

SIMULATE-3K LINKAGE WITH REACTOR SYSTEMS CODES

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Abstract

SIMULATE-3K is Studsvik Scandpower's best-estimate three-dimensional core kinetics code. SIMULATE-3K has been coupled to several best-estimate reactor systems codes including, RELAP5-3D, RELAP5-3.3, TRACE V5.0, and RETRAN-3D. The coupled codes can be applied to existing reactors and to advanced reactor designs.

The S3K linkage to each of the systems codes is a direct, explicit coupling of the two codes on a synchronous time-step basis. The coupling provides an execution method for the S3K three-dimensional neutronic model using the Nuclear Steam Supply System (NSSS) boundary conditions calculated by the systems code. Also, it allows the S3K calculated total core power and core power distributions to drive the 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. A coupled analysis of these transients is important because the core behavior is closely tied to the NSSS system. For example, 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, requires a detailed plant model.

The Peach Bottom 2 turbine trip transient is used to assess the accuracy of the coupled code calculations. Comparisons of the important plant parameters to results from RELAP5-3D, RELAP5-3.3, and TRACE V5.0 calculations are shown and discussed.

The MSLB benchmark is also used to demonstrate the capabilities of the coupled code systems. Comparisons of the calculated reactor power to the reference data are shown and discussed.

The comparisons demonstrate the applicability of S3K, either standalone or coupled with a system analysis code, to properly model system response during accident scenarios.

Keywords: SIMULATE-3K, coupled codes, safety analysis.

Introduction

SIMULATE-3K (S3K) is Studsvik Scandpower's neutron kinetics code [1]. S3K is currently used for reactivity initiated accidents (RIA) and BWR stability.

S3K has its own thermal-hydraulics solver, but it is in general more limited than that available in most reactor systems codes in use today. In addition, the range of control systems that can be modeled are much more limited than those available in most systems codes.

In order to take advantage of S3K and of the capabilities of the main reactor systems code (TRACE [2], RELAP5-3D [3], and RELAP5-Mod3.3 [4]), Studsvik has developed a generic interface between those codes and S3K. In principle, any reactor systems code can be interfaced with S3K using the generic interface.

The generic interface handles all of the additional input required to couple S3K with a systems code and all of the data transfers necessary to complete the coupling.

This paper briefly describes S3K and the three systems codes (TRACE, RELAP5-3D, and RELAP5-Mod3.3) that have been coupled with S3K. In addition, comparisons of the NEA MSLB benchmark and the Peach Bottom Turbine Trip benchmark are presented for each code system.

1. S3K 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 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 Nuclear Fuel Industries correlations used by the FRAPCON code. 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.

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. 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.

S3K version 2.03.10 is used for these calculations.

2. RELAP5 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, ATWS, and operational transients such as loss of feedwater, loss of offsite power, station blackout, and turbine trip.

RELAP5-3D, the USDOE of RELAP5, 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. RELAP5-3D version 2.3.7 is used for these calculations.

RELAP5-Mod3.3, the USNRC version of RELAP5, is very similar to the RELAP5-3D code and has compatible input, so the same RELAP5 input models can be used in both codes with little or no modification. Patch level 3 of RELAP5-Mod3.3 is used for these calculations.

3. TRACE overview

TRACEv5.0 is the latest in a series of advanced, best-estimate reactor system codes developed by the U.S. Nuclear Regulatory Commission (with the involvement of Los Alamos National Laboratory, Integrated Systems Laboratory (ISL), The Pennsylvania State University (PSU) and Purdue University) for analyzing transient and stationary neutronic/thermal-hydraulic behaviour of Light Water Reactors (LWRs). The code is a result of a consolidation of the capabilities of previous USNRC supported codes, such as TRAC-PF1, TRAC-BF1, RELAP-5 and RAMONA. The most important models of TRACE include multidimensional two-phase flow, non-equilibrium thermodynamics, generalized heat transfer, reflood, level tracking, and reactor kinetics.

