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S. R. Novascone
J. D. Hales
B. W. Spencer
R. L. Williamson

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S. R. NOVASCONE, J. D. HALES, B. W. SPENCER, R. L. WILLIAMSON

Fuels Modelling and Simulation Department, Idaho National Laboratory

PO Box 1625, MS 0340, Idaho Falls, ID 83415-0340 - USA

ABSTRACT

Since 2008, the Idaho National Laboratory (INL) has been developing a next-generation nuclear fuel performance code called BISON. BISON is built using INL's Multiphysics Object-Oriented Simulation Environment, named MOOSE. MOOSE is a massively parallel, finite element-based framework to solve systems of fully coupled non-linear partial differential equations using the Jacobian-Free Newton Krylov (JFNK) method. MOOSE supports the use of complex two- and three-dimensional meshes and uses implicit time integration, which is important for the widely varied time scales in nuclear fuel simulation.

BISON has been applied to various nuclear fuel problems to assess the accuracy of its 2D and 3D capabilities. The benchmark measurements presented here are from two well-known and documented reactor experiments. The first is the Risø3 Fission Gas Project, referred to as "Bump Test GE7", which was performed on rod ZX115. Bump Test GE7 consists of a base-irradiation period of a full-length rod in the Quad-Cities-1 BWR. The base irradiation test is followed by a "bump test" of a sub-section of the original rod, where the power is increased rapidly, which takes place in the test reactor DR3 at Risø in a water-cooled HP1 rig under BWR conditions. The second test is of a segmented PWR rod base-irradiated in the Electricity of France (EDF) commercial reactors. The segment was then re-fabricated and ramp-tested in the French Alternative Energies and Atomic Energy Commission (CEA) OSIRIS reactor to investigate PCMI resistance. These experiments were chosen because they allow for an evaluation of several aspects of the code, including fully coupled thermo-mechanics, contact, and several nonlinear material models.

BISON simulations of the bump tests were run using smeared pellet geometry. Comparisons between these calculations and experimental measurements are presented for clad diameter after the base irradiation and after a large power ramp. Preliminary comparisons between calculations and measurements are favourable, supporting the use of BISON as an accurate multiphysics fuel simulation tool.

1. Introduction

Fuel rod failure due to pellet-clad mechanical interaction (PCMI) can happen for a variety of reasons and is an active area of research and investigation. When used properly, nuclear fuel performance simulation software can be helpful in studying PCMI and fuel rod design. An assessment of how well simulation software performs can be effective for determining how best to interpret and use simulation results. In this paper, a relatively new fuel performance code called BISON [1] is used to simulate two well-documented experiments that include PCMI measurements of fuel rod diameter before and after a large power ramp.

BISON is a nuclear fuel performance code developed at the Idaho National Laboratory (INL), which is built on the MOOSE (Multi-physics Object Oriented Simulation Environment) code [2], also developed at the INL. MOOSE is a massively parallel, finite element-based framework that is used to solve systems of coupled non-linear partial differential equations using the Jacobian-Free Newton Krylov (JFNK) method. MOOSE supports the use of complex two- and three-dimensional meshes and uses implicit time integration, which is important for the widely varied time scales in nuclear fuel simulation. BISON can be used to solve the fully coupled energy, species diffusion, and solid mechanics equations. BISON includes material models for thermal expansion, swelling, densification, creep (thermal and irradiation), plasticity, relocation, smeared cracking, and fission gas production and release. Algorithms for mechanical (non-penetration) and thermal contact are used to simulate gap conditions between fuel pellets and clad.

In this paper, pre- and post-power ramp measurements of fuel rod diameter from two well-known experiments are presented with corresponding calculations from BISON. The tests considered are the Risø3 GE7 and OSIRIS J12 experiments, which both include measurements of rod outer diameter following PCMI resulting from a power ramp late in fuel life. These tests were chosen because they allow for evaluation of several aspects of the code. The data are believed to be reliable since they were selected as priority cases for the FUMEX-III [3] (FUel Modelling at EXtended Burnup-III) project.

