Application of the Coupled RELAP5/PANTHER Codes for PWR Steam Line Break Accident Analysis

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ABSTRACT

A dynamic coupling between the existing 1-dimensional thermal-hydraulics system code RELAP5 and the 3-dimensional neutronics code PANTHER is applied via the transient analysis code linkage program TALINK. An interface between PANTHER and the subchannel thermal-hydraulic analysis code COBRA 3C allows direct evaluation of the Departure from Nucleate Boiling Ratio in parallel with the coupled PANTHER/RELAP5 simulation.

The coupled codes are applied to develop a Final Safety Analysis Report (FSAR) accident analysis methodology for the major Steam Line Break (SLB) accident at hot zero power in a typical three-loop pressurised water reactor. In this methodology, the uncertainties related to the plant, core thermal-hydraulic and neutronic parameters are combined in a deterministic bounding approach based on sensitivity studies. The results of coupled thermal-hydraulic and neutronic analysis of SLB are presented and discussed. It is shown that there exists an important margin in the traditional FSAR accident analysis for SLB, which can be attributed by the conservatism's introduced by de-coupling the plant sub-systems.

Introduction

The major Steam Line Break (SLB) accident in a Pressurised Water Reactor (PWR) is classified as a design basis accident, for which a safety analysis must be performed to demonstrate the effectiveness of the reactor protection and safeguard systems in preventing the damage of the fuel rods [2].

In the traditional Final Safety Analysis Report (FSAR) SLB analysis, the core neutronic response is calculated in a de-coupled manner using a core sub-system separated from the Reactor Coolant System (RCS). The core response considered in the thermal-hydraulic analysis is based on parameters computed beforehand considering simplified and enveloping assumptions. The same de-coupled approach is applied to the calculation of the DNBR. The most penalising boundary conditions for each separate sub-system (system thermal-hydraulics, core neutronics and core thermal-hydraulics) can be different. Considering inconsistent bounding boundary conditions is an intrinsic conservatism of any de-coupled analysis. Moreover, extra penalties are generally added to some of the parameters transferred between two sub-systems to cover postulated uncertainties and approximations related to the needed discretisation.

The decoupled approach is merely the consequence of practical requirements rather than resulting from a lack of physical understanding. Indeed, it allows using dedicated simulation codes for both types of accidents. Although this uncoupling is very practical, it introduces large unquantifiable conservatism's in current methodologies which makes the analysed SLB one of the most limiting accident.

The SLB accident in a PWR is characterised by an asymmetric reactor coolant system thermal-hydraulics and a multi-dimensional core neutron kinetics. A best estimate simulation of the complex interactions between the core behaviour and plant dynamics needs the incorporation of a full three-dimensional modelling of the reactor core into the thermal-hydraulic system codes. Recent understanding of the physics and advances in computer technology make it feasible to develop coupled thermal-hydraulics and neutronics analysis codes for such applications [5] [7] [11].

In Tractebel Energy Engineering (TEE) coupled codes package, a dynamic coupling between the existing 1D thermal-hydraulic system code RELAP5/mod2.5 [10] and the 3D neutronic code PANTHER [6] is implemented via the transient analysis code linkage program TALINK [9]. In this approach, both client codes perform their calculations in separate operating system processes, while the TALINK program controls the data transfers between the two processes. Such data transfers are performed using the industrial standard TCP/IP protocols, which can be easily changed and checked. The TEE approach favours the dynamic coupling of existing codes that allows making the best use of confirmed expertise together with a limited validation effort on the coupling process itself while the codes are approved separately [4].The coupling enables the codes to be executed in parallel and provides an integrated plant system thermal-hydraulics and neutron kinetics model.

