal distribution and having the same reference as the sodium density reactivity.

14. HT-9 thermal conductivity of duct

In this research, the thermal conductivity of HT-9 material, which consists of only the duct, is only considered except for that of the sodium. The uncertainty of this parameter is $\pm 10\%$ uniform distribution following expert opinions.

15. Wire-wrapped pressure drop model

The simplified Cheng and Todreas (1986) model is used as the friction factor for the rod bundles in the core as follows [9], [10]:

For the laminar region where $\mathrm{Re} < Re_L$,

$$\mathrm{f} = C_{fL}/Re \tag{5}$$

For the turbulent region where $\mathrm{Re} > Re_T$,

$$\mathrm{f} = C_{fT}/Re^{0.18} \tag{6}$$

For the transition region where $Re_L \le Re \le Re_T$,

$$\mathrm{f} = \left(C_{fL}/Re\right)(1-\psi)^{1/3} + \left(C_{fT}/Re^{0.18}\right)\psi^{1/3} \tag{7}$$

where,

$Re_L = 300\left(10^{1.7(P/D-1.0)}\right)$
$Re_T = 10{,}000\left(10^{0.7(P/D-1.0)}\right)$
$\psi = \log(Re/Re_L)/\log(Re_T/Re_L)$
$C_{fL} = (-974.6 + 1612.0(P/D) - 598.5(P/D)^2)(P/D)^{0.06-0.085(P/D)}$
$C_{fT} = (0.8063 - 0.9022(\log(P/D)) - 0.3526(\log(P/D))^2)(P/D)^{9.7}(H/D)^{1.78-2.0(P/D)}$

The uncertainty range of this equation in Cheng and Todreas (1986) is $\pm 30\%$ for the laminar region and $\pm 14\%$ for the turbulent region respectively. However, the uncertainty range was derived by Chen et al. (2014) at $\pm 15\%$ (1σ) for all regions. Based on the literature and expert opinions, the uncertainty range of this parameter is $\pm 30\%$ with a normal distribution.

16. Primary pump coastdown curve

The coastdown curve is one of the important factors directly related to the primary flowrate during an accident. In the MARS-LMR code, the velocity table of the pump according to the time can be entered as an input. The uncertainty range of this parameter is $\pm 10\%$ uniform distribution obtained following expert opinions.

17. Core inlet form loss

The core inlet form loss is also related to the primary flowrate similar to the primary pump coastdown curve. The uncertainty range of this parameter is from 0.5 to 2.0 and is the correction factor based on log-uniform distribution according to OECD report NEA/CSNI/R (97) 35 Vol. 2 [5].

18. Heat capacity of reactor vessel material

The material of the reactor vessel is SS 316 and this parameter is related to the internal structure heat transfer. The uncertainty of this parameters is $\pm 10\%$ uniform distribution determined by expert opinions.

19. Convection in PHTS and IHTS

Except for the core region, the heat transfer model for the PGSFR primary system is considered an Aoki correlation [11]. This correlation is derived from the application of eddy diffusivities for heat and momentum transfers. The correlation is shown in Eq. 8 below. The uncertainty range of this parameter is $\pm 20\%$ normal distribution follwing expert opinions.

$$\mathrm{Nu} = 6.0 + 0.025\{0.014Re^{1.45}Pr^{1.2}(1 - e^{-71.8}Re^{-0.45}Pr^{-0.2})\}^{0.8} \tag{8}$$

20. Wall roughness of IHX shell side

Although the wall roughness related to the pressure drop of the IHX shell side is determined based on criteria discussed in the OECD report [4], the uncertainty range applied to the

ATHLET code did not match the range of the initial value in the MARS-LMR code. Therefore, the uncertainty range of this parameter is from 10^{-5} to 2.0×10^{-4} with uniform distribution following expert opinions.

21. Spacer grid form loss

To determine the pressure drop of the IHX shell side, the spacer grid form loss considers the wall roughness. Additionally, the uncertainty range is from 0.5 to 1.5 with uniform distribution having the same reference as the wall roughness.

