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SIMULATE-3K/RELAP5-3D, A COUPLED CODE SYSTEM

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INTRODUCTION

The best-estimate neutron kinetics code, SIMULATE-3K (S3K),[1] and the best-estimate nuclear systems analyses code, RELAP5-3D,[2] have been interfaced to provide a best-estimate coupled code system for performing plant transient calculations with reactivity feedback from a detailed core model. New reactor and fuel designs require more detailed methods for assessing the behavior of the core and NSSS during ATWS, ejected or dropped control rods, and Steam-line breaks.

S3K/RELAP5-3D COUPLED CODE

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. 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.

Typically, the RELAP5-3D core thermal-hydraulics nodalization does not include a flow channel for each fuel assembly, but from 1 to 12 effective flow channels. These flow channels comprise from a few up to several hundred 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. A brief description of the linkage between RELAP5-3D and S3K follows.

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 S3K 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 core bypass channel must also have the same number of axial subdivisions as the active core region.

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 is described below.

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 only approximate the radial power distribution (especially for the hot assemblies or controlled assemblies) unless a large number (>100) of RELAP5-3D channels are modeled. The resultant power distribution is then collapsed back to the coarse core nodalization used by the RELAP5-3D model.

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. The resultant power distribution is then collapsed back to the coarse core nodalization used by the RELAP5-3D model.

The additional inputs required by RELAP5-3D to activate this linkage are the definitions of the channels, S3K input and output filenames, initial power level, and trip signals. The additional input required by S3K is the channel assignments and the coupling option selection.

S3K also has a pin-by-pin fuel temperature calculation that may be activated for any of the coupling options. This option permits the calculation of temperature, enthalpy, DNB, CHF, etc. for every fuel pin in a set of user selected fuel assemblies. This module also compares the calculated fuel pin parameters to user specified criteria, i.e. melting temperature, and will provide information on the number of fuel pins that exceed the specified criteria. These safety criteria may be dependent on the fuel burnup in each fuel pin.

BENCHMARK CALCULATIONS

The Forsmark 3 plant underwent a transient in 1994 that resulted in a fault in the reactor protection system. This fault cased a pump runback to minimum speed, a failure of the hydraulic scram system, a turbine trip, and a screw insertion of the control rods. The screw insertion takes 4 minutes for the rods to completely insert. The operators were able to perform a manual hydraulic scram of a partial scram group (8 control rods) after 112.2 seconds. The following reactor vessel boundary conditions were defined for the model, feedwater flow and temperature, steam dome pressure, and pump speed. In addition, the screw insertion and the partial scram were initiated at the same times that occurred at the plant. The screw insertion started at 1.35 seconds into the transient at a rate of 1.513 cm/sec. The partial scram occurred at 112.2 seconds at a speed of 108.0 cm/sec. The APRM response from the calculation is shown in Fig. 1.

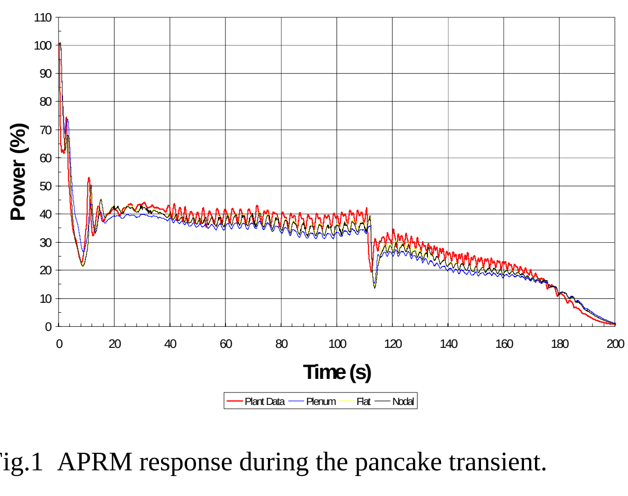

Fig.1 APRM response during the pancake transient.

SUMMARY

This coupling of S3K and RELAP5-3D 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 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.

REFERENCES

1. “SIMULATE-3K Models and Methodology,” SSP-98/13 Rev. 3, Studsvik Scandpower, (2006).
2 “RELAP5-3D Code Manual,” INEEL-EXT-98-00834, Revision 2.3, Idaho National Laboratory (2005).