The coupled PANTHER and RELAP5 codes were applied to simulate the major steam line break accident in a typical three-loop PWR. A method was developed to take account for the uncertainties related to the neutronic and thermal-hydraulic parameters. An interface between PANTHER code and the subchannel thermal-hydraulic analysis code COBRA 3C was developed in order to perform online calculation for Departure from Nucleate Boiling Ratio (DNBR). It has been shown that the model is capable of simulating the coupled core-plant interactions during a SLB transient. Sensitivity studies demonstrated the importance of certain conservatism's in the traditional FSAR accident analysis.

The final objective is to qualify and to use the coupled codes for licensing non-symmetric FSAR accident analysis. A methodology for such applications is under review of the Belgian Safety Authorities.

In this paper, a short description of the SLB accident is first given, followed by a presentation of the main models used by the different codes for SLB simulation. A detailed description of the coupling process between the two codes is to be found in [13]. Finally, the results of coupled thermal-hydraulic and neutronic analysis of SLB are presented and discussed.

Definition of the SLB accident

The main scenario for this main steam line rupture accident assumes a double-ended break to occur in one main steam pipe, upstream of the main steam isolation valves while the reactor is in hot shut-down condition. The steam relief through the break causes a supplementary steam flow which then diminishes as the steam pressure decreases. The increased steam flow causes an increased energy removal from the reactor coolant system and results in a reduction of the primary coolant temperature and pressure.

Due to the negative moderator coefficient, this cooldown causes an increase in core reactivity, in this case a reduction of the shutdown margin. Eventually, the shutdown margin may be lost and lead to core return to power.

When the primary pressure is sufficiently low the Safety Injection (SI) pumps introduce boricated water into the primary circuit. The boron concentration initially increases by steps as the boron front passes trough the whole system before coming back for the SI outlet. This causes oscillations in the nuclear power before the increasing boron concentration induces a steady power decline.

The RELAP5 plant model

All T&H system calculations are performed with RELAP5/mod2.5 [12] that is the version currently approved by the Belgian Safety Authorities.

Our standard plant model for a typical Belgian 17x17, 2775 MWth, 3-loops Westinghouse-type PWR is used but for some specific modifications introduced for use with the coupled codes package.

Split vessel model and mixing

Since the transient is not symmetric, a split vessel is simulated to account for differing conditions in the 3 loops cold outlet. The reactor vessel (downcomer, lower plenum, core and upper plenum) is divided into three regions. Partial mixing is possible in the lower and upper plenum. No mixing across the core region is allowed. The mixing coefficients are chosen to be bounding with respect to experimental results [1]. The junction area's and pressure losses are adapted in order to obtain the desired mixing coefficient in steady-state conditions at the inlet and outlet of the core. During the transient, the mixing ratio may show slight variations according to the upstream and downstream pressures in the connecting volumes.

Safety injection line purge volume

It is assumed that the boron concentration of the pipe situated between the no flow line of the SI pump and the cold leg, i.e. the purge volume, is the same as in the primary system. To avoid an artificially fast boron transport to the core by numerical diffusion, no boron is injected in the system before the integrated flow injected by the SI pump exceeds the purge volume.

Core fuel rod model

Three equivalent fuel rods are simulated, one in each split core region. The physical and geometrical properties are made coherent with their sector averaged equivalent in the PANTHER model.

The PANTHER core model

Neutronic model

The reactor calculations are carried out using the WIMS/PANTHER code package for PWR reactors [6]. For this SLB analysis, the nodal diffusion code PANTHER solves the 3D neutron kinetics transient over a full core representation using 1 node per assembly and 12 uniform axial layers over the active height.

The core characteristics corresponds to the end of a reference cycle (0 ppm) at equilibrium in hot zero power (HZP) all rods in (ARI) less most reactive rod configuration. In this particular case the most reactive rod is located in M:4 position.

The core is radially decomposed into 3 batches (B1, R2 and G3), each one corresponding to a RELAP5 split core region. Sector B1, where the most reactive rod is located, contains 53 fuel assemblies while sector R2 and G3 contain 52 fuel assemblies each. In this arbitrary decomposition, it has been assumed that the sector associated to the affected loop is centred on the most reactive rod, regardless of the true layout of the loop piping in the reactor.

