SAMPSON PWR Validation by TMI-2 Accident

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Following the severe accidents at the Fukushima Daiichi Nuclear Power Station, benchmark studies have been performed to assess BWR models of SAMPSON severe accident code against the three accidents leading to significant improvements in the code BWR version. However, despite that notable improvements have also been made in the PWR version of the code, such as for the thermally induced rupture of reactor cooling system and for applications to ATWS scenarios, it's likely that some thermohydraulic models of core degradation still need improvements. Therefore, simulation in the code of TMI-2 accident is being performed in order to assess core reflooding models during the quenching phase. Preliminary results show good agreement with TMI-2 data regarding zircaloy oxidation and hydrogen generation.

Keywords: SAMPSON code, severe accident, Three Mile Island Unit-2, zircaloy oxidation, fission product release, hydrogen production

1. Introduction

SAMPSON is an integral severe accident analysis code which has been developed by the Nuclear Power Engineering Corporation (NUPEC) with the support of the Japanese Ministry of International Trade and Industry [1]. The main purpose of the code is to simulate all the severe accident phenomena during light water reactor severe accidents from an initiating event till containment failure and release of fission products to the environment. The modelled phenomena include: molten core concrete interaction, fission product transport, steam explosion and hydrogen detonation. The code can be used also for design basis accidents as well as for Level 1 and Level 2 PRA [2].

However, the code validation by TMI-2 International Standard Problem [3] is still necessary. Therefore, the present study is a first validation attempt of SAMPSON PWR version through simulation analysis of TMI-2 accident.

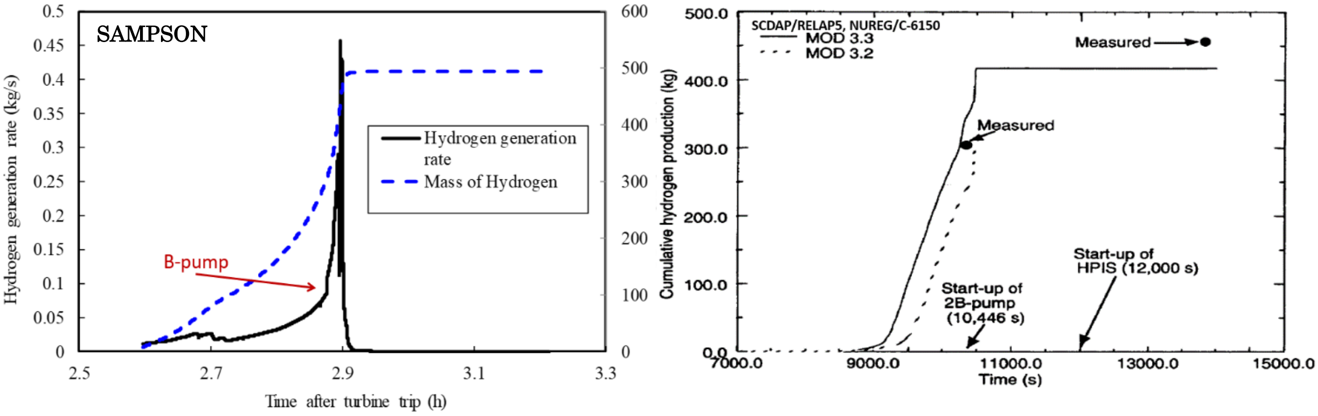
2. Analysis Description

Major components of TMI-2 reactor coolant system are represented in SAMPSON model which uses the same nodalization as the original model of SCDAP/RELAP5. SAMPSON Thermal-Hydraulics Analysis module (THA) is used to simulate the thermal-hydraulics of the reactor coolant system. The core is divided into 5 parallel channels and 10 axial nodes. Steam generator secondary side coolant mass flowrates, pressures, and feedwater temperatures, and primary side makeup and letdown flowrates are supplied as boundary conditions. The Fuel Rod Heat-up Analysis module (FRHA) of SAMPSON is used to simulate TMI-2 reactor core. FRHA core input model is obtained by direct conversion of the original SCDAP core input model.

SAMPSON TMI-2 model verification was made through steady state analysis which showed good comparison between the calculated initial conditions and TMI-2 data. The transient calculation covers the three phases of the accident.

3. Results and Conclusion

TMI-2 analysis allowed for comparison of the model predictions in SAMPSON to plant data. This exercise was, therefore, valuable for verifying and assessing the models in the code. The major trends in the TMI-2 accident are reasonably well predicted by SAMPSON especially the hydrogen production during core reflooding as shown in the comparison Figure below between SAMPSON, SCDAP/RELAP5 and TMI-2 data. Based on this comparison, it can be concluded that SAMPSON is therefore validated for PWR severe accident analysis.



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