PGSFR Core Thermal Design Procedure to Evaluate the Safety Margin

Sun Rock Choi*, 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

The Korea Atomic Energy Research Institute (KAERI) has performed a SFR design with the final goal of constructing a prototype plant by 2028. The main objective of the SFR prototype plant is to verify the TRU metal fuel performance, reactor operation, and transmutation ability of high-level wastes.

The core thermal design is to ensure the safe fuel performance during the whole plant operation. Compared to the critical heat flux in typical light water reactors, nuclear fuel damage in SFR subassemblies arises from a creep induced failure. The creep limit is evaluated based on the maximum cladding temperature, power, neutron flux, and uncertainties in the design parameters, as shown in Fig. 1. In this work, the core thermal design procedures are compared to verify the present PGSFR methodology based on the nuclear plant design criteria/guidelines and previous SFR thermal design methods.

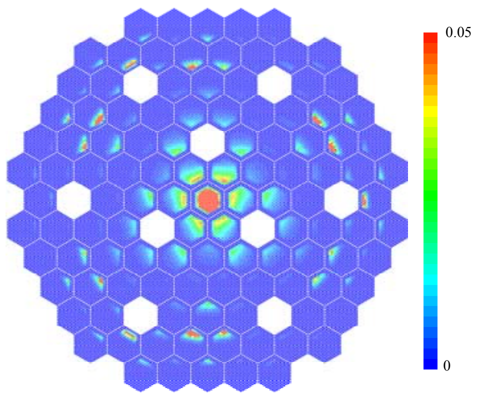
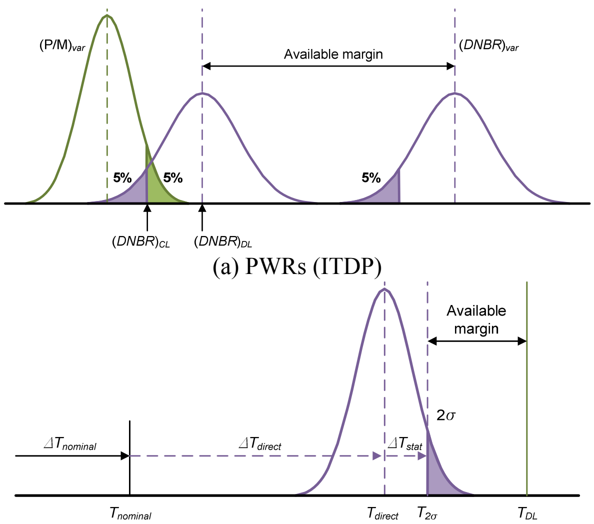


Fig. 1. Pin-wise distribution of cumulative damage function including uncertainties over the whole core at EOL

2. Core Thermal Design Criteria and Guidelines

The general design criterion (GDC) for the core thermal design is provided in GDC 10 of the NRC 10CFR50 Appendix A. Criterion 10—Reactor design [1]. This criterion is also endorsed in the liquid metal reactor GDC 3.2.1 of ANSI/ANS 54.1 1989 [2]. The criterion states that the reactor core and associated coolant, control, and protection systems shall be designed with an appropriate margin to assure that specified acceptable fuel design limits (SAFDL) are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences. Typical SAFDL employed in an LWR design is a departure from the nucleate boiling ratio (DNBR). Since the dominant design mechanism in the present SFRs is thermally induced creep, the cumulative damage function (CDF) is generally employed to be the SAFDL in the core thermal design.

The standard review plan of NRC and KINS provides guidelines to ensure the general design criterion in LWRs [3]. The acceptance criteria for the evaluation of fuel design limits provides assurance that there be at least a 95-percent probability at the 95-percent confidence level that the hot fuel rod in the core does not experience a DNB or transition condition during normal operation or AOOs. In addition, uncertainties in the values of the process parameters (e.g., reactor power, coolant flow rate, core bypass flow, inlet temperature and pressure, nuclear and engineering hot channel factors), core design parameters, and calculation methods used in the assessment of thermal margin should be treated with at least a 95% probability at the 95% confidence level. The review also evaluates the uncertainties associated with the combination of variables. For example, improved/revised thermal design procedures (ITDP/RTDP) in the Westinghouse are generally utilized to combine the design variable uncertainties in LWRs.



