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Review

Coupled 3D thermal–hydraulic and neutron kinetic Relap5 model for transient analysis of a 1200-MW$_e$ nuclear PWR plant

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Abstract This paper deals with the assessment of the fully integrated three-dimensional thermal–hydraulic and neutron kinetic modeling capability of the Relap5-3D system code. The interest in integrated system codes is growing in the field of safety analyses, where coupled three-dimensional simulations can significantly improve the level of understanding of spatial core phenomena in accident conditions. For this purpose, an asymmetric design basis accident, namely a main steam line break, has been simulated for a commercial 1200-MW$_e$ pressurized water reactor. The simulation has been performed with a one-dimensional plant model, including all the major primary, secondary and safety systems included in advanced PWR designs, upgraded with a three-dimensional thermal–hydraulic representation of the reactor vessel and a three-dimensional neutron kinetic core model. The analysis demonstrated the code capability to capture 3D core phenomena, not achievable with 1D plant model based on point kinetic.

1 Introduction

The interaction between reactor core thermal–hydraulic (TH) and neutron kinetic (NK) still represents a hurdle in the design optimization of nuclear power plants (NPP). Safety analyses are typically performed with 1D system codes, with a 1D representation of TH plant and core phenomena, and space-independent NK (so-called point kinetic). This provides an average prediction of spatial core phenomena and it is adequate for the simulation of accidents where spatial power distribution remains constant. Conversely, it is less representative for asymmetric accidents where spatial core phenomena are relevant, e.g., a control rod ejection. In that case, the common approach is to decouple core and plant simulations: Dedicated core analyses are performed with 3D NK codes based on lumped TH, and the results input the 1D

Disclaimer: "The Westinghouse AP1000™ has been considered as reference plant, while core neutron kinetic parameters have been derived from available literature data on Three Miles Island NPP (TMI-1). It shall be clarified that the purpose of this report is limited to show 3D TH–NK coupled calculation capabilities under commercial size PWRs representative operating and design conditions, as well as under conservative assumptions made to highlight local phenomena. The results here presented shall not be considered in any way applicable or associated to the AP1000™ plant, for which the safety demonstration publicly available applies.

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plant analyses. These multiple steps introduce uncertainties, translated into conservatisms by designer.

The potential of coupled 3D NK and TH simulations has been exploited in the last decades by several international activities [1]. A large number of transient analyses, as well as code-to-code benchmarks, has been performed, coupling widely used TH codes (e.g., RELAP, ATHLET, TRAC) and NK codes (e.g., PARC, NESTLE, QUABOX). The analyses dealt with relevant 3D accidents for both pressurized water reactors and boling water reactors (PWRs and BWRs), such as main steam line break (MSLB), loss of coolant accident (LOCA), loss of feedwater (LOFW), control rod ejection as well as a series of anticipated transient without scram (MSLB-ATWS, LOFW-ATWS, and LOCA-ATWS). The simulations campaign demonstrated the capability to predict spatial core phenomena, increasing the appeal of TH-NK coupled analyses, with the prospect of moving from conservative to best-estimate simulations as well as of uncertainties reduction and design optimizations.

Coupling can be performed in two different ways: serial integration and parallel processing. Serial integration requires significant code modifications by implementing a NK subroutine directly into the TH system code. Parallel processing allows TH and NK codes to be run separately and exchange data during the calculation; this approach requires few code modifications but suffer numerical instabilities and slow convergence. Nowadays, advanced system codes aim at integrating 3D core TH and NK model, allowing for direct system simulation. Relap5-3D [1], a system code developed at Idaho National Laboratory for transients and accidents analyses of NPP, is moving in that direction providing the fully integrated 3D TH and NK modeling capability (serial integration).

The scope of work presented in this paper is to assess the 3D code features, performing a commercial PWR accident simulation. The reference scenario is the MSLB with the failure of the most reactive control rod, a PWR asymmetric design basis accident, for which 1D core modeling is supposed not to be sufficiently representative. The reference plant is the Westinghouse AP1000™, a 1200-MWe commercial PWR [2].

2 Code description

Relap5-3D [1] is an outgrowth of the 1D code version (Relap5/MOD3) developed at INL for the US Nuclear Regulatory Commission. The code analyses the thermal–hydraulic behavior of PWR under operational and accidental conditions, and its use is widespread in the nuclear community.

The equation set gives a two-fluid system simulation using a nonequilibrium, nonhomogeneous, six-equations representation; energy/conduction heat transfer equations are solved within solids (heat structures). Special models necessary for the simulation of NPP components like pump and separator or specific physical models like critical flow or convection heat transfer are embedded into the code. The code includes NK models, coupled to the TH field equations, permitting simulation of the feedback between the TH and the neutronics inside the reactor core.