The set of coupled partial differential equations, together with the necessary closure relationships, are solved in a staggered (momentum solved at cell edges) finite difference mesh. Heat transfer is treated semi-implicitly, while the hydrodynamic equations (1, 2 and 3 Dimensional) make use of a multi-step time differencing scheme (SETS) that allows the material Courant limit to be violated, thus resulting in large time step sizes for slow transients, and fast running capabilities. The system of coupled non-linear PDEs is solved by means of a Newton-Raphson iterative method, which results in a set of linearized algebraic equations in pressure, whose results is obtained by direct matrix inversion. A full two-fluid (6-equations) model is used to evaluate the gas-liquid flow, with an additional mass balance equation to describe a non-condensable gas field, and an additional transport equation to track dissolved solute in the liquid field.

TRACE V5.0patch02 is used for these calculations.

4. System code coupling with S3K

Typically, the system code core thermal-hydraulics (TH) 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 the system code 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 system code core region is that it must contain the same number of axial planes in the active fuel region as the S3K model. A brief description of the linkage between system code and S3K 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 S3K three-dimensional neutronics using the Nuclear Steam Supply System (NSSS) boundary conditions calculated by the system code thermal hydraulics

code. It allows the S3K calculated total core power and core power distributions to “drive” the system code system model core. The system code uses its normal time step logic. S3K specifies a fixed time step over ranges of elapsed time. These S3K time steps must be no smaller than the maximum system code time step and must be an integer multiple of the system code maximum time step if it is not equal to the system code maximum step. S3K does not compute power until the accumulated time since the last calculation is greater than or equal to the S3K time step. If the accumulated time is greater than the S3K time step, the accumulated time is used as the S3K time step so that the S3K elapsed time is synchronized with the system code elapsed time.

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 behaviour of NSSS components, such as the separators, pressurizer, steam generators, and steam lines. A coupled analysis of these transients is important because the core behaviour is closely tied to the NSSS system.

The thermal hydraulic conditions in the core and plenum regions are passed to the S3K model which performs a calculation of detailed core power, which is then passed back to the system code model to use for the next time step. There are three different coupling options that are available for the linkage system code and S3K, “plenum”, “flat”, and “nodal”. Each of these options is described in the following paragraphs.

The “plenum” coupling option utilizes the S3K thermal-hydraulics calculation. The inlet flow and enthalpy to the core and the exit pressure in the upper plenum is provided by the system code model for each core channel. S3K 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 system code model and provided to system code. This option performs quite well provided that the core flow is always positive.

The “flat” coupling option does not utilize the S3K thermal-hydraulics calculation. Each fuel assembly in a system code channel receives the same fuel temperature, coolant density, and boron concentration at a given axial plane from the system code 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 system code channels are modelled.

The “nodal” coupling option is a variation of the “flat” option. Once again, the S3K 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 system code 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 system code channel.

5. Main steam line break benchmark comparisons

The reference problem chosen for simulation in a PWR is a MSLB, which may occur as a consequence of the rupture of one steam line upstream of the main steam isolation valves. The reference design for

the PWR is derived from the reactor geometry and operational data of Three Mile Island Unit 1 NPP (TMI-1) [5].

The transient starts with a doubled-ended break of one main steam line at the tie with the cross-connection line. The 24 inch (60.96 cm) main steam line and 8 inch (20.32 cm) cross-connect rupture results in the highest break flow assumption and maximizes the Reactor Coolant System (RCS) cool-down. The worst single failure is the failure in the open position of the feedwater regulating valve to the broken SG. This failure in the open position causes feedwater flow from the intact SG to cross over to the broken SG across the common header and maximizes the feedwater flow to the broken SG. The feedwater flow is eventually terminated by closure of the feedwater block valve, which is conservatively assumed to close 30 seconds after the break occurs. Subsequent to break initiation, and following reactor scram, the steam line turbine stop valves are assumed to slam shut, isolating the intact SG. The 8 inch cross-connect between the two steam lines of the broken SG remains open.

