Improved Fuel Performance Evaluation of SLB with Common Mode Failure for OPR1000 Reload Core

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1. Introduction

The CMF+SLB is beyond DBA* that assumes both Common Mode Failure of the digital safety related I&C system with Steam Line Break accident. The acceptance criteria of CMF+SLB is to prevent fuel damage by accident. The fuel damage is checked by maximum local power density (LPD) corresponding to the fuel melting temperature. According to SRP BTP 7-19 [1], vendor or applicant should analyze postulated CMF for each event in the accident analysis section of SAR using best-estimate methods (realistic assumptions) or SAR analysis method.

Currently, the fuel performance evaluation for CMF+SLB is performed with consideration of the physics characteristics of reload core, core axial power distribution shift by the change of core thermal hydraulic conditions and additional reactivity insertion by the xenon distribution. It is difficult to meet the acceptance criteria considering those conservative conditions

This paper introduces the improved methodology using the coupled code (NSSS simulation code and nuclear core design code) to solve the current situation. Also the fuel performance evaluation result with improved methodology using maximum LPD are described.

* Design Basis Accident

2. Improving Fuel Performance Evaluation for CMF+SLB

Analytical methodologies have been developed by optimizing the calculation of total reactivity in CMF+SLB accidents. The procedures of existing methodology and improved methodology are described in Fig. 1 and Fig. 2, respectively. The improved methodology removes the excessive conservatism of existing methodology through the application of optimized parameters including fuel rod information. In the event of CMF+SLB, the transient thermal hydraulic condition calculated by the NSSS simulation code is used as a boundary condition to the nuclear core design code that performs the 3-D reactivity calculation.

The NSSS simulation code using point kinetics model has the excessive conservative assumption for reactivity calculation, so it calculates more severe total reactivity than 3-D core design code. Therefore, the improved methodology using 3-D core design code helps to mitigate the severe analysis results.