The most prominent attribute that characterizes the Relap5-3D code is the fully integrated, multi-dimensional TH and NK modeling capability. This removes most of the restrictions on the applicability of the code to the full range of postulated reactor accidents. The multi-dimensional TH component was developed to more accurately model the multi-dimensional flow behavior exhibited in key components of the plant, such as the reactor vessel. The component defines a one-, two-, or three-dimensional array of volumes and the internal junctions connecting them; the geometry can be either Cartesian or cylindrical.

Table 1 AP1000™ main design and operational data

Plant parameters	
Thermal/electrical power	3415 MW_t/1200 MW_e
Reactor type	PWR 2-loop
RCS operating pressure	$155 \cdot 10^5$ Pa
Cold/hot leg temperature	554/594 K
RCS flow rate	$15 \cdot 10^3$ kg/s
NSSS operating pressure	$59 \cdot 10^5$ Pa
FW/main steam temperature	500/546 K
Steam flow rate per SG	943 kg/s
Core data	
Core dimension (active $L \times D$)	4.3×3 m
fuel assemblies	157
FA type/array	Square/17×17
Fuel Rods	41,448
Fuel	UO_2
^{235}U enrichment	2.35–4.45 w/o
Linear power	18.7 kW/m
Control rods	69 (8 banks)
Reflector assemblies	64

The multi-dimensional NK model in Relap5-3D is based on the NESTLE code [3], which solves the two- or four-group neutron diffusion equations in either Cartesian or hexagonal geometry using the nodal expansion method and the nonlinear iteration technique. Three-, two-, or one-dimensional models may be used. Several different core symmetry options are available including quarter-, half-, and full-core options for Cartesian geometry and 1/6, 1/3, and full-core options for hexagonal geometry. Zero flux, non-reentrant current, reflective, and cyclic boundary conditions are available. The steady-state eigenvalue and time dependent neutron flux problems can be solved.

3 Reference plant description

The Westinghouse Advanced Passive AP1000™ is a 1200-MWe two-loop PWR, based closely on the AP600 design. The main design parameters are reported in Table 1; a sketch of reactor systems is reported in Fig. 1. The thermal energy released in the core by the nuclear process is removed by the reactor coolant system (RCS) and transferred to the nuclear steam supply system (NSSS) through the steam generators (SG). The latter supply steam from the NSSS to the turbo-generator where about one-third of the generated thermal power is converted in mechanical power to the turbine shaft and finally in electric power at the generator. The waste heat is rejected to the final heat sink through the main condenser.

The reactor vessel is the principal component of RCS and contains the heat-generating core and associated supports, controls and cooling channels. The vessel is a 14-m-high cylinder of 4.5 m diameter, with a hemispherical bottom head and flanged and gasketed removable head. Two outlet and four inlet nozzles provide for exit of heated coolant and its return to the vessel fir recirculation through the core. The latter has an active height 4.3 m and diameter

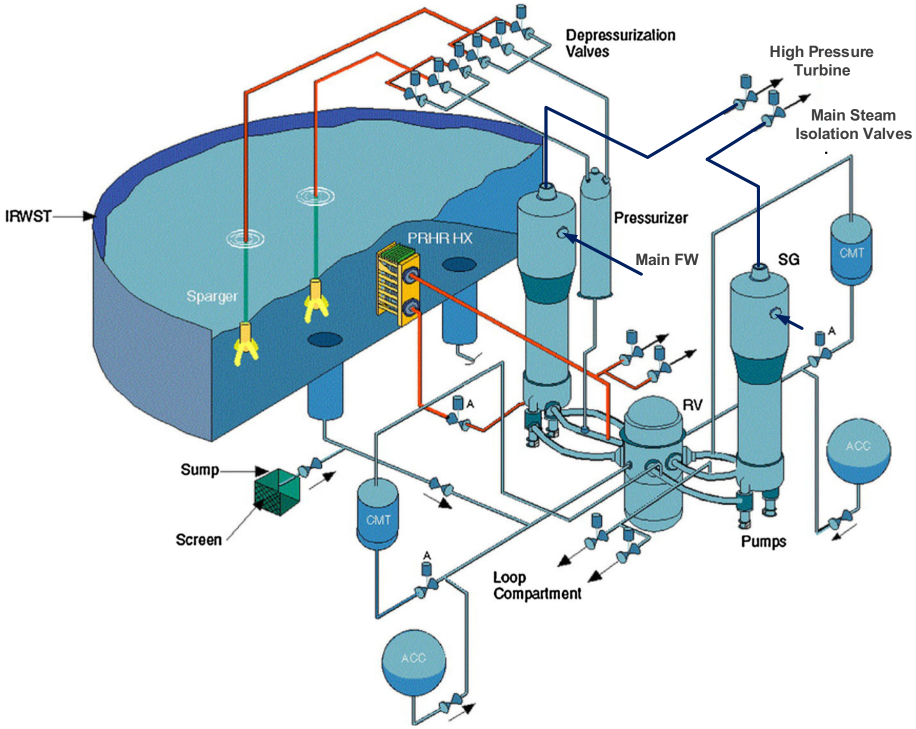

Fig. 1 Sketch of reactor coolant and passive core cooling systems

of 3 m and consists of 157 fuel and 64 external reflector assemblies, and 69 control rods. The fuel assembly is a square and open array of 17 × 17 rods, fuelled with UO_2 with a ^{235}U enrichment within 2.35–4.45 w/o.

