Extending the Capability of Nuclear Plant Systems Analysis with Advanced Tightly-Coupled Nuclear Fuels Performance

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INTRODUCTION

Traditional nuclear plant system analysis tools, such as RELAP5-3D [1], TRACE [2], and recently developed tools, RELAP-7 [3] and SAM [4], do not provide the physics and models to analyze fuel performance under normal reactor or accident conditions. These system analysis tools are generally restricted to internal core heat structures, yielding simplified thermal analyses, which were appropriate for analyzing nuclear reactor transients using an Appendix K methodology. However, the U.S. Nuclear Regulatory Commission (NRC) is considering updating the regulatory rule for emergency core cooling system (ECCS) success criteria, 10 CFR 50.46c. Implementation of the updated rule will introduce new and more restrictive performance-based requirements for nuclear fuel and will require re-analysis of many existing U.S. LOCA basis. Additionally, there is currently significant interest in analyzing accident tolerant fuel concepts (ATF) for various conditions and computing coping times in extreme cases. Furthermore, extension of the life cycle for Light Water Reactor (LWR) fuel is essential in guaranteeing economic viability in operating the current fleet of Nuclear Power Plants (NPPs).

Here, we will report on progress to tightly couple the BISON [5] nuclear fuels performance application to standard and next generation NPP systems/safety analysis codes. The intention here is to enhance and extend the capability of NPP systems/safety analysis codes to include multi-scale multi-physics fuels performance analysis. Preliminary results of normal and off-normal transients will be demonstrated.

SOFTWARE PACKAGES FOR FUELS PERFORMANCE ENHANCED SYSTEMS/SAFETEY ANALYSIS

Here we will briefly describe each NPP systems/safety analysis codes, RELAP5-3D, TRACE, RELAP-7 and SAM and the BISON nuclear fuels performance code.

NPP Systems/Safety Analysis Codes

RELAP5-3D

The RELAP5-3D [1] code has been developed for best-estimate transient simulation of light water reactor coolant systems during posted accidents. The code models the coupled behavior of the reactor coolant system and the core for loss-of-coolant accidents and operational transients, such as anticipated transient without scram, loss of offsite power, loss of feed water, and loss of flow. A generic modeling approach is used that permits simulating a variety of thermal hydraulic systems. Control system and secondary system components are included to permit modeling of plant controls, turbines, condensers, and secondary feedwater systems.

TRACE

The TRAC/RELAP Advanced Computational Engine (TRACE) [2] is the latest in a series of advanced, best-estimate reactor systems codes developed by the U.S. Nuclear Regulatory Commission (NRC) for analyzing transient and steady-state neutronic-thermal-hydraulic behavior in light water reactors. It is the product of a long-term effort to combine the capabilities of the NRC’s four main systems codes (TRAC-P, TRAC-B, RELAP5 and RAMONA) into one modernized computational tool. TRACE has been designed to perform best-estimate analyses of loss-of-coolant accidents (LOCAs), operational transients, and other accident scenarios in pressurized light-water reactors (PWRs) and boiling light-water reactors (BWRs). It can also model phenomena occurring in experimental facilities designed to simulate transients in reactor systems. Models used include multidimensional two-phase flow, non-equilibrium thermo-dynamics, generalized heat transfer, re-flood, level tracking, and reactor kinetics. Automatic steady-state and dump/restart capabilities are also provided.

RELAP-7

RELAP-7 (Reactor Excursion and Leak Analysis Program) [3] is a next generation nuclear systems safety analysis code being developed at the Idaho National Laboratory (INL). The most important development goals of RELAP-7 are to take advantage of the previous thirty years of advancements in computer architecture, software design, numerical methods, and physical models. The code is being developed based on INL’s modern scientific software development framework – MOOSE (Multi-Physics Object-
Oriented Simulation Environment) [6]. The four major improvements are 1) A well-posed seven-equation two-phase flow model (liquid, gas, and interface pressures); 2) Improved numerical approximations resulting in second-order accuracy in both space and time versus the first order approximations; 3) Implicit tightly coupled time integration for long duration transients, such as providing plant behavior for full life fuel cycle evaluations; 4) the ability to tightly couple to higher fidelity physics, such as the MOOSE-based BISON nuclear fuels performance application [5].

