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A New Best-Estimate Tool for Simulator Validation and Engineering Analysis Using SIMULATE-3K/RELAP5-3D

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ABSTRACT

The best-estimate neutron kinetics code, SIMULATE-3K (S3K), and the best-estimate nuclear systems analyses code, RELAP5-3D, have been interfaced to provide a best-estimate coupled code system for performing plant transient calculations with reactivity feedback from a detailed core model.

1. SIMULATE-3K OVERVIEW

The neutronic model used in S3K solves the transient three-dimensional, two-group neutron diffusion equations, including a six group model for delayed neutron precursors. S3K tracks dynamically nodal concentration of fission products and accounts for the extraneous neutron sources (SE) due to spontaneous fissions, alpha-n interactions from actinide decay, and gamma-n interactions from long-term fission product decay.

The S3K heat conduction in the fuel pin is governed by the one-dimensional, radial heat conduction equation. The material properties are temperature and burnup dependent. Temperature dependent conduction properties for UO2 and Zircaloy are tabulated based on data sets from the MATPRO code. The burnup dependence of the fuel conductivity is taken from Weisenack. The gap conductance model is functionalized versus exposure and fuel temperature. The heat source is the sum of two components, namely: the prompt fission heat and the decay heat. Decay heat is modeled by using the ANSI/ANS-5.1 23-group data. S3K takes into account the fact that fission energy is deposited both inside the fuel pellet and outside the pellet due to neutron slowing down and gamma attenuation.

The S3K hydraulic model uses a five-equation model, vapor and liquid mass conservation, vapor and liquid energy conservation and mixture momentum conservation. In addition to the conservation equations, closure relationships exist for each phasic density, defined as a function of the pressure and phasic enthalpy. It is important to mention that water properties are evaluated at the core exit pressure. The general drift formulation for the void fraction completes the set of equations to be solved. The concentration parameter and the void-weighted drift velocity are calculated using the EPRI correlations. The subcooled boiling model is taken from Lahey's mechanistic model.

2. RELAP5-3D OVERVIEW

The RELAP5 series of codes has been developed at the Idaho National Laboratory (INL) under sponsorship of the U.S. Department of Energy, the U.S. Nuclear Regulatory Commission, members of the International Code Assessment and Applications Program (ICAP), members of the Code Applications and Maintenance Program (CAMP), and members of the International RELAP5 Users Group (IRUG). Specific applications of the code have included simulations of transients in light water reactor (LWR) systems such as loss of coolant, anticipated transients without scram (ATWS), and operational transients such as loss of feedwater, loss of offsite power, station blackout, and turbine trip. RELAP5-3D, the latest in the series of RELAP5 codes, is a highly generic code that, in addition to calculating the behavior of a reactor coolant system during a transient, can be used for simulation of a wide variety of hydraulic and thermal transients in both nuclear and nonnuclear systems involving mixtures of vapor, liquid, noncondensable gases, and nonvolatile solute.

The RELAP5-3D computer code is formulated upon a transient, one-dimensional, two-phase flow model allowing both thermal non-equilibrium and non-homogeneous two-phase flow. The field equations are tightly coupled by efficient, semi-implicit or "Courant violating" nearly-implicit numerical solution schemes. A comprehensive two-phase flow regime map (e.g., bubbly, slug, mist, etc.) has been formulated to span all possible operating regimes. Once the local nodal flow regime has been determined, constitutive equations are used to compute interfacial surface area, heat transfer coefficients and interphase drag or slip. Likewise, wall-to-fluid heat transfer and friction coefficients couple the fluid behavior to structures. The code includes many "specialized components" like pumps, accumulators, valves, turbines and steam separators. Users have complete flexibility in interconnecting these components to represent the NSSS and/or Balance of Plant (BOP). RELAP5-3D contains other special process models like critical flow, non-condensable gas transport, thermal stratification and water level tracking.

RELAP5-3D solves the complete set of conservation laws for two-phase fluid dynamics. Specifically, the conservation of mass, energy and momentum are ensured for the liquid and vapor components individually within each node (control volume). This produces a "six-equation," full two-velocity non-equilibrium hydrodynamic model. Mathematical closure results from the interfacial phase "coupling" equations, which are flow-regime dependent. The six-equation set of field equations results in a numerically robust model that ensures the following:

- conservation of vapor mass (scalar)
- conservation of liquid mass (scalar)
- conservation of vapor energy (scalar)
- conservation of liquid energy (scalar)

- conservation of vapor momentum (vector)
- conservation of liquid momentum (vector)

A self-consistent set of interfacial closure equations determines the exchange of mass, momentum and energy between the vapor and liquid fields at each node. Additionally, transport equations are solved for:

- conservation of non-condensable gas
- conservation of material species (e.g., boron)

The above equation set represents the most comprehensive possible model for transient two-phase flows. Depending upon the flow regime in the node, the phases may be completely coupled or completely de-coupled, both in relative motion and energy interchange. It is precisely because of this "first principles approach" that RELAP5-3D has truly <u>predictive</u> capability. This results in a high-fidelity simulation of all transient behavior under normal, abnormal and emergency situations.