The AP1000™ design includes advanced passive safety features and extensive plant simplifications to enhance the safety, construction, operation, and maintenance of the plant. The AP1000™ design provides for multiple levels of defence for accident mitigation, resulting in the extremely low core damage frequency of $2.4{\cdot}10^{-7}$ event per reactor year. The safety systems for AP1000™ include passive single-failure proof safety injection, residual heat removal, and containment cooling. The passive core cooling system (PXS) provides the safety functions of core residual heat removal (RHR), safety injection, and depressurization. Core RHR function is performed by the passive residual heat removal heat exchanger (PRHR HX) which reject the core heat from the RCS to the atmospheric in-containment refuelling water storage tank (IRWST). The PXS uses three passive sources of water to maintain core cooling through safety injection in case of RCS leaks and ruptures, directly connected to two nozzles on the reactor vessel. These injection sources include the core makeup tanks (CMT), the accumulators (ACC), and the IRWST. The PXS provides for depressurization using four stages of the automatic depressurization system (ADS) to permit a relatively slow, controlled RCS pressure reduction. The passive containment cooling system (PCS) provides the safety-related ultimate heat sink for the plant. The steel containment vessel provides the heat transfer surface that removes heat from inside the containment and rejects it to the atmosphere. During an accident, the external air cooling is supplemented by the evaporation of water drained from a tank installed on top of the containment shield building.

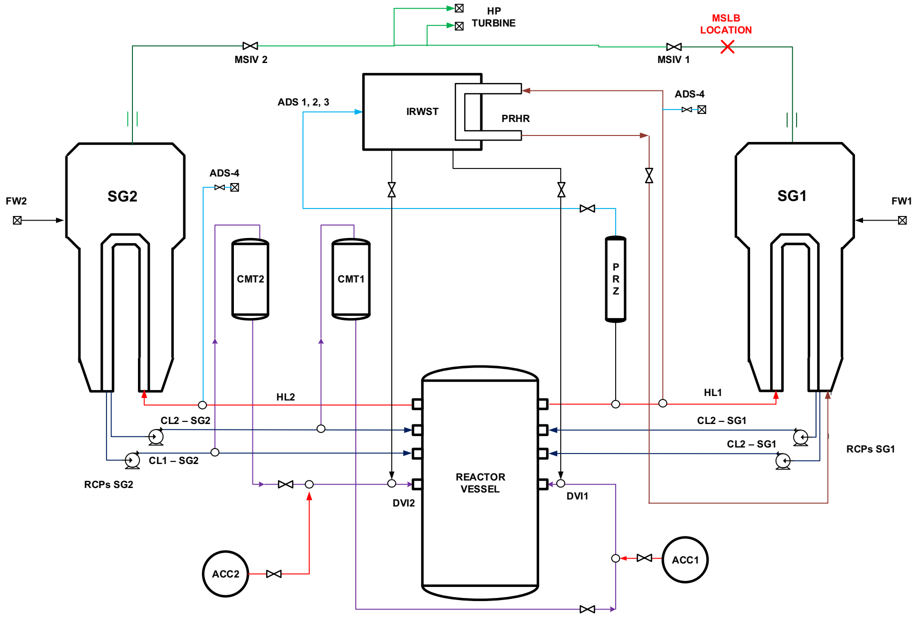

Fig. 2 Sketch of Relap5 model

4 Relap5-3D AP1000™ model

Ansaldo Nucleare developed a 1D Relap5 model of the Westinghouse AP1000™ based on point kinetic, including the Reactor Coolant System, the secondary system (FW line and HP turbine simulated as boundary condition) and the passive safety systems discussed in Sect. 3 (PCS excluded): the passive decay heat removal (PRHR) and passive safety injection systems (CMT, ACC, IRWST and ADS).

Engineered safety feature actuations are triggered by the plant safeguards actuation (S) signal, generated by specific initiating conditions such as low steam line or pressurizer (PRZ) pressure. The S signal triggers the CMT actuation, the reactor coolant pumps (RCP) trip and, on secondary side, the MWF line isolation and—in case of low pressure—the main steam line isolation. The CMT actuation calls the reactor and turbine trip and the actuation of PRHR. The ACC and IRWST subsequent injection is regulated by RCS pressure, respectively, at $49\bullet10^5$ Pa and containment pressure. The CMT low-level signal triggers the staged ADS valves actuation. (The first three stages are connected to the PRZ and discharge into the IRWST, and the fourth stage paths are connected to the hot legs of the RCS and discharge to containment.) The double-ended main steam line break (MSLB) has been simulated by the quick actuation of three fictitious valves: the closure of steam line internal connection (isolating the two sides of the break) and the opening of two valves, connecting the two sides of isolated steam line (two sides of the break) to boundary conditions, simulating the containment pressure. The model has been upgraded with a 3D reactor pressure vessel and NK model, to assess the multi-dimensional features of Relap5-3D. A sketch of plant model is reported in Fig. 2; reactor vessel model is reported in Fig. 3. The model is based on Relap5-3D Rev 2.4 [4] code version.