Thermal-hydraulic model

PANTHER built-in functions deal with core thermal-hydraulics in order to determine the thermal feedback's on transient core neutron kinetics. The model considers single thermal-hydraulic channels in isolation where coolant is treated as an equilibrium mixture of liquid and vapour. A sub-cooled nucleate boiling calculation takes into account of localised regions of vapour adjacent to the pins surface and the application of a slip correlation with drift provides an estimate of the difference between the mean vapour and axial velocities. This model is thus limited to low quality fluid mixture.

If this core T&H modelling is appropriate for core T&H feedback calculation in neutron kinetics, it is not suitable for accurate enough DNBR evaluations by not representing the effects on coolant crossflow or turbulent mixing between adjacent channels.

The COBRA 3C core thermal-hydraulics models

An extensive know-how has been built around the use of the COBRA 3C code [8] [12] for steady state and transient subchannel analysis of rod bundle nuclear fuel elements during both boiling and non-boiling conditions, including the effects of cross flow mixing. A dynamic link between PANTHER to COBRA 3C has been developed to insert ratings distributions and profiles from 3D core calculations into equivalent sub-assembly models for DNBR evaluation purposes. That link is only aimed to separate DBNR calculations and does not permit improvement in the PANTHER neutronic feedback calculations by enhanced COBRA core T&H modelling of crossflows.

The COBRA model designed for this subassembly analysis consists in 42 T&H sub-channels built around the T&H channel where peak power occurs. The average assembly powers and the pin power distributions of the hot assembly and its neighbours are merged automatically by the interface into a set of concentric rings that surrounds the hot cells with T&H subchannels of identical properties. The DNBR is evaluated using the W3 critical heat flux correlation.

Taking into account the actual power distribution calculated by the 3D neutronic code for the thermal-hydraulic assessment of fuel assembly performance allows to get an accurate picture of local effects within the reactor core.

Coupling of RELAP5 and PANTHER models for SLB

RELAP5 calculates the system thermal-hydraulic transient while PANTHER calculates the core neutron kinetics transient. The coupling is achieved by exchanging at each time step boundary condition values between RELAP5 and PANTHER through the TALINK interface [9]. That interface controls the data exchange and is capable of simple data processing.

The core inlet conditions are calculated by RELAP5 and retrieved by PANTHER directly from TALINK. Those are:

- enthalpy per sector (H_) : sector enthalpies calculated from internal energy, pressure and density in the core upstream volume;
- boron concentration per sector (B_): boron concentration in core upstream volume;
- mass flow rate per sector (W_) : inlet flow per sector converted into channel-wise mass flow;
- reactor pressure (P) : one uniform pressure since no pressure drop is calculated by PANTHER

They are applied uniformly to the corresponding sector (B1, R2 or G3).

The RELAP5 boundary conditions for each time step are computed by PANTHER and retrieved directly from TALINK :

- nuclear power profile per sector (Pz): the total nuclear power profile in each sector is axially distributed on the heat structures that ensures nuclear heat generation in each split core volume.

Batch averaging operations on all the assemblies within a region allow determining properties to each RELAP5 split core.

A schematic description of the data transfer process is illustrated in Figure 1.