(a) PWRs (ITDP)

(b) Typical SFRs (semi-statistical method)

Fig. 2. Thermal design procedures of PWR and SFR

2. Previous Procedures

The PWR core thermal design procedure considers uncertainties of both the fuel design limits and design parameters. Each uncertainty is evaluated with at least a 95% probability at the 95% confidence level. The main

SAFDL in LWRs is a DNBR. A DNBR of 1.3 generally satisfies the 95/95 guideline in typical design conditions. A combination of uncertainties is utilized by the improved thermal design procedure, as shown in Fig. 2. For a $DNBR_{CL}$ of 1.3, uncertainties of the design variables with a 95/95 tolerance limit are applied to evaluate the design limit of $DNBL_{DL}$. The difference between $DNBR_{DL}$ and best-estimated $DNBR_{var}$ determines the available safety margin.

The previous SFR core thermal-hydraulic analysis generally treats its uncertainty through the hot channel factor (HCF) method [4]. The hot channel factor, F_{ij}, is an absolute uncertainty ratio to its nominal value. Therefore, it is a positive number greater than unity. There are several methods to combine the hot channel factors. A semi-statistical method is generally employed to analyze the overall uncertainties. Figure 2 shows a schematic diagram to illustrate the semi-statistical method, where biased and random uncertainties are separately involved.

The biased or direct uncertainties assume that all hot channel factors affect the most unfavorable values at the same location and at the same time. The direct uncertainty is calculated as follows:

$$\Delta T_{uncertainty} = \sum_{i=1}^{m} \left\{ \prod_{j=1}^{n} \Delta T_i \cdot F_{ij} \right\}. \quad (1)$$

On the other hand, the random uncertainties are purely statistical. Therefore the propagated overall uncertainty is given by a square root sum of the individual random uncertainties

$$\Delta T_{uncertainty} = \sum_{i=1}^{m} \Delta T_i + 2\left[\sum_{j=1}^{n} \left(\sum_{i=1}^{m} \Delta T_i \cdot (F_{ij}-1) \right)^2 \right]^{1/2}, \quad (2)$$

where 2 is multiplied to address the 2-sigma uncertainty.

3. Analysis Codes

3.1 Thermal-Hydraulic Analysis

The current core thermal-hydraulic design was performed using the SLTHEN (Steady-State LMR Thermal-Hydraulic Analysis Code Based on ENERGY Model) code, which calculates the temperature distribution based on the ENERGY model [5]. To describe the cross-flow by the wire wrap of the fuel pin, a two-region model is employed. The axial velocities in the internal and wall regions of a subassembly can be obtained from the flow split method. This two-region approximation enables the momentum equations to be decoupled from the energy equations. Once the flow is split, the temperature and pressure drops are calculated along the axial node with the finite difference equations using a one-pass procedure instead of an iterative one. This simplification significantly reduces the computer storage and computing time.

The resulting energy transport equations for the two regions are then calculated by

$$\rho C_p U_{zI} \frac{\partial T}{\partial z} = (\rho C_p \varepsilon_I + \zeta k)\left(\frac{\partial^2 T}{\partial x^2} + \frac{\partial^2 T}{\partial y^2} \right) + Q \quad (3)$$

$$\begin{aligned} &\rho C_p U_s \frac{\partial T}{\partial s} + \rho C_p U_{zII} \frac{\partial T}{\partial z} \\ &= (\rho C_p \varepsilon_n + \zeta k)\frac{\partial^2 T}{\partial n^2} + (\rho C_p \varepsilon_s + \zeta k)\frac{\partial^2 T}{\partial s^2} + Q \end{aligned} \quad (4)$$

where the left and right terms represent convective heat transfer and conduction by the enhanced eddy diffusivity, respectively. Q, k and ζ are the volumetric heat source, coolant thermal conductivity coolant and conductivity enhancement ratio from the geometrical factor.