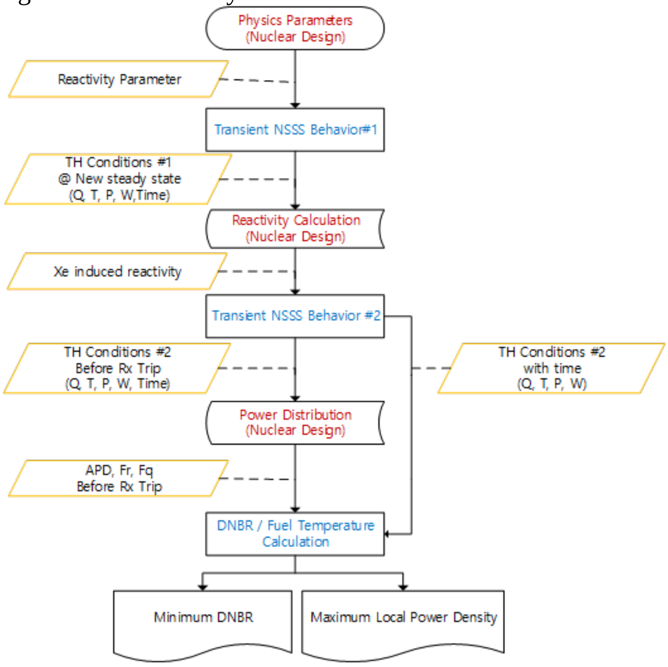

Fig. 1. Flow chart of existing methodology

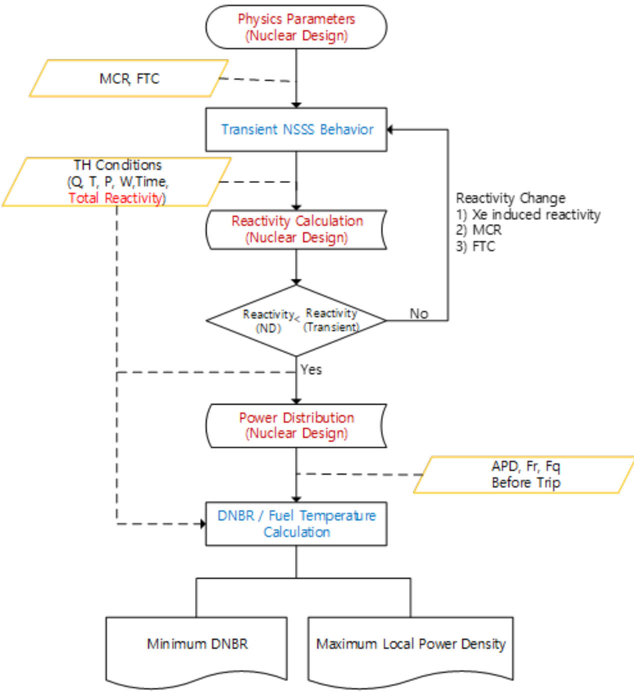

Fig. 2. Flow chart of improved methodology

3. Nuclear physics and fuel rod data

For fuel performance evaluation of CMF+SLB, the NSSS simulation code calculated power for post condition. This result is used in nuclear physics and fuel rod data generated by the nuclear core design and fuel behavior analysis code.

3.1 Nuclear physics data

Nuclear physics data (Moderator cooldown reactivity (MCR), Fuel temperature coefficient (FTC), Kinetic parameter, etc.) were produced with reload core. These reload core data is more conservative than the initial core.

3.2. Fuel rod data

The fuel rod data containing initial condition of fuel has been produced by fuel performance analysis code. To cover all fuel loaded in OPR1000 reload core, both average rod data and specific hot rod data are used for the fuel performance evaluation.

3.3. Determination of maximum local power density

The fuel performance evaluation is performed by checking the fuel melting temperature which decreases as burnup increases. Conventional fuel performance evaluation determines the maximum LPD and use this value as an acceptance criteria. Since fuel power generated at once and twice burnt fuel is smaller than fresh fuel, local power is decreased after once burnt fuel. The fuel performance evaluation verifies the maximum LPD results to meet the criteria.

4. Fuel Performance Evaluation for CMF+SLB

During the CMF+SLB transient, the maximum core power is decreased because of the optimized reactivity value by coupled code. The NSSS simulation code uses point kinetics and nuclear core design code uses 3D kinetics. The nuclear core design code can calculate the axial power distribution based on thermal-hydraulic condition change. The moderator temperature which was produced by the NSSS simulation code and the nuclear core design code are shown in Fig. 3. It minimize the decrease of the core moderator temperature. As a result, the core power is decreased as the decrease of positive reactivity due to the MCR added in the core during the CMF+SLB accident.

The calculation of maximum LPD during the CMF+SLB accident used in maximum core power, 3D peaking factor and core average linear power density. Since the maximum core power was reduced by improved methodology, the maximum LPD results is also decreased.

Although the results of maximum LPD may exceed the acceptance criteria with conservative nuclear physics data in consideration of reload core, the improved methodology was applied to meet the criteria.

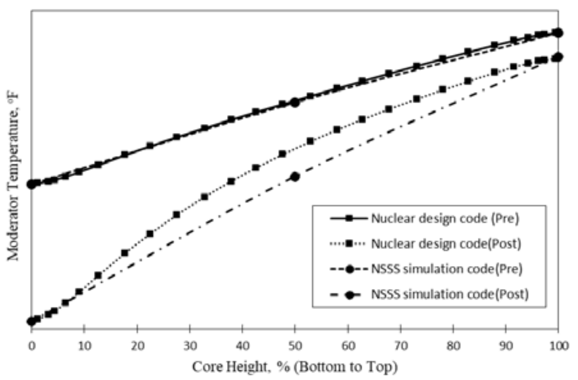

Fig. 3. Comparison of Moderator Temperature

5. Conclusions

The fuel performance evaluation with improving methodology during the CMF+SLB was performed for OPR1000 reload core.

Nuclear physics and fuel rod data were produced to carry out the fuel performance evaluation, and data were used to verify the melting of the fuel. During the response of CMF+SLB for OPR1000 reload core, it was confirmed that fuel melting is not occurred.

REFERENCES

[1] “Guidance for evaluation of diversity and defense-in-depth in digital computer-based instrumentation and control systems review responsibilities,” NUREG-0800 BTP 7-19 R7, Aug. 2016.