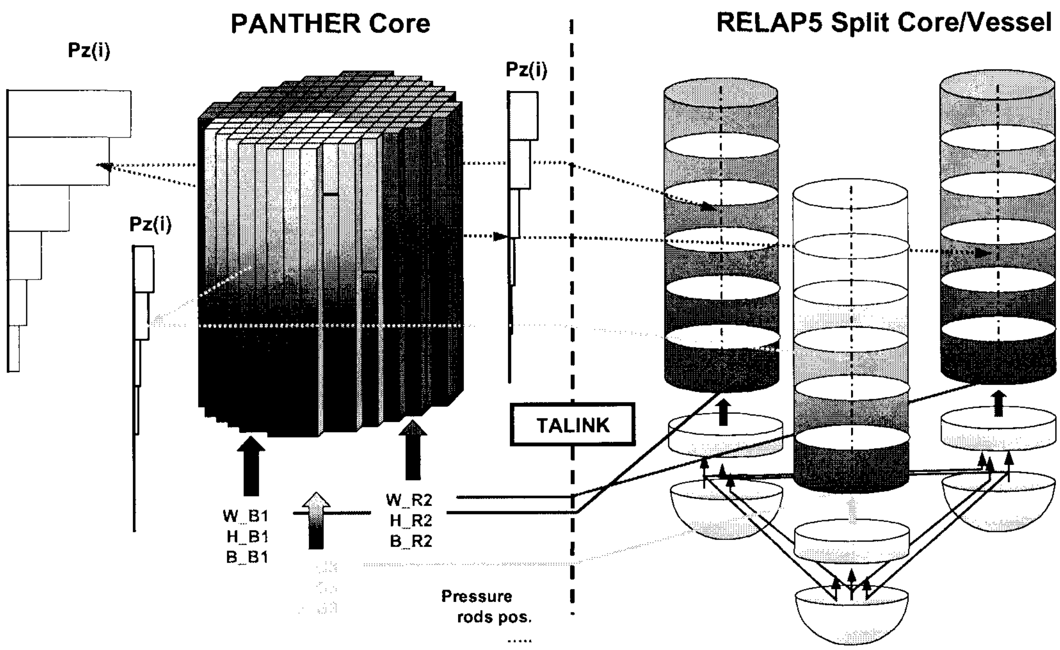


Figure 1 Schematic coupling between core neutronics and system T&H

Treatment of uncertainties in coupled calculations

In presently applied SLB methodologies, using a de-coupled approach, the penalties related to uncertainties or licensing conservatism's are independently applied at each calculation stage (ex. the RCS pressure could be minimised when evaluating the moderator neutronic feedback coefficient, nominal when simulating the system transient and minimised for the DNBR computation). In addition, when data are to be passed from one code to the next one, they are meant to conservatively cover all relevant transient conditions and are consequently penalised.

The use of a coupled codes package obviously improves the overall model accuracy by representing much more realistically interactions between system and core response. For example, certain simplifications are no more needed, such as the assumption of a fixed power profile during the whole transient or a unique set of neutronic coefficients in asymmetric conditions. However, the new methodology we propose does not take advantage of the improved accuracy brought by the new coupled codes package to reduce the various penalties taken into account (as for example a 20% increased moderator effect).

Results and discussion

Reference calculations for this MSLB transient at EOC HZP in ARI-1 configuration are carried out over 200 seconds (break occurs at t = 20 seconds). The data exchange is scheduled by the PANTHER time step that is tailored to the required updating frequency of the thermal-hydraulic inlet boundary conditions for the given neutron kinetics transient. The relative calculation times allow for a shorter RELAP5 time step.

Our method for determining most penalising transient conditions and applicable uncertainties follows a deterministic bounding approach [3] sequentially for 3D neutron kinetics, core thermal-hydraulics and system thermal-hydraulics.

In the first studied case, best estimate nuclear and system parameters are considered except for licensing assumptions such as most reactive stuck rod and single failure, and no return to power is observed. In a second case, the initial shutdown margin is reduced to the minimum value from the technical specifications, and a return to power is observed. This implies that if in actual core loading the shutdown margin is much higher than the minimal value required by the technical specifications, the Steam Line Break should not be a real safety concern.