22. Convection in IHX shell side

The heat transfer model for the IHX shell side is based on the Graber-Reiger correlation as shown in Eq. 9 below. The uncertainty range of this correlation is $\pm 12.2\%$ with a normal distribution based on Graber and Reiger (1973) [12].

$$\mathrm{Nu} = \left(0.25 + 6.2(P/D) + \left(-0.007 + 0.032(P/D)\right)Pe^{0.8-0.024(P/D)}\right) \tag{9}$$

The FOM includes all parameters used to judge the relative importance of the phenomena. Based on expert opinions, the FOM for the ULOF of the PGSFR is selected to be the fuel solidus temperature (1250 ℃), clad temperature (1075 ℃), and sodium boiling temperature. In the case of the sodium boiling temperature, the thermal margin of vaporization, which is the difference between saturation temperature and coolant temperature at the channel exit of hot pin, was considered and the saturation temperature was determined to be approximately 950 ℃.

Table 1 Model identification and ranking table (MIRT)

System	Phenomena		Related model	Distribution function	Uncertainty band [2σ]
Reactor core	Fuel rod heat transfer	F1	Fuel conductivity	Normal	±0.58 W/m·K
		F2	Convection	Normal	±20%
	Coolant density effect	F3	Sodium density reactivity	Normal	±32.6%
	Core radial expansion	F4	GP strain coefficient	Uniform	±10%
		F5	ACLP strain coefficient	Uniform	±10%
		F6	Reactivity coefficient	Normal	±30.6%
	Axial expansion of fuel and	F7	Fuel density reactivity	Uniform	±10%
		F8	Cladding strain coefficient	Uniform	±10%

	cladding	F9	Reactivity coefficient	Normal	±30.6%
	Control rod drive line expansion	F10	CRDL expansion reactivity coefficient	Uniform	±10%
		F11	RV expansion reactivity coefficient	Uniform	±10%
		F12	Control and shutdown rod worth	Normal	±19.8%
	Doppler reactivity feedback	F13	Doppler reactivity	Normal	±30%
	Inter assembly heat transfer	F14	HT-9 conduction	Uniform	±10%
	Core pressure drop	F15	Friction model	Normal	±30%
Primary heat transfer system (PHTS)	Pump Coastdown	F16	Coastdown curve	Uniform	±10%
	Natural convection	F17	Core inlet form loss	Log-uniform	0.5 – 2.0
	Internal structure heat transfer	F18	Heat capacity	Uniform	±10%
		F19	Convection	Normal	±20%
Intermediate heat transfer system (IHTS)	Tube side heat transfer	F20	Convection	Normal	±20%
	Shell side pressure drop	F21	Wall roughness	Uniform	10^{-5} – 2.0×10^{-4}
		F22	Spacer grid form loss	Uniform	0.5 – 1.5
	Shell side heat transfer	F23	Convection	Normal	±12.2%

3. Results

Figures 1 to 3 show the fuel centerline, cladding, and coolant temperatures, respectively, for the ULOF obtained using the MARS-LMR [13] and PAPIRUS [14] by completing the uncertainty propagation for 124 samples of the parameters. Wilks' formula was used for the BEPU evaluation, where the simulation models credit the third largest value from only 124 code calculations to satisfy the 95%/95% criterion. The maximum values for each sample calculation are presented in Figures 4 to 6. It was observed that the third largest value of the maximum fuel centerline, cladding and coolant temperatures for the ULOF are 1190 K, 1184.3 K, and 1170.5 K, respectively. Thus it was concluded that the thermal margin of the PGSFR for the ULOF does not exceed the safety acceptance criteria.