SAM
Systems Analysis Module (SAM) [4] is a modern system analysis tool being developed at Argonne National Laboratory for advanced non-LWR safety analysis. It aims to provide fast-running, whole-plant transient analyses capability with improved-fidelity for Sodium-cooled Fast Reactors (SFR), Lead-cooled Fast Reactors (LFR), and Molten Salt Reactors (MSR) or Fluoride-cooled High-temperature Reactors (FHR). SAM takes advantage of advances in physical modeling, numerical methods, and software engineering to enhance its user experience and usability. It is MOOSE-based and inherits MOOSE’s underlying meshing and finite-element library (libMesh) [8] and linear and non-linear solvers (PETSc [9]), in order to leverage the modern advanced software environments and numerical methods.

BISON: A Finite Element-Based Nuclear Fuel Performance Code

The Broadly Implicit Simulation of Nuclear fuels (BISON) [5] is a MOOSE-based finite element-based nuclear fuel performance code applicable to a variety of fuel forms including light water reactor fuel rods, TRISO particle fuel, mixed oxide (MOX) fuel, and metallic rod and plate fuel and for 1D spherical, 2D axisymmetric or 3D geometries. It solves the fully coupled equations of thermomechanics and species diffusion, for either two-dimensional axisymmetric or three-dimensional geometries. Fuel models are included to describe temperature and burnup-dependent thermal properties, fission product swelling, densification, thermal and irradiation creep, fracture, and fission gas production and release. Plasticity, irradiation growth, and thermal and irradiation creep models are implemented for clad materials. Models are also available to simulate gap heat transfer, mechanical contact, and the evolution of the gap/plenum pressure with plenum volume, gas temperature, and fission gas addition. BISON has been coupled to the mesoscale fuel performance code Marmot [7], demonstrating fully coupled multi-scale fuel performance capability. BISON is currently being validated against a wide variety of integral fuel rod experiments.

Numerical Approach for Coupled NPP Systems/Safety Analysis with 1D/2D Fuels Performance

Physics coupling approach between the BISON nuclear fuels performance code and the various NPP systems/safety analysis codes vary little. The main difference is that RELAP5-3D and TRACE have one-dimensional heat structures in the form of finite difference discretizations to represent fuel heat generation and heat transfer in the reactor core. RELAP-7 and SAM on the other hand incorporate a two-dimensional finite element representation of the core heat structure. BISON has both one-dimensional and two-dimensional finite element representations of the fuels performance, which make it ideal for this study. In all cases, the heat structure is coupled to the one-dimensional fluid flow by equating along a discrete axial distance (z) the clad heat flux, in terms of a heat transfer coefficient, to an internal energy source in the fluid. This coupling is illustrated in Figure 1.

![Graphical representation of 1D or 2D heat structures available in BISON for coupling to NPP systems/safety analysis codes.](image)

In all four coupling efforts presented here, the MOOSE MultiApp and Transfer system [6] was utilized to control the multi-physics transfer of information between the codes. For BISON/RELAP-7 coupling and BISON/SAM coupling, the codes are native MOOSE-based. Thus, the coupled software capability has full native support for all MOOSE features and is a straightforward effort.

For BISON/TRACE coupling and BISON/RELAP5-3D coupling, the approach consists the MultiApp system, which allows multiple MOOSE (or external) applications to run simultaneously in parallel and the Transfer system, which is designed to push and pull fields and data to and from MultiApps. Multiple codes are compiled into one executable, can efficiently exchange data in memory, and effectively converge all coefficients in an implicit iterative manner (Picard). A MOOSE-Wrapped App being developed...
for the NRC, called Blue CRAB (Comprehensive Reactor Analysis Bundle), designed to couple Department of Energy’s Office of Nuclear Energy (DOE-NE) codes with NRC codes was used for both the TRACE and RELAP5-3D coupling to BISON.

RESULTS

Here we demonstrate BISON/TRACE coupling with the MOOSE-Wrapped App, Blue CRAB, and BISON/RELAP-7 with native MOOSE coupling. Fig. 2 illustrates the Blue CRAB solution for BISON/TRACE coupling. The geometry is a 10cm rodlet from an average assembly pin during normal reactor operation. The blue line is the water temperature and the cyan line is the wall clad temperature.

![Fig. 2 Demonstration of normal pin conditions in BISON/TRACE coupling for 10 cm rodlet.](image)

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Fig. 3 is a preliminary result of near fuel failure after a station blackout (SBO) scenario on a full-length pin. The simulation was conducted with BISON/RELAP-7 and stopped before clad burst when the clad temperature, shown by the red line, reached 1150K. The blue line is steam volume fraction, which shows the core nearly boiled off. Steam temperature is shown with the green line.

![Fig. 3 BISON/RELAP-7 SBO simulation of a full-length pin.](image)

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REFERENCES