Effect of Rod Internal Pressure on Simulation of Halden Test IFA-650.9 with FE-based Fuel Analysis Code MERCURY

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1. Introduction

Analysis for the behavior of nuclear fuel in loss of coolant accidents in a nuclear reactor is being emphasized from the viewpoint of the characteristics of the nuclear fuel materials and the ability to maintain the coolable geometry. In particular, a nuclear fuel performance code based on the finite element method is being developed for the realistic nuclear fuel analysis. As a representative code, the finite element method-based BISON code [1] for the evaluation of nuclear fuel performance under steady-state and accident conditions has been developed in the CASL project of the US and the version is being continuously updated. In KAERI, a multi-dimensional code development project for nuclear fuel analysis of accident conditions is being carried out, and a code named MERCURY [2] is being developed.

In the process of developing the analysis code, it is necessary to validation for the various experimental data with the verification process. In this paper, the analysis results are presented in terms of the rod internal pressure of the MERCURY code for IFA-650.9, which is one of the IFA-650 series [3], which is the LOCA simulation test of the Halden Research Reactor for in-pile test data.

2. Halden Test, IFA-650.9

In the Halden Research Reactor, the IFA-650 series to understand the behavior of nuclear fuel during LOCA was carried out as an OECD-NEA project. The IFA-650.9 test was carried out in April 2009 using a high burn-up PWR rod of 89.9 MWd/kgU from AREVA. The test was carried out using the low fission power to achieve the desired conditions for high temperature, ballooning, and oxidation. The results showed that the ballooning in the lower part of the rod occurred and was followed by a cladding burst and fuel relocation as shown in Fig. 1 [4]. The IFA-650.9 is also presented as a benchmarking problem of IAEA CRP FUMAC [5]. In this project, the importance of thermal-hydraulic boundary conditions was emphasized in the analysis of nuclear fuel performance, and well-turned thermal-hydraulic boundary conditions were presented through the SOCRAT code of Russia for common boundary conditions [5].

In this study, the axial temperature of the cladding surface was given as a user input of the MERCURY code and the results of the internal behavior of the fuel rod from the cladding surface were evaluated. Since the MERCURY code does not analyze the coolant outside the fuel rod, it is reasonable to bring and use the appropriate boundary conditions to evaluate the characteristics of MERCURY code.

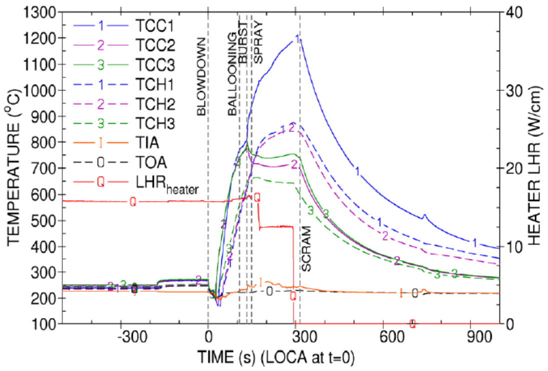

Fig. 1 Cladding and heater power in IFA-650.9 test [4]

3. Results of MERCURY Code

3.1 Geometry Mesh and Boundary Conditions

Since the MERCURY code is based on the finite element solution, the geometry is composed of 2126 nodes and 576 elements for two volumes of the fuel pellet and cladding. In the analysis model, the axial length of the fuel is 480 mm, and the outer diameter and thickness of the cladding are 10.75 mm and 0.725 mm respectively. Displacement in the axial direction was fixed as a boundary condition.

As a boundary condition for thermal analysis, the outer surface temperature of the cladding provided by the IAEA was assigned as shown in Fig. 2. One cladding surface thermocouple was located bottom region and another two thermocouples were attached a same level of upper region. Since the wall temperature was measured at two axial levels in the actual experiment, 21 temperature distributions along the axial direction were obtained analytically based on the experimental results.

3.2 Rod Internal Pressure and Ballooning

The internal pressure of the fuel rod is determined by the free volume, temperature, composition of the gases in the fuel rod. The free volume consists of the upper plenum with the gap between the fuel pellet and the cladding. In the MERCURY code, the calculation of the thermal-hydraulic condition is not performed, so the outermost boundary condition is the temperature on the cladding surface given by the user. In a nuclear reactor, since the temperature of the upper cladding and coolant are almost similar, it may be appropriate to assume the plenum temperature based on the temperature of the cladding. In experiments that measures pressure like the IFA-series, however, the gas temperature of the pressure line reaching the pressure sense should also be considered. Therefore, assuming the temperature of the plenum gas from the temperature of the cladding can lead to large errors. In the Fig. 3, the comparison of the results of the case where the internal pressure was calculated using the default option of the MERCURY code, which is to add 5 K to the cladding surface temperature and the case where the measured internal pressure was applied as the fixed condition. When the default option is used, the free volume temperature is applied too high, and the internal pressure is predicted to be significantly higher than the pressure measured in the experiment.

