Methodology for Coupling Physics and Systems Codes for Accident Simulations

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Abstract

Westinghouse has developed the RAVE™ three-dimensional core transient analysis system for PWR non-LOCA accident analysis. Methodologies have been developed for the use of this code and approved by the US NRC. This paper discusses the linkages between the codes for nuclear and thermal-hydraulic analysis of the core and the system analysis code along with the associated software and methodology decisions.

KEYWORDS: Transient Analysis, Coupled Codes, PWR

Introduction

Westinghouse has developed the RAVE™ three-dimensional core transient analysis system for Pressurized Water Reactor (PWR) non-LOCA accident analysis. Methodologies have been developed for the use of this code and approved by the US NRC. A summary of the usage of the system is presented in Reference 1 and its use to analyze the Organization for Economic Cooperation and Development (OECD) main steamline break benchmark is summarized in Reference 2. This paper will discuss the linkages between the codes and the associated software and methodology decisions.

A key requirement for the code was the ability to accurately analyze the transients in a reasonable amount of computer time. But the different parts of the system have different time constants and the coupling needs to take this into account. There are three main calculations that are needed to be coupled:

- The nuclear analysis of the core
- The thermal-hydraulic analysis of the fuel rods and coolant within the core
- The thermal-hydraulic analysis of the primary (outside of the core) and secondary systems.

Core Nuclear Analysis

The traditional time constant for the nuclear core is very short since the prompt neutron lifetime is so small. However, the Stiffness Confinement Method (SCM) has been implemented in the SPNOVA code which allows the time step size to be significantly larger in the nuclear calculation. [3] This is very important since a detailed 3-D analysis of the core is a time-consuming calculation and the number of time steps used needs to be minimized.

In discretizing the time dependence of coupled differential equations, the time step size tends to be controlled by the shortest time constant of the equations. For neutron kinetics equations, the prompt neutron lifetime is smaller by three of more orders of magnitude than the delayed neutron precursor lifetime. Thus the time step size used in conventional

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methods of numerically solving neutron kinetics equations must be very small for the solutions to be accurate; this is known as the stiffness problem. SCM, devised to alleviate the problem, was first introduced for point kinetics [4], and later generalized to space-time kinetics. [5] The method is based on the observation that, for a reactivity insertion, despite the very sharp initial change in the prompt neutron response, the delayed neutron precursor response is nevertheless very smooth throughout. As explained in Reference 5, this phenomenon can be understood in terms of the basic physical constants. It reflects the fact that the precursors are physically insensitive to the stiffness. Therefore, it should be possible to decouple the stiffness from the precursor balance equations. SCM implements this idea through the use of dynamic frequencies, defined as the instantaneous rate of the fractional change of population. The use of dynamic frequency provides the same effect as the classical prompt jump approximation without introducing any approximation.

Core Thermal-Hydraulic Analysis

The thermal-hydraulic analysis of the core has three different time constants. The first is associated with the heat generation in the fuel rod which is essentially instantaneous with the power change, and through Doppler feedback has an immediate impact on the physics calculation. The second is the heat transfer from the pellet to the coolant which is typically on the order of seconds. The third is the fluid flow vertically up the core coolant channels which typically takes about a second to flow through the core. The water density changes up the channel are important for the physics calculation.

Depending on the transients, the thermal-hydraulic core response is more of a steady-state condition than a transient property. Only for the extremely rapid events, such as a rapid rod bank withdrawal or the PWR rod ejection, do the time constants have a major significance.

Another factor to consider for the thermal-hydraulic analysis of the core is the differentiation between the average rod and the peak rod for each node. The nuclear analysis code needs node average parameters to calculate the resonance effective temperature and water density for the nuclear cross section adjustment. However, the evaluation of the transient requires analyzing the peak fuel pins, not the average pin. The peak fuel pin is evaluated for the heat flux to calculate the Departure from Nucleate Boiling Ratio (DNBR), the fuel centerline temperature for fuel melt projections and fuel enthalpy increase for clad rupture evaluations. The nuclear transient is generally more limiting with lower temperatures because of the negative reactivity coefficients; however, the peak pin is typically more limiting with higher temperatures. Thus, the calculations of the average pin and the peak pin typically utilize different parameters which tend to pessimize the calculations. The average pin information is important for the overall transient profile. However, the peak pin analysis can be performed after the core transient analysis calculation.

