Analysis of SMART RCP Shaft Break at a Full Power Accident Using a MARS-KS Based BEPU Approach

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1. Introduction

SMART (System-Integrated Modular Advanced Reactor) is an advanced small integral PWR developed by KAERI [1]. It has a compact size and a relatively small power compared to a conventional reactor. The main components of SMART such as helical steam generators and cannel-motor reactor coolant pumps (RCPs) are placed inside the reactor vessel so that there are no large pipe systems penetrating the vessel. Since the RCPs have a much smaller rotational inertia compared to the RCPs of commercial reactors, the RCP coastdown time is very short and then the events of decrease in reactor coolant flow rate caused by RCP malfunctions become more severe. Hence, this study considers a single pump shaft break with loss of offsite power (LOOP) at a full power–a postulated non-LOCA accident.

According to the guidance for the review of the methods used in transient and accident analyses (NUREG-0800, Section 15.0.2), uncertainty analyses addressing all important sources of uncertainty should be performed for best-estimate analyses to confirm that the combined code and application uncertainty is less than the design margin for the safety parameter of interest (e.g., peak cladding temperature, departure from nucleate boiling–DNBR). This requirement is typically important for small modular reactors (SMRs) since best-estimate evaluation approaches have been mainly developed for commercial reactors. The used mathematical models are mostly applicable to high-pressure high-flow conditions whereas SMRs usually operate under a lower pressure and a lower flow rate. Our previous study confirmed a low-accuracy prediction of critical heat flux models over the range of low-pressure low-flow conditions [2], and hence a large bias is expected for DNBR calculations.

Since 1990's, the best-estimate plus uncertainty (BEPU) methodology has been commonly utilized for LOCA analyses [3-5] and now spread to non-LOCA analyses [6-7]. SMRs with the compact integral design with less pipe systems show a reduced possibility of LOCA events, and non-LOCA events remain for consideration. In this study, a BEPU approach based on MARS-KS system thermal hydraulic code is applied to the analysis of a SMART RCP shaft break at full power accident. The main purpose of this work is to validate the applicability of the BEPU method to non-LOCA analyses.

2. Methodology

Among all the available uncertainty analysis methods, the probabilistic input uncertainty propagation method, which is most widely used in nuclear safety analysis, was selected to couple with the MARS-KS code because of its simplicity, robustness, and transparency [7]. The uncertainties of key input parameters are propagated to the MARS-KS simulation outputs via sampled data from known or assumed distributions.

First, a MARS-KS analysis model is developed as a reference case. Then, the input uncertainty parameters including manufacturing tolerances, boundary conditions, thermal properties, and heat transfer coefficients (see Table 1) are selected based on the previous studies [5,7]. Next, a Python script was developed to generate these input uncertainty parameters randomly and to generate MARS-KS input decks. Finally, a statistic analysis is performed on output uncertainty parameters. The sample size is selectively set to 124 (i.e., 124 code runs).

Table 1. Input uncertainty parameters

	Parameters	PDF	Mean	SD
Manufacture	Cladding OD (mm)	N	9.5	0.01
	Cladding ID (mm)	N	8.357	0.01
	Cladding roughness (μm)	N	0.8	1
	Fuel roughness (μm)	N	1.8	1
	Filling gas pressure (MPa)	N	6.653	0.05
B.C.	Coolant pressure (MPa)	N	15.5	0.075
	Coolant inlet temperature (°C)	N	295.5	1.5
	Core flow rate (kg/s)	N	2507	125.4
	Initial PZR liquid volume (%)	U	65.25	5
	Reactor power (MWt)	N	365	2.482
Prop.	Fuel conductivity	N	1	0.05
	Cladding conductivity	N	1	0.05
	Gap conductance	U	0.95	0.55
Heat Transfer Models	Groenevel CHF LUT	N	1	0.125
	Zuber CHF correlation	N	1	0.125
	Chen nucleate boiling correl.	N	1	0.125
	Chen transition boiling correl.	N	1	0.125
	Dittus-Boelter liquid convection correlation	N	1	0.125
	Dittus-Boelter vapor convection correlation	N	1	0.125
	Bromley film boiling correl.	N	1	0.125
	Pump 2-f head multiplier	U	0.5	0.5
	Pump 2-f torque multiplier	U	0.5	0.5

