Evaluation of FRAPCON3.5 Enhancement with IFA-597 experiment

Young-jun Kim*, Hyo-chan Kim, Yong-Sik Yang
LWR Fuel Technology Division, Korea Atomic Energy Research Institute,989-111 Daedeok-daero, Yuseong-gu, Daejeon, Korea, 305-353
**Corresponding author: yjkim05@kaeri.re.kr*

1. Introduction

The fuel performance is closely related to safety and economy of nuclear power reactor. In a commercial reactor, the long-term and high burn-up reactor operating has been performed to increase economics. The fuel performance test data for long-term and high burn-up are necessary to secure the fuel integrity. For this reason, a power ramp test has been performed to observe behavior of high burn-up fuel in a short-term using a re-instrument fuel rod. The re-instrument fuel rod is made of the spent fuel discharged from the commercial reactor and equipped with measurement devices.

The re-instrument fuel rod can be easily simulated using a re-fabrication option newly added in the FRAPCON-3.5.

The purpose of this study is to understand the re-fabrication option and evaluate the in-pile behavior of fuel rods for long-term and high burn-up.

The re-instrument fuel which was base-irradiated in commercial reactor and re-irradiated in the Halden reactor for power ramp test has been simulated using FRAPCON-3.5 code. Based on the simulation results, the in-pile behavior of re-instrument fuel rods has been studied.

2. FRAPCON-3.5 Code

FRAPCON-3.5 is the latest version of the NRC`s fuel performance code for the calculation of steady-state thermal-mechanical behavior of light-water reactor (LWR) oxide fuel rods for long-term and high burn-up.

Enhancements of FRACON-3.5 related this study are as follows: 1) Increasing number of time step and axial node; 2) Ability to input the specified axial nodes length; 3) Ability to specify different axial nodes with different central hole radii, or no central hole; 4) Added re-fabrication ability [1].

From these enhancements, power histories can be simulated in more detail and it is possible to input the height of axial nodes as a variable. A part of axial node can be set as the re-fabricated fuel after a certain time step using the re-fabrication option. Also, initial conditions of re-fabricated fuel rod such as plenum length, spring dimension, fill gas pressure and gas composition can be input.

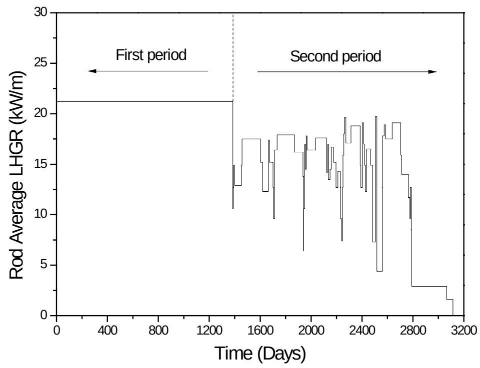
3. In-pile test data

3.1 IFA-597.2/3 rod 8 experiment

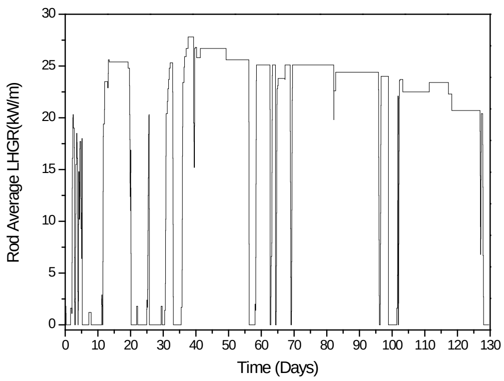
For the simulation, the in-pile test database IFA-597.2/3 experiment was prepared. Three segments from ABB 8x8 BWR fuel rod discharged from Ringhals 1 power plant have been re-instrumented. Two segments were equipped with fuel centerline thermocouples and bellows pressure transducers. These two segments are re-irradiated in the Halden reactor as IFA-597.2. After about two weeks of irradiation, one of fuel rods failed. The failure rod was removed and replaced with a third segment which was re-instrumented with a cladding elongation transducer. The assembly was reloaded in IFA-579.3.

3.2 Data preparation

In this study, the in-pile test database of rod 8 irradiated in IFA-597.2/3 was used. The rod 8 was base-irradiated up to 68MWd/kgU and re-irradiated in the Halden reactor for power ramp test [2]. The base and power ramp irradiation histories were shown in Fig. 1(a) and Fig. 1(b), respectively.



