Transactions of the Korean Nuclear Society Autumn Meeting
Changwon, Korea, October 20-21, 2022

Control Rod Ejection Analysis Methodology based on 3-Dimensional core transient simulation

Jin-Woo Park *, Song-Kee Sung, Dae-Gwang Hong, Sung-Ju Kwon, Chan-Do Jung, Han-Bit Noh, Jae-Don Choi
KEPCO NF, 242, 989 beon-gil, Daedeokdae-ro, Yuseong-gu, Daejeon, Korea
**Corresponding author: jinwoo@knfc.co.kr*

1. Introduction

The control rod ejection with high reactivity worth causes the sudden insertion of reactivity into the core. Immediately, the nuclear power of the reactor dramatically increases in an exponential behavior until the Doppler feedback effect becomes dominant to turn the reactivity balance and power down to lower levels. Although this happens in only within a few seconds, the energy generation causes fuel failures.

The KNF(KEPCO Nuclear Fuel)'s current safety analysis methodology is based on numerous conservative assumptions with the point kinetics model. It would predict the more adverse consequences than the phenomena in the real reactor system during the control rod ejection.

For that reason, KNF developed the safety analysis methodology based on the multi-dimensional core simulation in 2019 and its licensing process has been completed [1].

In this paper, the multi-dimensional safety analysis methodology is briefly introduced and its analysis results are also described.

2. Methods

2.1 Code System

Three-dimensional core neutron kinetics code ASTRA, sub-channel thermal-hydraulic analysis code THALES, and fuel transient analysis code FROST were coupled by using message passing interface(MPI) method[2]. The use of MPI method has benefits of not only variable data transfer but also parallel computation. Therefore a multi-dimensional full core transient analysis which needs the number of rod calculations was performed using multi-CPU parallel processing.

2.2 Methodology

Fig. 1 shows both the current methodology using point-kinetics(or 1-Dimensional) model and the newly developed methodology based on 3-D model.

The biggest difference of the 3-D methodology comparing to the current methodology is how to generate transient rod power. In current methodology, only hot rod transient power is generated by the combination of constant peaking factor and core average power, whereas the transient power information of all rods can be generated in the 3-D methodology. As a result, some safety margin of rod ejection analysis is obtainable through both the improvement of constant peaking factor assumption(Fig. 2) and the use of reasonable core average power(Fig. 3).

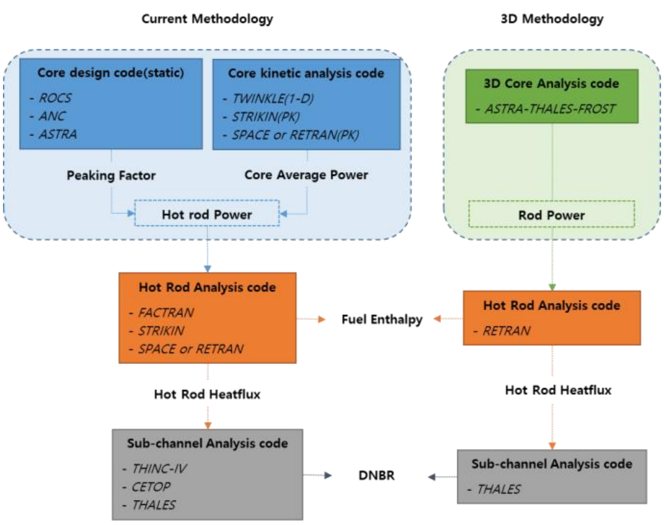

Fig. 1. The comparison of the methodology for rod ejection analysis.

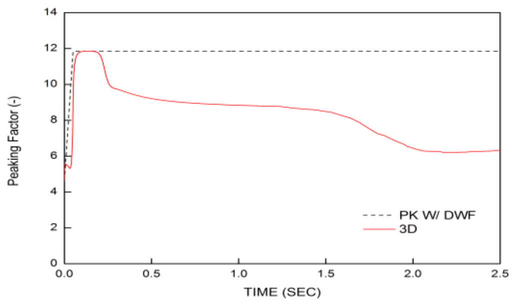

Fig. 2. Peaking factor of HZP condition

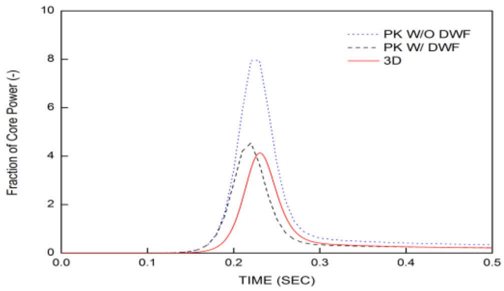

Fig. 3. Core average power of HZP condition

The additional characteristics of the 3-D methodology are described below.

