Simulation of Criticality Accident Transients in Uranyl Nitrate Solution with COMSOL Multiphysics

C. Hurt[1], P. Angelo[2], R. Pevey[1]

[1]Department of Nuclear Engineering, University of Tennessee, Knoxville, TN, USA
[2]Y-12 National Security Complex, Safety Analysis Engineering, Oak Ridge, TN, USA

Abstract

Numerical analysis of criticality accident transients is used to characterize fission source term development for emergency planning and to evaluate and confirm accident scenario understanding. Power excursions are a highly multiphysics phenomena, due to reactivity feedbacks based in physical parameters (such as temperature and void volume) which perturb the power history. Historically, 1D and 0D approximations to accident-related physics have been employed to analyze criticality excursion transients for particular geometries. However, enhanced computing and multiphysics capabilities (i.e. COMSOL) allow for a greater range of detail and flexibility in criticality transient simulations. Thus applicable physics phenomena (neutron kinetics, heat transfer, fluid flow, radiolytic gas transport) were developed in COMSOL for two configurations of fissile uranyl nitrate solution: a rectangular, open container and the SILENE annular reactor (Figure 1).The Conjugate Heat Transfer, Global ODEs, Transport of Diluted Species and PDE Coefficient physics are utilized in a 2D axisymmetric and 3D geometry to model the SILENE reactor and container criticality transients, respectively. A total of 7 global equations are solved to simulate neutron point kinetics in the solution; one for the prompt neutron balance and six for delayed neutron precursor production and decay. Conjugate Heat Transfer physics is used for all solid and fluid domains in both models, with highly conductive layers being used for thin stainless steel walls, convective cooling at all external surfaces, and a volumetric heat source due to fission. The fluid node equations for momentum and mass balance are solved for the fluid domains. Radiolytic gas physics consist of molecular transport and bubble formation and dispersion. Molecular transport of the radiolytic gas utilizes either the Transport of Diluted Species application or six axial global equations, with natural convection to the solution surface and a reaction rate determined by fission. Bubble formation and dispersion physics are modeled using either the PDE Coefficient or Transport of Diluted Species applications or six axial global equations. Natural convection is through the outlet at the solution surface and bubble formation is governed by fission rate and the critical concentration of the molecules in the solution. The MUMPS direct solver is used for the time-dependent studies and sequential study steps are utilized to employ varying time-stepping for the more rapid transient periods. The criticality experiment benchmark SILENE LE1-641 was modeled in which a 2 dollar reactivity ramp insertion over 20 seconds initiates a prompt critical transient and series of excursions. The fission rate transient over the first 300 seconds is shown in Figure 2 for the results from COMSOL and measured LE1-641 values. Figure 3 shows the reactivity contributions for the first 200 seconds of the transient. Radiolytic gas concentration distributions during the transient are shown in Figure 4. Figure 2 demonstrates the accuracy of the excursion model with

existing COMSOL applications. Verification of this approach encourages extension in COMSOL to other accident scenarios (i.e. different geometries and fissile materials) as well as the development of additional physical effects (e.g. solution sloshing and boiling).

Reference

1. COMSOL, Inc., “COMSOL Multiphysics User’s Guide,” Version 4.3, Burlington, MA (2012).
2. “Nuclear Criticality Accident Emergency Planning and Response,” ANSI/ANS-8.23-2007, American Nuclear Society, La Grange Park Ill (2007).
3. Y. Miyoshi, et. al., “Inter-code Comparison Exercise for Criticality Excursion Analysis,” NEA No. 6285, Nuclear Energy Agency, Organisation of Economic Co-operation and Development (2009).
4. B. Basoglu, T. Yamamoto, et. al., “Development of a New Simulation Code for Evaluation of Criticality Transients Involving Fissile Solution Boiling,” JAERI-Data/Code-98-011, Japan Atomic Energy Institute (1998).
5. D. Mather, A. Buckley, et. al., Critex: “A Code to Calculate the Fission Release Arising from Transient Criticality in Fissile Solutions,” SRD R 380, AEA Technology (1994).
6. “Assessment of the TRACE Code Using Transient Data from Maanshan PWR Nuclear Power Plant,” NUREG/IA-0241, U.S. Nuclear Regulatory Commission, Washington, D.C. (2010).
7. J. Gomes, C. Pain, et. al., “A Computational Framework for Complex Multi-Physics Modelling”, 1st International Conf. on Computational Methods for Thermal Problems - ThermaComp2009, Naples, Italy, pp. 205-309, Giannini Editor, (2009).
8. T. Mclaughlin, S. Monahan, et. al., "A Review of Criticality Accidents: 2000 revision” Report LA-13638, Los Alamos National Laboratory (2000).
9. W. Stacey, Nuclear Reactor Physics, p. 147, WILEY-VCH, Berlin, Germany (2007).
10. D. Chandler, “Spatially-Dependent Reactor Kinetics and Supporting Physics Validation Studies at the High Flux Isotope Reactor,” PhD Diss., University of Tennessee (2011).
11. B. Kiedrowski, et. al., MCNP5-1.6, Feature Enhancements and Manual Clarifications, LA-UR-10-06217, Los Alamos National Laboratory, Los Alamos, N.M (2010).

