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A BEST-ESTIMATE COUPLED CODE FOR REACTOR SAFETY ANALYSES USING SIMULATE-3K AND RELAP5-3D

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ABSTRACT

The best-estimate neutron kinetics code, SIMULATE-3K[1], and the best-estimate nuclear systems analyses code, RELAP5-3D[2,3,4,5], have been coupled to provide a best-estimate coupled code system for performing plant transient calculations with reactivity feedback from a detailed core model. The coupling of the two well known codes provides for a robust and flexible system capable of analyzing current and proposed Light Water Reactor designs. Comparisons to plant data for two transients and a calculation of a Main Steam Isolation Valve Closure with failure to SCRAM transient are presented.

1. INTRODUCTION

This paper discusses the coupled SIMULATE-3K (S3K) and RELAP5-3D code system and a series of calculations performed with the Columbia BWR coupled models to benchmark the coupled models versus plant data and a calculation of the Main Steam Isolation Valve (MSIV) Anticipated Transient without SCRAM (ATWS) event.

2. 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 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.

3. 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, 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.

4. S3K/RELAP5-3D COUPLED CODE

Typically, the RELAP5-3D 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 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 axial planes in the active fuel region as the S3K model. A brief description of the linkage between RELAP5-3D 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 RELAP5-3D thermal hydraulics code. It allows the S3K 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.

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 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 S3K, "plenum", "flat", and "nodal". Each of these options are 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 RELAP5-3D 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 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 S3K 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 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 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.

5. COLUMBIA BWR MODEL

The RELAP5-3D and S3K models for the Columbia BWR were developed to benchmark the Columbia training simulator.

5.1 S3K Model

The S3K model was developed from the CASMO-4 and MICROBURN-2 (MB2) data used by Energy Northwest for their current simulator load. The CASMO-4 calculations were rerun with the standard Studsvik set of branch cases for full range calculations (S3C option). The S3K model was built from the MB2 model. The active core is modeled with 25 planes and each fuel assembly is modeled with one node in the radial plane. The core burnup is that used for Middle-of-Cycle (MOC, 10200 MWd/MTU). The axial and radial powers calculated by MB2 and S3K compare very well.

5.2 RELAP5-3D Model

The RELAP5-3D model was constructed from data provided by Energy Northwest and included the reactor vessel, the recirculation loops, the steam lines out through the turbine control and bypass valves, and the safety/relief valves on the steam lines. Fig. 1 shows the nodalization of the reactor vessel. The reactor core is modeled with 33 radial flow channels. The central 5 control cells (20 fuel assemblies) are represented with a single flow channel. The other thirty-two channels represent 4 rings in 8 octants with approximately 24 fuel assemblies in each flow channel. The outer ring in each octant contains all of the peripheral fuel assemblies with smaller flow orifices. This geometric layout is a compromise based on flow characteristics and a reasonable representation of the radial void distribution in the core.

Several BOP systems interface with the reactor model. These are treated as simple flow boundary conditions in the RELAP5-3D model using a combination of time-dependent junctions (tmdpjun components) and time-dependent volumes (tmdpvol components). These systems are the Feedwater (FW) system, Control Rod Drive (CRD) system, the Reactor Water Cleanup (RWCU) system, the Residual Heat Removal (RHR) system, the High Pressure Core Spray (HPCS), Low Pressure Core Spray (LPCS), Low Pressure Coolant Injection (LPCI), the drywell, and the wetwell.

A few control systems are modeled to facilitate the execution of the transients. A pressure control system regulates the pressure in the steam header using the turbine control and bypass valves. A feedwater control system maintains the level in the reactor vessel by adjusting feedwater flow. A recirculation pump control system controls the pump speed and provides for pump runback when appropriate. A scram system monitors various reactor parameters and provides a scram signal to S3K to insert the control rods. The SRVs are opened and closed based on pressure setpoints. The safety injection systems, HPCS, LPCS, and LPCI are actuated based on pressure and level in the reactor vessel.