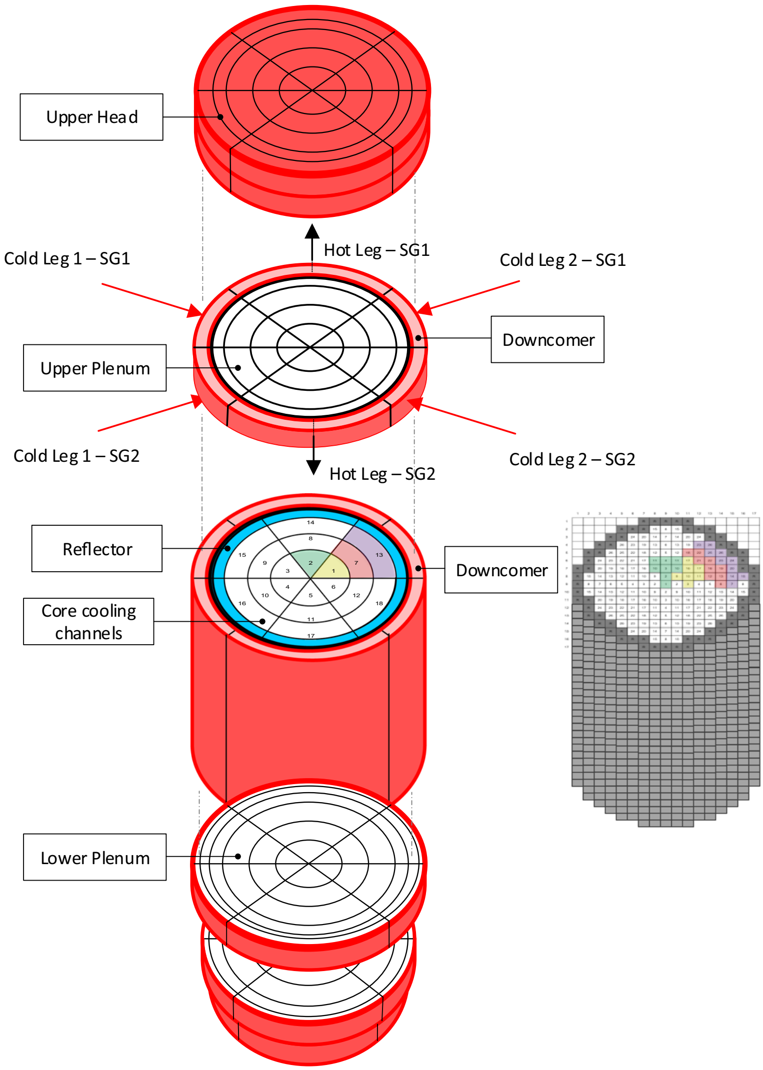

Fig. 3 3D TH model of the reactor vessel (left) and 3D NK model of core (right). On each axial plane, NK nodes are grouped in 18 zones and coupled to the 18 TH sectors (zones 1, 2, 7 and 13 have been highlighted as an example)

The reactor vessel is simulated by few 3D cylindrical components, representing the downcomer, the core lower and upper plenums, the core cooling and reflector channels, and the upper head. (See Fig. 3.) Six angular sectors have been simulated, each pertaining one coolant nozzle (four cold and two hot legs).

The 157 core cooling and 64 reflector channels have been simulated by one 3D cylindrical component, which consists of four radial zones (core center, middle, peripheral and unfuelled reflector), six angular sectors and 26 axial planes. The fuel elements are simulated by 18 vertical heat structures, discretized over the 26 axial planes and thermally connected to the 18 equivalent cooling channels.[1]

The NK mesh consists in a Cartesian matrix of $17 \times 17 \times 26$ nodes, representing a fuel or reflector assembly over 26 axial planes, accounting for material changes in the fuel design and burn-up. Over each plane, the NK nodes are grouped and coupled to the corresponding heat structures and the equivalent cooling channels, releasing the fission power in the fuel elements and obtaining reactivity feedback from fuel and moderator.

The code solves the NK equations (two prompt and six delayed groups) spatially discretized over about 6000 nodes, each of which characterized by a nuclear composition, depending on fuel assembly design (^{235}U% w/o, burnable absorber rods) and local burn-up. Compositions include both nuclear data (i.e., diffusion coefficient, macroscopic absorption, fission and scattering cross sections, and buckling) and reactivity coefficients (accounting for changes in moderator density, boron concentration, and fuel temperature); compositions must be defined for both thermal and fast neutron groups and, where applicable, must take into account for control rod insertion. Overall, the core is represented by 438 nuclear compositions, accounting for 1/8th core symmetry (i.e., 26 compositions defined per plane) and axial repetitions. Nuclear compositions of AP1000™ core were not available at the time of the analysis: the composition of Three Miles Island NPP (TMI-1) core [5], adapted to the AP1000™ [6] have been used. Compositions refers to the core at end of cycle.