RELAP5-3D has a general-purpose heat conduction model to calculate the energy state (temperatures) of metal mass (structures) throughout the system. Based upon Fourier's Law, the model is transient and generally one-dimensional with three possible coordinate frames (rectangular, cylindrical, and spherical) for each structure. Under conditions of core reflooding, the usually one-dimensional conduction solution becomes two-dimensional to more accurately predict quenching phenomena.

RELAP5-3D heat structures may have variable thermo-physical properties and internal heat sources. Boundary conditions can be convective, boiling, radiative, adiabatic, specified temperature and/or specified heat flux. Structural (wall) boundaries can be coupled directly to the fluid flow field; this allows for representation of internal walls and plates, as well as the external environments of primary and secondary containment atmospheres.

RELAP5-3D computes the heat transfer coefficient at a structure's surface with a very comprehensive "boiling map" that considers all possible heat transfer regimes, such as sub-cooled single phase, nucleate, CHF/DNB, transition and film boiling. In addition, the influence of noncondensables on these regions is mechanistically considered. This "map" has been subjected to extensive experimental validation under a broad spectrum of typical reactor conditions and thermodynamic states (e.g., pressures). A well-validated model for metal-surface (e.g., fuel rod) "rewetting" has been developed.

3. S3K/RELAP5-3D COUPLED CODE

Typically, the RELAP5-3D core thermal-hydraulics nodalization does not include a flow channel for each fuel assembly, but only 6 to 50 effective flow channels. These flow channels comprise from a few up to 100 fuel assemblies each. Therefore, the coupling of S3K and RELAP5-3D must provide fuel assembly based TH parameters to S3K for thermal feedback effects during a transient. The only firm requirement for the nodalization in the RELAP5-3D core region is that it must contain the same number of planes in the active fuel region as the S3K model. A brief description of the linkage between RELAP5-3D and SIMULATE-3K follows.

The linkage is a direct, explicit coupling of the two codes on a synchronous time-step basis. The coupling provides a method of executing the SIMULATE-3K three-dimensional neutronics using the Nuclear Steam Supply System (NSSS) boundary conditions calculated by the RELAP5-3D thermal hydraulics code. It allows the SIMULATE-3K calculated total core power and core power distributions to “drive” the RELAP5-3D system model core.

Detailed calculations from the component codes result in a methodology for analyzing limiting transients such as steam line breaks, rod drops/ejections, and ATWS scenarios. These transient events require detailed three dimensional core data and information about the behavior of NSSS components, such as the separators, pressurizer, steam generators, and steam lines. A coupled analysis of these transients is important because the core behavior is closely tied to the NSSS system. A detailed plant model is required in order to capture the timing and characteristics of the important thermal-hydraulic phenomena and/or operations events, such as valve closures, safety injection, or control system interactions.

The primary requirement to utilize the coupled code system is that the number of active planes in the core region must be the same for both the SIMULATE-3K and RELAP5-3D models. The RELAP5-3D model may have only one active core channel, but that channel must have the same number of axial subdivisions in the active fuel region.

The thermal hydraulic conditions in the core and plenum regions are passed to the SIMULATE-3K model which performs a calculation of detailed core power, which is then passed back to the RELAP5-3D model to use for the next time step. There are three different coupling options that are available for the linkage between RELAP5-3D and SIMULATE-3K, “plenum”, “flat“, and “nodal“. Each of these options are described below.

The “plenum” coupling option utilizes the SIMULATE-3K thermal-hydraulics calculation. The inlet flow and enthalpy to the core and the exit pressure in the upper plenum is provided by the RELAP5-3D model for each core channel. SIMULATE-3K will use this data to perform its own thermal-hydraulic calculations in the core region. These thermal-hydraulic results are only used to provide feedback values on a nodal basis for the cross section evaluation. The resultant power distribution is then collapsed back to the coarse core nodalization used by the RELAP5-3D model and provided to RELAP5-3D. This option performs quite well provided that the core flow is always positive.

The “flat” coupling option does not utilize the SIMULATE-3K thermal-hydraulics calculation. Each fuel assembly in a RELAP5-3D channel receives the same fuel temperature, coolant density, and boron concentration at a give axial plane from the RELAP5-3D calculation. This option is very robust, but it will approximate the accurate radial power distribution (especially for the hot assemblies or controlled assemblies) unless a large number (>100) of RELAP5-3D channels are modeled.

The “nodal” coupling option is a variation of the “flat” option. Once again, the SIMULATE-3K thermal-hydraulics calculation is not performed. However, an estimate of the true three-dimensional density and fuel temperature distributions is made utilizing the current nodal powers. The fuel temperature is estimated from the coarse value calculated by RELAP5-3D using a weight factor that is the ratio of the nodal power to the average power in the channel in that plane. The density for a given fuel assembly is calculated using a simple enthalpy rise calculation and the same weight factor described for the fuel temperature calculation. The density calculation also includes a normalization step that preserves the mass of liquid for each RELAP5-3D channel.