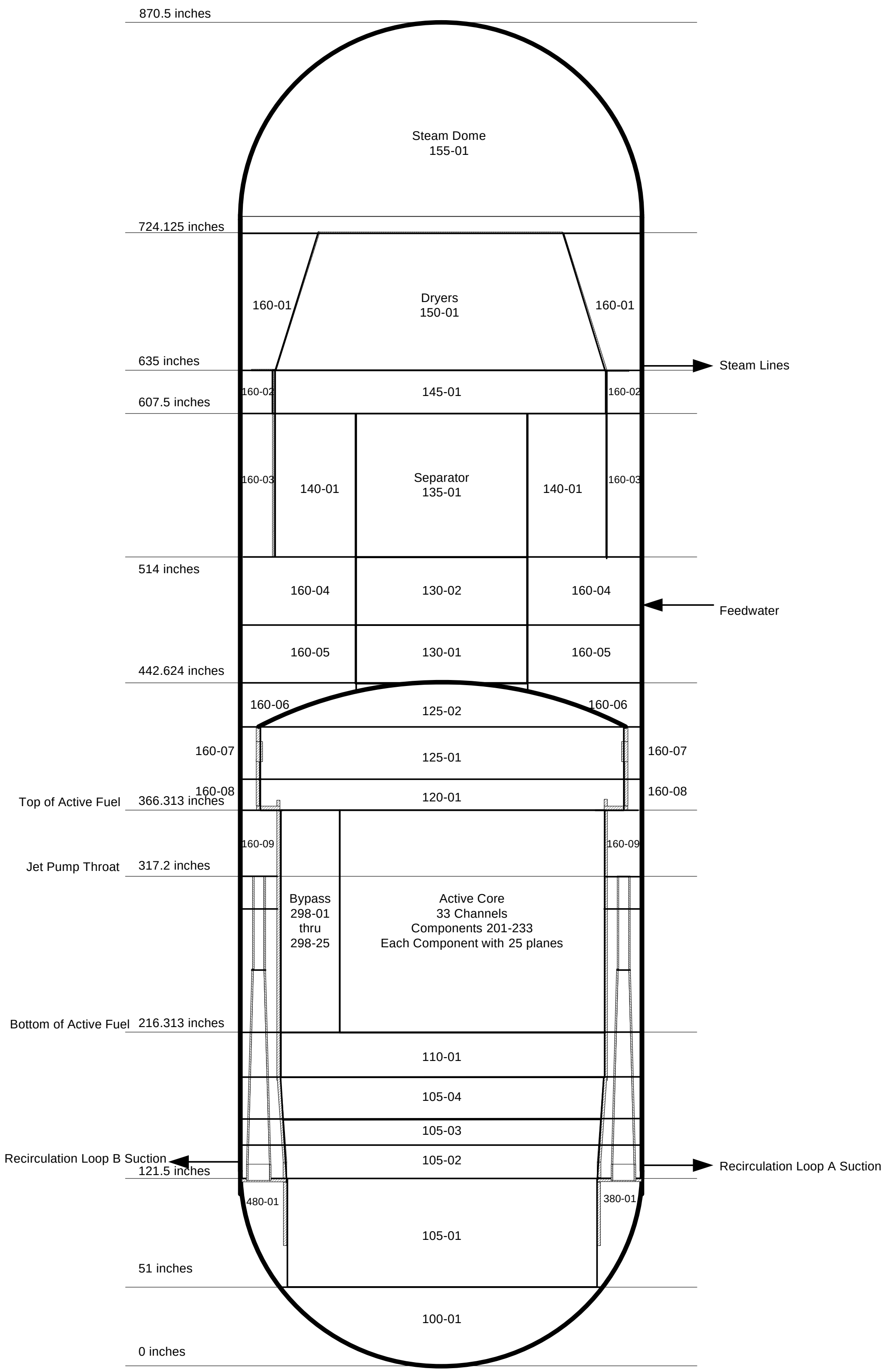

Fig. 1 Columbia reactor vessel nodalization

5.3 Comparisons to Plant Data

S3K/RELAP5-3D coupled calculations were performed for a normal reactor scram and a single recirculation pump trip. These two sets of plant data provide a good range of conditions for validating the models for transient conditions of the Main Steam Isolation Valve (MSIV) closure without scram transient (ATWS). The primary parameters of interest for the MSIV ATWS event are the reactor pressure and the reactor power. The manual scram event is used to demonstrate the adequacy of the S3K/RELAP5 model for calculating the reactor pressure. The single recirculation pump trip is used to demonstrate the adequacy of the S3K/RELAP5 model for calculating the