The models used for this comparison model the core only and use boundary conditions provided in the benchmark to represent the core inlet flow and enthalpy and the core exit pressure. A constant time step of 0.025 seconds is used for all the system codes and S3K for these calculations. The calculations were terminated after 100.0 seconds of simulation time.

5.1 TRACE and RELAP5 model description

The system code models (TRACE, RELAP5-3D, and RELAP5-Mod3.3) model the core region only. The boundary condition data for Exercise 2 in Reference [7] were used for core inlet flow and enthalpy and core exit pressure.

There are 18 active fuel flow channels modelled. Each channel is modelled with a pipe component and a heat structure for a given channel. The power generated by the S3K model is assigned to the appropriate heat structure and hydraulic cell based on the mapping information contained in the S3K model. The S3K mapping algorithm also extracts the thermal-hydraulic data from the systems code that is necessary to provide the feedback terms for the cross section model in S3K.

5.2 S3K model description

The nuclear data from the reference were used and converted to the format necessary for use by the core analysis code SIMULATE-3. The S3K core model relied on Studsvik's codes to generate the cross section data, and the history data. All data pertaining to the core (cross-sections, assembly dimensions, pin enrichments, etc.) were taken from the specification. The control rods were positioned as specified in the benchmark specification with the control rod in location N-12 stuck in a full out position for the duration of the transient.

The core is modeled with 24 planes in the active fuel with a uniform axial plane height of 6 inches. Each fuel assembly is modeled with 4 radial nodes (standard Studsvik PWR modeling guideline) which require hydraulic coupling information is input with 30 rows and columns. Figure 1 shows the assignment of the hydraulic channels modelled in the systems code to the fuel nodes in the S3K core model. Note that four nodes per assembly are used to model the core and this is reflected in Figure 1. The fuel assemblies on the X and Y axes are split between flow channels to preserve quadrant symmetry.