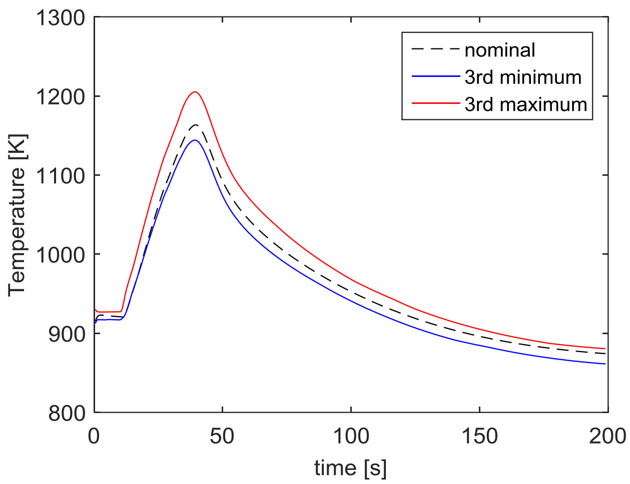

FIG. 1. Fuel centerline temperature distribution

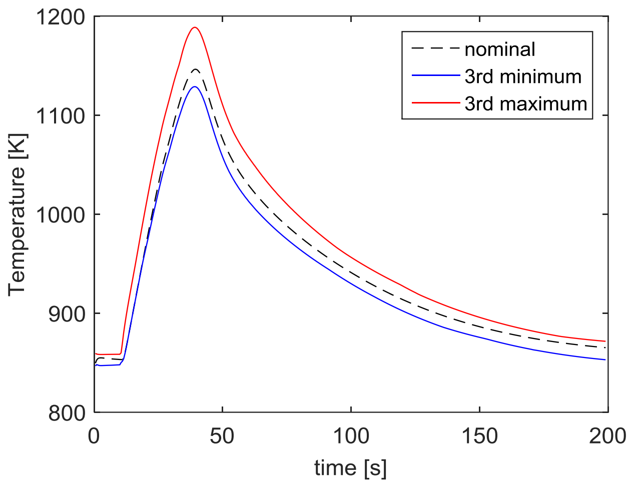

FIG. 2. Cladding temperature distribution

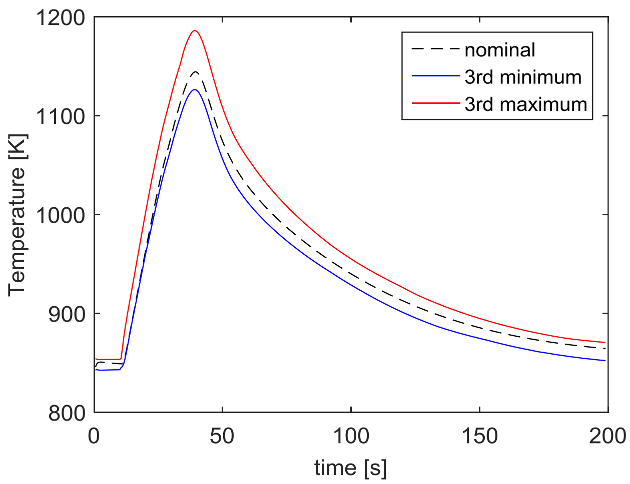

FIG. 3. Coolant temperature distribution

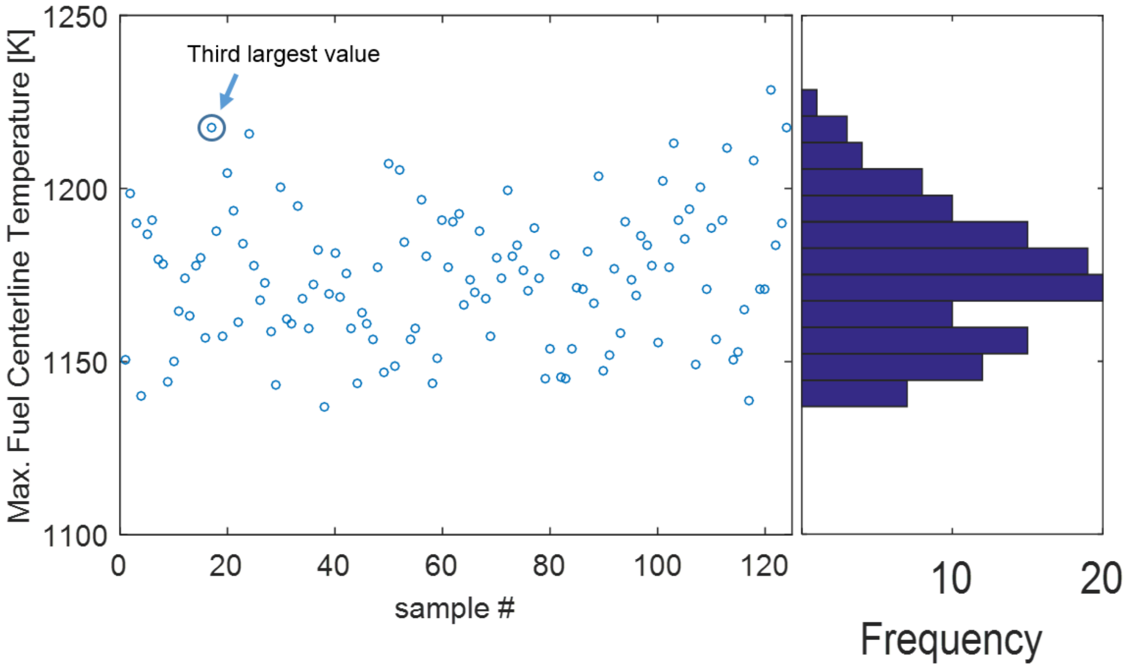

FIG. 4. Maximum fuel centerline temperature

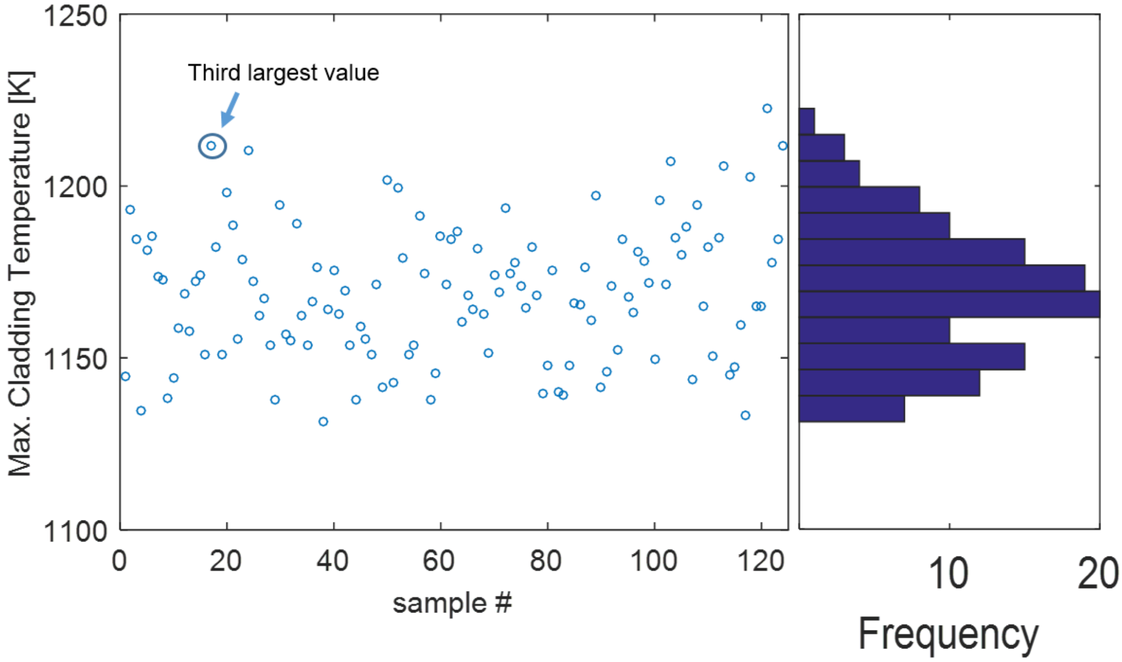

FIG. 5. Maximum cladding temperature

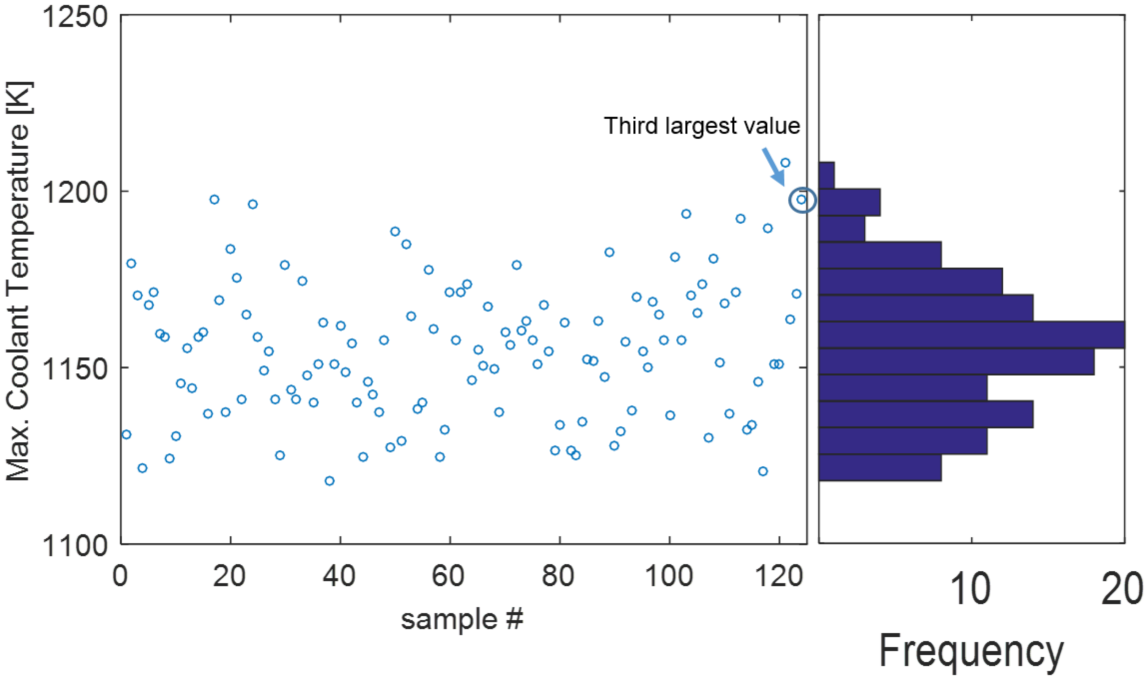

FIG. 6. Maximum coolant temperature

4. Conclusions

The uncertainty propagation was performed by mapping the uncertainty bands of the model parameters through the MARS-LMR to determine the distributions for the fuel centerline, cladding, and coolant temperatures for the ULOF. The results indicate that the simulation results for the temperatures satisfy the safety limits.

5. Acknowledgment

This work was supported by the National Research Foundation of Korea (NRF) grant funded by the Korea government (MSIP) (No. 2015M2A8A4046778).

References

