Analyses of Trace-parcs Coupling Capability

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ABSTRACT

In the nuclear reactor analysis studies, the best-estimate thermal-hydraulic and three-dimensional neutron kinetic codes have given a valuable contribution in improving design, optimization and safety analyses. In order to model the interaction between neutron kinetic and thermal hydraulic phenomena and its spatial distribution in a more accurate way, advanced techniques have been developed by coupling best estimate thermal hydraulic and three-dimensional neutron kinetic codes. These techniques allow to reduce uncertainties and, giving a more realistic analyses of transient phenomena, to carried out a more realistic study of the accident sequence. Therefore, three-dimensional coupling analyses permit a better estimation of the safety margins and can increase the operational flexibility of a Nuclear Power Plant (NPP).

In the last years, in order to analyze the thermal hydraulic behavior of Light Water Reactors (LWR), the USNRC is developing an advanced best estimate thermal hydraulic system code called TRAC/RELAP Advanced Computational Engine (TRACE) that can be run coupled with the 3D reactor kinetics Purdue Advanced Reactor Core Simulator (PARCS) code.

This paper presents a study of the coupling capability of the TRACE and PARCS codes by analyzing the “Main Steam Line Break (MSLB) benchmark problem”, consisting in a double-ended MSLB accident assumed to occur in the Babcock and Wilcox Three Mile Island Unit 1. The analyses of the TRACE/PARCS calculated data show that the codes are able to predict the expected phenomena typical of this transient and the related thermal hydraulic neutronic feedback. The graphical capability of SNAP has been useful to have a direct visualization of selected calculated data.

1 INTRODUCTION

In the nuclear reactor analysis studies, the best estimate thermal hydraulic and three-dimensional neutron kinetic codes have given a valuable contribution in improving design, optimization and safety analyses. In the last years the technologic development has provided evolved computers that, coupled with evolved computational methods, have increased the possibility of producing more realistic simulations of the complex phenomena taking place in a nuclear reactor, permitting a better analyses of the multidimensional phenomena as well. In
this framework, in order to model the interaction between neutron kinetic and thermal hydraulic phenomena and their spatial distribution, taking place during steady and transient condition, in a more accurate way, advanced techniques have been developed by coupling best estimate thermal hydraulic and three-dimensional neutron kinetic codes. The resulting coupled calculated data give the possibility of developing more accurate studies and reduce the related calculation uncertainties. This produces more realistic analyses of the main phenomena, typical of a nuclear reactor, permitting a more complete and a better study of the transient sequence following a postulated accident. Typical transients that could require a coupled analysis are characterized by strong interaction between the core neutron kinetic and the thermal hydraulic of the reactor cooling loop and by asymmetric power excursion. In this framework, the use of coupled best estimate thermal hydraulic and three-dimensional neutron kinetic codes can give a valuable contribution in improving design, optimization and safety analyses, permitting a better estimation of the safety margins and can increase the operational flexibility of a NPP [1-7].

In the last years, in order to analyze the thermal hydraulic behavior of LWRs, the USNRC is developing the advanced best estimate thermal hydraulic system code TRACE that can be run coupled with the 3D reactor kinetics code PARCS.

This paper presents an analysis of the capability of the coupled TRACE and PARCS codes to simulate transients characterized by a strong and space-dependent thermal hydraulic and neutronic interaction. The work also tests the behaviour of their thermal hydraulic/neutronic code coupling by analyzing the “MSLB benchmark problem” consisting in a double-ended break in one of the two steam lines upstream the main isolation valve in the Three Mile Island Unit 1.

2 REFERENCE REACTOR AND MSLB SCENARIO

2.1 Reference Reactor

In order to test and evaluate the best estimate simulating capability of coupled codes, three different international benchmarks have been sponsored by Organization for Economic Co-Operation and Development (OECD)/Nuclear Energy Agency (NEA) in the past years: “PWR Main Steam Line Break in TMI-1”; “BWR Turbine Trip in Peach Bottom”; “VVER1000 coolant transient” [1].

In this paper, in order to analyze the best estimate capability of TRACE/PARCS coupled codes to simulate transients, characterized by a strong thermal hydraulic and neutronic interaction and test their thermal hydraulic/neutronic code coupling, the simulation of the OECD/NEA PWR MSLB benchmark problem is proposed [1].

The reference reactor is the Three Mile Island unit 1, a typical Babcock and Wilcox 800 MWe PWR. The reactor consists of two loops; each loop is composed by a single Hot Leg (HL) two Cold Legs (CL) and a Once Through Heat Exchanges (OTSG) characterized by superheated steam at its outlet [1].

2.2 Phenomena of interest characterizing the MSLB transient

This paper is focused on the analyses, by using the TRACE/PARCS coupled codes, of a OECD/NEA PWR MSLB benchmark problem consisting in a double ended break in one of the two steam lines upstream the main isolation valve.