The additional input required by RELAP5-3D to activate this linkage are the definitions of the channels, SIMULATE-3K input and output filenames, initial power level, and trip signals. The additional input required by SIMULATE-3K is the channel assignments and the coupling option selection.

The coupled S3K/RELAP5-3D code system has been utilized to perform a series of calculations for the Forsmark 1 BWR nuclear plant in Sweden in support of the PRA. These calculations were done with the failure of a variable number of control rods to insert after a SCRAM. That model will also be used to assess the performance of a planned automatic boron injection system for Forsmark 1.

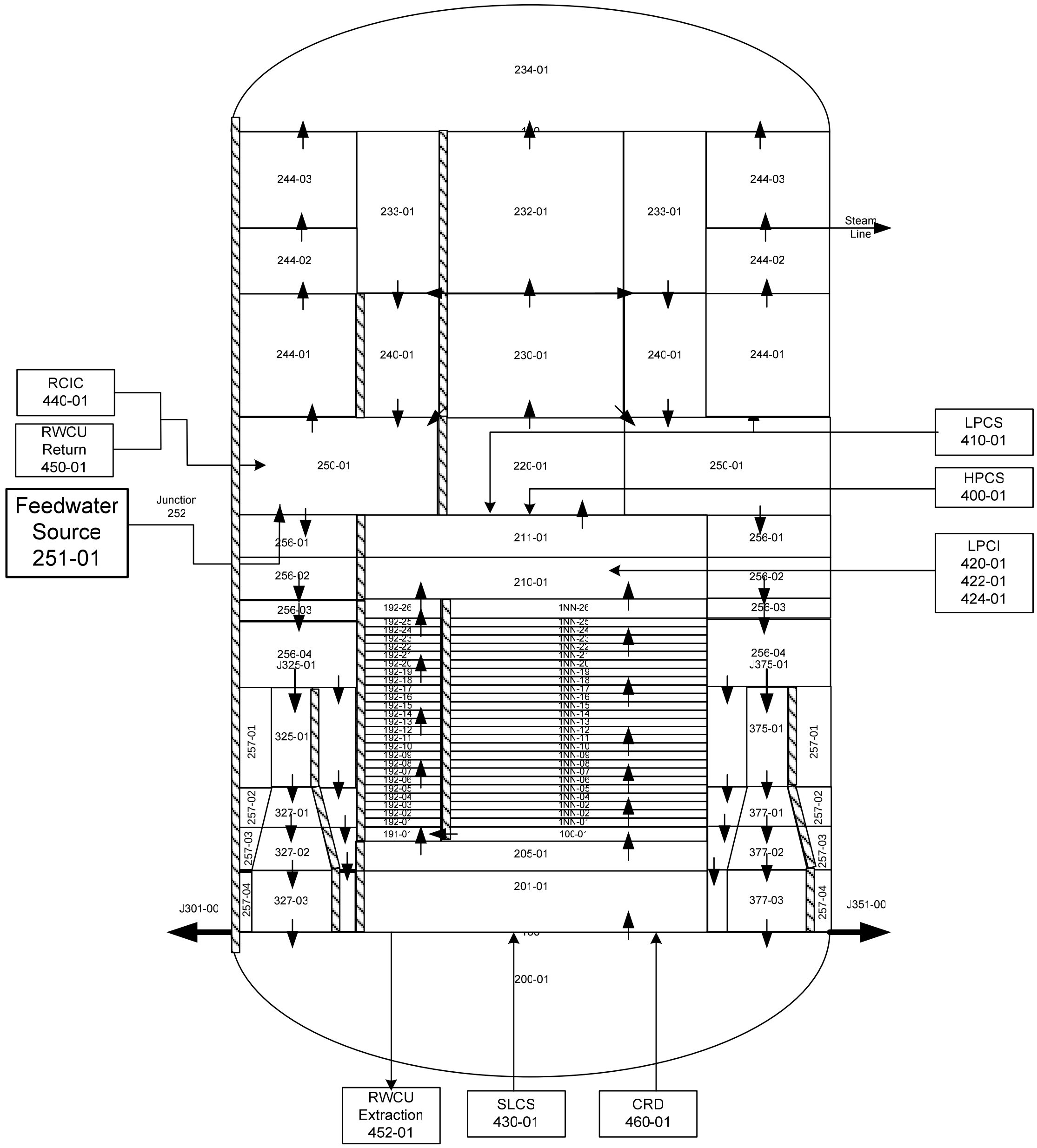
Simulator benchmarking calculations for a BWR6 plant are underway using the coupled S3K/RELAP5-3D code system. The RELAP5 model was initially developed by the utility and provided to Studsvik. The model was modified to incorporate a SIMULATE-3K interface and the vessel was renodalized to include the latest RELAP5 modeling guidelines. The scope of the model includes the reactor vessel, both recirculation loops modeling individually (a lumped jet pump is used for each loop), and four separate steam lines out through the steam header. All turbine control valves were lumped into one effective control valve and all steam dump valves were lumped into one effective dump valve. A control system to position the control and dump valves was included that controlled the pressure in the reactor and permitted a trip of the turbine. Each safety/relief valve (SRV) was modeled individually. The feedwater flow was modeled as a flow/temperature boundary condition with a three-element controller.