reactor power during transient conditions without scram. All calculations are performed with the "flat" coupling option described in section 4.

Fig. 2 shows the comparison plant data and the S3K/RELAP5-3D calculation of the reactor steam dome pressure during the manual scram transient. The pressure drops more in the RELAP5 model due to the different feedwater response of the standalone RELAP5 controller versus the plants digital control system. In addition, the operators manually controlled the pressure to approximately 905 psig instead of the "hands off" approach used for the RELAP5 calculation which controls the steam dome pressure to 965 psig. In general the response is good.

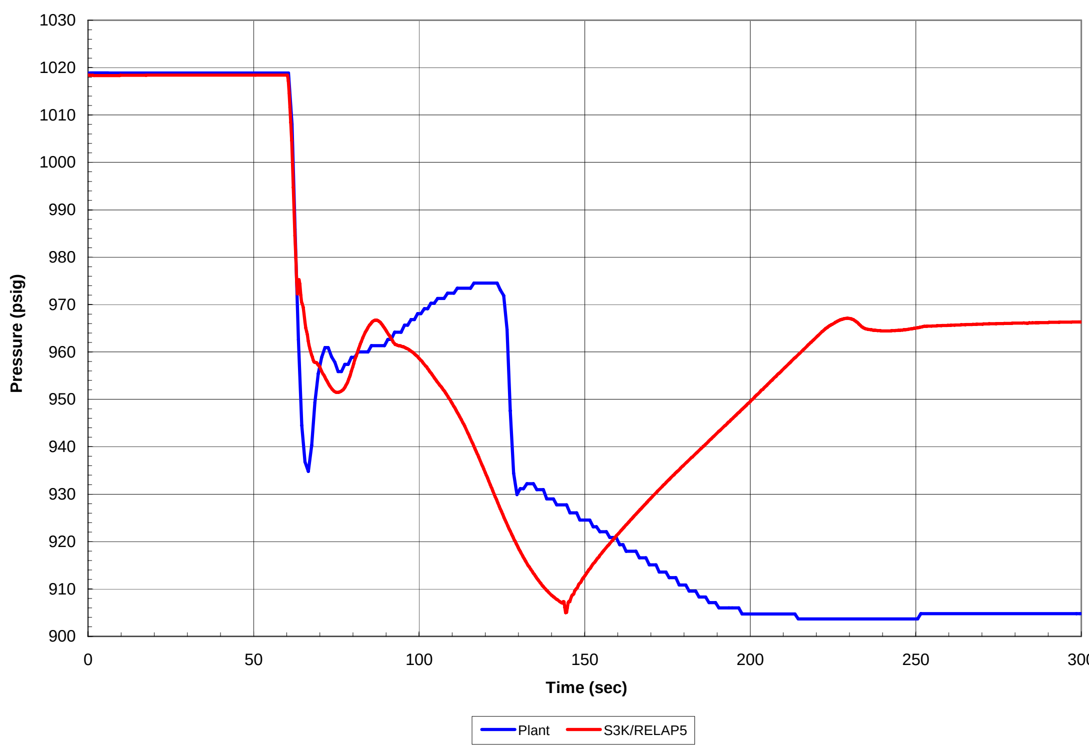

Fig. 2 Reactor pressure for the manual scram.