The core response to some transients can be very localized, such as PWR rod ejection or zero power steam-line break, as oppose to loss of flow events. It is desirable to use the same core model for all the transients. Thus a detailed core model was chosen which models the core using the same nodal configuration as the neutronic calculation for the average node (a typical 3-loop Westinghouse plant model contains more than 15000 nodes in the core). The hot pin analysis can utilize simpler coupling models to generate just the hot pin parameters.

The Westinghouse RAVE system utilizes the VIPRE-W [6] code for the core thermal-hydraulic analysis. The flexibility of VIPRE allows the input of fuel parameters which can be calibrated to best estimate or conservative fuel temperatures.

System Analysis

The reactor coolant system (RCS) is more complex and has a longer time constant. On the primary side, it typically takes more than 10 seconds for the fluid to complete a circuit. The RCS components, including the reactor vessel, pressurizer, steam generators, reactor coolant pumps and the relief and safety valves are modeled. In addition, the reactor control and protection systems utilizing the excore detector signals from the SPNOVA code, the Engineered Safety Feature Actuation System (ESFAS), safety injection system, charging and letdown flows, and secondary side models, including main/auxiliary feedwater flows and steamline and feedwater line valve actuations are modeled.

The Westinghouse analysis utilizes the RETRAN-02 code [7] for the system analysis with a detailed system nodalization. The size of the core nodes used in the system model affects the time step size used in the calculations due to the Courant limitation. Thus it is necessary to have relatively large nodes representing the nuclear core than the detailed model used for the 3-D core neutronic analysis. The core is actually modeled with non-conducting heat exchangers, with the heat flux supplied by the VIPRE core calculation.

Core Interfaces

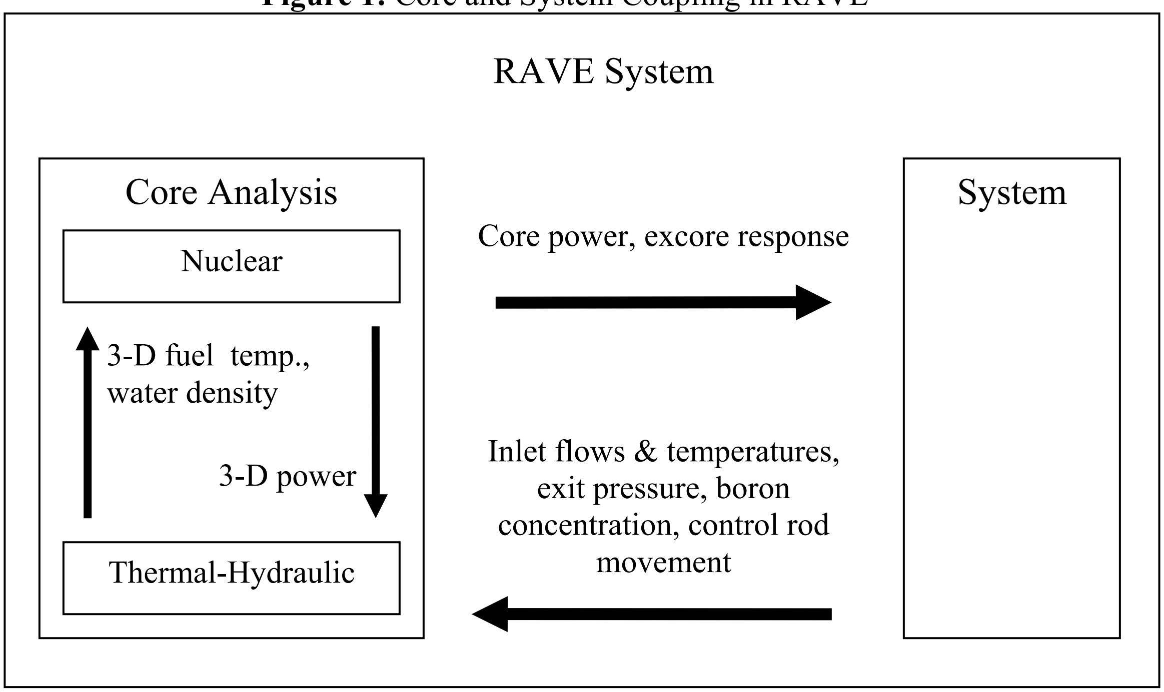
The interfaces between the core physics and core thermal-hydraulics are extensive since the prompt power and decay heat production are input to the thermal-hydraulic model and the effective fuel resonance temperature and moderator density are passed back to the physics calculation for the thermal feedback effect on the cross sections. Because of the close relationship, the physics model and the core thermal-hydraulic model utilize the same geometrical mesh structure. The coupling can utilize an explicit method (beginning of time step values) or an iterative method in the nuclear code to update the cross sections based on the end of step time values. The latter is more accurate, but can significantly increase the computational time. However, the explicit method gives appropriate results with reasonable sizing of the time step length.

