Assessment of Core Status of TEPCO's Fukushima Daiichi Nuclear Power Plants
(85)Analysis of Fukushima Daiichi Unit 2 "3-peaks region" by the SAMPSON code based on the conclusions obtained from CORA-17 experiment validation

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Abstract

The aim of this work is to perform a detailed analysis on the "3-peak region" observed during the Fukushima Unit 2 accident transient after the depressurization of the RPV. Due to the complex phenomena involved in the core reflooding process, the SAMPSON code has been validated using CORA-17 experiment. Then, the code is used to analyze the "3-peak region" in the accident transient.

K**eywords:** Severe Accident, Reflooding, CORA-17, "3-peak" unit 2

1. Introduction

The "3-peak region" observed in the 1F2 transient is still under discussion. The causes of the pressure peaks that occur after the RPV depressurization are still unknown. Several assumptions have been made; including the core slumping and hydrogen generation due to reflooding, being the later the most consistent with the data onsite. The complex phenomena involving the reflooding process presents a challenge to the several severe accident codes which none of them has been able to reproduce it accurately.

For the SAMPSON code validation purposes, a more controlled environment has been used employing the CORA experiments data. CORA-17 includes a relooding phase that injects water into a damaged BWR bundle. Using the temperature and hydrogen data from the experiment, the SAMPSON code capabilities can be checked, and, if it is necessary, update models can be developed.

Once the code is validated, it is used to analyze the 1F2 accident focusing on its performance during the "3-peak region".

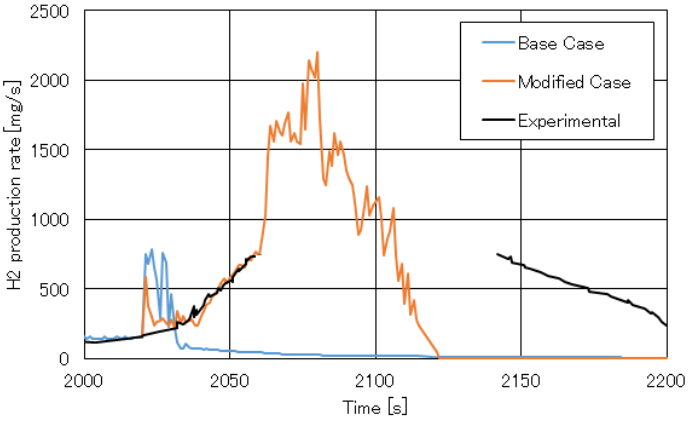

Fig 1. CORA-17 H_2 generation during reflooding [1]

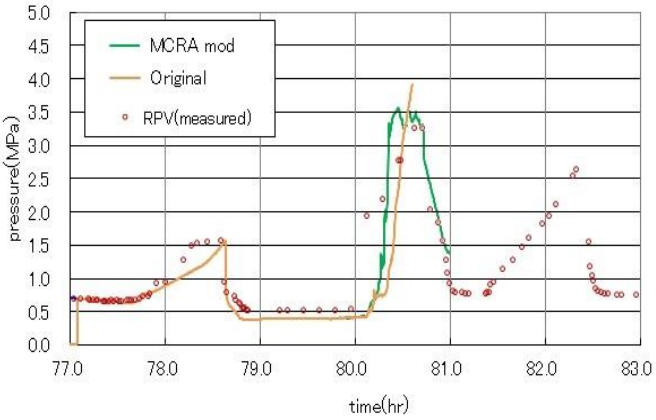

Fig 2. SAMPSON "3-peak region" simulation results with MCRA modification [2]

2. Results and discussion

The initial results of the SAMPSON code showed a large divergence with the CORA-17 data. The code computed a small and more spontaneous hydrogen release once the water refloods the core. In order to overcome the difference, new models have been tested and implemented on MCRA and FRHA modules in order to compensate the Zr mass that interacts with the steam. The newest results show an improvement and a better agreement with the data and the post bundle images of the experiment [Fig.1]

The 1F2 analysis is more complex, mainly due to the difference between the controlled environment of CORA and the real accident conditions. Thus, the influence and importance of the developed models have to be tested in a full plant analysis. In the preliminary analysis shown in [Fig. 2], the original results show a RPV failure at the summit of the second peak meanwhile, with the modifications, the results show better agreement with the data.

3. References

[1] Sepold,L. et. al. *Behavior of BWR-type Fuel Elements with B4C/Steel Absorber Tested under Severe Fuel Damage Conditions in the CORA Facility*. Forschungszentrum Karlsruhe. Jan 2009
[2] Information Portal for the Fukushima Daiichi ACCIDENT ANALYSIS and Decommissioning Actvities https://fdada.info/en/home2