Proof of Concept Validation Results of the FAST Fuel Element Model

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Summary

The Fuel And Sheath modelling Tool (FAST) is a fuel performance code that is being developed for both normal and transient operating conditions. The FAST code includes models for heat generation and transport, thermal-expansion, elastic strain, densification, fission product swelling, pellet relocation, contact, grain growth, fission gas release, gas and coolant pressure and sheath creep. These models are coupled and solved on a two-dimensional (radial-axial) geometry of a fuel pellet and sheath. This paper outlines the results of an ongoing proof of concept validation of the code conducted as part of the development process. The validation compared model predictions against both experimental data and results obtained from the ELESTRES and ELOCA fuel performance codes. Overall, the results show excellent model performance except in cases of strong axial dependence.

1. Introduction and Description of the FAST code

The Fuel And Sheath modelling Tool (FAST) is a nuclear fuel element simulation program that is being developed at the Royal Military College of Canada (RMC) with technical support from the Fuel and Fuel Channel Safety Branch of Atomic Energy of Canada, Chalk River Laboratories.

In order to accommodate limited computational resources, all models must always make simplifying assumptions which introduces idealization error. This has historically restricted fuel modelling codes to one-dimensional and quasi-two-dimensional representations of fuel-elements to reduce the computation expense of the models to manageable levels. Continuous advancements in both computer hardware and software have expanded modelling capabilities beyond many of the previous limitations. This research is an attempt to capitalize on these advancements to develop a fuel performance code with improved flexibility and predictive capability by making fewer simplifying assumptions (thus introducing less idealization error). The FAST code is intended as a tool to help inform the design of the next generation of fuel performance models.

The FAST code integrates models used for normal operating conditions and those for transient conditions into a single platform for modelling nuclear fuel under both normal and transient reactor conditions. Previously, the modelling in these regimes was handled by two different codes, ELESTRES-IST (v.1.2.0.1) and ELOCA-IST (v.2.1), based on different simplifying assumptions and considering different phenomena.

The FAST model has been developed on the Comsol Multiphysics (v.4.3a) finite-element platform which allows for a significant reduction in development cost compared to a stand-alone in house code. This platform reduces the need to develop tools for many tasks such as building model geometry and finite-element meshes, solving linear systems and post-processing results. The flexibility of the platform also allows for easy modification of the model for application to unique/experimental elements or development of new mathematical models of phenomena.

Individual phenomena such as heat-transfer are modelled mechanistically. These phenomena are then coupled using an iterative approach to obtain a converged simultaneous solution at each time step. Using this approach, the FAST code includes models for heat generation and transport, thermal-expansion, elastic strain, densification, fission product swelling, contact, grain growth, fission gas release, gas and coolant pressure and sheath creep. This model has evolved from previous treatments developed at the Royal Military College of Canada (RMC) by Morgan and Shaheen [1–4]. A more detailed description of the FAST code is available in references [5,6].

The results presented in this paper were produced using a geometry consisting of one half-pellet in the radial-axial plane (axisymmetric) with an accompanying sheath. The pellet geometry allows central holes, dishing and chamfering of one or both ends of the pellet. The model assumes that the pellet is representative of all pellets within the element (no strong axial dependence of the boundary conditions). This allows a periodic boundary condition to be applied which bounds the model in the axial direction. It is worth noting that the geometry can be modified in COMSOL to include more pellets or represent more fuel designs details as desired. An investigation into such expanded geometries is ongoing at RMC.

2. Validation Procedure and Results

The FAST code has undergone a proof of concept validation against both experimental data and results obtained from the ELESTRES and ELOCA fuel performance codes. It should be stressed that this validation is intended to demonstrate the potential of the model development technique. It is not intended to be compared to the industry recommended validation requirements. The validation of the FAST code has been done in two separate parts using experimental data provided by Atomic Energy of Canada Limited, Chalk River Laboratories.

The first validation exercise is a comparison of the predicted end-of-life condition of seven irradiated fuel elements which underwent post-irradiation examination (PIE). The cases were selected to cover a range of power, burnup and geometries. The maximum linear power and burnups for these cases ranged from 25 to 53 kW m^{-1} and 132 to 552 MWh kgU^{-1}, respectively.

The PIE provided measurements of the fission gas release volume, grain size, sheath strain, and circumferential ridge heights of each element. These irradiation tests have integrated many different phenomena, which makes it difficult to attribute any discrepancies in the results to a specific model or phenomenon. The temperature in these tests was also too low to initiate any high-temperature effects.

