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Status of optimal evaluation system for safety analysis of OPR1000 and Westinghouse nuclear power plants

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1. Introduction

The U.S.NRC(Nuclear Regulatory Commission) has conducted international joint study on the damage of LOCA cladding under high burnup conditions since 1998, raising the need to change the LOCA acceptance criteria in 2003 based on the results of experimental studies on Zirconium Based Alloy, which is widely used as a nuclear fuel rod cladding. The NRC is promoting legislation by announcing new ECCS(Emergency Core Cooling System) acceptance criteria reflecting the research results and opinions of public hearings[1]. ACRS(Advisory Committee on Reactor Safeguards), an advisory department for reactor safety experts at the U.S.NRC, recommended the final issuance of the Draft Final Rule, RG 1.222, RG 1.223, and RG 1.224, but it is not legislated as of 2021[2,3,4].

As domestic regulators(KINS) also want to legislate new ECCS regulatory requirements, since 2019, KHNP has been working on developing a safety analysis methodology for OPR1000 and Westinghouse-type nuclear power plants in preparation for new ECCS regulations. This paper describes the development status of SPACE-based LBLOCA and SBLOCA safety analysis methodology TR(Topical Report) that satisfy new ECCS requirements for OPR1000 and Westinghouse 3-Loop type nuclear power plant. It also includes the development status of SPACE-based Non-LOCA safety analysis methodology that satisfies new ECCS requirements for OPR1000 nuclear power plants.

2. Methods and Results

This section describes the LOCA, Non-LOCA safety analysis methodology, FFRD(Fuel Fragmentation, Relocation and Dispersal) model development, and safety analysis code quality assurance status.

2.1 LOCA safety analysis Methodology

The LOCA methodology for preparing for the new ECCS acceptance criteria is being developed as follows. The SPACE code is being modified in consideration of internal oxidation and CRUD thermal resistance effects of high burnup. And new uncertainty variables such as CRUD Thickness and CRUD Porosity were added. In addition, SPACE methodologies such as changes in SPACE core modeling, development of Burndown Curve based on nuclear design data, generation of nuclear fuel performance-based limitation, and FFRD model are being developed. Based on this, the inner oxidation model used in the new LBLOCA analysis methodology compared to the existing SPACE methodology is shown in Fig. 1.

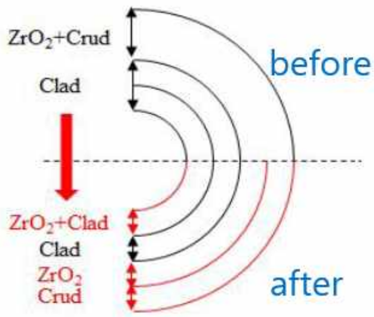

Fig. 1. Inner oxidation model improvement

2.2 Non-LOCA safety analysis Methodology

The Non-LOCA safety analysis methodology is being developed by two organizations. One organization develops methodology for FSAR chapter 15.1, 15.2, 15.5, 15.6, the other organization develops methodology for FSAR chapter 15.3, 15.4. In the case of FSAR 15.3,4, the current APR1400 SPACE Non-LOCA methodology is extended to OPR1000 while changing the Hot Channel model from RETRAN to SPACE and applying new RIA(Reactivity Initiated Accident) acceptance criteria to produce additional related data for CEAE(Control Element Assembly Ejection) analysis. FSAR 15.1,2,5, and 6 used the domestic code system by using SPACE and THALES codes instead of CESEC-III and CETOP-D, which were previously used. Representative accidents are shown in Table I, and the evaluation results are generally similar to the exiting results.