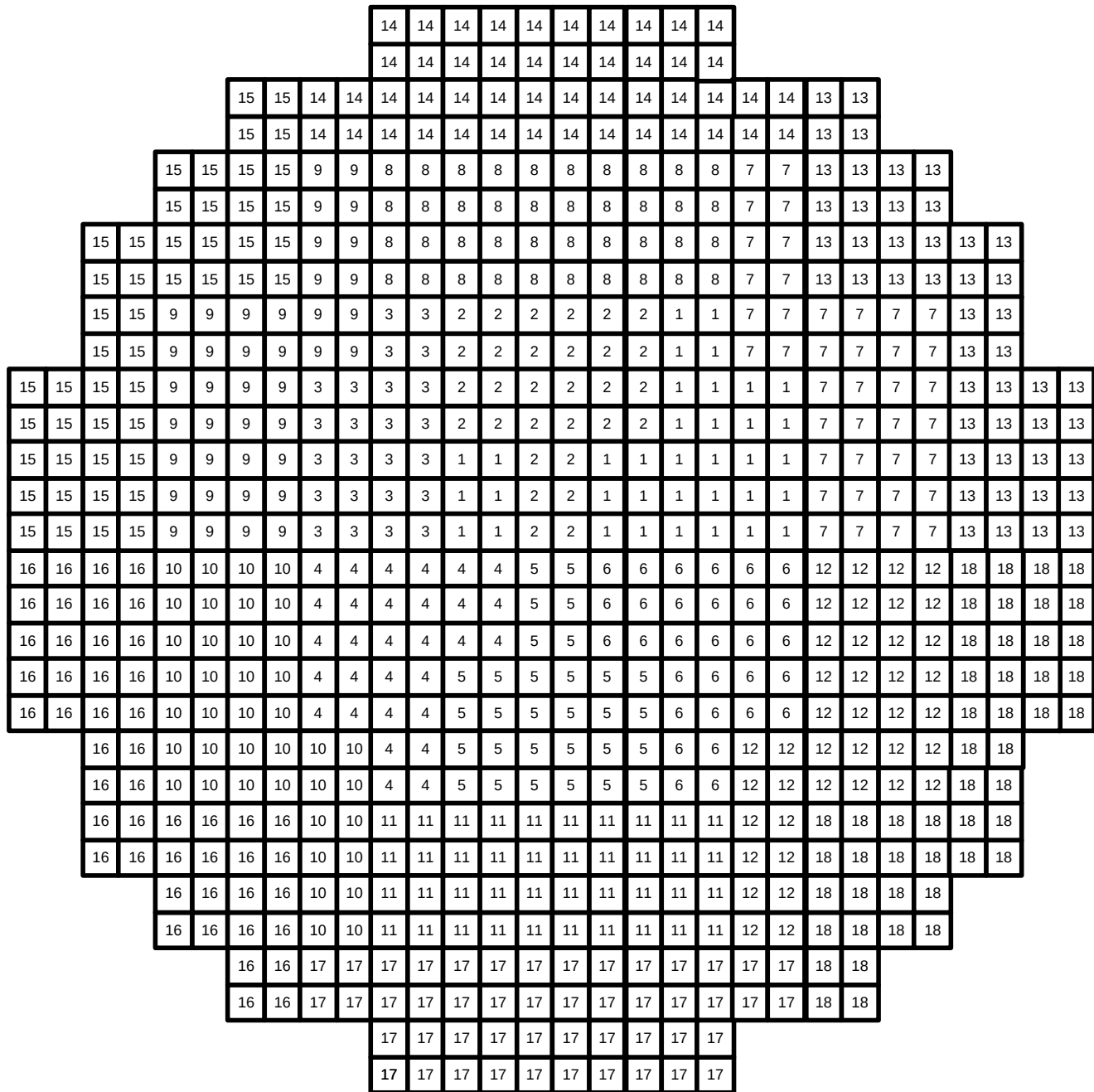


Figure 1 Assignment of fuel assemblies to system code flow channels for the MSLB benchmark.

S3K has input components that define the system code hydraulic cells and heat structures associated with each channel and plane in the core.

5.3 Results

The primary variable for comparison for the core-only models is the reactor power response. Figure 2 shows the comparison of the reactor power response for each of the coupled models. The agreement on the peak power is quite good between the different system code models. There are slight differences in the timing of the scram between the codes.

