Code Extension to Simulate RIA Tests for the Zr-U Metallic Fuel Rod

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1. Introduction

A code for evaluating the temperature of Zr-U metallic fuel rod has been developed to cover the Condition I and II events [1]. Further efforts have been made to extend the program to treat heat transfer crisis during the tests for reactivity-initiated accidents (RIA).

2. Models and Results

The Zr-U fuel rods have new features compared with commercial oxide fuels; no gap between Zr-U meat and Zr-1%Nb clad, and square cross-section with fins at four corners.

Temperature is calculated using finite element (FE) method. Heat transfer models for the program are described. Finally simulations for RIA tests are conducted.

2.1 FE Model for Temperature Calculation

To estimate the temperature of the fuel rod, the heat conduction equation is solved in two dimensions. Finite element method is adopted to represent the complex geometry of the fuel rod.

The thermal conductivity of the meat depends on temperature and burnup, and heat transfer coefficient is affected by the surface and fluid temperatures. In this way, the developed program for the Zr-U metallic fuel rod corresponds to a nonlinear transient heat-transfer problem.

Time integration of the transient problem is performed by an implicit method, and iterative calculation for the non-linear problem by Newton-Rapson method or direct substitution method. It uses a sparse matrix solver for FE equations during iterations at every time step. It can also treat a power distribution in the fuel meat due to neutron self-shielding.

The verifications of the developed program were conducted using the ABAQUS code that shares the same user subroutines as in the developed program. Steady state and transient problems were analyzed for 1/8 rod model due to the symmetry of the fuel rod and full model. From the evaluation of temperature for the 1/8 rod model at steady state, maximal error of 0.18 % was present relative to the ABAQUS result. Analysis for the transient problem using the full rod model resulted in the same as the variation of centerline temperature from the ABAQUS code during a hypothetical power transient. The distribution of heat flux for the entire cross section and surface was almost identical for the two codes. The frame for the present RIA simulation is based on the identical FE model.

2.2 Heat Transfer Model

The heat transfer models on clad surface are properly selected by investigating previous researches for oxide fuels [2-4].

Sub-cooled liquid cooling is modeled by a natural convection correlation. Nucleate boiling heat transfer is calculated by Thom model. Critical heat flux is estimated by Kutateladze correlation valid for sub-cooled pool boiling conditions. In the film-boiling regime Labuntzov model is used to estimate heat transfer coefficient, which is modified to account for the sub-cooling condition [2-4]. Heat transfer coefficient between departures from nucleated boiling and film boiling is estimated by an interpolation. The minimum stable film boiling temperature is determined by an empirical correlation. The moment of clad rewetting can also be controlled by the input of rewetting time.

2.3 Calculation for RIA Tests

The power pulse half-width during the RIA tests is less than 5 msec which is similar to that in the Japanese Nuclear Safety Research Reactor (NSRR). The RIA tests for fresh fuel rods are typically conducted under atmospheric pressure and pool heat transfer condition.

The variation of temperature on time is calculated. The heat inputs for the calculations are in the range of 60 to 200 cal/g. The temperatures at fuel center and surface are investigated.

It is estimated that the centerline temperature at early stage is close to that in the adiabatic condition due to very short pulse width. The maximal centerline temperature increases monotonously in proportional to heat input. The centerline temperature decreases with heat transport to clad surface.

For the same heat input, the average temperatures on clad surface are raised earlier and higher than those observed in the case of oxide fuels with radial gap between pellet and cladding. The non-existence of gap and higher thermal conductivity of Zr-U meat facilitate the heat transport to the clad surface, which leads to earlier departure from nucleate boiling. The duration at high clad temperature is, however, much shorter than that appeared in the oxide fuel case. Figure 1 shows the temperature variation during a RIA simulation for the case of rewetting time = 0.8 sec. It is possible to reproduce the abrupt transition of heat-transfer mode, when the clad surface reaches the minimum stable film boiling temperature or the time becomes equal to the specified rewetting-time. There appears a certain heat input below which the clad temperature does not reach the film-boiling regime.

The computation time required for a RIA simulation is less than 35 sec for the calculation up to 3 sec on a 2.5 GHz Pentium IV machine. Convergence problems have not been observed during the simulations.

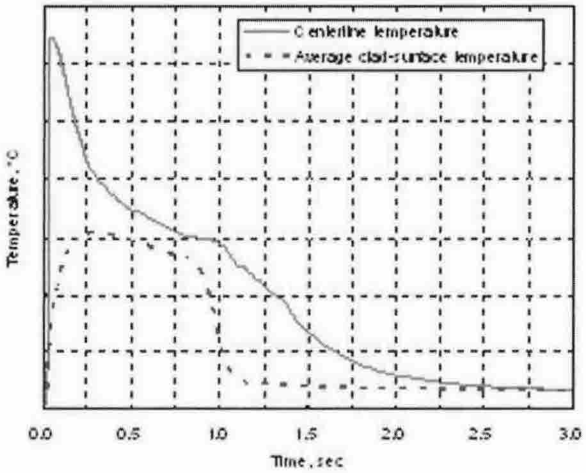

Figure 1. Temperature as a function of time which is calculated with an assumption of rewetting time = 0.8 sec.

3. Conclusion

The code to calculate the temperature of Zr-U metallic fuel rod is extended to be capable of simulating the RIA tests. The calculations have been accomplished within reasonable time and without convergence problems. The tendency of the temperature of the fuel rod during the RIA tests is well modeled for almost adiabatic condition at early stage as well as for the transition of heat-transfer modes on subsequent cooling.

Acknowledgements

The Ministry of Science and Technology (MOST) of the Republic of Korea has sponsored this work through the Mid– and Long–term nuclear R&D Project.

REFERENCES

[1] J.S. Cheon, B.H. Lee, Y.H. Koo, J.Y. Oh, J.S. Yim, D.S. Sohn, Development of Program for Evaluating the Temperature of Zr-U Metallic Fuel Rod, Proceedings of the Korean Nuclear Society Spring Meeting, Gyeongju, Korea, October 2003 (in Korean).

[2] K. Mikitiouk, A. Shestopalov, K. Lioutov, L. Yegorova, G. Abyshov. Modification of IPSN's SCANAIR Fuel Rod Transient Code for High Burnup VVER Fuel: NUREG/IA-0165, IPSN 99/09, NSI RRC 2181, May 1999.

[3] U. Rohde, The Modeling of Fuel Rod Behaviour under RIA Conditions in the Code DYN3D, Annals of Nuclear Energy, Vol. 28, p. 1343-1363, 2001.

[4] http://loop2.boom.ru/hydro/water-e.html