3. Results

The limiting case for transient DNBR with the conservative initial and boundary conditions was selected as the reference case, as given in Table 2. The core power and core flow rate are 103 % and 95 % of the designed values. A bottom-skewed axial power distribution with an axial shape index (ASI) of -0.35 was applied. The radial power peaking factor of the hottest rod is 1.524. The fuel temperature coefficient and moderator density coefficient are bounding values expected for whole fuel cycles.

Table 2. Initial and boundary conditions

Parameters	Value	Error (%)
Core power [MWt]	376	0.0
SG inlet temperature [°C] (primary)	324.3	0.06
PZR pressure [MPa]	15.67	0.0
Core flow rate [kg/s]	2371.6	-0.42
PZR level [%]	77.5	0.0
SG pressure [MPa]	4.95	1.02
Linear heat rate [kW/m] (average/hot/hottest)	18.71/ 18.09/ 12.28	
CEA worth [%Δρ]	-7.35	
ASI [-]	-0.35	
Doppler reactivity	Most	
Moderator density reactivity	Least	

Table 3. Sequence of events

Time (s)	Events	Setpoint
0.0	Single RCP shaft break	
0.06	Low reactor coolant flow trip condition reached	81.3 %
1.17	Reactor trip signal generates - Turbine trip - LOOP - RCP coast down starts - FW pump trip - MSIV/MFIV 1,3,4 start to close	
-	Maximum fuel temperature	
1.59	PRHRS signal generated upon the low FW flow signal	4.46 %
1.68	CEAs start to insert	
2.70	MSIV/MFIV 2 start to close	
2.23	MDNBR	1.53/2.53
6.17	MSIV/MFIV 1,3,4 completely closed	
8.11	z	
187.0	Maximum PZR pressure	~16 MPa
324.0	Maximum SG pressure	8.9 / 9.7 MPa
10,251	Shutdown cooling temperature	215 °C

The transient initiated with a single RCP shaft break (see Table 3). Due to the RCP malfunction, the reactor coolant flow quickly reduced, reaching the low flow setpoint of 81.3% (see Fig. 1) and generating a reactor trip signal at ~1.2 seconds. LOOP was assumed at the reactor trip causing turbine trip, feedwater pump trip, RCP coastdown, and MSIVs/ MFIVs closing.

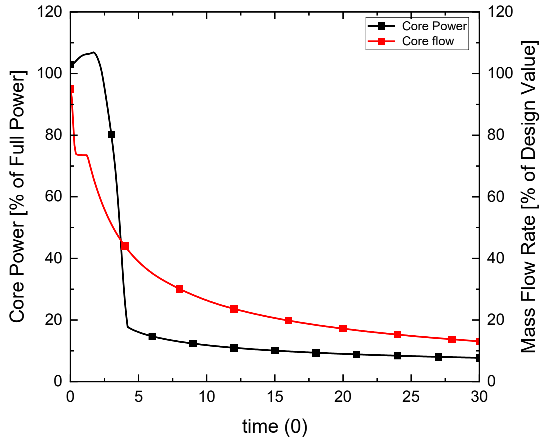

Figure 1. Core power and core flow rate

Due to the core flow reduction, the heat removal by steam generators (SGs) reduced leading to a slight pressurization in the primary side during a short period and an increase in the core power (see Fig. 1) due to positive moderator density feedback. Also, the SG pressure started to increase due to continuous heat absorption form the primary side, and the passive residual heat removal systems (PRHRS) actuated by a low FW flow signal to bring the systems to the safe shutdown cooling condition.

The minimum DNBR values at the hottest rod is plotted in Fig. 2. The DNBR starts to reduce from ~3.5 at the beginning of accident to the MDNBR of 2.53 at ~2.2 seconds due to the loss of core cooling caused by the reduction of core flow and to the return-to-power caused by the reactivity feedbacks. The calculated MDNBR is relatively high, possibly relating to initial core inlet temperature and reactivity feedback.