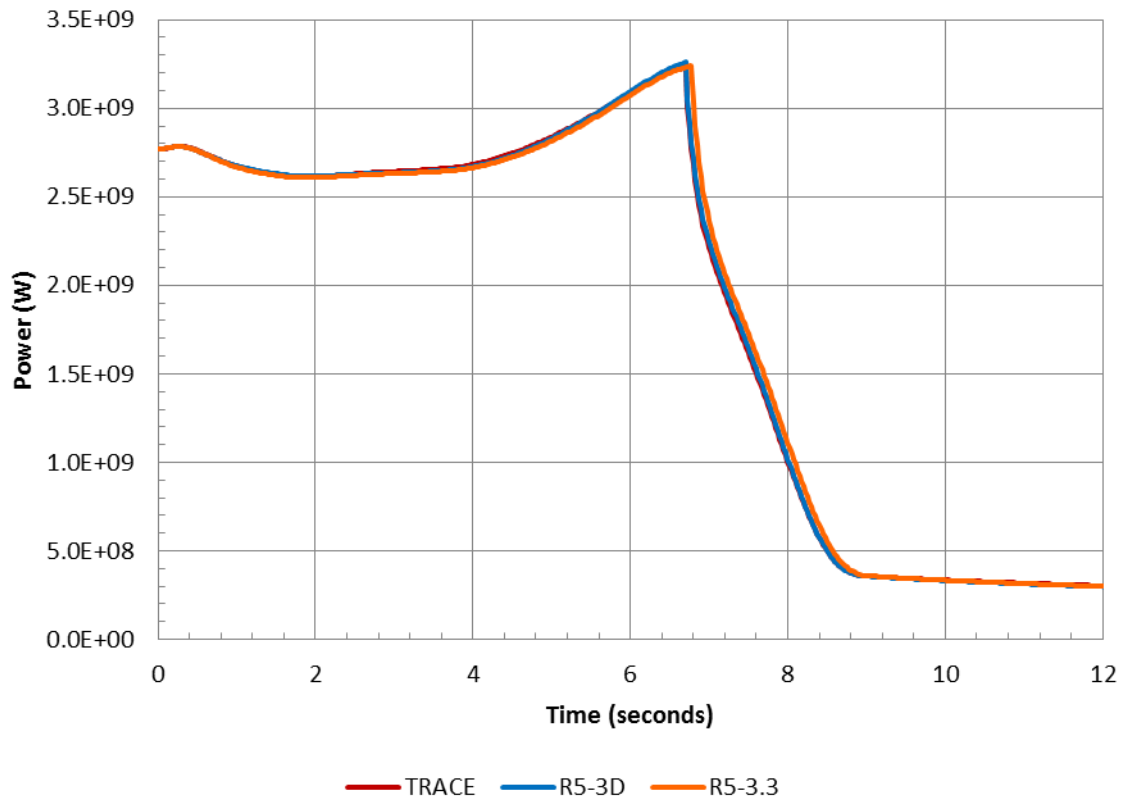


Figure 2 Comparison of coupled code reactor power response for the MSLB benchmark.

The calculations were performed on an INTEL i7 CPU running at 2GHz and the 64-bit Windows 7 Home Premium operating system. All codes were compiled with Visual Studio 2008 and INTEL FORTRAN 12.1.127. Table 1 shows the CPU times for each calculation.

Table 1 CPU Times for the MSLB Calculations

System Code	CPU Time (seconds)
TRACE V5.0p2	654.5
RELAP5-3D	1359.5
RELAP5-3.3	604.8

6. Peach Bottom turbine trip comparisons

The PB2 TT2 starts with a sudden closure of the turbine stop valve (TSV) followed by the opening of the turbine bypass valve. From a fluid-flow phenomena point of view, pressure and flow waves play an important role during the early phase of the transient. This is because rapid valve actions cause sonic waves, as well as secondary waves, which are generated in the pressure vessel. The pressure oscillation generated in the main steam piping propagates with relatively little attenuation into the reactor core. The induced core pressure oscillation results in changes to the core void distribution and fluid flow. The magnitude of the neutron flux transient in the BWR core is affected by the initial rate of pressure rise (caused by the pressure oscillation) and has spatial variation. The detailed description of the test is available in Ref. [6].

The simulation of the power response to the pressure pulse and subsequent void collapse requires a 3-D core modeling supplemented by a 1-D simulation of the remaining reactor coolant system. A turbine trip transient in a BWR reactor is considered one of the most complex events to be analyzed because it involves the reactor core, the high pressure coolant boundary, associated valves and piping in highly complex interactions with rapidly changing variables. As mentioned earlier, the transient begins with a sudden TSV closure that initiates a pressure wave in the main steam system, which is quickly transmitted to the reactor pressure vessel. While the TSVs are closing, the bypass system valves are designed to open, which allows for steam release and, thus, pressure relief. Safety relief valves (SRVs) eventually begin to open at re-established set points, providing additional pressure relief. The pressure wave requires a detailed nodalization modeling of the steam system and its associated valves to correctly capture the timing effects and pressure wave magnitude. This assures the availability of a pressure history on each valve, allowing adequate modeling of steam flow through the valves. A constant time step of 0.001 seconds for the first 1.0 second of the transient and 0.005 seconds for time greater than 1.0 seconds is used for all the system codes and S3K for these calculations.

6.1 TRACE model description

The TRACE modelled was developed by PSI and described in Reference [7]. It was originally setup to use with a PARCS core model and was adapted to use an S3K model developed by Studsvik [8]. The active core region is modelled with 33 flow channels and 24 axial planes in the active fuel regions. Figure 3 shows the layout of fuel assemblies in the S3K model to flow channels in the TRACE model along with the location of control blades and the outer orifice region.

6.2 RELAP5 model description

The RELAP5 model was developed by SSP based on information from References [6] using standard modelling guidelines for best estimate calculations for a BWR. The RELAP5 model uses the same 33 flow channels to model the core fuel assemblies as the TRACE model described previously.

