A Fuel Rod Performance Analysis and Design Code –ROPER

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1. Introduction

A fuel rod performance analysis and design code, ROPER, has been developed adopting the recent findings and achievements through many researches and studies on the fuel rod in-pile behaviors. The major thermal and mechanical models of the code are described and the prediction results by the code are compared with the measured data to show that the code is properly developed to predict the steady-state fuel rod performance.

2. Methods and Results

The ROPER code iteratively calculates the interrelated effects of temperature, pressure, cladding elastic and plastic behavior, fission gas release, and fuel densification and swelling under the time-varying irradiation conditions. Also, its calculations include those for the temperature distribution through the fuel and clad, stresses and deformations in the fuel and clad, fission product release, pellet-clad interaction and clad corrosion.

2.1 Fuel Rod Performance Model

The clad outer surface temperature is calculated using the correlation with heat flux. In this calculation, the temperature drop between clad outer surface and reactor coolant is considered as groups of single phase forced convection and sub-cooled nucleate boiling regions, both of which considered different correlation for heat flux, i.e., Dittus-Boelter equation [1] for forced convection and Bergles and Rohsenow's equation [2] for sub-cooled nucleate boiling, respectively.

In order to calculate the pellet temperature, the steady state integral form of the heat conduction equation is solved using finite difference method by considering the energy balance for the control volume, in which the fuel temperatures are calculated with iteration procedure since the fuel thermal conductivity depends on the temperature. Thermal conductivity degradation with burnup of fuel pellet is considered by using modified NFI model [3].

The steady state fission gas release is calculated using FORMAS algorithm [4] and saturated grain boundary concentration [5].

The clad deformation includes the effect of thermal expansion, elastic, creep, plasticity and irradiation growth at internal and external pressure and prescribed temperature. Stresses in the cladding are calculated from the approximated equilibrium equation as a finite difference equation at the clad mid-wall with boundary conditions. The plastic and creep strain increments are determined by the effective stress and the Prandtl-Reuss flow rule and the elastic stress/strain by the general isotropic form of Hook's law. The creep strain increments are calculated from a strain hardening creep model, in which the creep strain rate is correlated to stress, temperature, fast neutron flux and accumulated creep strain. The method of successive elastic solution is used for solving the non-linear equations. The solution also gives the contact pressure between the fuel and cladding.

The pellet densification/swelling, clad corrosion, creep and irradiation growth models, which have been calibrated to the irradiation data, are used to simulate the fuel rod behaviors.

2.2 Verification and Application

The ROPER code has been calibrated with the fuel performance database such as HRP's IFA, IFPE (High Burnup Effects Program (HBEP)), Superramp Project and Overramp Project, etc. and verified with the in-reactor performance data from the various PWR plants.

Figures 1~4 show some of the comparison of predicted by ROPER and measured values for steady state fission gas release, fuel centerline temperature, clad oxide thickness and clad OD. The uncertainty of each model has been determined to bound at least 95% of the data.

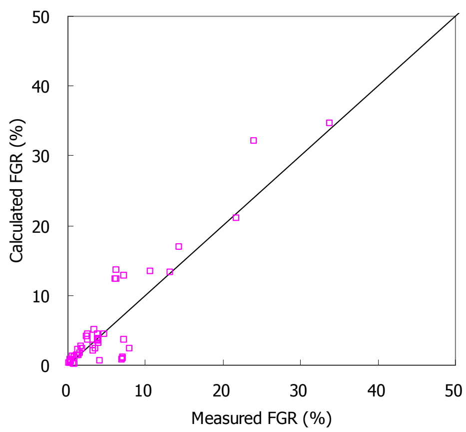

Fig. 1 Measured vs. predicted steady-state fission gas release.

