DNBR Evaluation for SLB analysis by Multi-Dimensional Safety Analysis

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1. Introduction

Steam line break accident (SLB) in the nuclear reactor is one of the representative Non-LOCA accidents in which thermal-hydraulics and neutron kinetics are strongly coupled each other. Departure of Nucleate Boiling Ratio (DNBR) is a key parameter to examine safety issue Thus, there have been enormous effort to investigate SLB analysis regarding to the examination of DNBR. System thermal-hydraulics codes have been widely used for safety analysis. However, multi-dimensional approach with multi-physics methodology has been used to make it possible to visualize the nuclear reactor fuel assembly region realistically. [1, 2]

Recently, CUPID-RV, has been developed for three-dimensional reactor thermal-hydraulics analyses. The purpose of the CUPID-RV is to quantitatively examine a safety margin for hypothetical accidents such as LOCA and SLB and consequently secure the integrity of the NPP design. In addition, neutron kinetics code has been coupled for realistic behavior of reactor power output and system Thermal-hydraulics code is coupled for thermal-hydraulics behavior of the rest of reactor coolant system (RCS). In this paper, we would like to examine the DNBR in subchannel-scale analysis and suggest insight for necessity of subchannel-scale safety analysis in terms of an enhancement of safety margin.

2. Numerical Methodology

2.1 System modeling

Region	Code	features
Reactor core	MASTER	subchannel-scale resolution
RPV	CUPID-RV	3D neutron diffusion
RCS	MARS	system-scale thermal-hydraulics

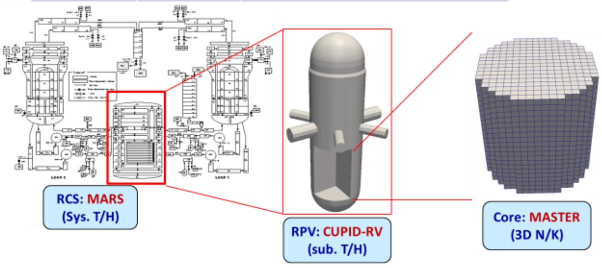

Fig. 1. System configuration for MSMP safety analysis

In this system, entire reactor coolant system (RCS) is considered with respect to multi-scale and multi-physics (MSMP) methodology. Figure 1 shows the system configuration for MSMP safety analysis. RCS except reactor pressure vessel (RPV) is modeled by system thermal-hydraulics code, MARS. The thermal-hydraulic behavior inside of the RPV is calculated by the CUPID-RV code. The RPV is modeled as a body-fitted mesh with subchannel resolution [3].

It is inevitable to consider a neutron kinetics in the reactor core during SLB accident because sudden injection of cold coolant induces abnormal power distribution. In this MSMP methodology, nodal-based neutron kinetics code, MASTER, is coupled.

2.2 Subchannel-scale physical models

The reactor core model of CUPID-RV includes pressure drop model for precise visualization of the flow distribution in the reactor core. The turbulent mixing between neighboring subchannel can be occurred due to the turbulent fluctuation and the flow disturbance by structures such as grid spacer mixing vane. In this study, Equal volume exchange and void drift (EVVD) model is applied. Table 1 shows the subchannel-scale physical models model adopted in CUPID-RV code for the RPV thermal-hydraulics analysis

Table 1. Subchannel T/H model

Model	x	y	z	Note
Friction factor			O	$\Delta P = -\frac{1}{2}\left(\frac{f}{d_{hy}} + K'\right)\left(\frac{G_k^2}{\rho_k}\right)$
Form loss	O	O		$\Delta P_L = -\frac{K_G}{2}\left(\frac{W_{IJ,k}\lvert W_{IJ,k}\rvert}{l_{IJ}\rho_k s_{IJ}}\right)$
Turbulent mixing and void drift			O	• EM (Equal Mass exchange) • EVVD (Equal Volume exchange and Void Drift)
Grid spacer			O	$\Delta P = -\frac{K}{2}\left(\frac{G_k^2}{\rho_k}\right)$
Mixing vane	O	O		$M_k = f^2 u_l \rho_l A \times u_l$

3. Results

Multi-physics calculation is carried out for normal operation condition of the Pressurized Water Reactor (PWR) OPR1000. System T/H code and neutron kinetics code have fast-running capability. Thus it takes most of computational time is to compute the subchannel-scaled T/H by CUPID-RV. However, CUPID-RV code are fully parallelized so that more than 300 processors are used in this simulation. In this paper, we would like to compare the minimum DNBR obtained 1D system T/H code and multi-dimensional MSMP methodology and investigate why this approach has an advantage of enhancing safety margin.

Table 2. Comparison of MDNBR

Methodology	DNBR
FSAR [4]	1.068
1D System TH	2.020
Multi-D MSMP	2.729

First of all, table 2 shows the minimum DNBR prior to the reactor trip. FSAR [4] shows the lowest minimum DNBR because the computational result reported in FSAR was produced by as many conservative assumptions as possible. 1D system safety analysis code, MARS with standard OPR1000 SLB input deck, shows enhanced minimum DNBR. lower than realistic three-dimensional simulation. Compared with those 1D approach, multi-dimensional MSMP approach produces quite enhanced minimum DNBR results. In this paper, the reasons why enhanced MDNBR is obtained are investigated.