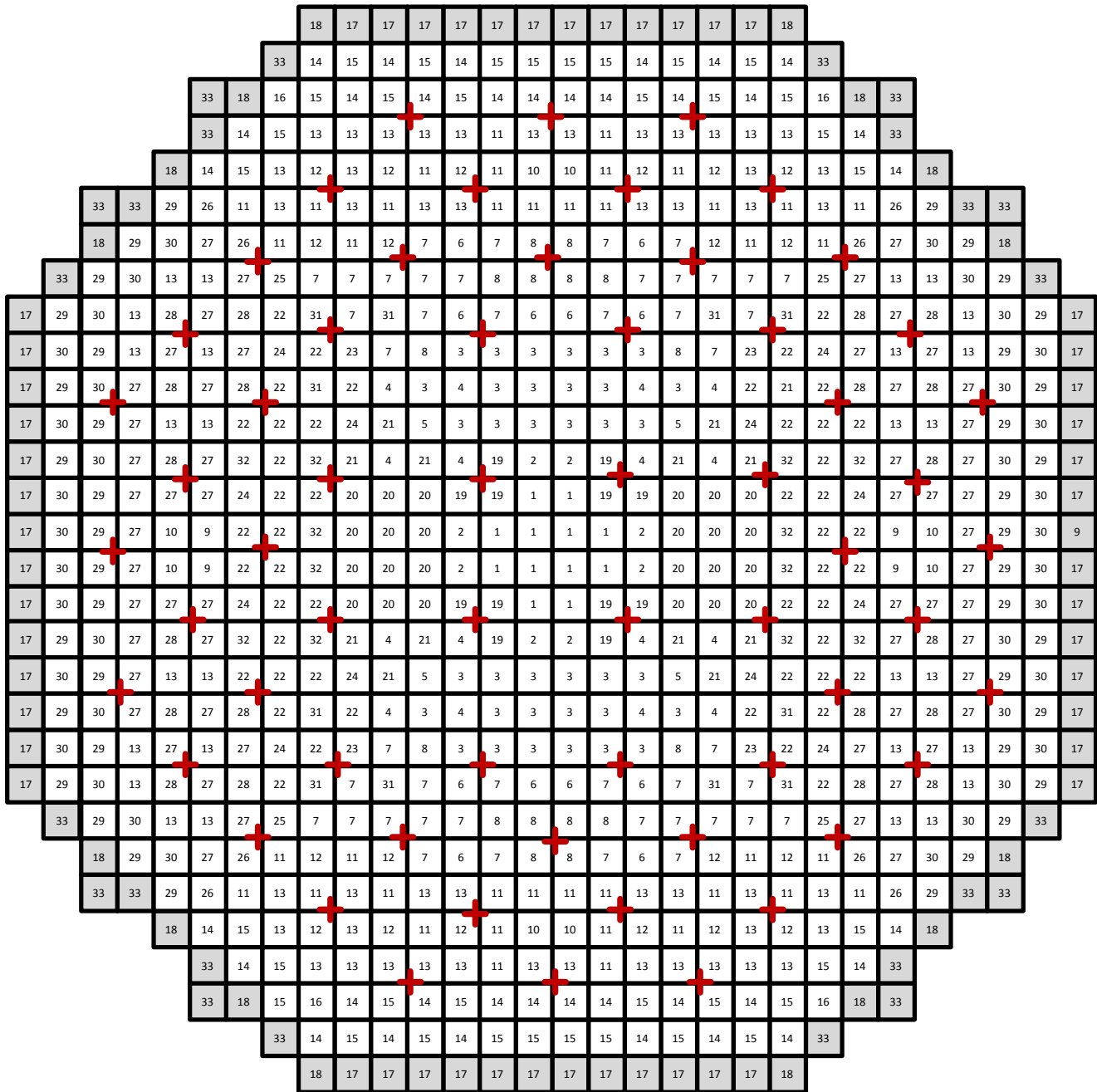


Figure 3 Assignment of fuel assemblies to TRACE flow channels for Peach Bottom.