The above coupling is for the feedback in the physics calculation and thus represents the average rod in each node. The analysis for the radially averaged peak fuel enthalpy should utilize a model for the hot rods, not an average rod. The average rod thermal-hydraulic analysis impacts the physics transient, so it is tightly coupled. However, the hot rod is calculated based on the statepoints derived from the transient evaluation and can be performed at a later time. In this case, a model for the hot rod is developed and the heat input versus time is provided separate from the core transient. The hot rod model and the average rod models will differ since different conservatisms need to be applied, and in some cases multiple different hot rod models with different conservatisms may be used depending on the parameter of interest.

System and Core Interfaces

The system code provides the required boundary conditions (flow rate, temperature, pressure and soluble boron concentration) and the control rod movement/trip information. The detailed core thermal-hydraulic model provides the heat input into the system and the excore Nuclear Instrumentation System (NIS) detector values. A hybrid coupling is used between the RCS and 3-D core models. In the RCS code, a heat flux boundary is used to model the core, allowing a closed system for analysis which essentially eliminates the stability issues of the coupled code calculations; however, it requires additional checks to confirm the validity of the core outlet conditions against the more detailed core thermal-hydraulic model. The system model has the appropriate pressure drop and the heat input is taken from the detailed core thermal-hydraulic calculation. This allows the system code to be run and converge independent of the core calculation. The core calculation is run using the boundary conditions from the system code at the beginning of the time step. The coupling is summarized in Figure 1 below. One important detail in the coupling occurs when there is a loop imbalance causing one of the core sectors to be significantly different than the others. This should be translated into variable inlet conditions to the assemblies in the core. Thus the 2-D (core inlet and outlet) and 3-D (core nodes) mapping of the parameters become important, particularly the 2-D core inlet temperature distribution (core inlet mixing model) for steamline break transient.

Figure 1: Core and System Coupling in RAVE



This hybrid coupling allows the two calculations to be performed independently over a combined time step. There is no actual need for the codes to utilize the same time step sizes, only that they synchronize for information transfer. In fact, the RCS code typically utilizes a smaller time step size than the core analysis. The effectiveness of this coupling is easily checked by comparing calculations using different time step sizes. If they give the same results, then a sufficiently small time step has been defined which is consistent with the coupling being used.

General Considerations

The code coupling was performed in a manner as to not physically couple the codes. After much experience, Westinghouse has found this to a more effective, especially for software maintenance over time. The individual codes are linked using the Parallel Virtual Machine (PVM) software which allows the individual programs to pass information while executing on the same or different computers.[8] This coupling was also implemented in a manner which allows each of the codes to be executed in a stand-alone mode. Thus the same version of the software can be utilized in a coupled or stand-alone mode. Thus if any one of the codes is updated, then the modification is automatically incorporated into the RAVE coupled code system.

Conclusion

In summary, the Westinghouse RAVE code utilizes a hybrid-coupled set of core and RCS codes to perform a coupled core-system analysis. Methodologies have been developed for these analyses and have received US NRC approval. The system and the methodologies have performed well, and efficiently provide realistic transient simulations which can be utilized to gain DNB margin.

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