A diagram of the reactor vessel nodalization is shown in the figure below.

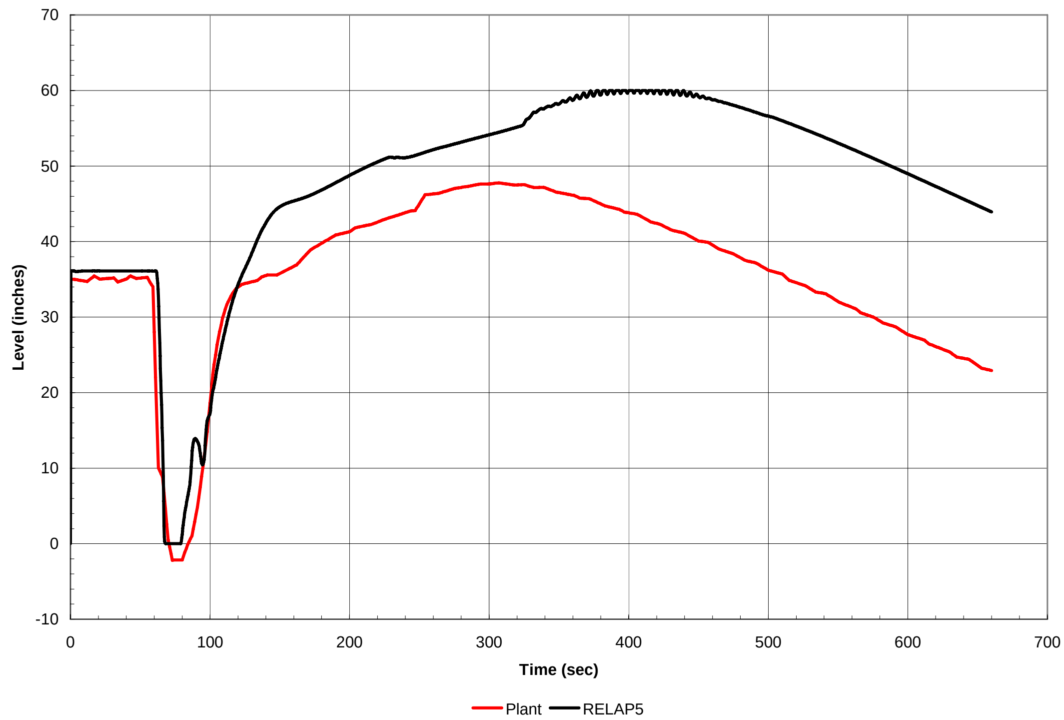
Preliminary comparisons to two plant transients, a scram and a downpower maneuver, have been performed. Graphs of selected parameters are shown below.

Reactor Scram Transient

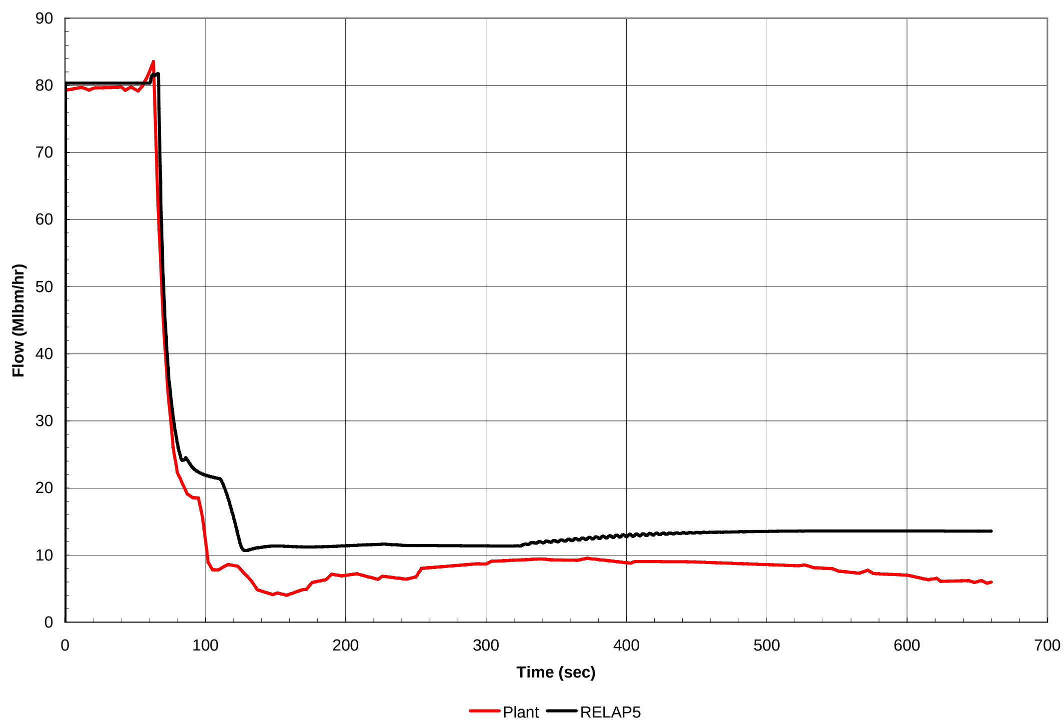
The model was set up to perform a reactor scram at 60 seconds and all control and safety systems were allowed to operate as designed. The response of the model compares quite well with the plant data. The only significant difference is the pressure at which the model stabilizes 2-3 minutes after the scram. The steam dump system is controlling the system to a different pressure setpoint than the plant.