The results of this comparison exercise have been summarized in a series of graphs using the case number on the horizontal axis as illustrated in Figure 1. The average experimental value has been included along with the maximum measured value for each element. This provides a sense of scattering in the experimental results. The FAST calculation was performed with and without the incorporation of a circumferential crack model.

The second validation exercise was to validate the high-temperature transient components of the model. This was done by comparing model predictions to measurements from an irradiated fuel experiment, FIO-131, conducted at CRL and previously released to the Nuclear Energy Agency of the Organization for Economic Co-operation and Development (NEA-OECD)[7]. In the experiment, the primary coolant loop of an instrumented fuel element was depressurized during high power operation thereby simulating Loss Of Coolant Accident (LOCA) conditions. This data set includes in reactor, time dependent measurements of pellet and sheath temperatures, internal gas pressure and external coolant pressure. A post irradiation analysis provided measurements of the sheath deformation and Zircaloy oxidation behaviour.

The FIO-131 experimental results showed significant axial dependence due to thermal hydraulic and neutronic effects along the length of an element. In the ELOCA code, this was accounted using a three-axial segmentation feature. This capability, however, is not currently supported in the FAST code. The element was therefore modelled as three independent elements each representing a third of the element (with no communication between elements). These results are displayed in Figure 2.

3. Discussion

Under NOC conditions (see Figure 1), the FAST code shows better predictive behaviour than the existing IST code with some exceptions. The FAST code predicts larger fission gas release volumes than the ELESTRES and ELESIM codes for all cases. This prediction is an improvement in the two cases 1026 and 1281 (which showed significant gas release), however, it resulted in a false prediction of gas release in case 1282 (with case 1283 also indicating a similar trend although with lower burnup and power). The cracked pellet model is found to predict slightly lower fission gas release. This result is believed to be caused by increased pellet to sheath interaction, which leads to improved gap heat transfer. This intern results in lower pellet temperatures, reducing the diffusion of gas atoms in the lattice, and thus trapping more gas atoms in the pellet.

The predicted sheath hoop-strain at the mid-pellet and at pellet-to-pellet interface show significant improvement, particularly when using the cracked pellet model. This trend also exists for the circumferential ridge strain results. However, in virtually all cases, all of the models are consistently under predicting the measured strains. The ridge height prediction from the FAST code (cracked pellet) was found to be closer to the mean measurements in five of the seven cases. In general, FAST was found to under predict the ridge heights, while ELELSTRES over predicted the average measured values.

In the transient test (see Figure 2), the FAST predictions agree well with the experimental results for both the centerline and fuel periphery temperatures. The FAST code predictions of gas pressure from the three separate element models was found to be too high compared to the measurements. Despite over estimating the internal gas pressure, which acts as a driving force for sheath creep, the FAST code significantly under predicted the sheath strain for the high strain segments. This suggest a problem with the sheath creep model implemented in FAST, which results in an under prediction of sheath creep at high temperatures. This also may account the over prediction of the internal gas pressure since insufficient sheath creep results in a lower free gas volume in the element resulting in higher internal gas pressure.