Following the double ended guillotine break in the main steam line, chocked flow phenomena takes place in each of the two main steam line broken parts. The OTSG depressurization, characterizing this part of the transient, causes an increase of the steam flow
through this OTSG that, with the contextual blow down, determines an increase of the OTSG removed power and a consequent generation of a colder primary coolant plug. This plug causes a positive reactivity peak in the core. An high neutron flux signal scram takes place. The turbine trip, together with the closure of the turbine stop valves and of the Feed Water (FW) control valve, takes place as well. The HPIS injection could be caused by a Reactor Coolant System (RCS) low pressure signal due to the large overcooling of the primary system. Since this transient is characterized by an asymmetric core cooling, a best estimate coupled thermal hydraulic and neutron kinetic analyses is necessary to have a realistic simulation of the interaction between thermal hydraulic and neutronic.

As the most important concern of this kind of transient is an increase of power in the second part of the transient after the scram (the so called “return to power”), different conservative assumptions have been done in the simulation [1-9].

3 CODE APPLICATION

3.1 TRACE Code

In order to analyze the thermal hydraulic behavior of LWR reactors, USNRC has maintained four codes, the RAMONA, the RELAP5, the TRAC-B and the TRAC-P. In the last years, USNRC is also developing an advanced best estimate thermal hydraulic system code, by merging the capabilities of the previous codes into a single code. This new code is called TRACE, and is a component-oriented code designed to perform best estimate analyses for LWRs. In particular, this code is developed to simulate operational transients, LOCAs, other transients typical of the LWR and to model the thermal hydraulic phenomena taking place in the experimental facilities reproducing the behaviour of reactor systems.

TRACE is a finite volumes, two-phase flow code, with 3D capability, allowing the user to model heat structures and control systems that interact with the component models. It can be run coupled with the 3D reactor kinetics code PARCS [10,11].

3.2 Symbolic Nuclear Analysis Package (SNAP)

TRACE can be used together with an user-friendly front end, Symbolic Nuclear Analysis Package (SNAP), able to support the user in the development and visualization of the nodalization, to show a direct visualization of selected calculated data, and accepts existing RELAP5 and TRAC-P input. The TRACE/SNAP architecture is shown in Fig. 1.

SNAP is a suite of integrated applications including a “Model Editor”, “Job Status”, the “Configuration Tool” client applications and a “Calculation Server”. In particular, the “Model Editor” is used for the nodalization development and visualization and for the visualization of selected calculated data by using its animation model capabilities. The codes currently supported in SNAP are CONTAIN, COBRA, FRAPCON-3, MELCOR, PARCS, RELAP5 and TRACE [11,12].

3.3 PARCS Code

PARCS code is a three-dimensional core simulator solving steady-state and transient problems. The two-group neutron diffusion equations or the discrete ordinate transport equation are implemented in the code to predict the behaviour of the reactor following a perturbation. The analyses that it is possible to perform are “eigenvalue calculation”, “transient calculation”, “xenon/samarium calculation”, “decay heat calculation”, “pin power calculation”, “adjoint calculation”. It can be run in “stand-alone” mode or coupled to the RELAP5 and TRACE code [13-17].
3.4 TRACE/PARCS coupling

A best estimate TRACE/PARCS coupling calculation is achieved by the inter-process communication protocol, PVM. The TRACE and PARCS processes are loaded in parallel and the PARCS process transfers the nodal power data to the TRACE process that send the fuel and coolant temperature and density to the PARCS process. In the 2004 PARCS has been merged into TRACE [13,14,15]. The TRACE V5 Patch 1 and the merged PARCS V2.7 version are used in this paper [10].

3.5 TRACE/PARCS Nodalization description

The thermal hydraulic and neutronic nodalization of the reference circuit (Fig. 2a), made using SNAP, has been developed starting by the Babcock and Wilcox PWR TM1 TRACE/PARCS coupled model distributed by USNRC, in the Code Application and Maintenance Program (CAMP) framework. As can be seen, the two loops of the reactor are modelled separately; for each loop the HL, the two CLs and the OTSG are modelled. The PRZ and the related surge line are modelled as well. The two main steam lines for each OTSG are modeled separately for the broken SG (OTSG A) and are modeled lumped for the intact SG (OTSG B).

The vessel of the reactor is modelled by using the “3D vessel component”, available in TRACE code, and 15 axial cells, 5 radial rings and 6 azimuthal sectors have been used to nodalize it.

Radially, the first three vessel regions model the primary fluid flowing through the active core, the fourth region models the primary fluid flowing through the barrel-baffle region, whilst the last region models the primary fluid flowing through the downcomer of the reactor. Axially, the first three cells model the lower part of the reactor, the other 6 cells model the active core region, and the other 6 cells model the upper part of the reactor. The active heat structure are modelled with 18 average heat structures coupled with a power component.

The PARCS nodalization, simulating the core of the reactor, is axially divided into 28 cells, where the first and the last represent the reflector of the reactor, while the other represent the active core. For each axial cell, the core is radially divided into 177 homogeneous cells simulating each fuel element; radially, the outer cells model the reflector and are coupled with the TRACE hydraulic cells where the by-pass flow is hydraulically modelled. Fig. 2b shows, using SNAP, the radial thermal hydraulic volumes and neutronic nodes mapping.
3.6 Analyses of the calculated data

According to the MSLB case, the double-ended guillotine break is located in one of the two 24-inch steam lines upstream of the main steam isolation valve. An additional break is considered in the 8-inch cross-connect pipe in order to maximize the break flow. The plant is in hot full power status at the end of the cycle.