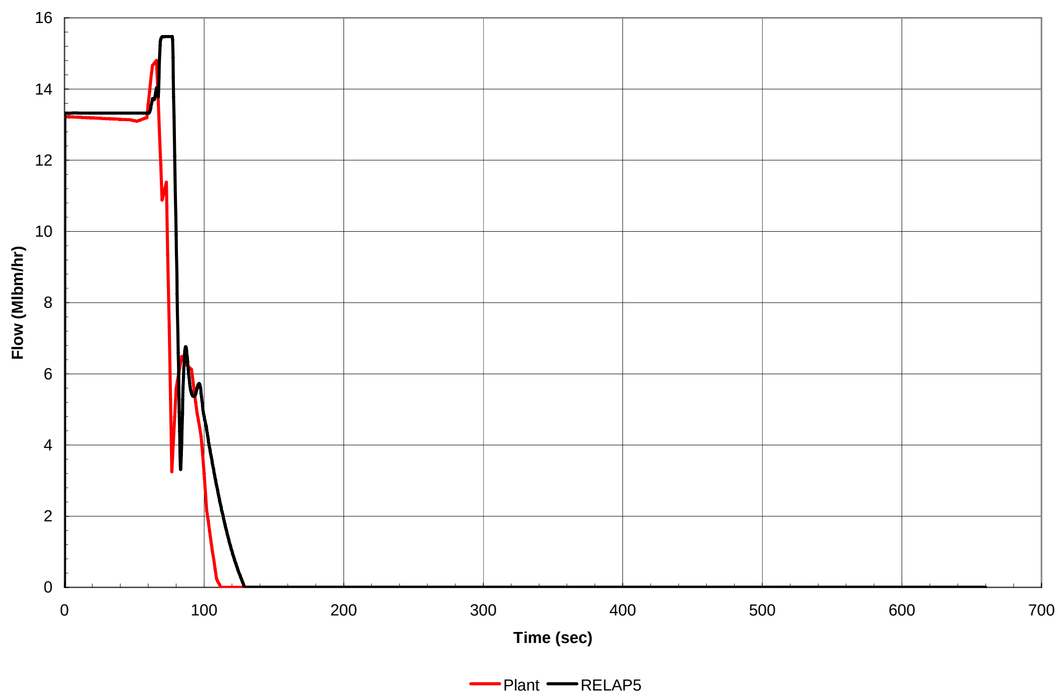
RELAP5 nodalization of the BWR6 reactor vessel



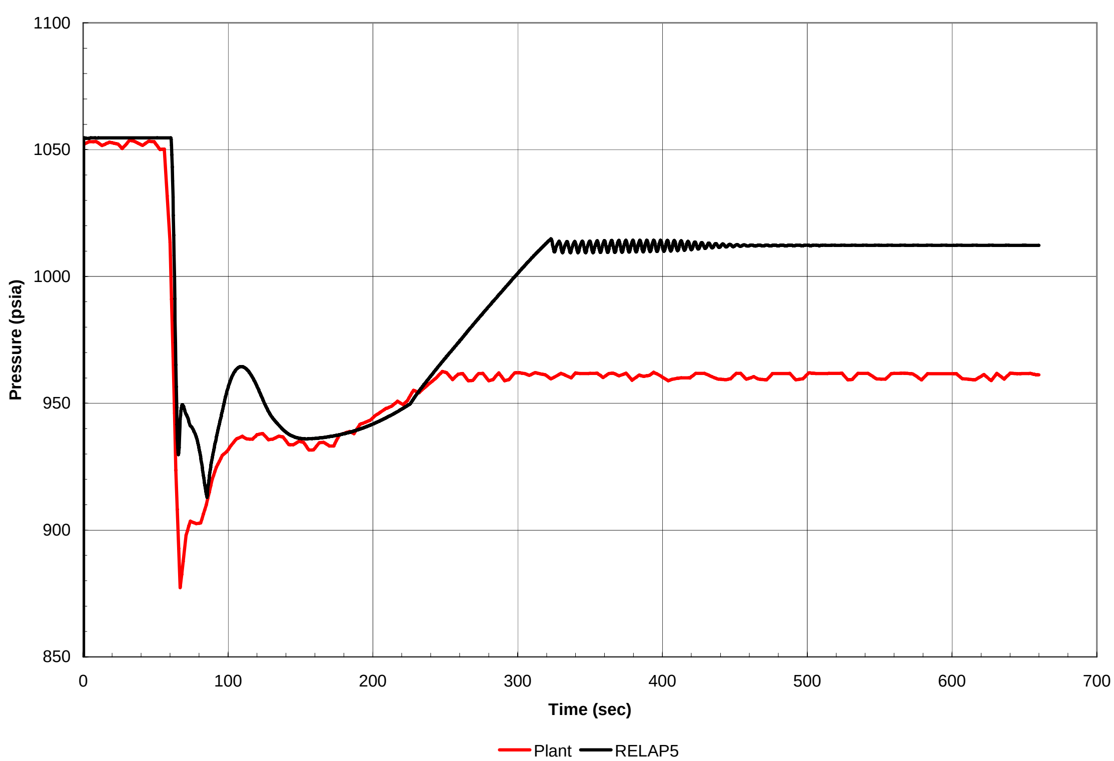
Narrow range level response during a reactor scram.