6.3 S3K model description

The neutronic core model is based on the S3K standalone 764-channels model described in details in Reference [8]. The S3K core model relied on Studsvik's codes to generate the cross section data, and the history data. All data pertaining to the core (loading, assembly dimensions, pin enrichments, etc.) were taken from the specifications. The cross section data for all lattices present in Peach Bottom cycles 1 and 2 were generated by means the lattice code CASMO-4 ("L"-library). A SIMULATE-3 model for Peach Bottom was set up and depleted for two cycles until the end of cycle 2 corresponding to the point at which the turbine trip transients were conducted.

The S3K input contains the mapping information that describes which system code hydraulic cells and heat structures are used for extracting thermal-hydraulic feedback and depositing calculated power.

6.4 Results

The primary parameters of interest in this study were the power and steam dome pressure response. The power response is shown in Figure 4. The comparison of the TRACE data to the measured data is quite good. The RELAP5-3D and RELAP5-Mod3.3 power peaks are close to the measured value, but delayed about 0.1 seconds.

Figure 5 shows the steam dome pressure response. Once again, the TRACE model compares well with the measured data. The RELAP5 models reach lower pressures primarily due to the lower peak power achieved.

The calculations were performed on an INTEL i7 CPU running at 2GHz and the 64-bit Windows 7 Home Premium operating system. All codes were compiled with Visual Studio 2008 and INTEL FORTRAN 12.1.127. Table 2 shows the CPU times for each calculation.

Table 2 CPU Times for the Peach Bottom Calculations

System Code	CPU Time (seconds)
TRACE V5.0p2	1046.1
RELAP5-3D	
RELAP5-3.3	863.3

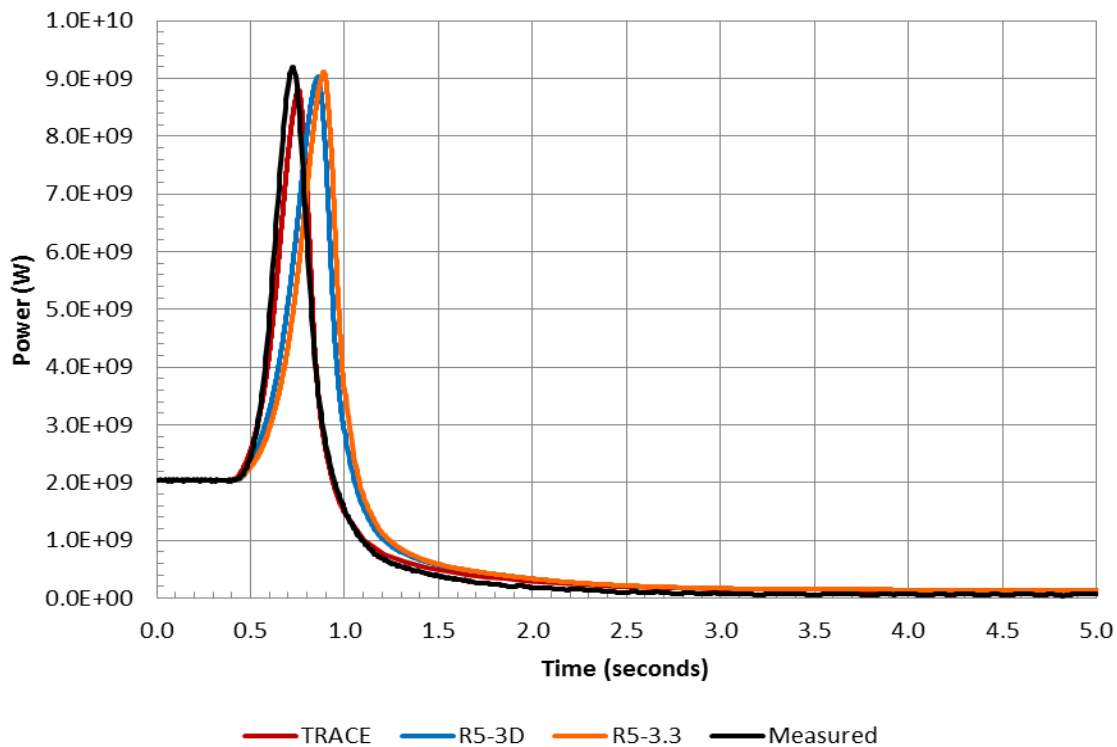


Figure 4 Comparison of the power response between the system code models for PB2 TT2.