[1] B. E. Boyack, R.B. Duffey, P. Griffith, G. S. Lellouche, S. Levy, U. S. Rohatgi, G. E. Wilson, W. Wulff, and N. Zuber, Quantifying Reactor Safety Margins: Application of Code Scaling, Applicability, and Uncertainty Evaluation Methodology to a Large Break Loss of Coolant Accident, NUREG/CR-5249, USNRC, Washington, DC , 1989.

[2] B. E. Boyack, I. Catton, R.B. Duffey, P. Griffith, K. R. Katsma, G. S. Lellouche, S. Levy, U. S. Rohatgi, G. E. Wilson, and N. Zuber, "Quantifying Reactor Safety Margins, Part 1-6," Nuclear Engineering and Design, **119**, 1, 1990.

[3] S. W. Lee, B. D. Chung, Y. S. Bang, and S. W. Bae, "Analysis of Uncertainty Quantification Method by Comparing Monte Carlo Method and Wilks' Formula," Nuclear Engineering and Technology, **46**, 4, pp. 481-488, 2014.

[4] NEA/CSNI/R(97)35/Vol. 2, 1998, Report of the uncertainty methods study for ad vanced best estimate thermal hydraulic code applications.

[5] Kazimi, M.S., Carelli, M.D., 1976. Heat transfer correlation for analysis of CRB RP assemblies, CRBRP-ARD-0034.

[6] Soon, H., Y.I. Kim, S.J. Kim, Y.J. Kim, 1998. Analysis of BFS-73-1 experiment, KAERI/TR/1133/98.

[7] Cacuci, D.G., 2010. Handbook of nuclear engineering, Springer.

[8] Ha, K.S., K.R. Lee, W.P. Chang, H.Y. Jeong, 2011. Validation of the reactivity feedback models in MARS-LMR, KAERI/TR-4395/2011.

