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Feasibility Study on the Rod Ejection Simulation with Point Kinetics Model

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1. Introduction

For the export of the nuclear fuel, Korea Nuclear Fuel (KNF) is developing the KNF-owned safety analysis method. As a part of this project, the alternative approach of rod ejection simulation compared to the Final Safety Analysis Report (FSAR) [1] is studied. Per the Kori units 3 and 4 FSAR, one-dimensional neutron kinetics model is used for the rod ejection simulation. Generally, RETRAN [2], which simulates the response of nuclear steam supply system for most of transients with point kinetics model, is not applied to simulate the fast reactivity-induced transient such as the rod ejection. However, KNF attempts to use RETRAN instead of the existing code for the rod ejection transient analysis to establish the Integrated Safety Analysis Methodology (ISAM). The essential process in this work is to review the applicability of RETRAN. In this paper, presented are the results of this feasibility study.

2. Analysis Methods

In this section, the important models used in this feasibility study are described.

2.1 Description of the Rod Ejection Transient

This transient is the result of the assumed mechanical failure of a control rod mechanism pressure housing such that the reactor coolant system pressure would eject the control rod and drive shaft to the fully withdrawn position. The consequence of this mechanical failure is a rapid reactivity insertion together with and adverse core power distribution, possibly leading to localized fuel rod damage.

If a rod ejection transient were to occur, a fuel rod thermal transient which could cause DNB (Departure from Nucleate Boiling) may occur together with limited fuel damage. The amount of fuel damage will be governed mainly by the worth of the ejected rod and the power distribution attained with the remaining control rod pattern. The transient is terminated by the Doppler reactivity effects of the increased fuel temperature and by reactor trip actuated by neutron flux signals, before conditions are reached that can result in damage to the reactor coolant pressure boundary, or sufficiently disturb the core, its support structures or other reactor pressure vessel internals to impair significantly the capability to cool the core.

2.2 Method of Analysis

Generally, the calculations of the rod ejection transient are performed in two stages; first an average core calculation and then a hot spot calculation. The average core calculation is performed using neutron kinetics methods to determine the average power generation with time including the various reactivity feedback effects; that is, Doppler and moderator feedback. The fuel enthalpy and temperature transients in the hot spot are then determined by performing a detailed fuel rod transient heat transfer calculation with the peak power at the hot spot. Peak power at the hot spot can be calculated by multiplying the average core power by the hot channel factor.

2.3 Reactivity Feedback Models

The most important reactivity feedback is the Doppler feedback. The Doppler reactivity feedback plays a decisive role to halt the power excursion due to the positive reactivity insertion from the ejected rod. To guarantee the desired Doppler reactivity feedback equal to the conservative design value, the iterative scheme to search the Doppler power defect is employed. To ensure the conservative Doppler reactivity feedback, the fuel temperature coefficient of RETRAN is adjusted.

The moderator temperature reactivity feedback is a bit insignificant because the time duration of the rod ejection transient is very short.

2.4 Fuel Rod Model

RETRAN fuel model employs six regions; four in the fuel, one for the gap, one for the clad and one for the coolant respectively. And the axial region is divided by twenty nodes; two for the inactive core and eighteen for active core. Thermal properties of fuel as function of temperature are used and the expansion of fuel during the transient is considered.

2.5 Reactor Scram Model

After the reactor trip signal occurred, the scram rods begin to drop into the core with some delay. The trend of the negative reactivity insertion of the reactor scram affects the energy deposition in the fuel rod. RETRAN needs to input the explicit scram curve in the form of the time versus reactivity table. In this study, the scram curve shown in the Fig. 1 is used. This curve is produced by the iterative method based on the comparison of the average core power after the reactor scram.

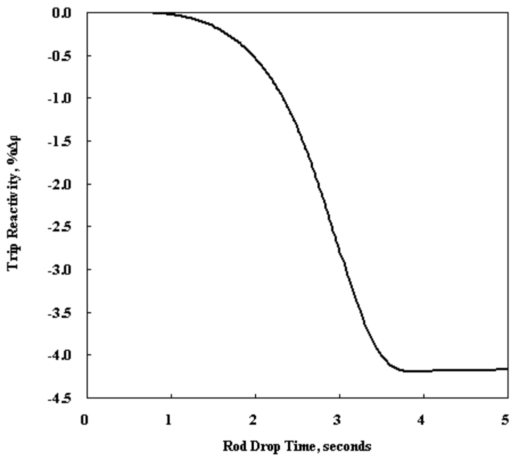

Fig. 1. Reactor scram curve.

3. Results of Comparison

In order to assure the suitability of RETRAN method, comparisons of key important parameters in the FSAR are performed. The values referred as "FSAR" in the figures are obtained from the FSAR using the converting tool.

First, the average core powers are compared to check the reactivity feedbacks and the overall transient behaviors. Per the comparison result as shown Fig. 2, there is a resemblance between two curves. This means RETRAN model can simulate the reactivity behaviors during the transient analogous to those in the FSAR.

Second, the fuel temperatures at the hot spot, which are key safety parameters to be checked with respect to the melting and the enthalpy deposition, are compared in Fig. 3. Comparison result shows a close similarity between them. Therefore, it is found that the fuel performance during the transient can be predicted properly.

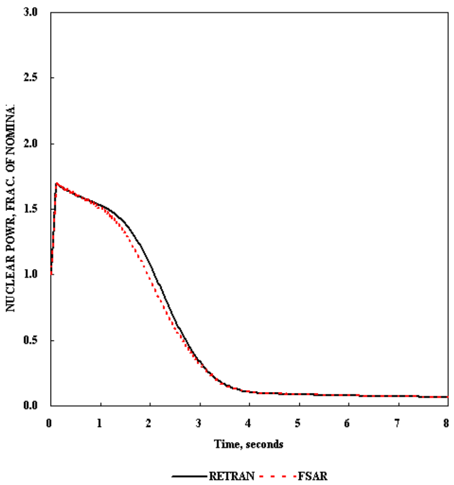

Fig. 2. Comparison of average core powers.

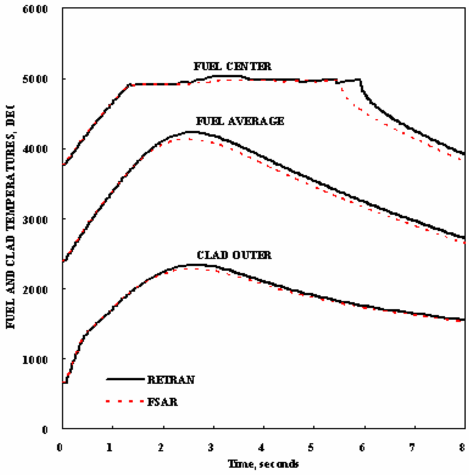

Fig. 3. Comparison of fuel temperatures at hot spot.

4. Conclusions

In order to investigate the fitness to adopt RETRAN as a tool for analysis of the rod ejection transient, the appropriate RETRAN model for the fast transient is developed. Comparisons of key important parameters between RETRAN and FSAR calculations are performed. According to the results, the RETRAN model can be used for the analysis of the fast transient such as the rod ejection.

REFERENCES

[1] "Kori Nuclear Power Plant Units 3 and 4 Final Safety Analysis Reports," Westinghouse Electric Company, January 1975.

[2] "RETRAN-3D - A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid-Flow Systems," Electric Power Research Institute, December 1997.