Fig. 3 shows the reactor level response during the manual scram. The level response differences are due in part to the feedwater controller differences. The RELAP5 model uses a standalone three-element proportional-integral controller and the plant uses a digital control system. Overall, the response comparison is quite good.

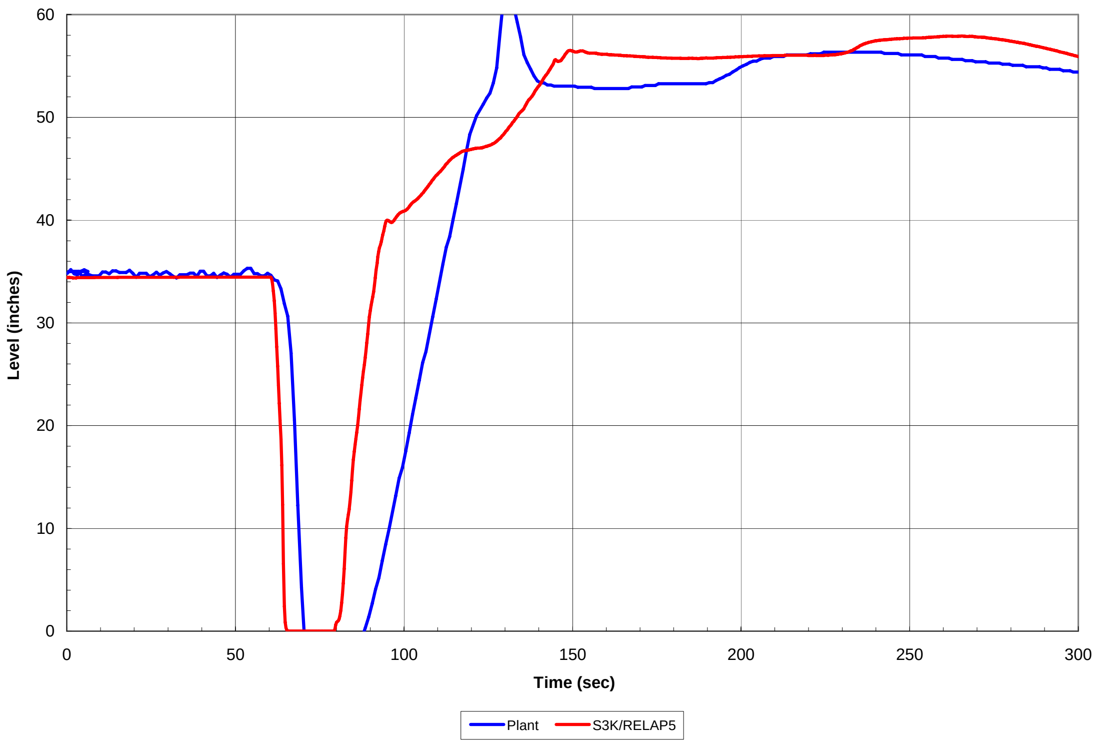

Fig 3 Reactor level for the manual scram.

Fig. 4 shows the comparison plant data and the S3K/RELAP5-3D calculation of the reactor power during the single recirculation pump trip transient. The overall comparison is good. The difference in the stable power level is due to the radial nodalization in the core regions of the RELAP5 model.

As can be seen from the comparisons in this section, the S3K/RELAP5-3D model of the Columbia BWR gives results that are comparable to the actual plant response for two transients. These comparisons give some confidence that we can apply the model to investigate the MSIV-ATWS transient for which there is no plant data.