5 Transient analysis

The event simulated is the main steam line break, a design basis accident for PWR. It consists in the double-ended guillotine break of the main steam line at SG1 outlet, upstream its main steam isolation valve (MSIV1), with the failure to insert the most reactive control rod (i.e., stuck in the quadrant pertaining to broken loop). This accident introduces a strong asymmetric perturbation at both plant (break of one loop) and core level (stuck rod), and is extremely representative for the 3D code capability assessment purpose.

The plant status relates to hot full power at the end of the cycle. Conservatively, no credit is given to boron negative reactivity insertion from CMT, to PRZ heaters and to CVCS.

The break is postulated at t = 100.0 s. A large steam flow rate is initially discharged from both sides of the break (Fig. 4), causing both SGs depressurization on secondary side (Fig. 5). In a couple of seconds, the low pressure generates the plant safeguards actuation (S) signal, triggering the SGs isolation on secondary side (i.e., MSIV1 and 2 closure and MFW isolation), the reactor coolant pumps (RCP) trip and CMT actuation; this latter calls for reactor and turbine trip and PRHR actuation.

Despite the most reactive control rod postulated failure, reactor SCRAM causes a sharp decrease in core power, due to the negative reactivity insertion (Figs. 6 and 9a). The mass release from steam line pertaining SG2 stops as soon as MSIV2 closes; SG2 shell, being isolated, recovers in pressure and reaches thermal equilibrium with primary system.

The mass discharge from SG1 steam line cannot be intercepted by MSIV1, since it is located downstream the break; SG1 shell progressively blowdowns, till its complete venting. The SG1 shell depressurization enhances the primary to the secondary system heat transfer due to the saturation temperature decrease on secondary side, causing a significant and

[1] For TH analysis, it is not necessary simulating each fuel assembly individually and it is convenient to group them by TH similarity, in collapsed equivalent assemblies. If a higher level of detail is locally required, the detailed simulation of the fuel assembly of interest should be performed. This is out of the scope of the present work and could be considered as a next step of the 3D code assessment activity.

