Transactions of the Korean Nuclear Society Spring Meeting
Jeju, Korea, May 19-20, 2022

Evaluation of the Fuel Performance for the Whole Rods of a Reactor Core

Changhwan Shin [a*], Jae-yong Kim [a], Yunki Jo[b], Deokjung Lee[b]
[a]*Nuclear Fuel Safety Research Division, Korea Atomic Energy Research Institute, 111 Daedeok-daero 989 beon-gil, Yuseong-gu, Daejeon, 34057, Republic of KOREA*
[b]*Department of Nuclear Engineering, Ulsan National Institute of Science and Technology (UNIST), Ulsan, 44919, Republic of Korea*
[*]*Corresponding author: shinch@kaeri.re.kr*

1. Introduction

In the OPR1000 nuclear reactor, the fuel typically goes through a three-cycle operation process every 18 months. The nuclear fuel assembly is burned at three locations for each cycle in the nuclear reactor. Each fuel rod in an assembly has an individual power history according to the location of the fuel rod and the characteristics of the surrounding assembly. This power history causes differences in the characteristics of individual fuel rods. The fuel performance analysis of nuclear fuel is used as an initial condition of nuclear fuel in case of an accident analysis, and when the fuel stores after operation it becomes an initial condition of a stored fuel rod. Therefore, it is necessary to calculate the fuel performance analysis of individual fuel rods by reflecting the neutronics analysis or real operation history. In this paper, we introduce a system for evaluating the fuel performance of whole fuel rods for the 1/4 core in the OPR1000 reactor using the inputs of FRAPCON automatically produced through the core analysis code, RAST-K [1].

2. Calculation Procedure

The Python code was developed to calculate the multiple inputs of FRAPCON[2] and visualize the calculated results. The execution of the FRAPCON is performed sequentially or in parallel by Python code. The output files of the FRAPCON consist of the summary file, the formatted data file for plotting of results, and restart file for the initial condition of the transient calculation code, FRAPTRAN[3]. The formatted results files of '*.plot' are recorded in a DB file of HDF5 format[4] format. The function to extract results from the stored DB, plot them as graphs, and save them as picture files is also include in this python code. The calculation and visualization procedure was shown in Fig. 1.

A reactor core of OPR1000 consists of 177 fuel assemblies of fresh, once burned, twice burned condition. Each fuel assembly consists of the 236 fuel rods, 4 guide tubes, and 1 instrumentation rod. Since the reactor core has a symmetrical shape of 1/8, in this analysis, the fuel rods of 12,272 were calculated for 52 fuel assemblies for 1/4 of the core. The loading pattern in the equilibrium core is shown in Fig. 2.

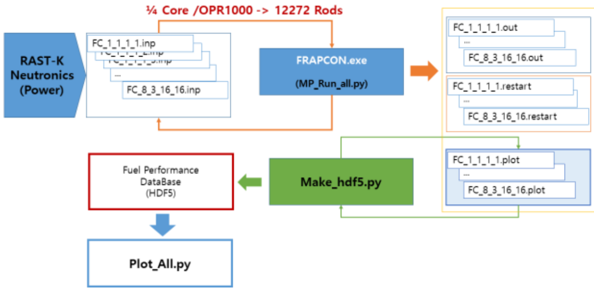


Fig. 1. The procedure of the fuel performance analysis for the whole rods in a reactor core.

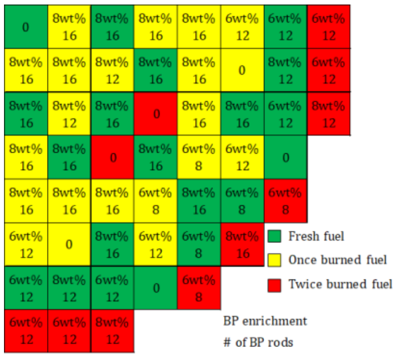
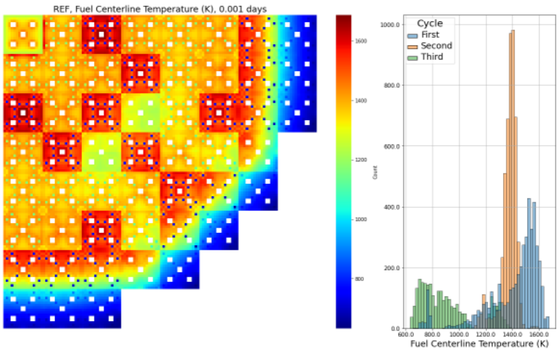


Fig. 2. The loading pattern in the equilibrium core.