Core flow response during a reactor scram



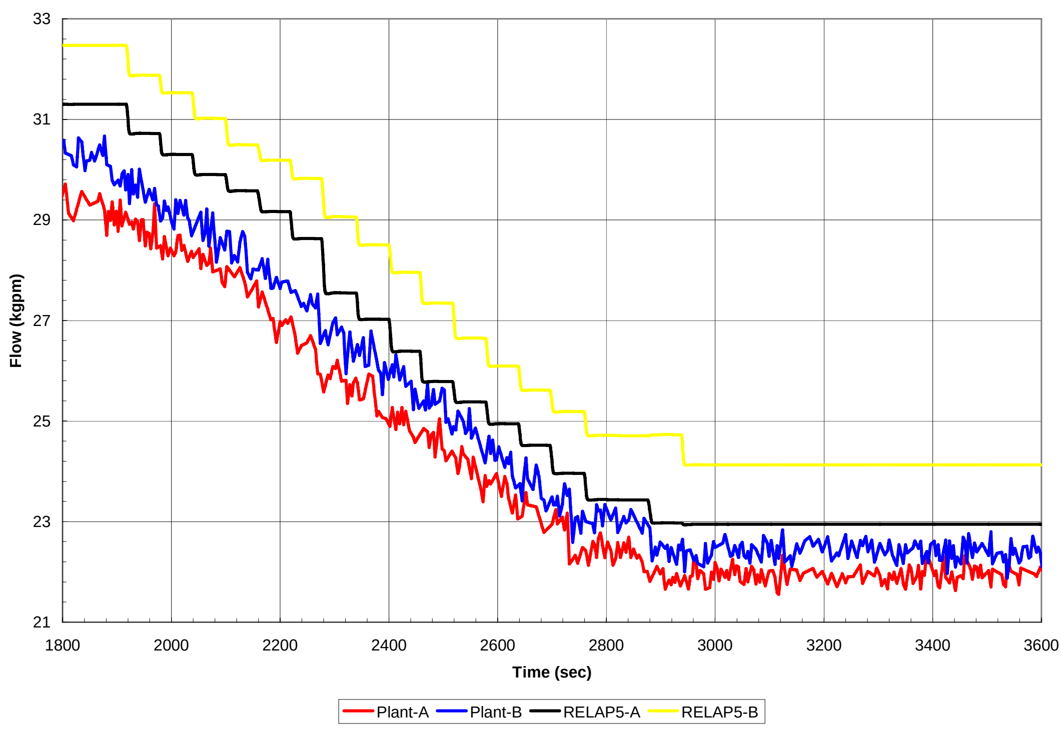
Feedwater flow response during a reactor scram.



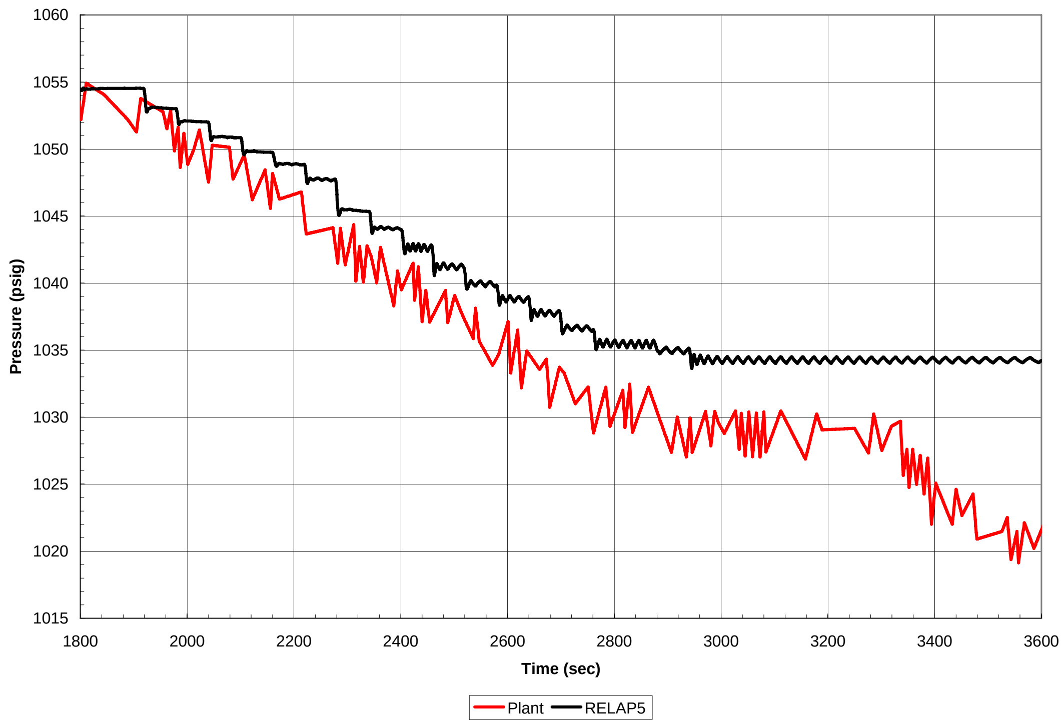
Steam Dome Pressure response during a reactor scram.