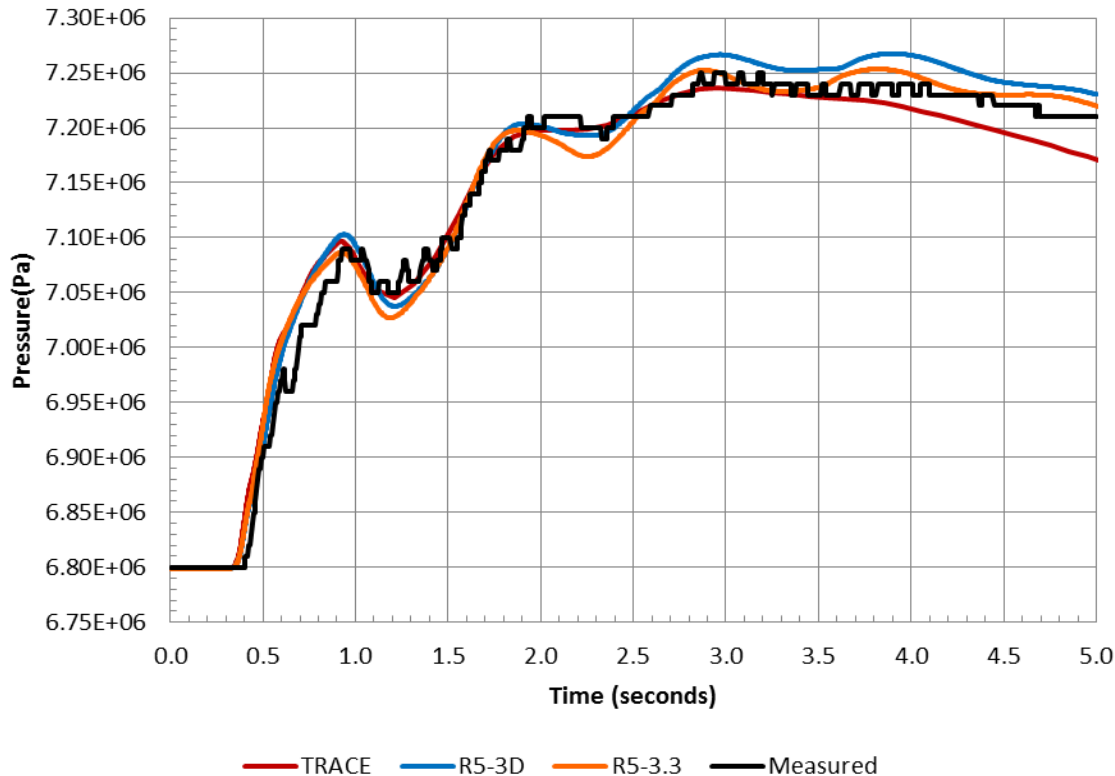


Figure 5 Comparison of the steam dome pressure response between the system code models for PB2 TT2.

7. Conclusion

In summary, the successful integration of S3K and each of the system codes 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 S3K and reactor systems codes (TRACE, RELAP5-3D, and RELAP5-Mod3.3). This coupling of S3K and standard reactor systems codes provides a tool for best-estimate calculations in support of safety analyses, PRA support, training simulator benchmarking, and just-in-time analysis of plant transients. The coupled S3K/ RELAP5-3D code has already been used to perform calculations to support PRA analyses of failed control rods in Scandinavia, automated boron injection systems for BWRs in Scandinavia, and benchmark calculations for training simulators in the U.S.

8. References

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