Transactions of the Korean Nuclear Society Autumn Meeting
Changwon, Korea, October 24-25, 2024

Development of Whole Rod Performance Evaluation System

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Keywords : fuel performance, flexible operation, iSMR

1. Introduction

There is currently significant growth in generation from renewable energy sources, because of strategies to reduce carbon emissions in order to meet government policies. There is, therefore, an increasing need to transition an existing nuclear power plant from baseload operation to flexible power operations (FPO). The FPO involves power variations and causes high stress on the fuel cladding during a power increase. This induces strong pellet-cladding interaction (PCI), which can result in failure of the fuel cladding in a corrosive environment[1].

FPO involves subjecting the fuel to complex time-varying power histories that could increase the duty on the fuel rods[2]. PCI risk is affected by time-varying power histories, it is necessary to evaluate many rod cases. In addition, in iSMR that use control rods, there can be a difference between fuel rod and core power. Therefore, it is necessary to evaluate the fuel performance of individual fuel rods using each power history quickly. In this paper, we introduce a system for evaluating the fuel performance of whole fuel rods using the ROPER.

2. Calculation Procedure

The Python program was developed to manage the input data, ROPER[3] code, HSAM(Hoop Stress Assessment Methodology)[4] and calculated results. The program is developed with a GUI. It reads the results of ASTRA[5] calculations and stores them in an HDF5[6] DB. And the geometry data and thermal-hydraulic data are inputted and stored in an DB according to the assembly location and cycle.

ROPER DLL evaluate the fuel performance using the interface data from the program and the calculation results stored in an DB. The ROPER DLL is developed to perform parallel computations using OpenMP(Open Multi-Processing). And the program can visualize calculation results for desired data in the form of graphs and organize data in tables. The system diagram was shown in Fig. 1. HSAM DLL is a program developed to evaluate PCI margin in real time. It performs rapid stress evaluation using correlations and parallel computation using OpenMP.

In order to evaluate fuel performance, nuclear data are generated under baseload operation and transient such as xenon oscillation. ASTRA code is used to calculate local power for the fuel rods of 1/4 core and generate output file and 3D power data file. Output file contains UO_2 enrichment, Gd_2O_3 content for each fuel type assembly and assembly configuration. 3D power data file contains local power, burnup and so on. The program read these out files, and organizes time-dependent power, burnup, enrichment, etc. according to the location of the fuel assembly and cycle, and stores them in an DB.

The program send interface data to ROPER DLL and retrieve the evaluation results such as temperature, gap conductance, hoop stress, gap thickness, oxide thickness, etc. DLLs enable parallel computing, facilitating rapid core-level calculations. And the program send interface data to HSAM DLL and retrieve the evaluation results such as hoop stress, local power limit etc. The program stores calculation results in an DB and can visualize by generating graphs or organizing them into tables.

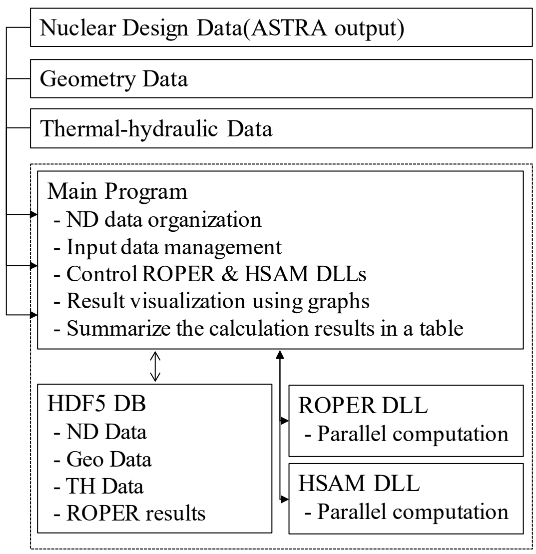
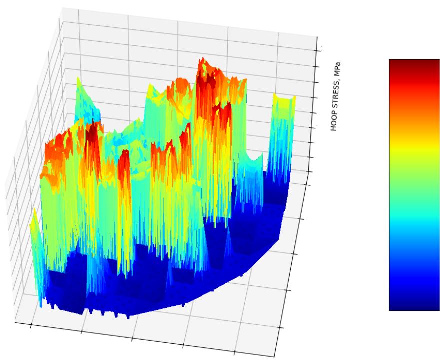


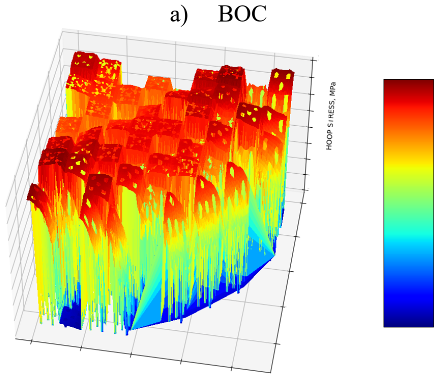
Fig. 1. The system diagram of the fuel performance analysis for the whole rods in a reactor core

3. Results

The hoop stress distribution at the BOC(begin of cycle) and the EOC(end of cycle) are shown in Fig. 2 (APR1400) and Fig. 3 (iSMR). The cladding is in a compressive stress state because fresh fuel has large gap at the BOC. However, pellet swelling and cladding creep cause the cladding stresses to change from compression to tension at the EOC.