Downpower Maneuver

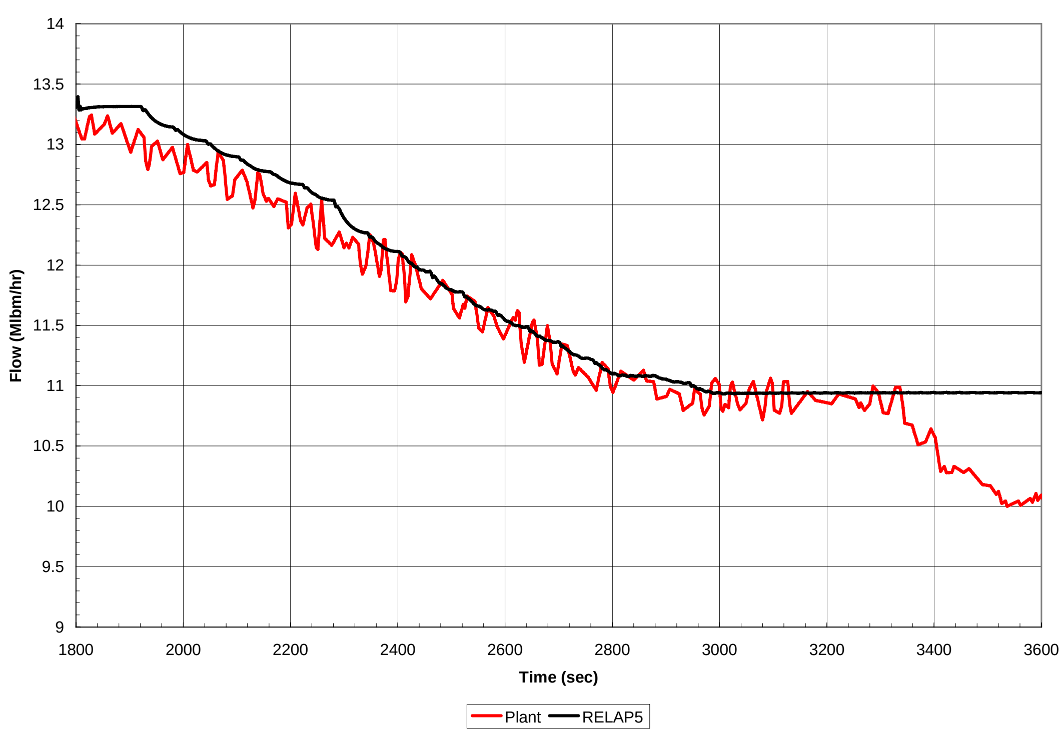
The downpower maneuver was set up to follow the plant flow control valve (FCV) positions versus time. The FCV positions started changing at 1920 seconds and all control and safety systems were allowed to operate as designed. The response of the model compares quite well with the plant data. The RELAP5 feedwater temperature was held constant during the transient, whereas the actual feedwater temperature will vary somewhat during the transient, but this is not expected to affect the results significantly.