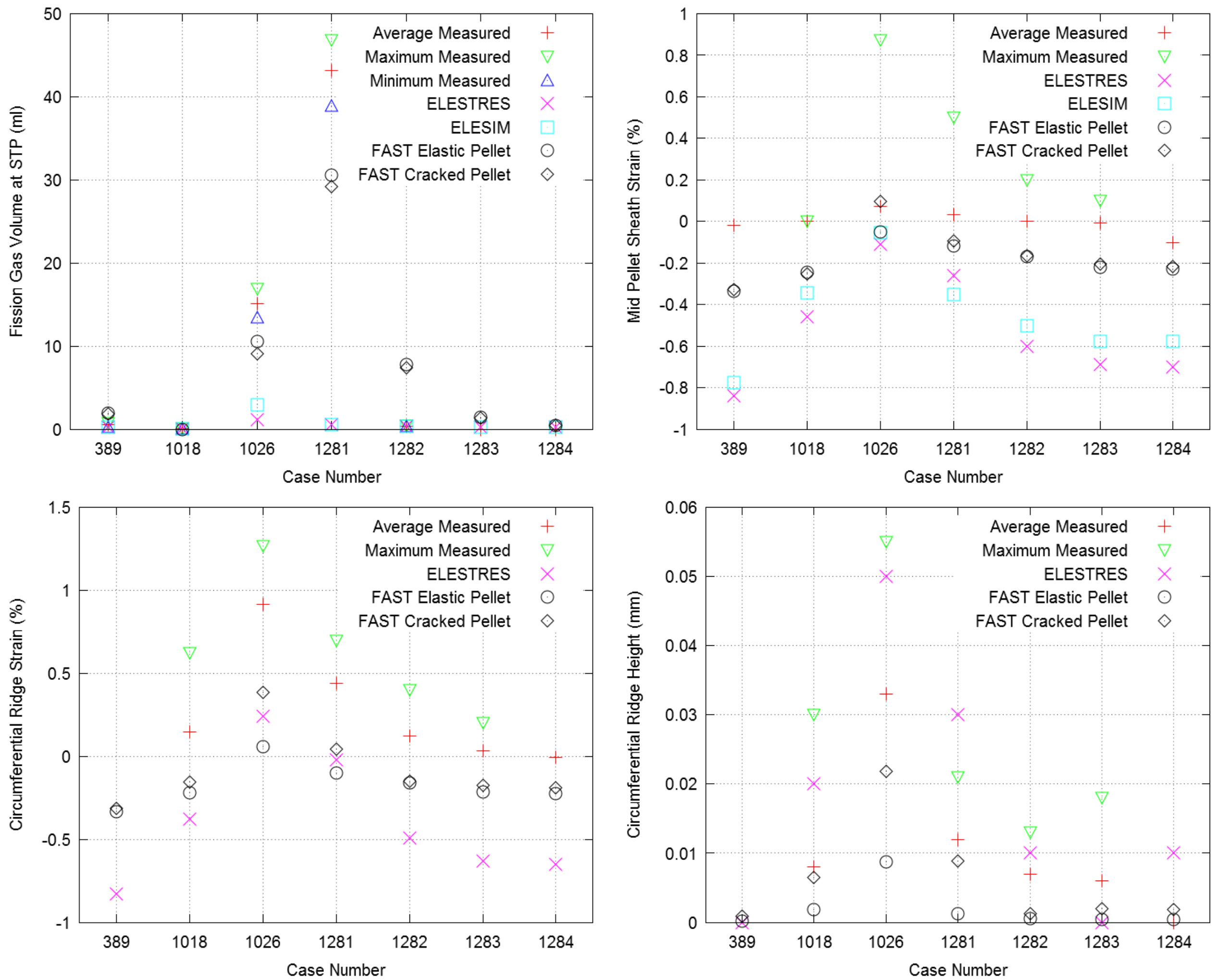

Figure 1 Results of the FAST validation for normal operating conditions.