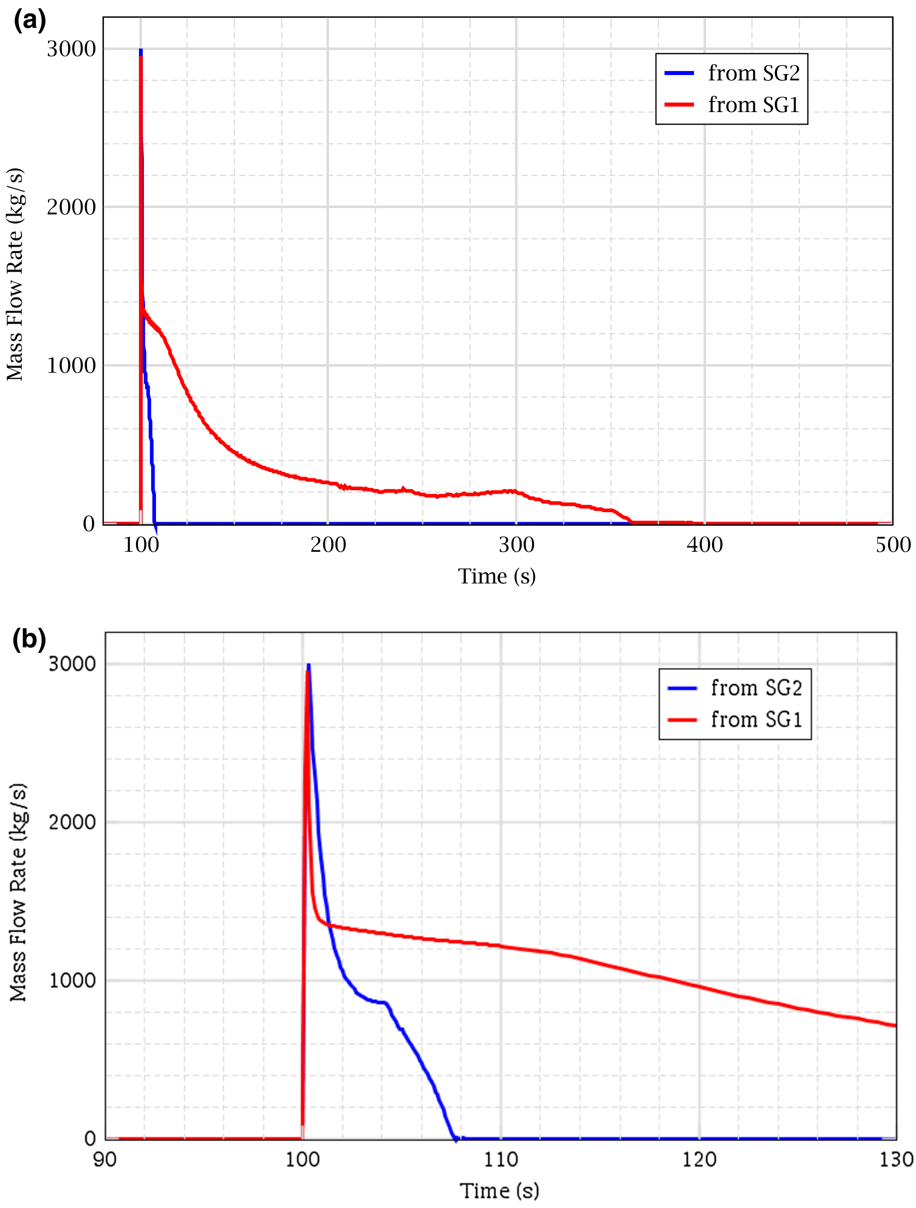

Fig. 4 Mass flow rate discharged to the containment from the two sides of the break (i.e., SG1 and SG2 sides). The flow from SG2 side of the break goes to zero as soon as MSIV2 closes, while flow from SG1 side of break cannot be intercepted by MSIV1 closure, since the break is located upstream—full transient (**a**) and details on first part up to MSIV actuation (**b**)

asymmetric RCS cooldown: The cold legs from SG1 experiences a large decrease in temperature, due to the high power exchanged, while legs from the isolated SG2 stabilize to an equilibrium value (Fig. 7). During the first part of the transient, the coolant mixing inside the reactor vessel is inefficient and the RCS temperature unbalance at cold leg nozzles results in

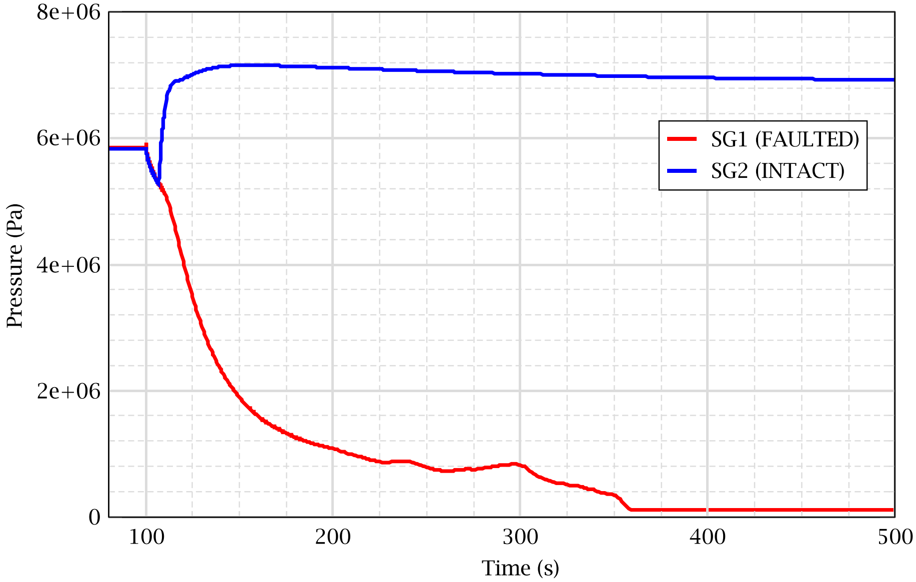

Fig. 5 Steam generators secondary side pressure

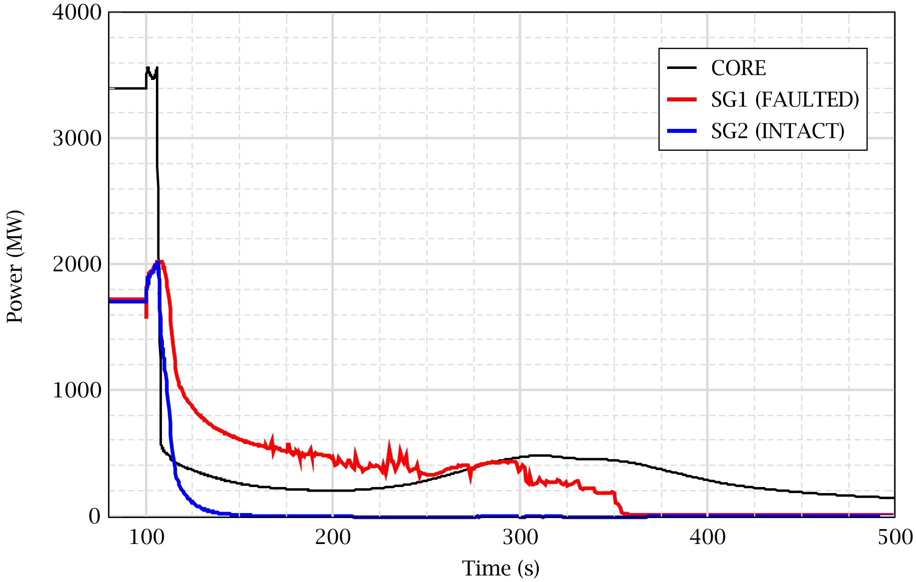

Fig. 6 Total reactor core power and power removed from the SGs

the unbalance at core inlet (Fig. 10a). The spatial difference in moderator density, combined with the control rod failure, causes asymmetry in the radial core power shape, which is higher in the cold half of the core, and peaked in the stuck rod area (Fig. 10b).