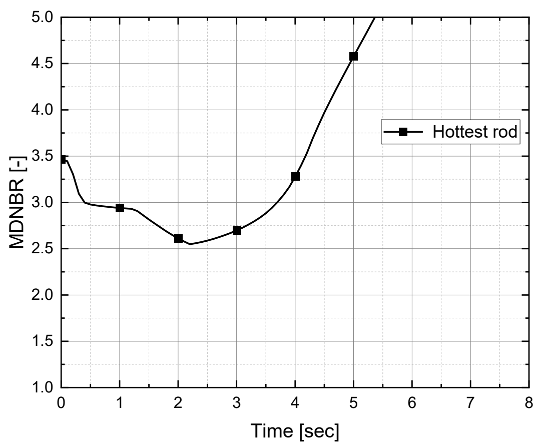

Figure 2 Core power

To confirm the DNBR calculation, a parallel simulation of 1/4 SMART core was performed using the CTF subchannel code. The initial and boundary conditions applied to the CTF model were taken from the MARS-KS analysis results. A radial core power distribution and a core inlet flow distribution were assumed. An asymmetrical core inlet flow distribution is also assumed considering the single RCP malfunction (see Fig. 3a). Consequently, the coolant temperature distribution was asymmetrical (Fig. 3b). Boiling was shortly observed in the hot fuel assemblies, downstream of the core (Figs. 3c).

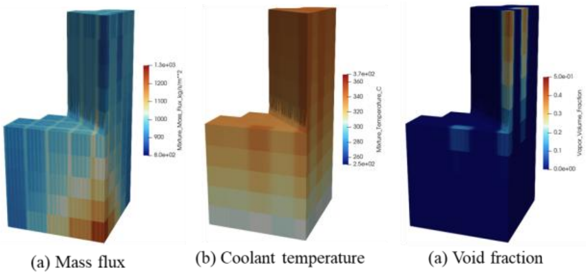

Figure 3 CTF simulation results at the MDNBR condition (2.2 seconds)

The transient DNBR along the hottest fuel rod was plotted in Fig. 4 The MDNBR was observed at 0.6 m ~ 0.8 m (around the axial power peaking location), and it started to decrease from 4.3 at 0.0 second to 2.5 at 2.2 seconds and then turned to increase. The CTF MDNBR closes to the MARS-KS calculated value.

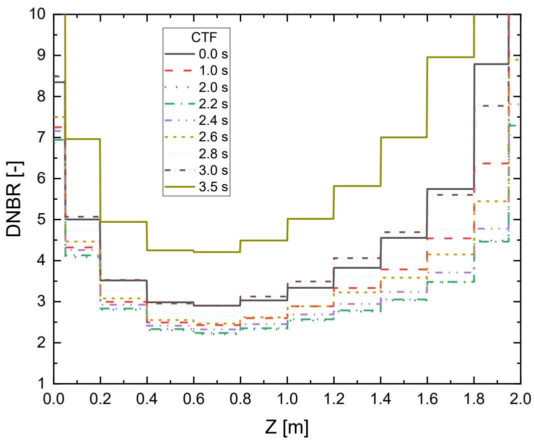

Figure 4 CTF simulation results: Transient axial DNBR

Figure 5 shows the MARS-KS uncertainty analysis result for MDNBR. The smallest MDNBR is 1.38, slight lower that the SMART SSAR DNBR and below the acceptable criterion of 1.5. The SMART SSAR limiting case mostly bounds the calculated DNBR values from the below.

The statistical analysis results of MDNBR are shown in Fig. 6. The calculated MDNBR has a normal distribution shape wit the mean of 2.26 and the standard deviation of 0.31. The MDNBR 95/95 was estimated to be 1.66. That means, the MDBNR acceptance criterion was not violated for the SMART design.