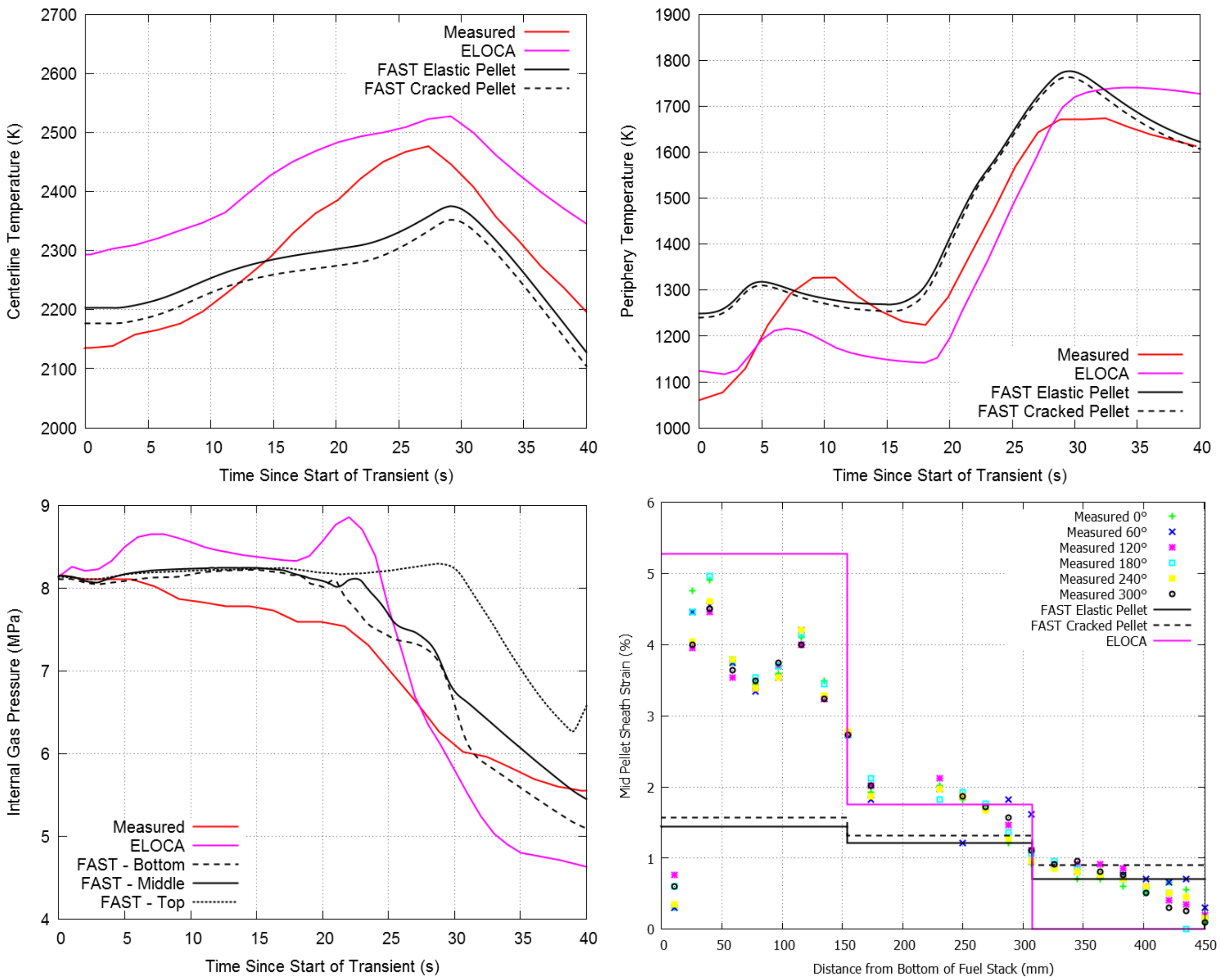

Figure 2 Results of the FAST validation for normal operating conditions.

4. Conclusions

Results from the proof of concept validation of the FAST fuel performance code were presented for both normal and transient reactor conditions. The NOC results demonstrated an improved predictive capability as compared to the ELESTRES code, particularly for the prediction of sheath strain. The transient test demonstrated a successful continuous transition from NOC to accident conditions. The model showed good agreement with pellet temperature measurements; however, a problem was identified with the sheath creep model which severely under predicted the creep rates at higher temperatures. An investigation into the source of this discrepancy is currently underway.

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6. References

[1] D. Morgan, "A thermomechanical model of CANDU fuel," MASc Thesis, Royal Military College of Canada, Kingston, Ontario, 2007.

[2] K. Shaheen, "A mechanistic code for intact and defective nuclear fuel element performance," Royal Military College of Canada, Kingston, Ontario, 2011.

[3] K. Shaheen, J. S. Bell, A. Prudil, and B. J. Lewis,, "Modelling CANDU Fuel Element and Bundle Behaviour for Performance of Intact and Defective Fuel," in *Proceedings of the International Conference on the Future of Heavy Water Reactors*, Ottawa, Ontario, 2011.

[4] K. Shaheen and B. J. Lewis, "Platform-Based Modelling of Intact and Defective Fuel Behaviour," in *Proceedings of Top Fuel 2009 Conference*, Paris, France, 2009.

[5] A. Prudil, B. Lewis, and P. Chan, "Development of the FAST Code for Modeling CANDU Fuel," presented at the Annual Canadian Nuclear Society Conference, Saskatoon, 2012.

[6] A. Prudil, B. Lewis, and P. Chan, "Requirements for Extending the FAST Code for Transient Simulation of Nuclear Fuel," presented at the CNS/CNA Student Conference 2012, Saskatoon, 2012.

[7] Atomic Energy of Canada Ltd. Chalk River Laboratories, "IFPE/CANDU-FIO-131, CANDU experiment FIO-131 Fuel Behaviour under LOCA Conditions," Nuclear Energy Angency (NEA) Organization for Economic Co-Operation, URL: http://www.oecd-nea.org/tools/abstract/detail/nea-1783 NEA-1783 IFPE/CANDU-FIO-131, Apr. 2007.