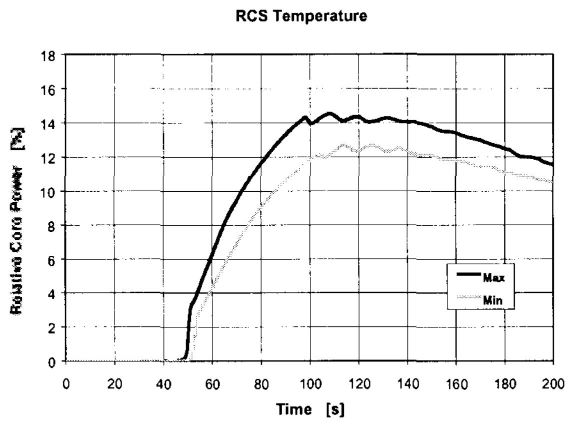
The most relevant sensitivities on neutron kinetics and core T&H cases are presented in [13]; they involve the initial reactivity state (shutdown margin), the moderator density effect and the Doppler power effect.

The most penalising single failure has been determined to be the failing to start of the Safety Injection train in the affected loop. This condition is considered in all described calculations.

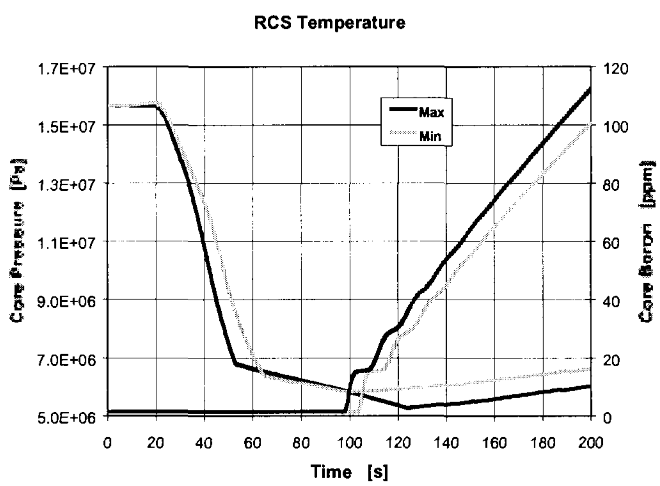
The sensitivity studies on system thermal-hydraulics initial and boundary conditions showed that variations on particular parameters had no significant impact on the transient before the minimum DNBR state-point is obtained. The parameters are the pressuriser location; the Main Feed Water flow rate; the Main Feed Water temperature; the RCS initial pressure; the initial Steam Generator level and the Auxiliary Feed Water flow rate and temperature. Three parameters were determined to have a significant impact on the transient evolution: the initial RCS temperature, the core coolant flow rate and the initial pressuriser level. They are discussed in the following paragraphs.

Initial RCS temperature

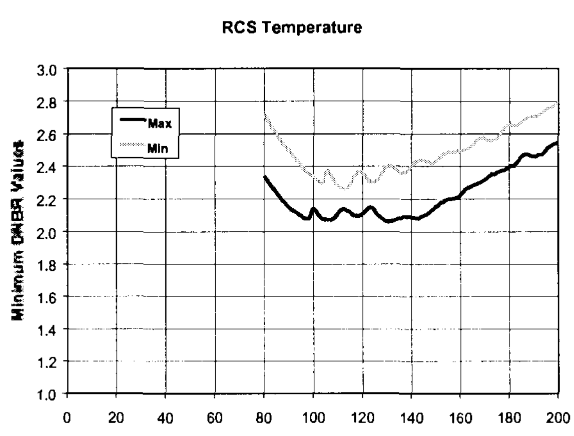
Figure 2 SLB transient for min/max initial RCS temperature