Table I: Non-LOCA analysis accident

FSAR	Accident
15.1	Inadvertent Opening of SG ADV Steam Line Break
15.2	Loss Of Condenser Vacuum Feed Line Break
15.3	Complete Loss of Reactor Coolant Flow Reactor Coolant Pump Locked Rotor
15.4	Uncontrolled RCCA Withdrawal at a Subcritical or Low Power Startup Condition Uncontrolled RCCA Withdrawal at Power Single RCCA Falling Spectrum of RCCA Ejection Accidents
15.5	PZR Level Control System MalFunction
15.6	Let Down Line Break SG Tube Rupture

2.3 FFRD Model

As nuclear fuel burnup increased, issues about fragmentation, location, and dispersion of nuclear fuel pellets were raised. The FFRD model has not been included yet in the U.S. ECCS new requirement(draft) 10CFR50.46c, but the FFRD model is being developed as there is a possibility that it may be included in the domestic new ECCS requirement. The developed models are being compared with other code calculation results and the experimental results, and the experimental verification results are shown in Fig. 2.

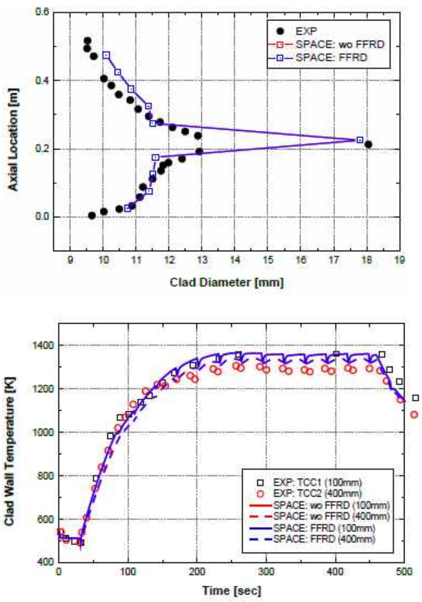

Fig. 2. Halden and SCIP Studsvik experimental verification results

2.4 Safety analysis code Quality Assurance

In the SPACE code used in this study, additional models were developed compared to the 2017 licensing version(SPACE 3.0), and existing models were also improved. In addition, it is necessary to establish the latest SQA(Software Quality Assurance) system as the latest SQA is applied not than when the license was obtained. Accordingly, KHNP has established and is operating the latest SQA system based on ASME NQA 2008 and 2009a. The KHNP SQA system is shown in Fig. 3.

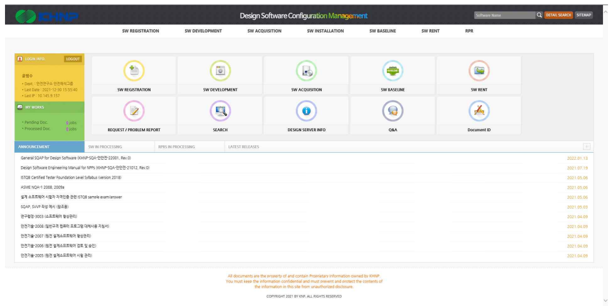

Fig. 3. KHNP SQA system.

3. Conclusions

In accordance with the new proposed ECCS requirements, it was necessary to analyze the performance of fuel rods according to the oxidation of nuclear fuel cladding and develop an optimal evaluation methodology using SPACE codes. This paper describes the contents of the development of LOCA and Non-LOCA safety analysis methodology and the establishment of a safety analysis code system for FFRD models as he contents of the progress of the study.

In the future, based on the above technology development contents, the safety analysis methodology that can respond to new ECCS requirements will be developed and published as TR to promote licensing.

REFERENCES

[1] Advisory Committee on Reactor Safeguards, “Proposed Draft Rule for 10 CFR 50.46c, ‘Emergency Core Cooling System Performance During Loss-of-Coolant Accident’,” January 26, 2012.

[2] U.S. Nuclear Regulatory Commission, Regulatory Guide 1.222 (draft), “Measuring Breakaway Oxidation Behavior,” January 19, 2016.

[3] U.S. Nuclear Regulatory Commission, Regulatory Guide 1.223 (draft), “Determining Post Quench Ductility,” January 19, 2016.

[4] U.S. Nuclear Regulatory Commission, Regulatory Guide 1.224 (draft), “Establishing Analytical limits for Zirconium-Alloy Cladding Material,” January 19, 2016.