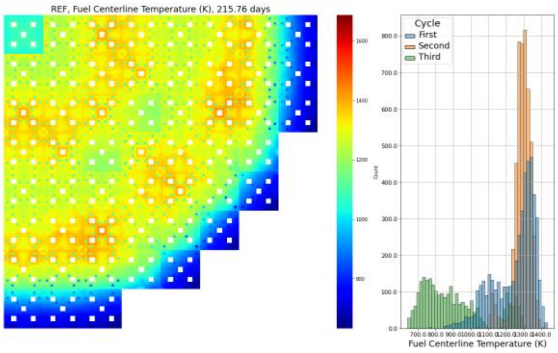
3. Results

The maximum temperature distribution of the pellets in the reactor core for the BOC (begin of cycle), MOC (middle of cycle), and the EOC (end of cycle) is shown in the Fig. 3. As shown in histogram of right side, the 1st cycle assemblies in equilibrium condition have higher temperature over a wide range compare to that the 2nd cycle assemblies are existed in a narrow range. Since the 3rd cycle assemblies are loaded outside in general, the power is low and the temperatures are distributed low.

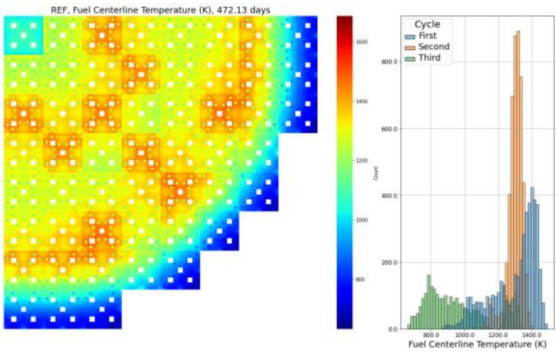
The rod internal pressure distribution is shown in Fig. 4. The internal pressure of the fuel rod is determined by the release of fission gas, the change in the free volume due to the deformation of the pellet and the cladding, and the temperature. The history over whole life cycle of fuel rods in the peak fuel assembly of the internal pressure was shown in Fig. 5. The BOC in this article is the start of the 3rd cycle. In the Fig. 6, the history of the fission gas release, the total void volume, and the plenum gas temperature are illustrated. The void volume



(a) BOC



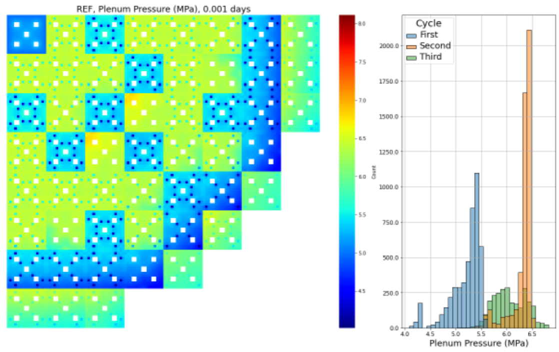
(b) MOC



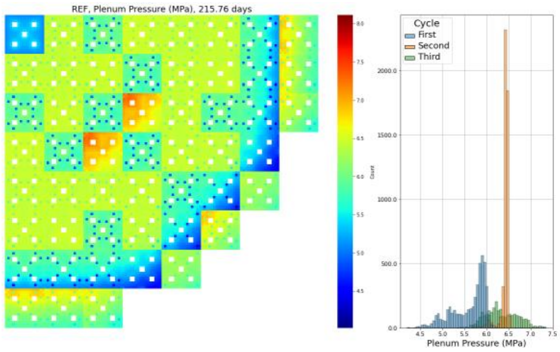
(c) EOC

Fig. 3. The maximum temperature distribution of fuel pellets in the reactor core.

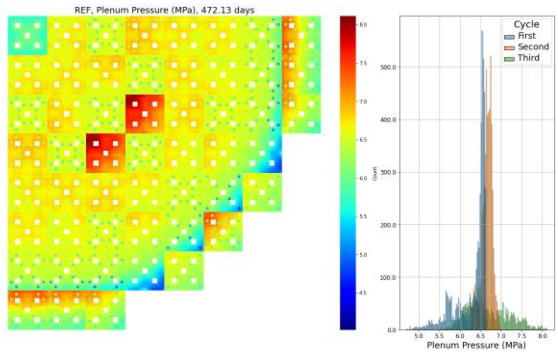
tends to decrease sharply in the 1st and 2nd cycle, while it increases almost slightly in the 3rd cycle due to contact between the pellet and cladding. The temperature of the plenum gas is also slightly decreased. It can be seen that the fission gas release increases slowly until the middle of the 2nd cycle, but increases rapidly after that, and is influencing as a major factor that increases the pressure in the fuel rod. The distribution of the rod internal pressure within one fuel assembly has a wide range of 6.3~8.1 MPa at the end of life. This trend is similarly observed for another performance parameters such as the hydride of cladding. In the accident analysis using the best-estimate plus uncertainty (BEPU), the fuel state at the time of the accident as the initial condition should be evaluated the fuel performance analysis based on the neutronics analysis. The realistic estimation will be possible by statistically evaluating the individual performance of the fuel rod distributed in the core.