a) relative core power



b) core pressure and boron concentration



c) minimum DNBR

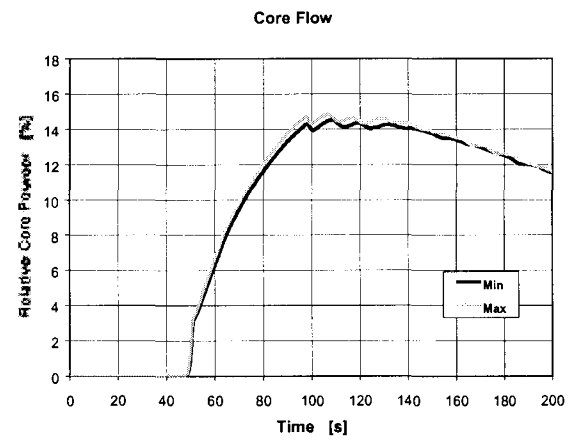
In the transient first phase, before the return to power, it can be seen (Fig. 2b) that a higher initial temperature means a faster depressurisation which is explained by a larger dilatation coefficient and a larger break flow as the initial secondary pressure is higher.

The shutdown margin is lost sooner (Fig. 2a) because the moderator effect increases with temperature while the initial antireactivity is identical in the two cases. The larger neutronic retroaction is also the cause for a steeper slope of the neutronic power evolution.

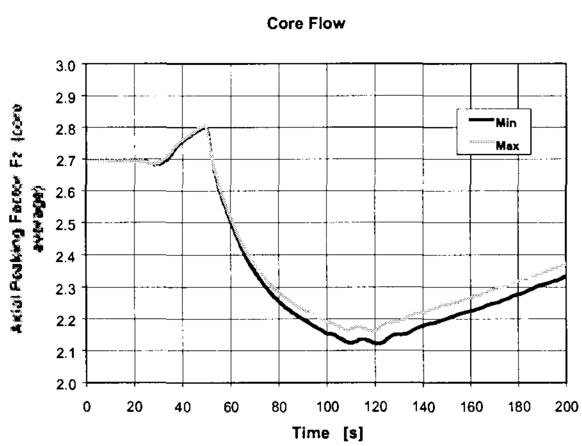
The faster initial pressure drop allows for a sooner boron arrival in the core. However the maximum power reached is significantly higher in the case of a maximum initial temperature. This has a direct penalising effect on the minimum DNBR (Fig. 2c).

Core coolant flow

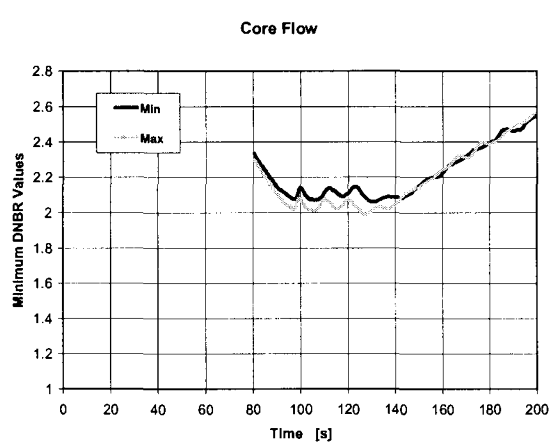
Figure 3 SLB transient for min/max core flow rate



a) relative core power



b) axial peaking factor



c) minimum DNBR

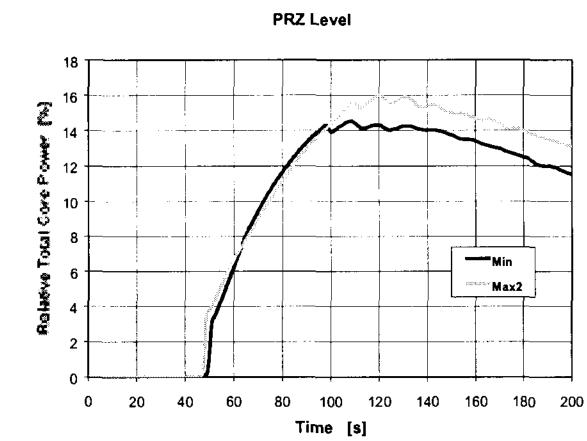
Traditionally, when the licensing parameter is the minimum DNBR, the reactor coolant thermal design flow is used (i.e. minimum). Indeed, everything else being equal, a lower vessel flow at the state point results in a lower DNBRmin.