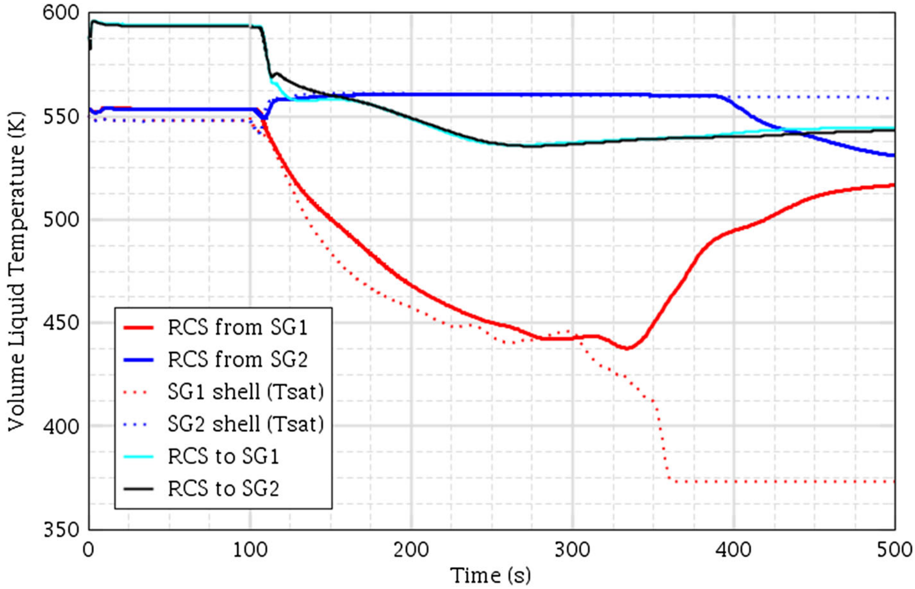

Fig. 7 RCS versus SG shell temperatures

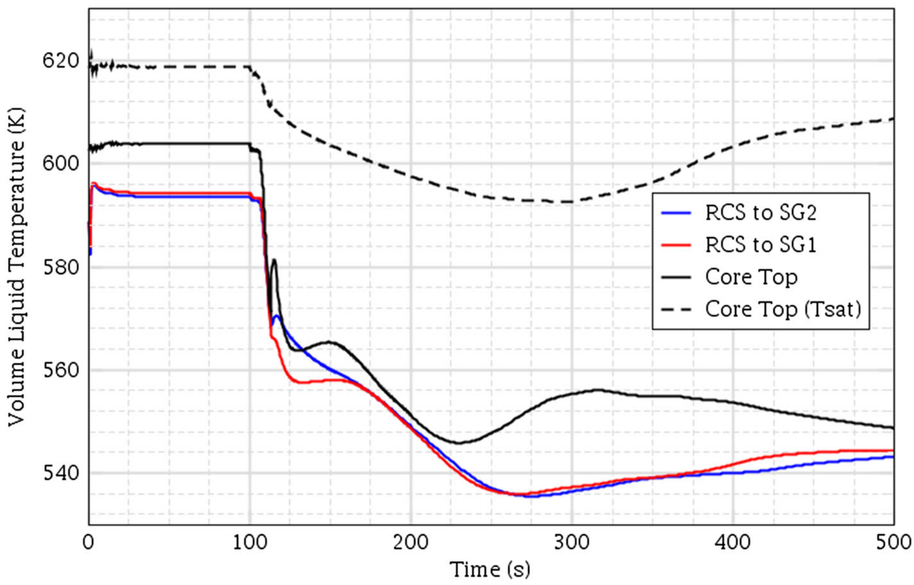

Fig. 8 RCS temperatures: core outlet versus saturation

As transient evolves, RCS flow from the isolated SG2 significantly decreases, promoting the in-vessel mixing; core inlet temperature evenly decreases, following the RCS temperature profile from SG1 (Fig. 8).

The progressive decrease in core coolant temperature causes a positive reactivity insertion and the consequent local return at power (Figs. 6 and 9a); core power reaches a second peak of about 500 MWth (15% of full power)[2] at 320 s. Fission power is markedly localized in the upper part of the core, around the stuck rod (Fig. 11). Despite the local return at power, core coolant temperature is well below saturation (Fig. 8) and no voids are predicted. Moreover, it should be noted the predicted return to power is enhanced by the conservative assumption to not simulate boron injection by CMT (Figs. 10 and 11).