(a) BOC



(b) MOC



(a) EOC

Fig.4. The rod internal pressure distribution of fuel rods in the reactor core

4. Conclusions

In this paper, we show the development and results of the analysis system for whole fuel rods in a nuclear reactor. The power history of the fuel rod varies depending on the loading pattern. It was confirmed that the diversity of these powers results in the diversity of the performance results of fuels. Recently, it is trying to estimate the burn-up effect of the fuel rod in the safety analysis such as LOCA. In the safety analysis methodology, the representative fuel rod and the fuel assembly are assumed conservatively. In this case, the representativeness according to figure of merits needs to be statistically evaluated for all core fuel rods. The developed system can be used as an important tool to evaluate the fuel performance in normal operation and also build an initial condition DB for fuel storage.

Acknowledgement

This work was supported by the Korea Institute of Energy Technology Evaluation and Planning(KETEP) grant funded by the Korea government(MOTIE) (20217810100050, Development of Safety Enhanced Nuclear Core Technology for APR).

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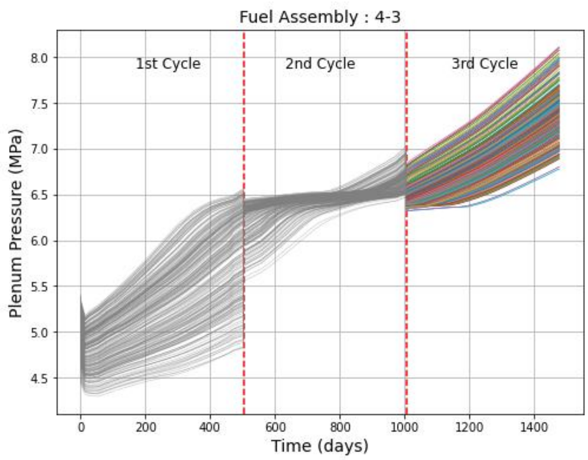
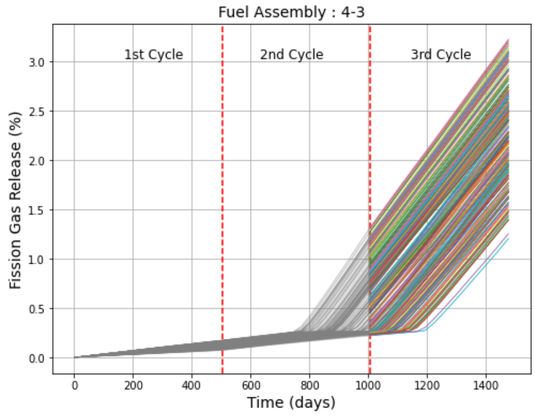
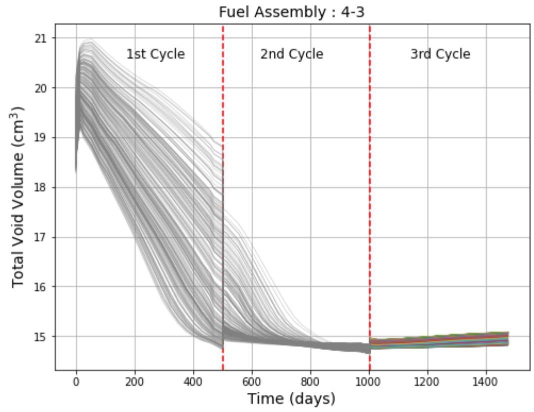


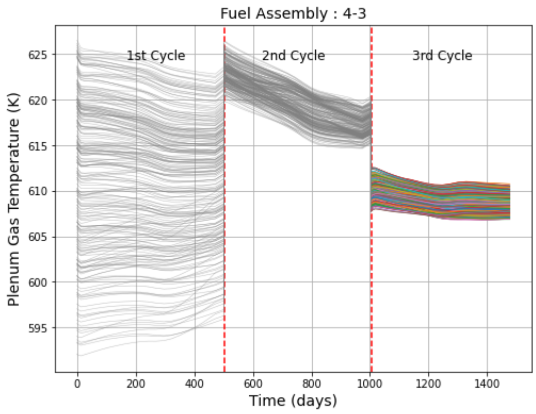
Fig. 5. The rod internal pressure of each rods in the peak fuel assembly (raw #3 and column #4).



(a) Fission gas release



(b) Total void volume



(c) Plenum gas temperature

Fig. 6 The major parameters affecting the rod internal pressure