FUMEX is an international project with the objective of evaluating fuel performance codes. Since 1981, the International Atomic Energy Agency (IAEA) has sponsored a series of Coordinated Research Projects (CRP) in the area of nuclear fuel modelling. The objectives of the projects have been to assess the maturity and predictive capability of fuel performance codes, support interaction and information exchange between countries with code development and application needs, build a database of well-defined experiments suitable for code validation, transfer a mature fuel modelling code to developing countries, and provide guidelines for code quality assurance and code application to fuel licensing. Interactions with international fuel modelling researchers via FUMEX-III played a significant role in the BISON evolution, particularly influencing the selection of material and behavioural models which are now included in the code.

The FUMEX-III cases are generally high burnup integral fuel rod experiments and thus involve complex coupled multiphysics behaviour. A mature fuel performance capability is needed to have any hope of reasonable comparison to experimental data.

2. Information common to both simulations

Rod segment data was modelled assuming 2D axisymmetry, based on the geometry specified in the FUMEX-III [3] (Risø3 GE7 and OSIRIS J12) project and the Risø technical report [4] (Risø3 GE7).

For UO_2 fuel, the temperature dependent thermal conductivity was defined using the correlation for unirradiated material suggested by Fink [5], modified to account for the effects of irradiation and porosity using a series of multipliers, as outlined by Lucuta et al [6]. Solid and gaseous swelling was described using empirical correlations from MATPRO [7], with densification and relocation specified using models from ESCORE [8]. Relocation strains were applied to the fuel in the radial direction only. Thermal and irradiation creep was prescribed using the model from MATPRO [7]. The Forsberg-Massih two-stage model [9] was used to predict fission gas release. The Zircaloy clad was modelled with typical constant thermal properties for heat conduction. Secondary thermal creep and irradiation creep were specified using the models of Hayes and Kassner [10] and Hoppe [11], respectively, and irradiation growth was described using the model from ESCORE [8]. For these simulations, the cladding thermal and irradiation creep model was combined with a J2 plasticity model to simulate rapid deformation during the power ramps. The material and behavioural models used here are described in greater detail in reference [1]

A fully coupled thermo-mechanical analysis was performed, simulating base irradiation and power ramp with the following assumptions. For simplicity, the pellet stack was simulated as a single smeared fuel column. Cladding oxidation was not considered for either case, so it was expected that calculations would under-predict final clad outer-diameter. Mechanical contact between the fuel and clad was assumed to be frictionless. According to the technical reports for both experiments, fission gas release was very small. The fission gas release in the BISON simulations was insignificant. Zircaloy-2 and Zircaloy-4 were the cladding types in the GE7 and OSIRIS-J12 experiments respectively. A review of the material properties and models for each clad material suggested that a single set of properties and models could be used for the two simulations.

3. Risø bump-test GE7

The GE7 bump test was carried out during the third Risø Transient Fission Gas Release Project in 1989 [4]. The test pin (ZX115) was the lower middle segment of four approximately 0.975 m segments assembled to a stringer. The cladding was stress relieved Zircaloy-2 with a bonded zirconium liner. The fuel segment was base irradiated in the Quad Cities-1 BWR over four reactor cycles.

The bump irradiation was performed in a water-cooled rig under BWR conditions. The power history during the bump included a 6-hour conditioning period at approximately 23 kW/m, a 15 minute power ramp, and then a 4-hour hold, where the peak power at the end of the hold period was 35.5 kW/m. Note that the tabular power data supplied with the test documentation indicated a 2 kW/m power rise during the final hold period. The bump included a strong axial profile, with a peak axial factor of 1.48.