(a) Linear heat generation rate history of base irradiation



(b) Linear heat generation rate history of power ramp

Fig. 1 Power history of the test rod (IFA-597.2/3 rod 8)

In the base-irradiation, it is assumed that axial profile of the test rod is flat according to database report. In ramp test, the various axial power profiles were condensed into the twenty axial profiles. The schematic diagram of axial node and representative axial profile for ramp test are shown in Fig. 2.

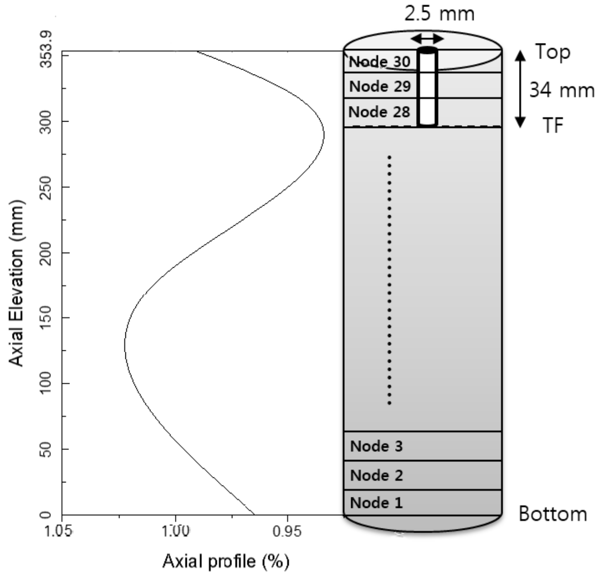


Fig. 2 Schematic diagram of axial node and representative axial profile for power ramp test

The code input information referred to the Halden reactor project (HRP) report [3-5] and test fuel data bank (TFDB). The coolant mass flux and inlet temperature are given. However, those conditions were slightly adjusted to match the given cladding temperature because all conditions of test reactor are not provided.

4. Results and discussion

The in-pile data base and the HRP report include the fuel centerline temperature recorded by thermocouple and fission gas release (FGR).

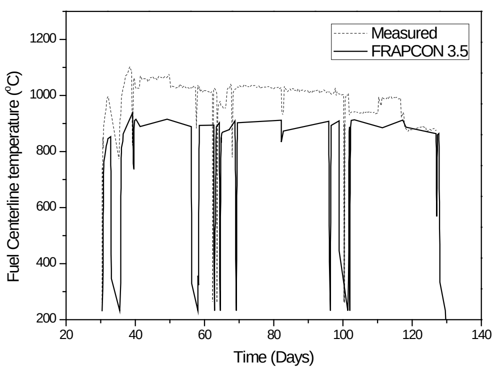


Fig. 3 Fuel centerline temperature for power ramp test

Fig. 3 shows the comparison of the fuel centerline temperature recorded by thermocouple with calculated by FRAPCON-3.5. The results show that calculated data are under-estimated. To understand the discrepancy clearly, the fuel centerline temperature versus the linear heat rate during first four ramps and prior to the final shutdown are shown in Fig. 4 and Fig. 5, respectively. During the first four ramps, the fuel centerline temperature is almost same at low linear heat rate, however, the temperature difference increases with increasing linear heat rate.

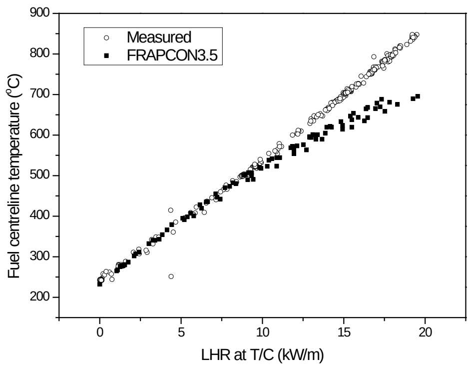


Fig. 4 Fuel centerline temperature versus linear heat rate during first four ramps

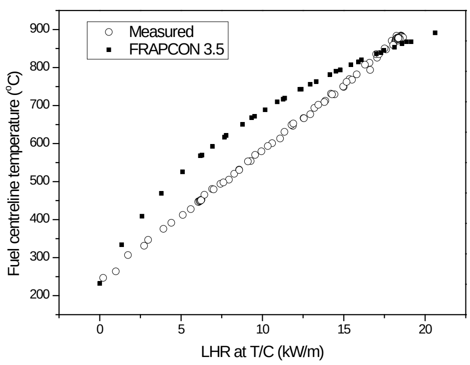


Fig. 5 Fuel centerline temperature versus linear heat rate prior to final shutdown

The temperature difference between the measured and calculated data is also presented in the HRP report [5]. The HRP report described that the temperature difference resulted from the thermal degradation of the measured system and the pellet rim effect.