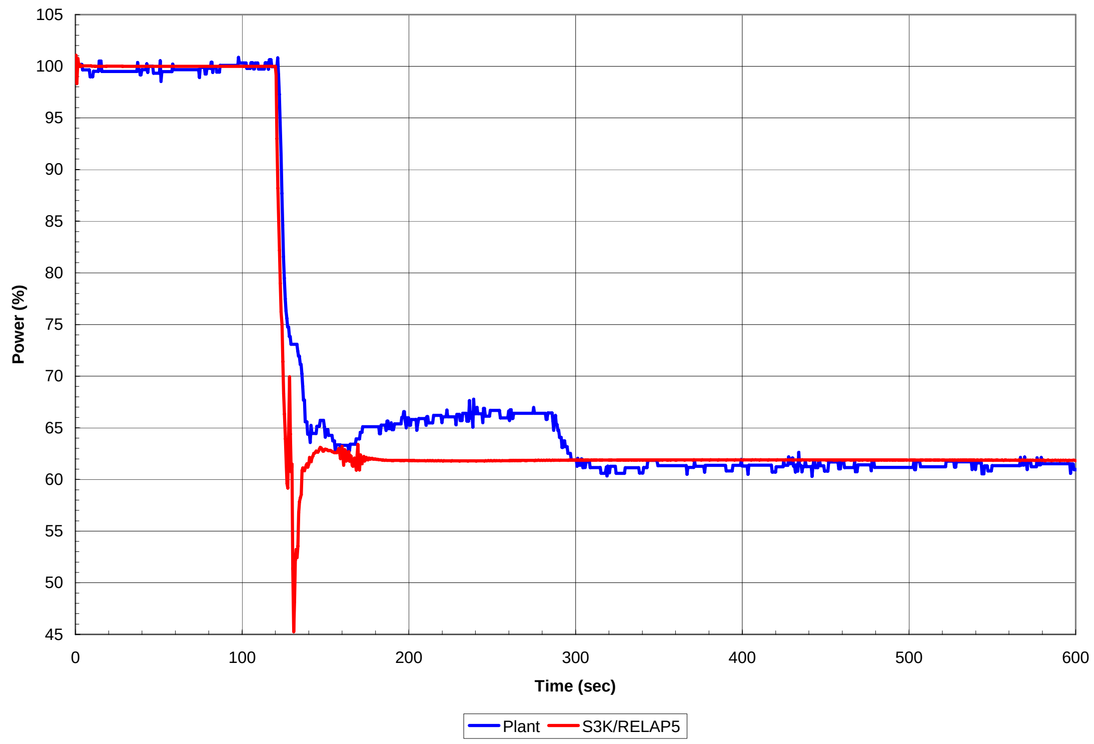

Fig. 4 APRM power for the single recirculation pump trip.

5.4 MSIV Closure Calculations

A calculation of the Main Steam Isolation Valve (MSIV) closure transient without scram was performed with the previously described S3K/RELAP5-3D model. The transient is initiated by a simultaneous closure of all 4 inner MSIVs with a closure time of 1 second. The standalone control systems for feedwater control and pressure control are allowed to work as designed. The Safety/Relief valves were assumed to operate at their relief settings. The valve hysteresis is modeled in the RELAP5 control system that positions the relief valves. Each relief valve is modeled separately and is located on the appropriate steam line.

Fig. 5 shows the reactor power response during the MSIV-ATWS transient. There is an initial spike to 370% power due to the initial pressurization and is turned around when all 18 safety/relief valves open to drop the pressure. The reactor level (see Fig. 6) drops low enough to initiate a recirculation pump runback which drops the power to a level consistent with the reduced core flow at the lower pumps speed. The power cycles along with the pressure after the pump runback as two safety/relief valves cycle to control pressure between 1020 and 1090 psia (see Fig. 7). The reactor power, pressure, and level all oscillate with the same frequency. The level is out of phase with the power and pressure and swells when the safety/relief valves are open and pressure is dropping and shrinks when the valves are closed and the reactor is pressurizing.