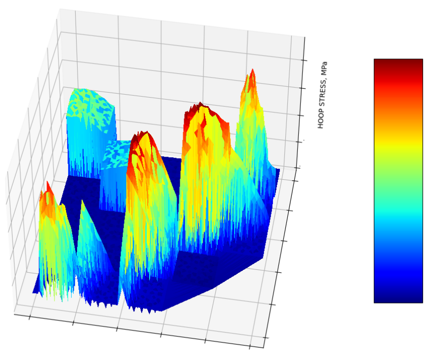


a) BOC

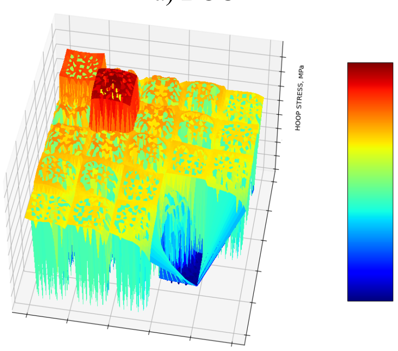


b) EOC

Fig. 2. Hoop stress distribution for APR1400 Plant



a) BOC



b) EOC

Fig. 3. Hoop stress distribution for iSMR Plant

The hoop stress in the assembly at the F6, an external coordinate of iSMR, is higher than that in other position at EOC. Fig. 4 shows the hoop stress according to time for the rod which has max. hoop stress in the assembly at the F6. Upon review of the calculation results, a tabular summary of the calculation data for the selected fuel rods can be generated. Reviewing the local power history, this rod exhibits a tendency for relatively higher power increase at the end of the cycle. This could be an effect of the iSMR power plant using control rods

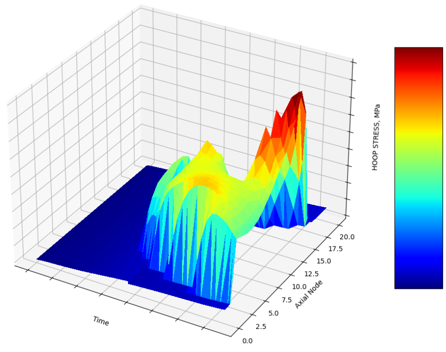


Fig. 4. Max. hoop stress in the assembly at the F6

Local power increases due to axial offset shifts can occur with control bank movement and with Xenon oscillation. It causes the hoop stress increase. Fig. 5 shows the hoop stress distribution during Xenon oscillation. The most restrictive fuel rods were selected by reviewing the calculation results summarized in a table, and their local power and hoop stress were compared in Fig. 6.

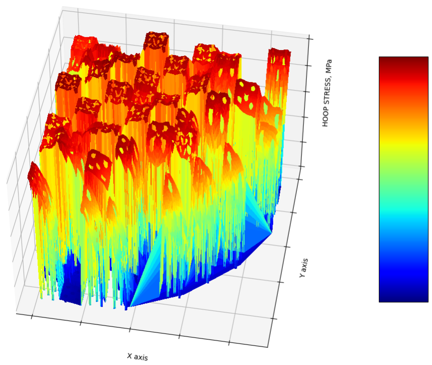


Fig. 5. Hoop stress distribution during Xenon oscillation for APR1400

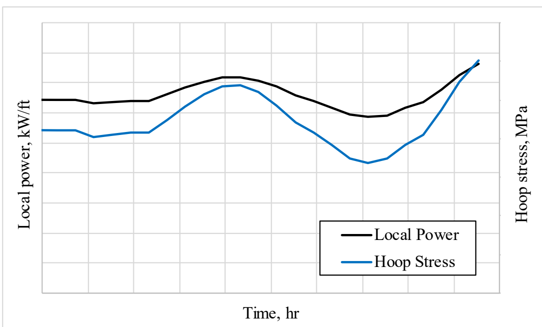


Fig. 6. Local power and hoop stress during Xenon oscillation for APR1400

4. Conclusions

In this paper, we show the development and results of the fuel performance evaluation system for the whole rods of a reactor core. This system with a GUI utilizes parallel computing for enhanced speed, effectively manages data using database, and facilitates easy analysis of data on the entire core and individual fuel rods through figures and tables.

This system is valuable for evaluating local behavior based on the power history of individual fuel rods and is intended for use in PCI assessments. It will be useful for analyzing fuel performance of both APR1400 and iSMR designed for boron-free operation.

Acknowledgement

This work was supported by the Innovative Small Modular Reactor Development Agency grant funded by the Korea Government (MOTIE) (No. RS-2023-00259289).

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