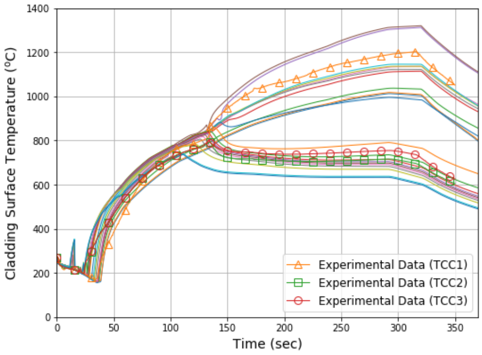

Fig. 2 The nodal temperature on cladding surface for boundary condition. (The line plots are nodal temperatures given from SOCRAT analysis).

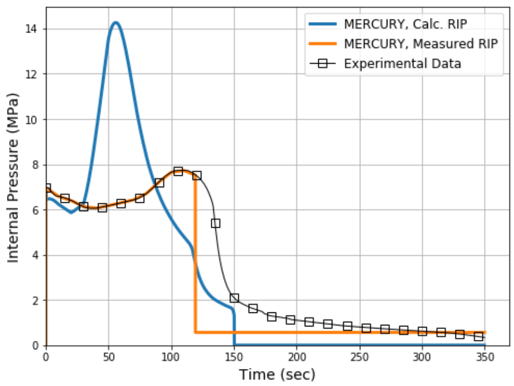

Fig. 3 The rod internal pressure calculated using default option of MERCURY and experimental measured data.

The cladding deformation at the time of cladding rupture as a result of the MERCURY analysis is shown in Fig. 4. In the two cases, the difference in rupture time occurred, resulting in a clearly different deformation shape. Since the creep model that analyzes large deformation is composed of a function of temperature, the shape of deformation is determined by the time of rupture and the temperature of the surface at that time.

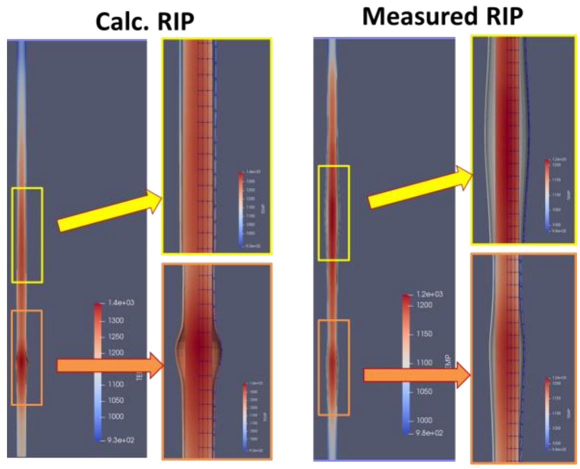

Fig. 4 Fuel temperature and deformation shape at the time of rupture.

Fig. 5 shows the temperature of the cladding surface at the time of rupture. When the default option was used, high temperature was concentrated in the lower part and large deformation occurred concentrated in the lower part, whereas when the internal pressure was fixed as the experimental value, two high temperature regions appeared, which resulted in two deformation peaks.

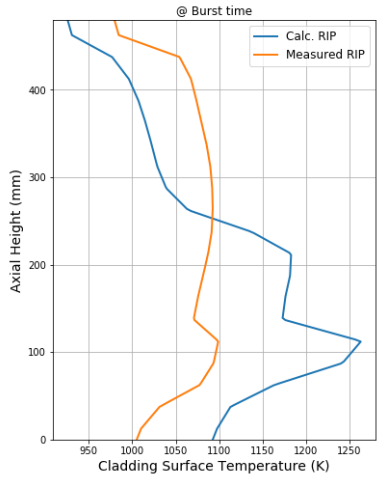

Fig. 5 Cladding surface temperature at the time of rupture

4. Conclusions

The analysis of LOCA simulation test, IFA-650.9 was performed using the accident condition nuclear fuel analysis code, MERCURY. In the analysis of IFA-650.9, the evaluation of the fuel rod internal pressure has a great influence on the cladding deformation. As an important factor in predicting the internal pressure of fuel rods, an appropriate assumption about the gas temperature is required. In Mercury, an option to apply the temperature of the free volume as a function of time by a user input has been added, and the results of the temperature effect through this will be summarized and presented later.

Acknowledgement

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