The thermal degradation of measured system causes the increase in the fuel centerline temperature with burnup. There was an increase in measured temperature of about 80℃ for a linear heat rate of 20 kW/m.

Initially much of power in the Halden reactor is generated in the pellet rim due to fission of the Pu, which was generated during the base irradiation. A power decrease due to the fuel depletion affects the fuel centerline temperature. The fuel centerline temperature decrease is related to position of the fuel depletion in pellet. The power mostly decreases in the rim, which means less decrease of the fuel centerline temperature than a more evenly distributed one.

Unlike first four ramps, the fuel centerline temperature prior to final shutdown is over-estimated at low linear heat rate. It is possible that this result is caused by the fission gas release. Fig. 6 shows the fission gas release for power ramp test. The fission gas release calculated by FRAPCON3.5 is over-estimated than the measured and the PIE data. Fig. 7 shows the fuel centerline temperature versus power of the first four ramps. In this result, no fission gas release is assumed for final shutdown during the power ramp test. The high fission gas release causes a degradation of gap conductance due to low gas conductivity of Xe and Kr. The fuel centerline temperature increases at low linear heat rate. The forsberg-massih fission gas model used this calculation is not appropriate for this case. To predict the centerline temperature during the power ramp, the fission gas release should be also studied through various in-pile test data.

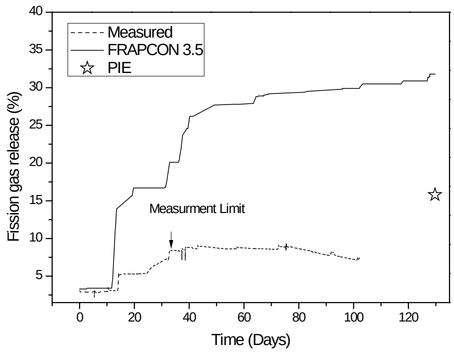


Fig. 6 Fission gas release for power ramp test

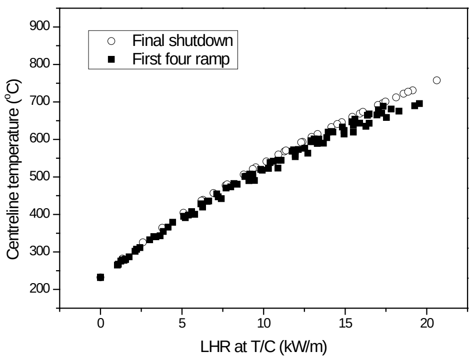


Fig. 7 Fuel centerline temperature versus power of first four ramp and final shutdown assuming no fission gas release during power ramp test

5. Conclusion

The in-pile behavior of the fuel rods for a long-term and high burn-up was evaluated by simulating the re-instrument fuel rod using the re-fabrication option newly added in FRAPCON-3.5. The calculated and measured data have difference because of distinct characteristic of in-pile test conditions and the fission gas model used in calculation. However, through re-fabrication option, re-instrument fuel rod can be easily simulated by FRAPCON-3.5. In addition, the reactor power history and the fuel rod shape can be simulated in more detail using newly added option. For the future, more case studies will be conducted to evaluate the in-pile behavior of the fuel rods for a long-term and high burn-up.

Acknowledgement

This work has been carried out under the Nuclear R&D Program supported by the Ministry of Science, ICT and Future Planning (NRF-2012M2A8A5002224).

REFERENCES

[1] K.J.Geelhood, W.G.Luscher, FRAPCON-3.5 : A Computer Code for the Calculation of Steady-State, Thermal-Mechanical Behavior of Oxide Fuel Rods for High Burnup, NUREG/CR-7022, Vol.1 Rev.1, May 2014

[2] K.Malen, A.Micski, D.Schrire, B.Nilsson, PIE of high burn-up BWR fuel rod IFA-597.3(Rod 8), HRP-356/18.

[3] Vankeerberghen,M., The integral fuel rod behaviour test IFA-597.2: Precharacterisation and analysis of the measurements, HWR-442, March 1996.

[4] Vankeerberghen. M., Fuel rod failure in the integral fuel rod behaviour test IFA-597.2, HWR-466, May 1996.

[5] I.Matsson, J.A.Turnbull, The integral fuel rod behaviour test IFA-597.3: Analysis of the measurements, HWR-543, January 1998.