Boundary conditions were typical for a boiling water reactor. The coolant pressure was constant at a value of 7.24 MPa. The clad outer wall temperature was fixed at 564 K. Helium was the rod fill gas, having an initial pressure of 0.29 MPa. The axial power profile during the base irradiation was relatively flat, varying over the range of 0.96 to 1.027. The fast neutron flux in the clad was supplied via input using data supplied with the experiment.

A comparison of the predicted and measured rod outer diameter is shown in Figure 1. The dashed line is the as-manufactured rod diameter, prior to irradiation. The experimental data, shown as symbols, indicate the measured average rod diameter at both the end and middle fuel pellet locations, giving an indication of rod ridging due to pellet hour glassing. The solid line is the predicted rod diameter following the power bump.

Due to the large axial power profile during the power bump, the upper portion of the rod remained relatively cool and did not undergo plastic deformation as a result of the bump. The final rod diameter in this region is thus a function of clad creep during the base irradiation. BISON over-predicts the clad creep-down, giving a rod diameter approximately 40 μm less than measured. The prediction, however, is reasonable. Permanent clad deformation during the bump is observed over roughly the bottom two-thirds of the rod. BISON predicts the shape of this deformation nicely but under-predicts the magnitude.

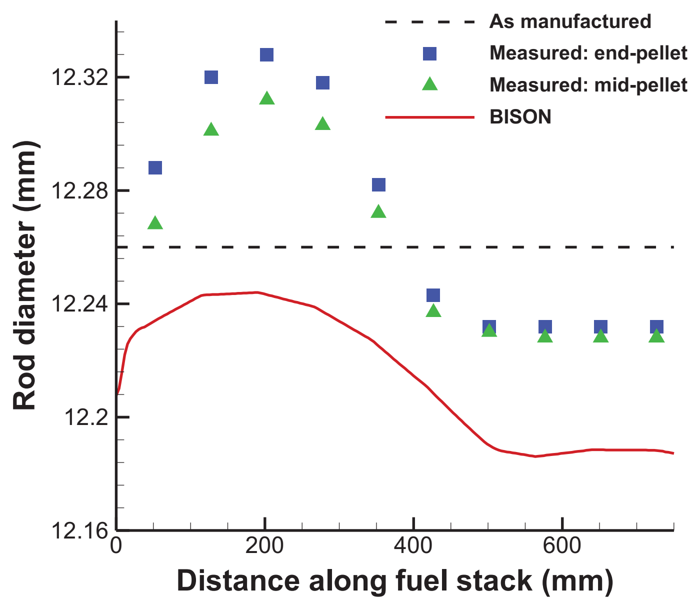

Figure 1 Risø3 GE7 bump test experimental measurements and BISON calculation results.

4. OSIRIS

Detailed information concerning the OSIRIS J12 experiment was obtained through participation in the FUMEX-III project [3]. The data were gathered from a segmented PWR rod base-irradiated in the Electricity of France (EDF) commercial reactors, then the segment was re-fabricated and ramp tested in the French Alternative Energies and Atomic Energy Commission (CEA) OSIRIS reactor to investigate PCMI resistance [3].

The approximately 0.522 m segmented Zircaloy-4 clad rod was irradiated for 2 cycles in the EDF Gravelines-5 PWR to a final discharge burn-up of 23.852 MWd/kgU. The average powers in the 2 cycles were approximately 16 and 23 kW/m. The rod segment designated J12-5, which was irradiated in the fifth span from the lower end of the assembly, was re-fabricated with new end plugs without altering either the fuel column or the internal fill gas. After a conditioning period of 762 minutes at 21 kW/m, the power was increased quickly (9 kW/m/min.) and held at 39.5 kW/m for 739 minutes. The axial profile was flat during base irradiation and varied from approximately 0.75 at the ends of the segment to 1 at the centre during the bump test.

The initial fill-gas (Helium) pressure was 2.6 MPa, and the coolant pressure was 15.5 MPa. The external clad temperature was assumed uniform over the rod length and specified at approximately 585 K during base irradiation and 615 K during the ramp. The fast neutron flux in the clad was provided via input using data supplied with the experiment.

