Regulatory Audit Activities on Nuclear Design of Reactor Cores

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1. Introduction

Regulatory audit analyses are initiated on the purpose of deep knowledge, solving safety issues, being applied in the review of licensee's results. The current most important safety issue on nuclear design is to verify bias and uncertainty on reactor physics codes to examine the behaviors of high burnup fuel during rod ejection accident (REA) and LOCA, and now regulatory audits are concentrated on solving this issue. KINS develops regulatory audit tools on its own, and accepts ones verified from foreign countries. The independent audit tools are sometimes standardized through participating the international programs.

2. Regulatory Audit Activities on Nuclear Design

2.1 Development of COREDAX

The COREDAX code is a three-dimensional reactor core simulator for auditing nuclear safety analysis and design that is developed by KAIST under sponsorship of KINS. COREDAX is based on the analytic function expansion nodal (AFEN) method [1], and is written in FORTRAN90. COREDAX runs steady and transient standalone calculations, and also is coupled to the thermal-hydraulic system code MARS-KS [2] which provides the temperature and flow to COREDAX during the transient calculations. COREDAX Version 1.8 is now developed [3].

The AFEN method is successfully applied to the steady-steady and kinetics reactor analysis. The AFEN method outperforms the conventional nodal methods not only in determining the core criticality but also in the flux distributions, especially in the region where large flux gradients exit near strong material discontinuity. The superior performance of AFEN is mainly attributed to the fact that AFEN is based on no transverse integration and expansion in non-separable analytic basis functions.

The COREDAX code has been verified through various sample problems, and was ascertained to be precise for the international benchmark problems. Figs. 1~2 show some verification and validation results.

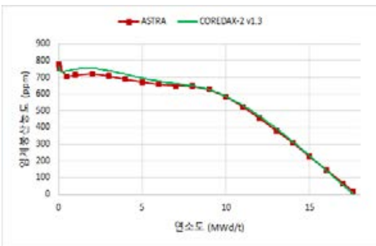


Fig. 1 CBC of APR1400 Initial Core

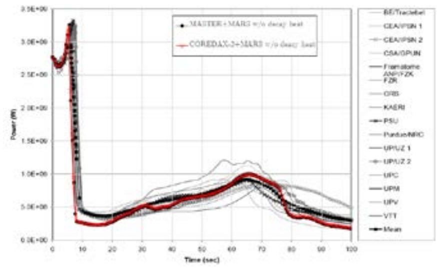


Fig. 2 OECD/NEA MSLB Benchmark Problem

2.2 Power Behaviors of High Burnup Fuels

Performance degradation of high burnup fuels is not clarified yet, especially during transients such as REA and LOCA. For evaluating high burnup fuels, 3-dimensional power simulation is required. A cross section set of the core is generated by SCALE [4], and PARCS [5] and COREDAX codes are used for power calculations of the fuels.

APR1400 core is modeled for the analysis. The cross section sets generated by SCALE are verified through comparing k-inf from SCALE with that from Monte-Carlo calculations. As shown in Fig. 3, they agree very well.

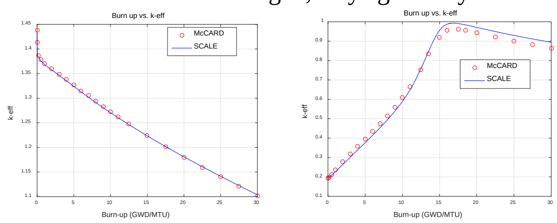


Fig. 3 Verification of SCALE calculations

REA is simulated by using PARCS with cross section sets generated by SCALE. For preliminary analysis, it doesn't take account of any uncertainties such as control rod worth and etc. Fig. 4 shows the power behaviors during rod ejection and full core power distributions with pin by pin.

