Preliminary Analysis of Experiment for Thermo-Mechanical and Hydraulic Behavior during LOCA

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1. Introduction

Licensing and safety criteria of nuclear fuel are changing due to the adaptation of design extension conditions (DEC) and safety issues related to a high burn-up fuel. A proposed new LOCA acceptance criterion requires a multi-physics coupled safety analysis. In a domestic situation, a study for the multi-physics coupled safety analysis was not performed until now and a simple connection method for the thermal hydraulic code and foreign mechanical code is being progressed. Therefore, thermo-mechanically coupled experiments are required in order to assess a new LOCA acceptance criterion and to validate a multi-physics coupled safety code system.

A preliminary calculation for an experimental facility for thermo-mechanical and hydraulic behavior is required to validate the experimental conditions. Another purpose of preliminary calculation is to check the ability of recent thermal-hydraulic safety analysis code for the multi-physics coupled conditions.

2. Design of Test Facility

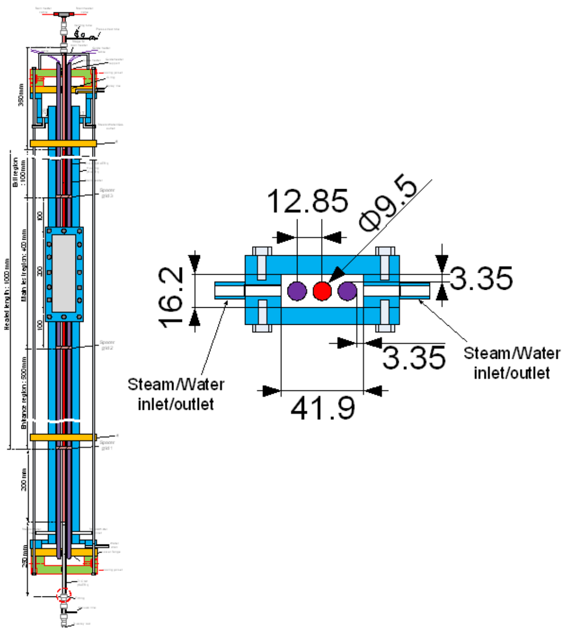

Fig. 1. Design of the Test Facility

Fig. 1 shows a design of the test facility to investigate the thermo-mechanical and hydraulic behavior during LOCA. Three heaters were considered for the multi-physics study as the 1st phase stage. As the second stage, 5X5 rod bundle geometry will be considered in the future. Among the three heaters, main heater is located at the center and two adjoint heaters were installed for the guide heating function. The current study referred to the 1x3 heater bundle design for the preliminary performance calculation.

3. Performance Calculation of the Test Facility

The MARS-KS 1.5 was utilized in the current calculation. In the experiment, the nuclear fuel is simulated by using heater which has a complicated internal geometry and various materials. The current calculation divides the heater structure into three parts which are heating section, gap and cladding. Among them, the cladding and gap is simulated without assumption for the geometry. The heating section representing nuclear fuel pellet is simplified by a homogeneous material having the same outer diameter of the heater. The average property of the nuclear fuel were analyzed in advance and applied to the current calculation.

Fig. 2 shows a node diagram applied to the current calculation, in which the major values of the hydraulic volumes and structures were also written together in the figure. For simplification, the heating section is considered to have 1.0m length and unheated section is positioned at downstream of the heated section. Three heaters were simulated in the calculation according to the test facility design. The guide heater does not have the cladding unlikely the main heater located at the center of the rod bundle. The same power was applied to three heater. The heated section was simulated using “pipe” component having five sub-volumes. The upper unheated volume was also simulated by “pipe” component which is to prevent the undesired numerical instability as well as to simulate the real situation.

The thermal-hydraulic boundary condition was determined in the range of the experiment requirements which was set based on the previous studies. The flow condition was controlled to get the desired heater deformation while the major thermal hydraulic status is controlled constant. In another word, the heater deformed or rupture condition was found by controlling the injected coolant flow under the fixed heater thermal power, fluid system pressure, cladding internal pressure and initial fluid and heater temperature.

For initial 100 seconds, the heater was maintained adiabatically. The flow and power were simultaneously applied at the instant of the 100 seconds.

The MARS gap conductance model includes the physics for the cladding as well as for the fuel rod pellet, and gap. Since the current study assumes no deformation of the fuel pellet, the gap conductance due to the deformation of the fuel was not considered. Only the properties of the nuclear fuel pellet such as volumetric conductance and heat capacity were considered in the calculation. The oxidation of the nuclear fuel was not considered either in this study.

Since the thermal hydraulic conditions adopted in this study induce an abrupt quenching phenomena, a drastic variation of the temperature on the heater surface is expected. Therefore, A thermal front tracking model was adopted for the volume V110 including the heater.