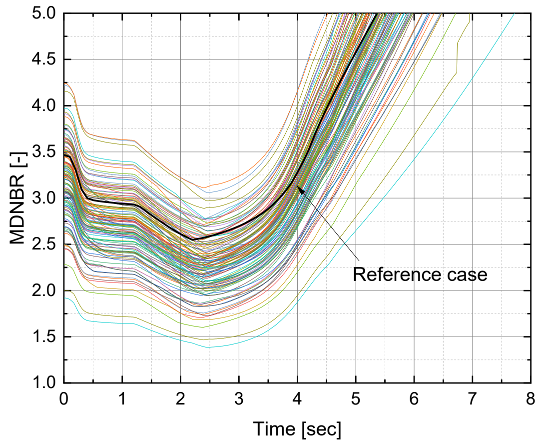

Figure 5. Uncertainty analysis results: MDNBR

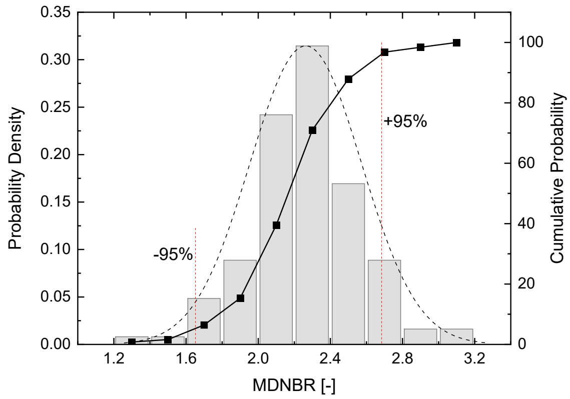

Figure 6. Statistical analysis of MDNBR

4. Conclusions

The MARS-KS calculation for the SMART RCP shaft break at a full power with LOOP was performed and compared with the CTF parallel simulation of 1/4 core. The MDNBR calculated by the MARS-KS was quite high ~2.53, and it could be related to the reactivity feedbacks and initial core inlet temperature. Dynamic behaviors of the SMART core could be observed at a high resolution with the CTF simulation.

An uncertainty qualification methodology was setup based on the probabilistic input uncertainty propagation method and the MARS-KS code. A set of 22 input parameters was selected to feed into the uncertainty analysis. The estimated MDNBR 95/95 is 1.66, not violating the SMART acceptance criteria.

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REFERENCES

[1] K. K. Kim, W. Lee, S. Choi, H. R. Kim and J. Ha, SMART: The First Licensed Advanced Integral Reactor, Journal of Energy and Power Engineering, Vol. 8, pp. 94-102, 2014.
[2] N. H. Hoang, S. S. Jeon, J. Lee and J. Y. Park, MARS-KS Analysis of Low Pressure Low Flow Critical Heat Flux Based on Annulus and Round-Tube Experimental Data, Transactions of the Korean Nuclear Society Spring Meeting, 2025.
[3] G. E. Wilson, Historical insights in the development of Best Estimate Plus Uncertainty safety analysis, Annuals of Nuclear Energy, Vol. 53, pp. 2-9, 2013.
[4] S. Y. Lee and C. H. Ban, Code-Accuracy-Based Uncertainty Estimation (CABUE) Methodology for Large-Break Loss-Of-Coolant Accidents, Nuclear Technology, Vol. 148(3), pp. 335-347, 2004.
[5] D. G. Kang, Analysis of LBLOCA using Best Estimate Plus Uncertainties for Three-Loop Nuclear Power Plant Power Upgrade, Annuals of Nuclear Energy, Vol. 90, pp. 318-330. 2016.
[6] J. Zhang, A. Kovtonyuk and C. Schneidesch, Towards a Graded Application of Best Estimate Plus Uncertainty Methodology for non-LOCA Transient Analysis, Nuc. Eng. Des. Vol. 354, pp.110189, 2019.
[7] O. Marchand, J. Zhang and M. Cherubini, Uncertainty and sensitivity analysis in reactivity-initiated accident fuel modeling: synthesis of organization for economic co-operation and development (OECD)/nuclear energy agency (NEA) benchmark on reactivity-initiated accidents codes phase-II, Nuc. Eng. Tech. Vol. 50, pp. 280-291, 2018.