Prediction of System Thermal-hydraulic Response following an ATWS Event

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Abstract

The present paper describes the evaluation of the capability of the RELAP5/MOD3 code to predict the system response following an Anticipated Transient Without Scram (ATWS) event. The experiment L9-3, a unique nuclear experiment simulating an ATWS event induced by loss of feedwater accident in Loss-of-Fluid-Test (LOFT), is calculated. The experimental condition and sequence are reviewed and a calculation modeling is developed with the important test-specific features. The RELAP5 calculation result is compared with the experimental data and the predictability of the system response of the RCS, the reactor power, and the SG secondary system is analyzed. As a result, it is shown that the RCS thermal-hydraulic response, the reactor power response, and the secondary system response following the LOFT L9-3 experiment can be reasonably predicted by the RELAP5 code under the current modeling scheme, and thus, that the code can be reasonably applied to the analysis of thermal-hydraulic response following the ATWS in real plant. Also it indicated that the further sensitivity studies to improve the predictability are needed on the effect of steam generator (SG) modeling, the pilot operated relief valves/safety relief valves (PORV/SRV) discharge modeling, and the moderator temperature coefficient (MTC) feedback on the system response.