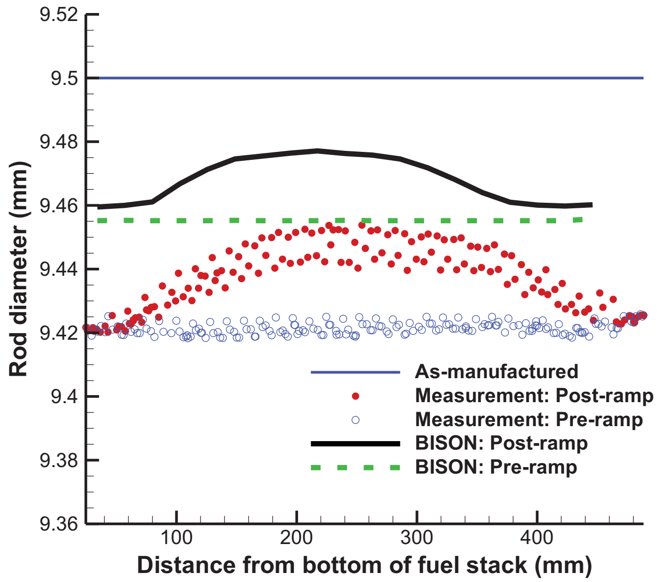

Figure 2 OSIRIS experimental measurements and BISON calculation results.

A comparison of the predicted and measured rod outer diameter is shown in Figure 2. The thin-solid line is the as-manufactured rod diameter, prior to irradiation. The experimental data, shown as solid (post-ramp) and open (pre-ramp) circles, indicate the measured average rod diameter at both the end and middle fuel pellet locations, giving an indication of rod ridging due to pellet hour-glassing. The thick solid line is the predicted rod diameter following the power bump and the dashed line is the predicted rod diameter prior to the ramp. BISON under-predicts the clad creep-down, resulting in a rod diameter about 35 μm larger than the measurement for both pre- and post-ramp diameter measurements. As with the Risø comparison, the predicted clad displacements occurring during the power ramp are very reasonable.

Another figure of merit for comparing simulation to experiment results is the gap between the fuel and the clad at the end of the test. Post irradiation examination data [3] indicated a post-test radial fuel to clad gap of 16 μm. BISON calculations of this gap were approximately 15.5 μm. Since clad creep-down was under predicted for this case and the original simulation-geometry was correct, one could conclude that the comparison of gap value between the measurement and the calculation is close because the combination of thermal expansion, swelling, creep, and relocation models over predicted the final fuel diameter.

5. Summary

Rod outer diameter measurements for the Risø3 GE7 and OSIRIS J12 experiments, both before and after a power bump, were presented and compared to BISON calculations.

For the base irradiation, BISON over-predicted clad creep-down in the Risø test, yet under-predicted clad creep-down for the OSIRIS experiment. The reason for over-prediction in one case and under-prediction in the other is not clear, but is likely related to assuming the same irradiation creep material parameters for both simulations. The Risø rod was constructed of stress relieved Zircaloy 2 and the OSIRIS J12 rod was Zircaloy 4. Since it is well known that irradiation creep is significantly affected by alloy and heat treatment [8], the differences in the predictions are not surprising. Further investigation of these differences is underway.

The predicted diameter change during the bump test is well predicted by BISON for both experiments.

Note that experimental error estimates are not available, which adds to the assessment difficulty. Also, no adjustments to input parameters were made to tune the simulation results to better fit the experimental data. The input parameters used in this study are consistent between each simulation and are based on consensus among the authors.

Follow-on work is planned using BISON to study PCMI phenomena. Such plans include the addition of an oxidation model and implementation of damage models to gain understanding about fuel rod conditions during PCMI when rod failure is impending. Since BISON can easily consider discrete pellet geometries, comparisons between smeared and discrete pellet geometry approximations are possible and will reported in the future.

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