At the Start Of the Transient (SOT), the breaks open and the codes predict the expected SG blowdown and the consequent broken secondary side depressurization, Fig. 3a. This way, following the MSLB benchmark problem, it is assumed that the FW regulating valve of the broken OTSG fails in open position, the OTSG depressurization causes an increase of the FW that, with the contextual SG blowdown, determines an increase of the OTSG removing power, Fig. 3b, and a consequent generation of a colder primary coolant plug. Since the main coolant pumps are supposed in operation, maximizing the primary/secondary heat transfer, the cold plug is pumped in the core causing a positive reactivity peak.

An high power signal scram takes place. Though it is conservatively assumed that the scram rod insertion is not completed because of one stuck withdrawn rod, the core is shutdown. After the scram a rapid decrease of the core power takes place due to the negative
reactivity insertion (Fig. 4a). The OTSG continues to cool the coupled loop; this phenomenon is incremented by the HPIS injection. This cooling causes a second power increase that will terminate when the OTSG loses its cooling capability.

During this simulation, a moderate so called “return to power” is predicted by the coupled codes (Fig. 4a), thought a re-criticality is not observed. One reason of the core power increase during the subcriticality period is the significant increase of the contribution of the precursors, due to the first power peak, to the power that can overwhelm the core subcriticality causing a moderate and temporary increase of power [8]. The severe power distortions, that affect the fuel assemblies located in the colder region of the core and near the location of the stuck rod, give a contribution as well. Fig. 4b shows the different reactivity contributions due to the control rod (CR), Doppler effect (Doppler), reactor coolant density (DENSITY) and their sum (TOTAL).

![Figure 4: a) Reactor power behavior; b) CR, Density, Doppler and TOTAL reactivity behavior.](image)

Figs. 5, 6a and b show some of the SNAP visualization calculated data capabilities. In particular, Fig. 5 shows the fluid condition of the plant at about 76 s after the SOT. It is easy to see the colder primary side loop A fluid condition.

![Figure 5: Fluid condition of the plant at about 76 s after the SOT.](image)
Considering the asymmetric cooling of the core, Fig. 6a shows the asymmetric fluid condition in the vessel at all its axial levels. It is easy to see the asymmetric behavior predicted by the coupled codes at the different elevations of the vessel nodalization. Fig. 6b shows the moderator temperature at the core exit. From this figure, it is easy to see the asymmetric moderator temperature associated with each fuel element of the neutronic nodalization.

![Image](image_url)

**Figure 6:** a) Fluid condition of the vessel at about 76 s after the SOT at all the axial cells; b) Temperature of the moderator associated with each fuel node at the core outlet at about 76 s.

### 4 CONCLUSIONS

In the last years, in order to analyze the thermal hydraulic behaviour of LWRs, the USNRC is developing the advanced best estimate thermal hydraulic system code TRACE that can be run coupled with the 3D reactor kinetics code PARCS in order to model the interaction between neutron kinetic and thermal hydraulic phenomena and its spatial distribution. This allows a more realistic study of the accident sequence permitting a better estimation of the safety margins and increasing the operational flexibility of a NPP. In order to analyze the capacity of TRACE/PARCS coupled codes to simulate transients characterized by a strong thermal hydraulic and neutronic interaction and test their thermal hydraulic/neutronic coupling, the best estimate coupled simulation of the OECD/NEA PWR MSLB benchmark problem has been here proposed. The results of the calculated data show that the expected phenomena typical of this kind of transient are reasonably predicted by the codes. The asymmetric behaviour, related to the asymmetric loop cooling, is reproduced and the so called “return to power”, that is the major concern related with this kind of transient, is predicted as well though a re-criticality is not observed. SNAP shows to be a mature tool for a code user in the development and visualization of the code model and in order to have an immediate visualization of selected calculated data, showing the time evolution through animation model as well, through its graphical interface.

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**ABBREVIATION**

BWR  Boiling Water Reactor  
CAMP  Code Application and Maintenance Program  
CL  Cold Leg  
HL  Hot Leg  
HPI   High Pressure Injection System  
IAEA  International Atomic Energy Agency  
LWR  Light Water Reactor  
MSLB  Main Steam Line Break  
NEA  Nuclear Energy Agency  
NPP  Nuclear Power Plant  
OECD  Organization for Economic Cooperation and Development  
OTSG  Once Through Steam Generator  
PARCS  Purdue Advanced Reactor Core Simulator  
PWR  Pressurized Water Reactor  
RELAP  Reactor Excursion and Leak Analysis Program  
SNAP  Symbolic Nuclear Analysis Package  
SOT  Start Of the Transient  
TRAC  Transient Reactor Analysis Code  
TRACE  TRAC/RELAP Advanced Computational Engine  
USNRC  United States Nuclear Regulatory Commission  
VVER  Russian Pressurized Water Type Reactor  

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