1) Conservatism of core kinetics parameters

Table I and Fig. 3 show core average power behavior corresponding to kinetics parameter assumptions in the 3-D methodology. The core kinetics assumptions used in point-kinetics model are also applicable to the 3-D methodology. To maintain conservatism of the current methodology, these assumptions are identically adopted in the 3-D methodology.

Table I: kinetics parameter assumptions

Case	Rod worth (pcm)	Fraction of delayed neutron (%)	Prompt neutron life time (μ sec)	Fuel temperature coefficient (pcm/√ K)	Coolant temperature coefficient (pcm/℃)	Scram worth (%Δ ρ)
1	536	0.557	26.8	-176	-44.1	-12.59
2	536	0.412	26.8	-176	-44.1	-12.59
3	536	0.412	15.0	-176	-44.1	-12.59
4	536	0.412	15.0	-126	-44.1	-12.59
5	536	0.412	15.0	-126	-15.0	-12.59
6	536	0.412	15.0	-126	-15.0	-5.0

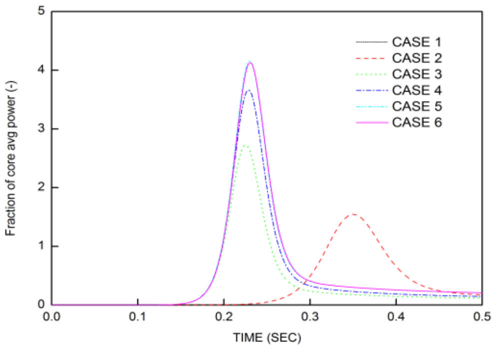

Fig. 3. Core average power with kinetics parameter assumptions

2) Hot rod evaluation method

The current hot rod evaluation method using the RETRAN code is also applied to the 3-D methodology. Therefore the current conservative approach for rod evaluation is still effective in the 3-D methodology.

3) Evaluation of fuel failure caused by DNB

The evaluation of the fuel failure caused by DNB is performed for all assemblies in core. Even if a hot rod in assembly is reaching DNB, the other rods are conservatively assumed to be failed.

4) Evaluation of fuel failure caused by PCMI

The fuel failure caused by PCMI(Pellet Cladding Mechanical Interaction) are directly evaluated using all rod power histories in the entire core. In this evaluation, the generated rod power is assumed to be accumulated in rod without the heat removal between rod and coolant.

3. Results

Table II shows the results of a control rod ejection analysis using the 3-D methodology. 3-D analysis results are satisfied with the criteria[3] for fuel failure and core cool-ability. And more safety margin compared to the current methodology can be obtained by using the 3-D methodology.

Table II: Results of rod ejection analysis

		Point kinetics	3-D	Remarks
A. Fuel rod cladding failure	A-1. High temperature cladding failure	< 12 %	< 2 %	Current limit for effective dose evaluation : 15%
	A-2. PCMI cladding failure	Not occurred	Not occurred	Lowest limit value[3] applied in 3-D
B. Core coolability	B-1. Enthalpy	< 150 cal/g	< 100 cal/g	Limit : 230 cal/g
	B-2. Fuel melting	0 % (<4800 °F)	0 % (<4000 °F)	Melting temperature : about 5.000 °F

4. Conclusions

KNF developed the safety analysis methodology based on the multi-dimensional core simulation in 2019 and its licensing process has been completed. From the results of the preliminary safety analysis using the 3-D methodology, additional safety margin(refer to Table II) can be obtained. It would be useful to resolve current safety issues and to develop the advanced safety analysis methodology based on realistic core transient simulation.

REFERENCES

[1] Korea Institute of Nuclear Safety, 특정기술주제보고서 “APR1400 원전 3 차원 노심해석 기반 제어봉집합체 이탈사고 해석 방법론” 심사결과보고서, 2021. 03.

[2] JW Park, GT Park, Code Coupling for Multi-Dimensional Core Transient Analysis, Transactions of the KNS 2015

[3] U.S. Nuclear Regulatory Commission, Regulatory Guide 1.236, “Pressurized-Water Reactor Control Rod Ejection and Boiling-Water Reactor Control Rod Drop Accidents,” 2020. 06.