The results show that a variation in RCS flow does not modify significantly the system thermal-hydraulics transient. Remarkably, no alteration of the primary to secondary heat transfer is observed.

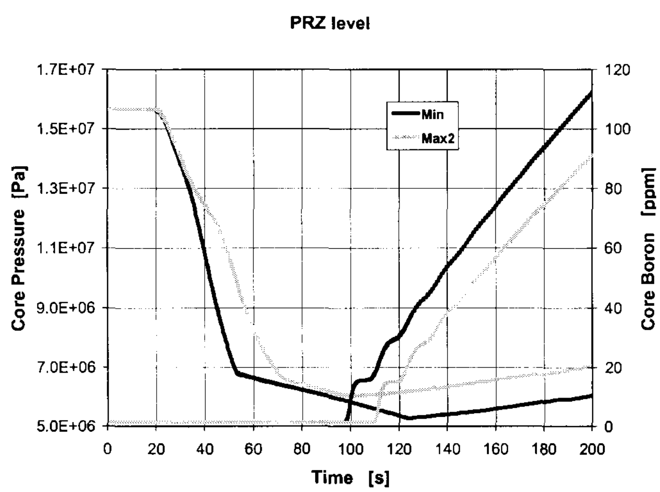
In contrast, a substantial impact is observed on the core response (Fig. 3a) and particularly on the peaking factor (Fig. 3b). As a matter of fact, a higher core flow rate let colder water penetrate further in the core. Through the moderator effect, the modified coolant temperature profile establishes a higher local power. The increased axial peaking factor (Fz) and total power results in a lower minimum DNBR value (Fig. 3c).

Initial pressuriser level

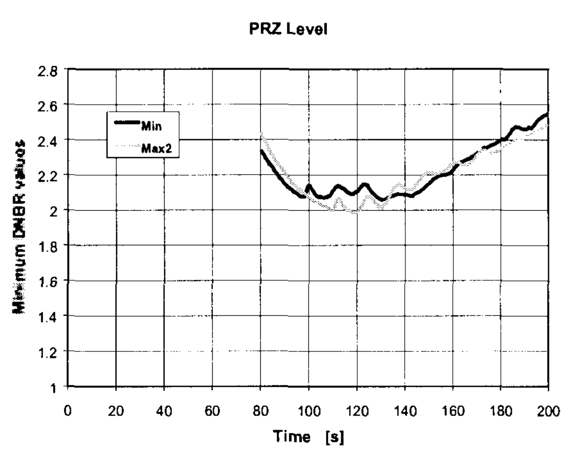
Figure 4 SLB transient for min/max initial pressuriser level



a) relative core power



b) core pressure and boron concentration



c) minimum DNBR

A higher initial pressuriser level increases the primary mass and volume inventory, which results in a slower depressurisation and a delayed safety injection (Fig. 4b) leading to a higher peak power (Fig. 4a). On the other hand, a higher pressure at the minimum DNBR state point is less penalising.

In the presented case, the two competing effects results in the maximum initial pressuriser level being the most penalising condition (Fig. 4c). This conclusion is highly dependent of many system parameters and could vary even for very similar situations. However the total effect on the licensing parameter does not exceed a 5% variation.

Increased licensing margin

Compared to similar simulation using a de-coupled methodology and a point-kinetics neutronic model, important margin to the licensing criteria is observed. As all applicable traditional licensing conservatism's and uncertainties were preserved in the new methodology, the increased licensing margin is credited to the disappearance of penalties introduced to cover discrete data transfer between codes (as the minimum DNBR state-point or axial power profile) or introduced by considering inconsistent penalising conditions across the full system (ex. reactor coolant flow rate).