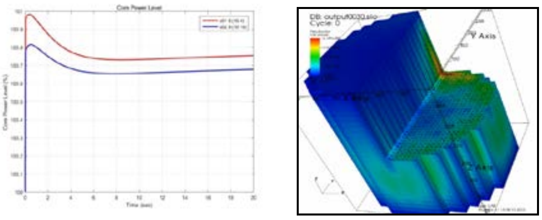
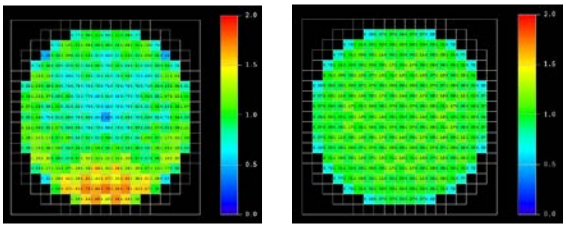


Fig. 4 Power Behaviors and Full core power distributions

COREDAX/SCALE results are shown in Fig. 5. 4-finger control rod assembly at core low part is ejected at zero power and full power conditions, the inserted reactivity of which is 0.121$ and 0.013$, respectively.



<Zero Power> <Full Power>

Fig. 5 Power Distributions at 3 seconds

2.3 Audits of MSLB

Main steam line break accident is a transient increasing reactivity of a core, which is required to model 1st system and 2nd system. PARCS/TRACE [6] is used for APR1400 core. For preliminary analysis using PARCS/TRACE coupling code system, 1-D TRACE model of the reactor core with PIPE components is used. In the steady-state calculation, k-eff by PARCS standalone and PARCS/TRACE is 1.00027 and 1.00086, respectively. 3-D TRACE model of the reactor pressure vessel with a VESSEL component has been developing for detailed coupling calculation. The results are shown in Fig. 6.

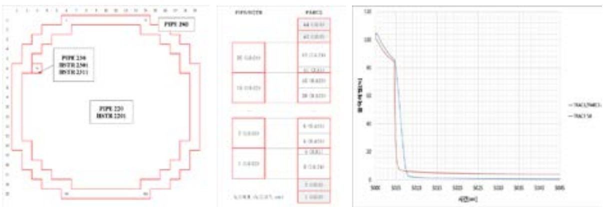


Fig. 6 MSLB using PARCS/TRACE

2.4 OECD/NEA UACSA Program

OECD/NEA expert group on UACSA has been doing activity to quantify the uncertainty [7]. The expert group has proposed benchmark problems for phases I~IV. Phase IV-a benchmark problem offered 21 criticality experiments based on LEU-COMP-THERM-007 and LEU-COM-THERM-039. Fig. 7 shows configuration of criticality experiment. The objective of this program is to compare the results of participants for sensitivity and uncertainty results of 5 scenarios, and finally to find out safety issues.

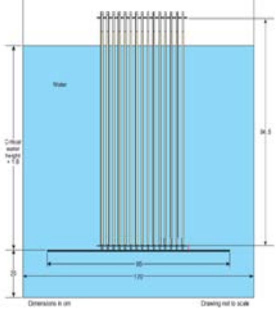
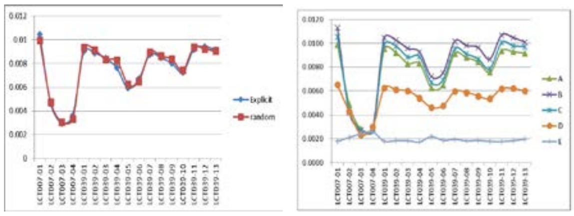


Fig. 7 Configuration of the Criticality Experiments

Fig. 8 shows the sensitivity and uncertainty results for the benchmark problem. The sensitivity result could have significant difference when the experiments have different configurations for several parameters such as rod pitch, and the difference in sensitivity also causes the difference in uncertainty.



<Sensitivity Results> <Uncertainty Results>

Fig. 8 Sensitivity and Uncertainty Analysis Results

2.5 Coupling with Fuel Performance Code

Safety criteria of high burnup fuels for REA and LOCA, which are set up in the near future, are evaluated by FRANCON/FRAPTRAN [8] and PARCS/COREDAX which gives the power distributions to FRANCON/ FRAPTRAN. Their evaluations are focused on dealing with uncertainties for the safety parameters of high burnup fuel.

3. Conclusions

New safety issues on nuclear design, reactor physics tests, advanced reactor core design are steadily raised, which are mainly drawn from the independent examination tools. It is some facing subjects for the regulators to find out the unidentified uncertainties in high burnup fuels and to systematically solve them. The safety margin on nuclear design might be clarified by precisely having independent tools and doing audit calculations by using them. SCALE-PARCS/COREDAX and the coupling with T-H code or fuel performance code would be certainly necessary for achieving these purposes.

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