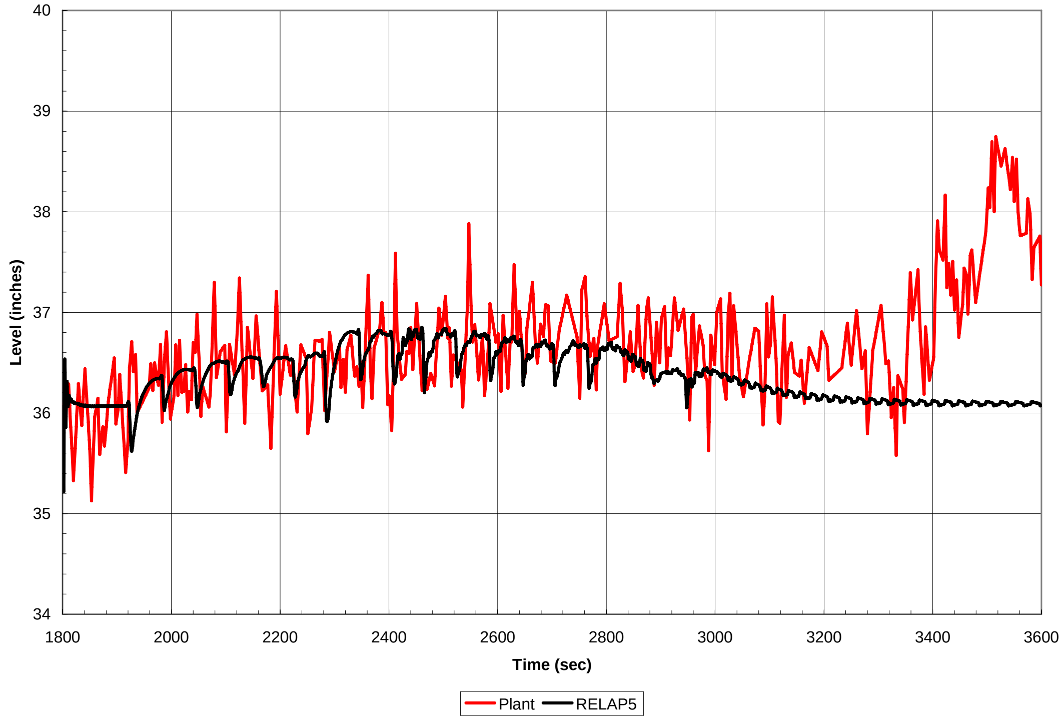
Recirculation drive flow during a downpower maneuver.



Steam dome pressure response during a downpower maneuver.



Feedwater flow response during a downpower maneuver.



Narrow range level response during a downpower maneuver.

4. Summary

All simulators are required to perform validation testing of their simulator models upon completion of the initial simulator construction or whenever significant changes are made thereafter. Many simulators, if not most, have performed at least one model upgrade since the initial completion of their simulator. However, many of those same simulators are relying on the original engineering calculations performed for benchmarking the initial simulator construction validation. Even those that have utilized updated engineering calculations are relying on very simplified engineering models. Specifically, most simulators benchmark to engineering analyses that were performed to satisfy their Safety Analysis requirements. The problem with this is that the models used for these safety analyses are much more limited in the extent of the plant that they model and, therefore, much more simplified in terms of modeling auxiliary systems that provide critical boundary conditions to the NSSS models. The result is essentially benchmarking a very sophisticated simulator model to a very simplified engineering model. The only reason this is acceptable is due to the pedigree of the engineering codes compared with the lack of pedigree of the simulator codes. So, we compromise by sacrificing complexity of the model for the pedigree of the computer codes.

Now this compromise is no longer required. With the integration of RELAP5-3D and S3K, nuclear power plant training departments can now benchmark their simulator models to pedigreed engineering codes. The result is a computer model of the plant of unparalleled complexity and sophistication that brings with it the pedigree of the two most widely used engineering codes in nuclear safety analysis and design.

Studsvik-Scandpower offers this unparalleled capability in several ways. First, Studsvik can sell the utility the license for the codes, build the initial benchmark models, and complete the initial validation testing of the simulator. At the end of the project, Studsvik would provide a detailed report of the validation results. The utility would then own the software and models so that they could perform all future validation testing in-house. Second, Studsvik can build the benchmark models, complete the validation testing of the simulator, and provide the detailed validation report without selling the license for the computer codes. In this approach, Studsvik is essentially providing a simulator validation service. Without the license to the codes, the utility would have to return to Studsvik for future validation testing if/when it became necessary. Third, the first approach can be followed with the exception that the utility accepts responsibility for performing the validation testing including producing the final report.

In the present simulator environment, more and more attention is being placed on the adequacy of simulator validation testing. Obviously, the best validation is to benchmark against actual plant data; however, for most abnormal and definitely severe transients, no plant data is available. For these transients, the integration of the S3K/RELAP5-3D models provide the best estimate of actual plant behavior available with the only available computer codes that have an engineering pedigree. The end-result will be the highest level of confidence in simulator fidelity and operator training possible today.

Another benefit would be the availability of an engineering analysis tool that could be used by engineering and operations to scope out design modifications to the plant or operating procedure changes. The licenses for the computer codes could be structured in such a way to permit a site-wide use of the codes and models for just these types of activities. The benefit to engineers and operators of being able to scope design modifications and procedure changes from their desks to eliminate unacceptable options and limit their efforts to the only the viable options is extremely valuable.

In summary, the successful integration of S3K and RELAP5-3D affords new opportunities with many benefits previously unavailable to the commercial nuclear power plant industry. Higher fidelity simulation and the availability of advanced analytical tools to engineers and operators are all now possible with the integration of these two best-estimate codes. Results of the coupled code system to date have shown very good comparison to plant transient data.