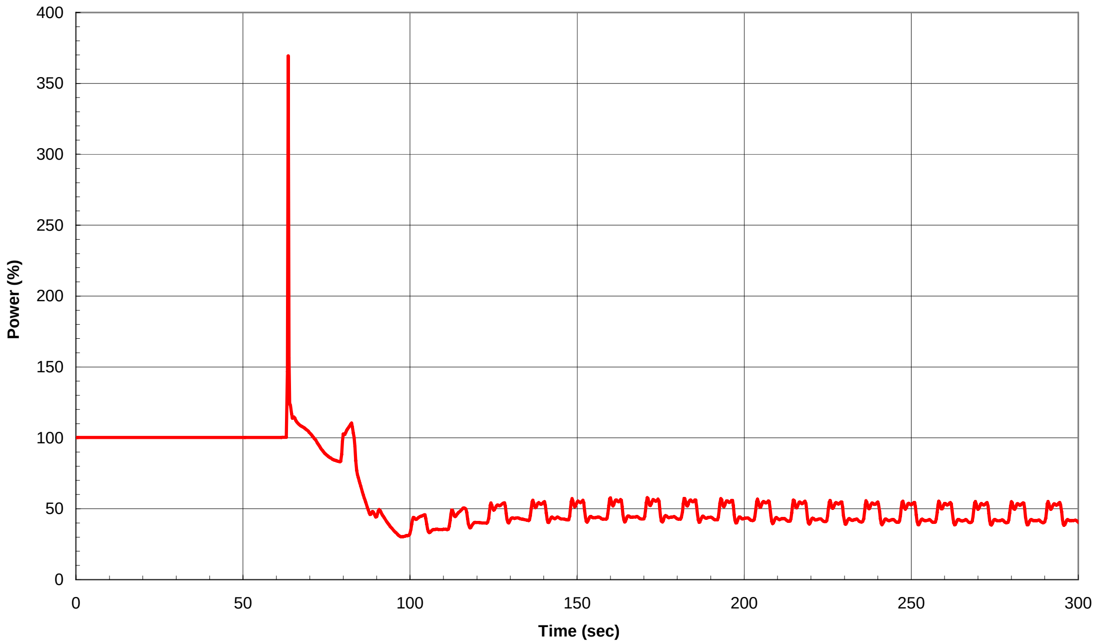

Fig. 5 Reactor power for the MSIV-ATWS transient.