Conclusions

The main objective of this work is to analyse coupling of best estimate thermal-hydraulic and neutronic codes for major steam line break simulation. The coupled PANTHER and RELAP5 codes were applied to simulate the SLB accident in a typical three-loop PWR. A method was developed to take account for the uncertainties and conservatism's related to the neutronic and thermal-hydraulic parameters. An interface between PANTHER code and the subchannel thermal-hydraulic analysis code COBRA 3C was developed in order to perform online calculation of minimum DNBR. It has been shown that the model is capable of simulating consistently the coupled core-plant interactions during a SLB transient.

The effect of the 3D core neutron kinetics and system thermal-hydraulic coupling with a direct link to DNBR calculations is to introduce consistent interactions between the respective models. Those interactive ties do smooth out the impact of the variation of each parameter separately on the global core/system response, and, hence, on final DNBR value.

It has been thus proven that the current coupled simulation is a robust tool for SLB transient analysis that allows the identification of large margins on DNBR compared to those traditionally observed in de-coupled methodologies. Moreover, as no supplementary hypothesis are to be made as the two codes exchange their boundary conditions, the validity domain of the coupled codes package is defined by the validity domain common to the two codes. As a consequence, the coupled codes package offers an extended versatility matching possible new regulatory requirements.

The final objective is to qualify and to use the coupled codes for licensing non-symmetric FSAR accident analysis. A methodology for such applications is to be submitted for review by the Belgian Safety Authorities.

References

[1] EPRI, Thermal Mixing in the Lower Plenum and Core of a PWR, EPRI NP 3545 Project 1749-1, May 1984

[2] USNRC, Standard Review Plan, 15.1.5, Steam system piping failures inside and outside containment (PWR). NUREG-0800 rev2, July 1981

[3] *Callaert, M. et al.* The Belgatom Methodology for Licensing of Non-LOCA transients by the Means of the Best Estimate Code RELAP5/MOD2. Proc. IAEA Technical Committee Meeting on Comparison of Best Estimate Methods for Judging Design Margins of AWCRs, Lyon, October 1994.

[4] *Delhaye, X. et al.* Coupling of the neutronics code Panther with Relap5 : the Belgatom approach to simulate the core response of non-symmetric reactivity transients. Proc. Relap5 International Users Seminar, College Station, May 1998.

[5] *Gose, G., Downar, T., Ott, K.*, 1998. Analysis of the reactivity during a pressurized water reactor main-steam-line-break transient. Nucl. Tech. 124, 284-290.

[6] *Hutt, P.K., et al.* 1991. The UK Core Performance Code Package. Nucl. Energy 30, 291-298.

[7] *Jackson, C.J., Cacuci, D.G., Finnemann, H.B.*, 1999. Dimensionally adaptative neutron kinetics for multidimensional reactor safety transients – I:New features of RELAP5/PANBOX. Nucl. Sci. Eng. 131, 143-163.

[8] *Masterson, R.E., Wolf, L.*, 1978. COBRA III P: an improved version of COBRA for full core light water reactor analysis. Nucl. Sci. Eng.48, 293-310.

[9] *Page, R.* Description of the Talink code for linkages of codes executing in parallel. C5166/TR/050, November 1996.

[10] *Rascle, P., Duval, C.*, 1999. Thermal-hydraulic neutronic coupling for PWR steam line break accident studies. Proc Ninth International Topical Meeting on Nuclear Reactor Thermal Hydraulics (NURETH-9)

[11] *Rowe, D.S.* COBRA III C: a digital computer program for steady state and transient analysis of rod bundle nuclear fuel elements. Battelle-Pacicfic Northwest Laboratories, 1973.

[12] *Ransom, V.H.* Relap5/Mod2 Code Manual, NUREG/CR-4312, December 1985.

[13] *Schneidesch et al.*, Thermal-hydraulic and neutronic analysis of steam line break accident with coupled Relap5 and Panther, proceedings of BE-2000 conference, Washington DC, October 2000.