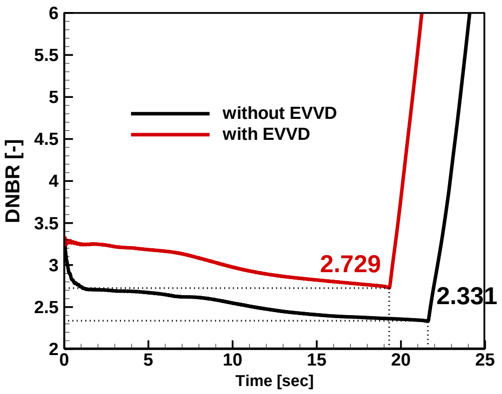

Fig. 2. DNBR profile

Conventional 1D safety analysis codes cannot consider a radial flow mixing behavior in the reactor core. However, three-dimensional analysis yields spatial flow mixing due to convection and diffusion phenomena. In addition, if realistic physical models such as radial turbulent mixing is taken into account, the radial flow mixing phenomena is realistically calculated and become positive effect to enhance minimum DMBR. Figure 2 shows the time history of DNBR from an initiation of steam line break. Even though any radial mixing model is not considered, the DNBR is enhanced, compared with those from 1D system TH approach. Moreover, by activating radial flow mixing model such as EVVD model, the DNBR is increased about 17%.

Figure 3 shows the radial distribution of liquid temperature and DNBR at an elevation the minimum DNBR occurs. When the turbulent mixing model (EVVD) is not considered, the coolant is not radial mixed properly. Thus, the temperature and DNBR distribution shows somewhat discrete pattern. On the other hand, turbulent mixing model yields well-mixed radial flow distribution, which shows lower local liquid temperature and higher MDNBR.

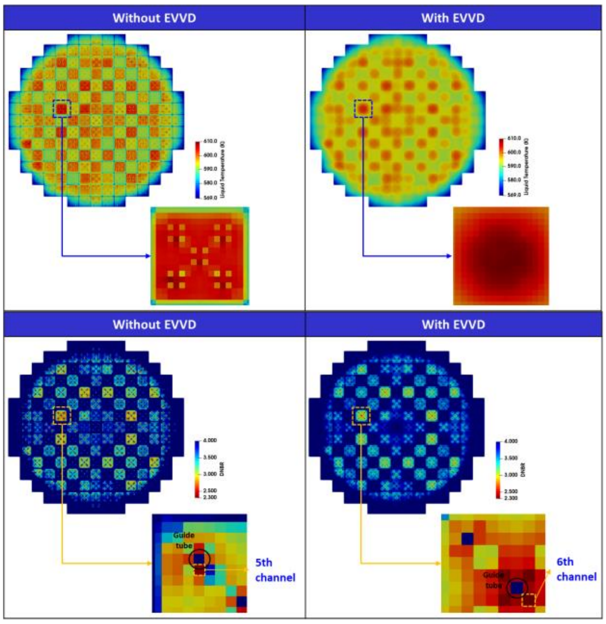

Fig. 3. Liquid temperature (top) and DNBR (bottom) distribution

In general, system-scale safety analysis code model 'hot-pin' to examine DNBR with conservative assumption that the minimum DNBR occurs at the hot pin. In MARS code input deck, the peaking factor of the hot-pin is 1.79. However, from the results of the co-simulated neutron kinetics code, the axially-averaged peaking factor of a rod which shows minimum DNBR is 1.4641. In addition, this rod is not highest power-peaked fuel rod. Figure 4 shows the comparison of peaking factor distribution between hottest fuel rod and the rod which shows minimum DNBR. From axial profile of peaking factor shown in Fig. 4(left), the local minimum DNBR location is not same as the location at which the local maximum peaking factor is obtained. Thus, it can be concluded that the 'hot-pin' assumption that conventional one-dimensional system-scaled safety analysis has been adopted can be examined realistically by multi-dimensional MSMP approach.

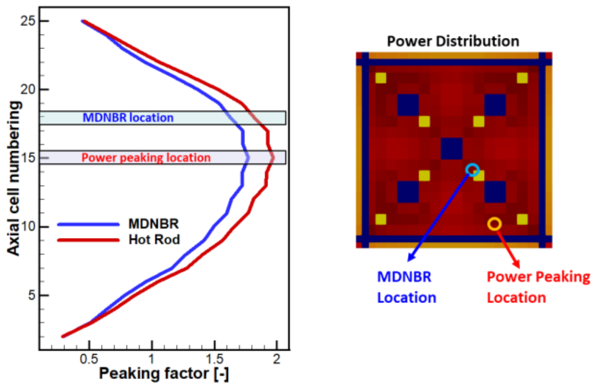

Fig. 4. Comparison for MDNBR and power peaking location

4. Conclusion

The SLB accident was simulated by multi-dimensional MSMP approach using the coupled CUPID-RV/MARS/MASTER code. The minimum DNBR from the FSAR and conventional system T/H code and MSMP approach is compared. Multi-dimensional MSMP methodology produced enhanced minimum DNBR. In addition, the necessity for realistic subchannel-scaled multi-dimensional safety analysis was pointed out in terms of radial mixing and local peaking factor.

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