Figures used in the abstract

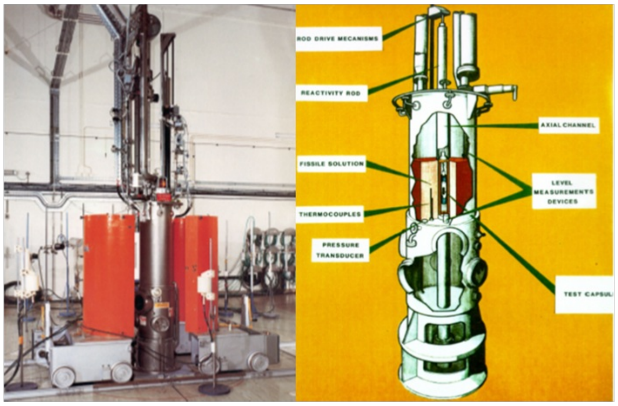

Figure 1: Photograph and Schematic of the SILENE Reactor at the Valduc facility in France.

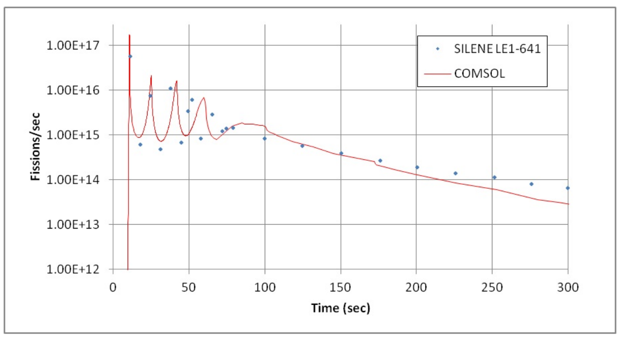

Figure 2: The excursion power (in fission/sec) during benchmark SILENE LE1-641.

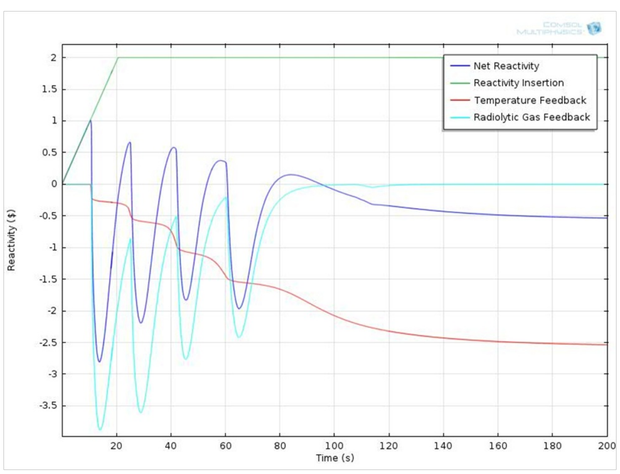

Figure 3: The reactivity contributions during benchmark SILENE LE1-641.

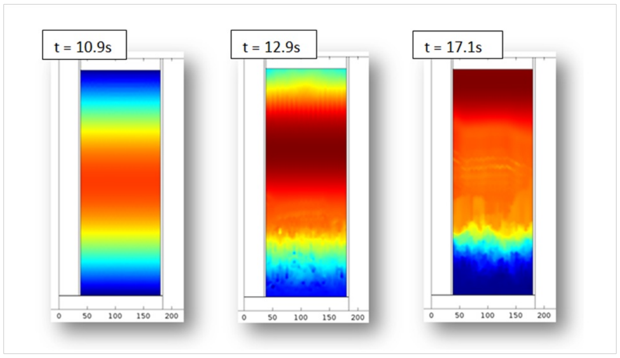

Figure 4: The radiolytic gas concentration during SILENE LE1-641 (dimensions in mm).