.

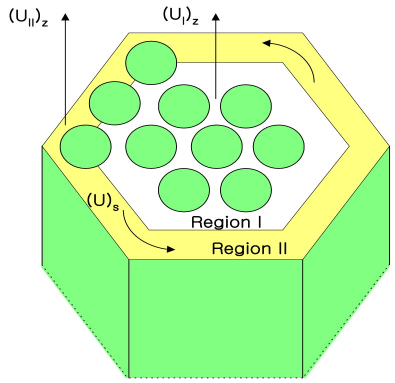


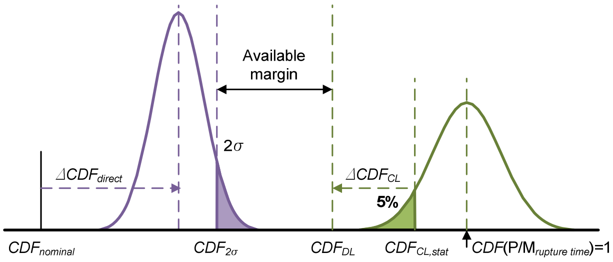
Fig. 3. Two region energy transport model in the SLTHEN code

3.2 Fuel Performance Analysis

MACSIS is a computer code for the thermal performance and dimensional characteristics of metal fuel pins under normal operating conditions of a Liquid Metal Cooled Fast Reactor [6]. It is necessary to have a method to accurately assess the thermal performance of a metal fuel pin in a fast neutron environment. The MACSIS computer program was developed as a design tool for a metallic fuel rod.

4. PGSFR Thermal Design Procedure

Since cladding mid-wall temperature indicates indirect information of thermally-induced fuel failure, the previous thermal design procedure in SFRs aims to minimize the maximum cladding mid-wall temperature including uncertainties over the whole core. However, the present PGSFR methodology directly evaluates the fuel cladding failure (thermal strain, CDF, hoop stress, etc.) including uncertainties to conduct a more precise performance analysis and assure more safety margin over the SAFDL. In addition, fuel performance analysis is conducted using the non-equilibrium cycle temperature history to reflect realistic wastage performances.



(c) PGSFR

Fig. 4. PGSFR thermal design procedure

Figure 4 shows the present PGSFR thermal design procedure. In the core thermal-hydraulic analysis, the cladding temperatures and pin power/flux are evaluated by the semi-statistical method using the related hot channel factors. Uncertainties from the fuel performance models (rupture time correlation, FCCI, OD wastage, thermal properties, etc.) are statistically combined. The difference between the maximum CDF and the design limit (0.05) determines the available safety margin. The present HCFs are mainly employed from the CRBR except fuel-related uncertainty such as an incorrect fuel distribution. PGSFR specific HCFs will be developed to evaluate the thermal margin.

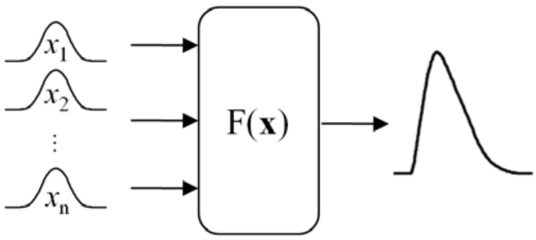


Fig. 5. Uncertainty propagation to output

The 95% one-side tolerance limit in LWRs corresponds to 1.96σ, which is similar to the present uncertainty of 2σ. Uncertainty combination methods such as ITDP and the Monte Carlo method (MCM) are compared to quantify the hot channel factor method. The probability density functions (PDF) for the input quantities propagate through the thermal model F(x) to offer the overall PDF for the output quantity, as shown in Fig. 5.