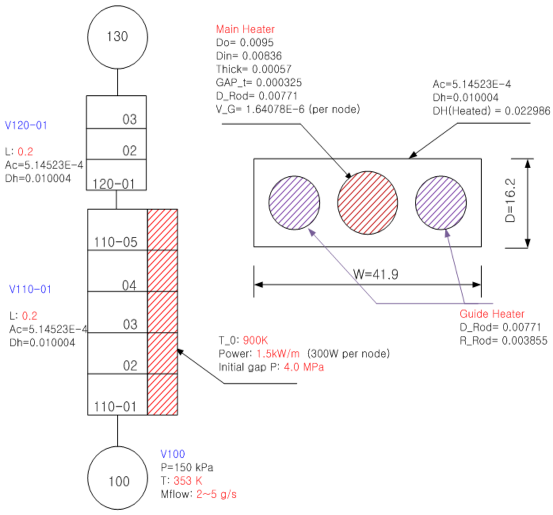

Fig. 2. Node Diagram of Problem

Table I shows the problem definition and major boundary conditions of the calculation. The current case shows a rupture behavior for the 3.0 g/s of injected coolant flow condition. If the flow is further reduced, the strain is enlarged and the rupture occurs more quickly. The cladding near the flow inlet is quenched just after flow is injected, in the meanwhile, the temperature of the upper part of the heater shows an increasing trend for a while and a quenching process were shown from the lower node of the heater as shown in Fig. 3. The cladding internal pressure has an unstable trend and shows a peak as the cladding temperature is abruptly increased, which accompanies a rupture. After the rupture, the internal pressure is set to be same values as the fluid volume. The strain has maximum value at the instant when the cladding internal pressure has maximum value, which is rupture condition. The maximum strain is maintains a constant which means the de-elastic deformation. Fig. 4 shows the cladding strain at the upper part of heater, 5th node in the node diagram. The outer radius shows a consistent change according to the cladding strain variation.

Table I: Problem Description and Major Boundary Conditions

Parameter	Value
Heater Power (kW/m)	1.5
Transient Initiation (sec)	100
Pressure (V130) (MPa)	0.14
Cladding Internal Pressure (MPa)	4.0
Injected Coolant Temp. (K)	353
Coolant Flow (g/s)	***3.0 g/s***
Coolant Initiation (sec)	100
Initial Heater Surface Temperature (K)	900 K

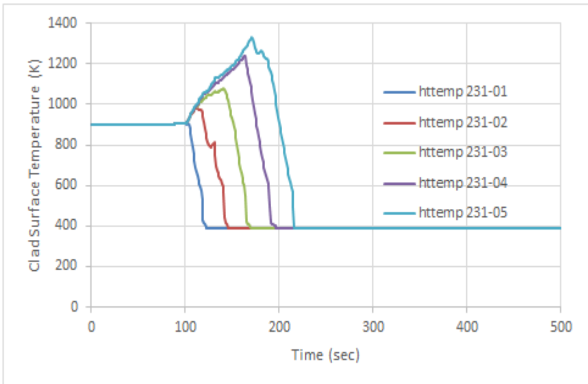

Fig. 3. Cladding Surface Temperature

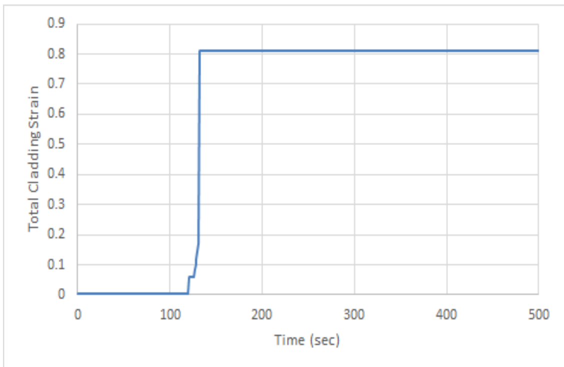

Fig. 4. Cladding Strain

4. Conclusions

An analysis for the multi-physics coupled phenomena expected at the test facility for the thermos-mechanical and hydraulic behavior during the LOCA were performed in the present study. Both of thermo-mechanical and hydraulic behavior were calculated for the preliminary results. It was concluded from the analysis that multi-physics phenomena can be predicted by using multi-physics coupled safety code. However, the further development should be performed based on the following information;

- More detail specification of the test facility
- Property variation according to temperature
- Real thermal hydraulic initial conditions including heat loss

During this task, the foundation for multi-physics coupled safety analysis was built up. The calculation results will be used to set the experimental conditions at early stage of experiments. Finally, the results will be used as a reference for the validation of safety analysis code with experimental data base.

ACKNOWLEDGMENTS

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