[9] Cheng, S.K., Todreas, N.E., 1986. Hydrodynamic models and correlations for bar e and wire-wrapped hexagonal rod bundles – bundle friction factors, subchannel friction factors and mixing parameters, Nucl. Eng. Des. 92, 227-251.

[10] Chen, S.K., Todreas, N.E., Nguyen, N.T., 2014. Evaluation of existing correlation s for the prediction of pressure drop in wire-wrapped hexagonal array pin bundle s, Nucl. Eng. Des. 267, 109-131.

[11] Aoki, S., 1973. Current liquid-metal heat transfer research in Japan, Prog. Heat Mass Transfer 7, 569-573.

[12] Graber, H., Rieger, M., 1973. Experimental study of heat transfer to liquid metal s flowing in-line through tube bundles, Prog. Heat Mass Transfer 7, 151-166.

[13] K.S. Ha, H.Y. Jeong, C.G. Cho, Y.M. Kwon, Y.B. Lee, D.H. Han, "Simulation of the EBR-II loss-of-flow tests using the MARS code," Nucl. Technol, **169**, pp. 134-142, 2010.

[14] J. Heo and K. D. Kim, "PAPIRUS, a parallel computing framework for sensitivity analysis, uncertainty quantification, and estimation of parameter distribution," Nuclear Engineering and Design, **292**, pp. 237-247, 2015.