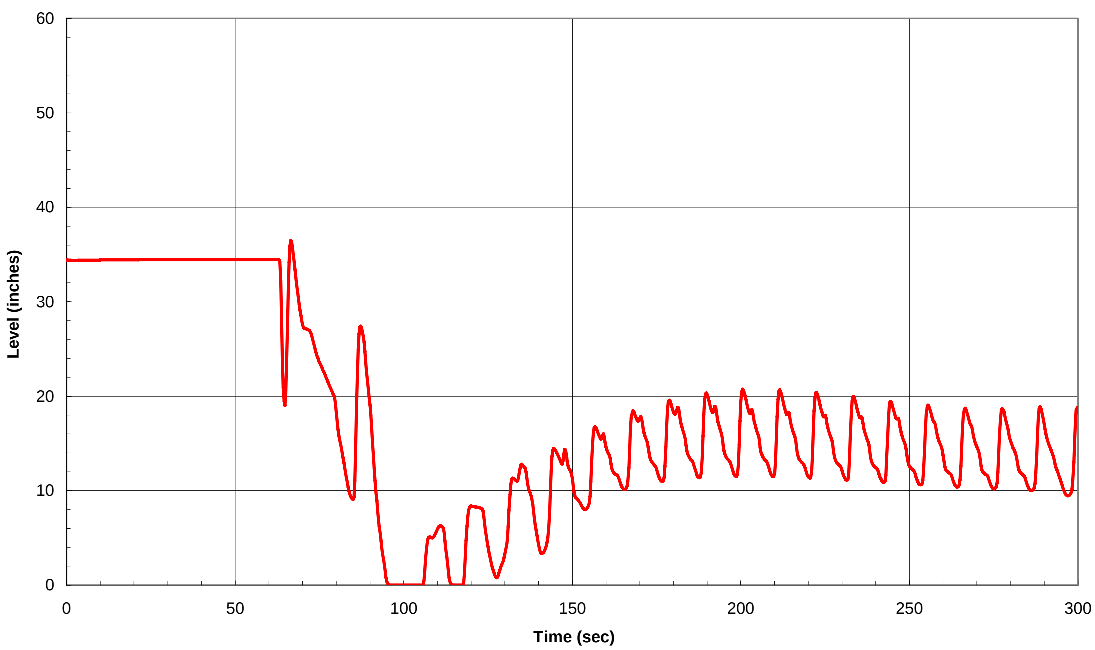

Fig. 6 Reactor water level for the MSIV-ATWS transient

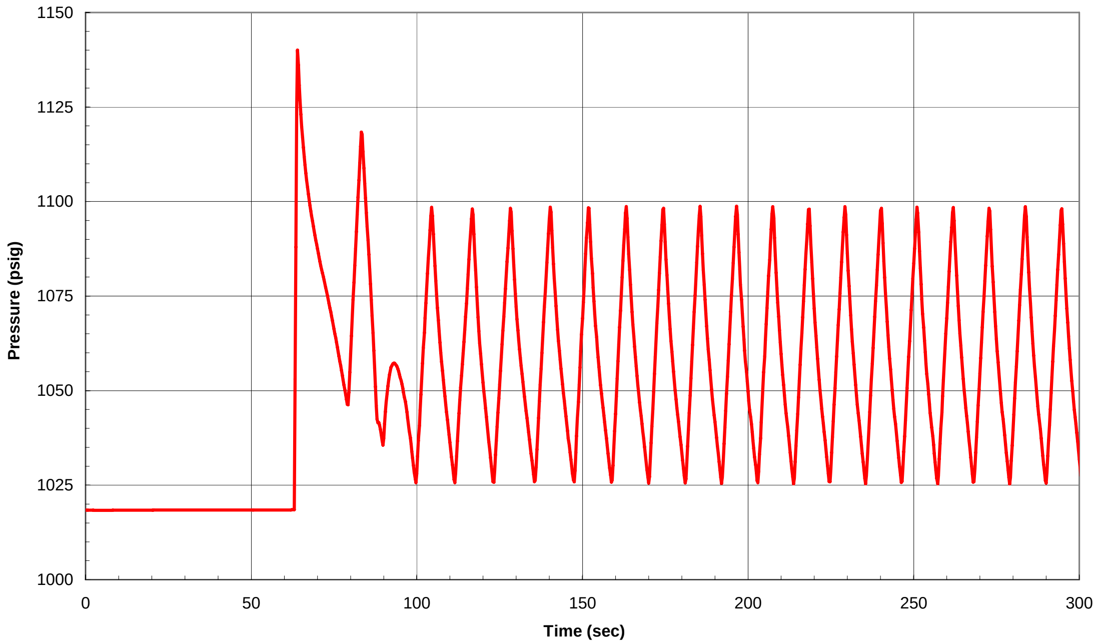

Fig. 7 Reactor pressure for the MSIV-ATWS transient

Overall, the response of the S3K/RELAP5 coupled model is consistent with expectations. The post runback power level is lower than the single recirculation pump trip value (45% versus 63%) because the core flow is lower (37 Mlbm/hr versus 60 Mlbm/hr). These comparisons give the analyst some confidence that the S3K/RELAP5 coupled model can be used for more severe transients than have occurred at the plant.

6. CONCLUSIONS

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. This coupling of S3K and RELAP5-3D provides a tool for best-estimate calculations in support of safety analyses, PRA support[6], training simulator benchmarking[7], and just-in-time analysis of plant transients. The coupled 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.

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