The HCFs, ITDP, and MCM reveal similar uncertainty propagation for cladding the mid-wall temperature for typical SFR conditions, as shown Fig. 6. Figure 7 shows the overall distribution of the maximum fuel centerline temperature with a comparison of the improved thermal design procedure. Figure 7 also exhibits the endpoint of at least a 95% probability as indicated by the vertical lines, reflecting the asymmetry propagation of the input uncertainties in the Monte Carlo method. The calculated results of the hot channel factor were close to those of the improved thermal design procedure like the mid-wall temperature calculation, owing to their similar statistical analyses.

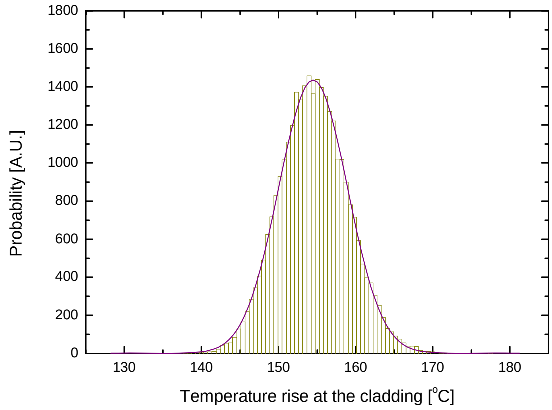


Fig. 6. Uncertainty propagation to mid-wall temperature

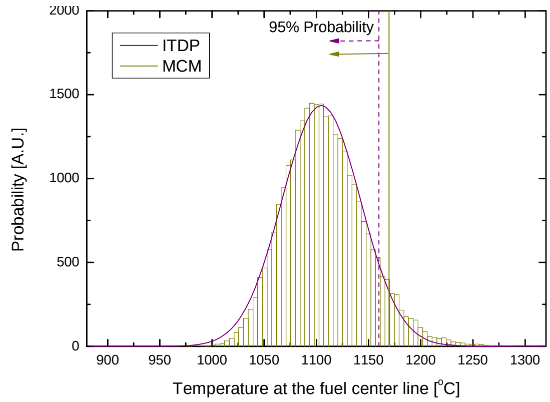


Fig. 7. Uncertainty propagation to fuel centerline temperature

5. Conclusions

The PGSFR core thermal design procedure is verified based on the nuclear plant design criteria/guidelines and previous methods in LWRs and SFRs. The present method aims to directly evaluate the fuel cladding failure and to assure more safety margin. The 2σ uncertainty is similar to 95% one-side tolerance limit of 1.96σ in LWRs. The HCFs, ITDP, and MCM reveal similar uncertainty propagation for cladding mid-wall temperature for typical SFR conditions. The present HCFs are mainly employed from the CRBR except the fuel-related uncertainty such as an incorrect fuel distribution. Preliminary PGSFR specific HCFs will be developed by the end of 2015.

ACKNOWLEDGEMENT

This work was supported by the National Research Foundation of Korea (NRF) grant funded by the Korea government (MSIP). (No. NRF-2012M2A8A2025622)

REFERENCES

[1] General Design Criteria for Nuclear Power Plant, 10CFR50 Appendix A., NRC, 2013.
[2]General Safety Design Criteria for a Liquid Metal Reactor Nuclear Power Plant, ANSI/ANS 54.1, 1989.

[3] Standard Review Plan 4.4 Thermal and Hydraulic Design, NUREG-0800, NRC, 2007.
[4] 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.
[5] W. S. Yang, An LMR Core Thermal-Hydraulics Code Based on the ENERGY Model, Journal of the Korean Nuclear Society, Vol. 29, pp. 406-416, 1997.
[6] W. Hwang, et. Al., MACSIS: A Metallic Fuel Performance Analysis code for simulating In-Reactor Behavior under Steady-State Conditions, Nuclear Technology, vol.123, p130, 1998.