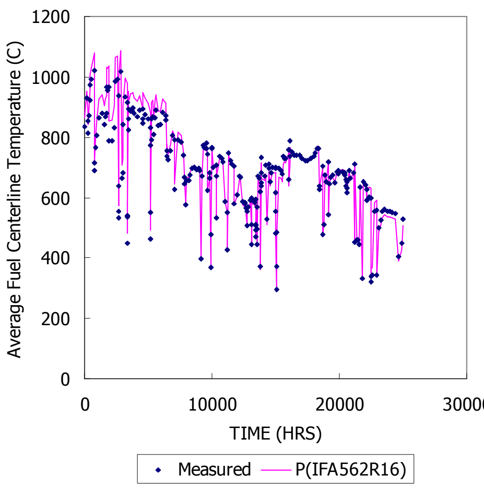

Fig. 2 Measured vs. predicted fuel centerline temperature of IFA562 rod 16.

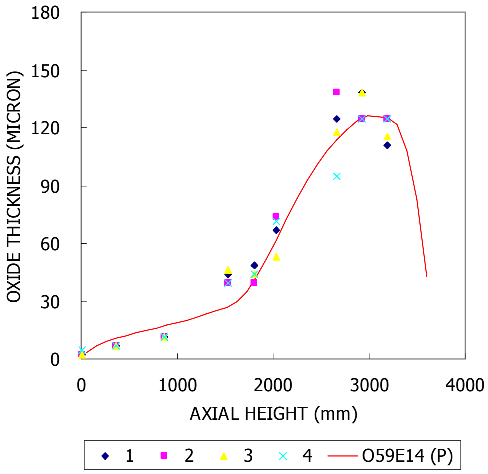

Fig. 3 Measured vs. predicted clad oxide thickness of 3rd cycle irradiated improved Zircaloy-4 clad.

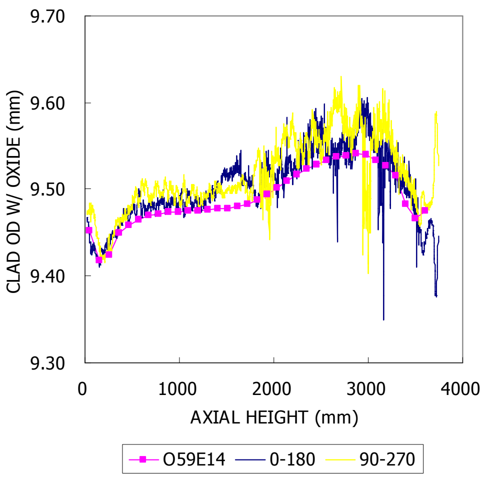

Fig. 4. Measured vs. predicted rod outer diameter of 3rd cycle irradiated Zircaloy-4 clad.

For the conservatism, the model uncertainties determined through verification procedure have been considered in the fuel rod design analysis along with fuel rod fabrication tolerances by using the SRSS (square root of the sum of the squares) method.

Figure 5 shows the fuel centerline temperatures calculated with and without burup degradation effect on the fuel thermal conductivity at 20KW/m. As expected, the fuel centerline temperature is increasing from unirradiated to high burnup conditions contrary to the no burnup dependency case.

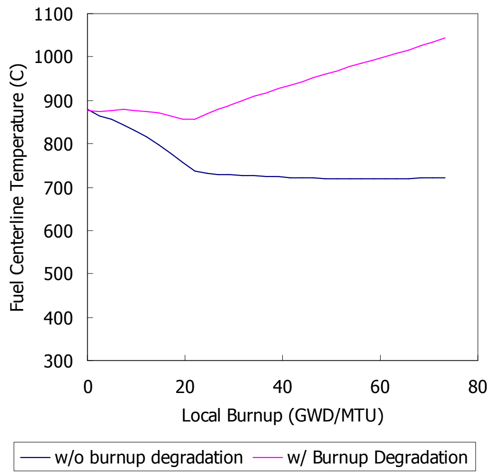

Fig. 5 Comparison of fuel centerline temperature calculated with and without burnup dependency of pellet thermal conductivity.

3. Conclusions

A fuel rod performance analysis and design code, ROPER, was developed incorporating updated fuel performance models. These works reveal that the code is appropriately organized to predict the in-reactor fuel rod behavior based on the currently available database. The incorporation of fuel themal conductivity models including burnup degradation effect might have impact on fuel rod design analysis.

Acknowledgments

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