Core velocities vector components (axial, radial and angular), at different axial planes, are reported in Fig. 12. Although the component in the main flow direction (axial) is dominant, not negligible cross-flows in transverse directions have been reported. This is reflected in the spread of axial velocities, reported since core center and particularly marked during the return at power. Axial velocities spread is due to a not uniform core coolant density distribution (RCS pumps are off and circulation is driven by axial density differences), mainly caused by the core local peak around the stuck rod and, with minor intensity, by different core composition/burn-up zones (as can be also seen in Fig. 11 at core top, overlapped with the power peak around the stuck rod). Axial velocities distribution at 310 s, at inlet and outlet core planes, are reported in Fig. 13. As can be seen, axial velocities at core inlet are homogeneous, confirming the mixing within downcomer and lower plenum, while core outlet velocities are peaked in the region of the stuck road, with a profile that is very similar to that reported in Fig. 11 at core outlet.

As soon as SG1 dries out, at about 350 s, it stops removing power from the RCS, causing a global increase in core coolant temperature. The latter, combined with the fuel doppler, causes the core power to decrease. From this time, the transient largely smooths; the PRHR system successfully provides the long-term residual heat removal safety function and the simulation is ended.

6 Conclusions

The simulation of a representative PWR asymmetric accident has been performed, namely a main steam line break with the failure to insert of the most reactive control rod. The analysis has been performed with a Relap5-3D model of the AP1000™ plant, including a 3D TH reactor vessel and a 3D NK core model.

The analysis demonstrated the code capability to capture 3D core phenomena: the unbalance of radial power shape due to different moderator temperature during the first part of the transient, the power peak in the stuck rod area and the axial distribution at the second power peak as well as the core velocities 3D components distribution, with a homogeneous flow distribution at core inlet—demonstrating downcomer and lower plenum mixing—and the flow spread developed along the core, due to density/local power unbalances.

These spatial phenomena cannot be simulated with point kinetic calculation, where nominal axial and radial power shape are imposed. An average core response would be obtained based on lumped core TH parameters and this would result in the underestimation of local power peaks. The drawback for the high level of detail provided by 3D NK analyses is the need for a large amount of input data to be managed; for a typical PWR core, several thousands of NK nodes with hundreds of different nuclear compositions need to be defined. Point

[2] Please note that the predicted core behavior is based on the conservative assumptions made to highlight local phenomena as well as on the application of physical properties derived from a different reactor. This is consistent with the purpose of the activity, that is to show 3D TH–NK-coupled calculation capability through the analysis of a commercial size PWR transient. The results here presented shall not be considered applicable to the AP1000™ reactor, for which the safety demonstration publicly available applies.

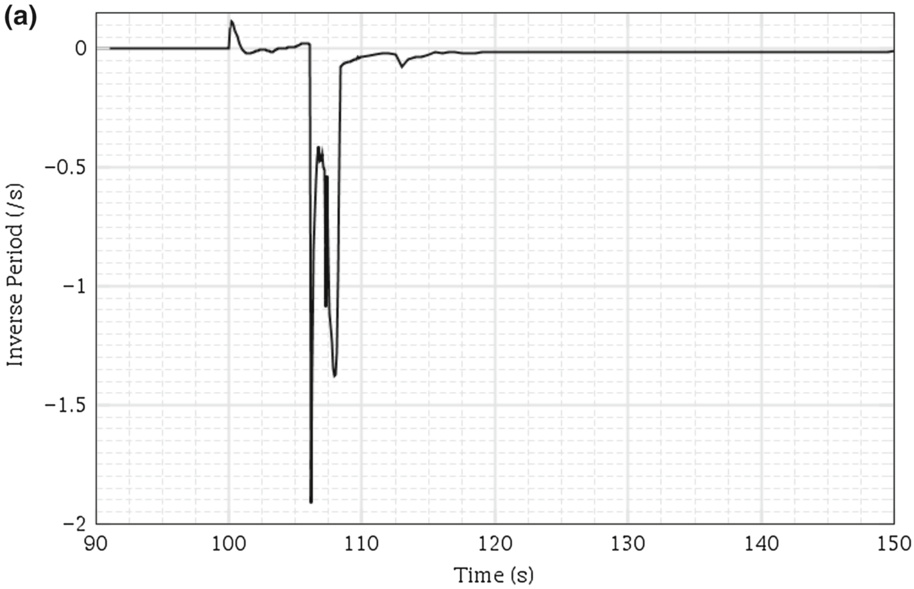

Fig. 9 Inverse of reactor core period: first 150 s (**a**) and from 150 s to transient end (**b**)

kinetic is still considered a handy and efficient tool for accident where core spatial effects are not relevant.

The 3D TH vessel and core modeling allows to properly simulate coolant flow distribution and mixing. This is a significant improvement with respect to 1D calculations where core is represented by few parallel channels only (periphery, middle, center of the core and hot channel) or fictitious 3D core models, made of large number of 1D cross-connected channels.

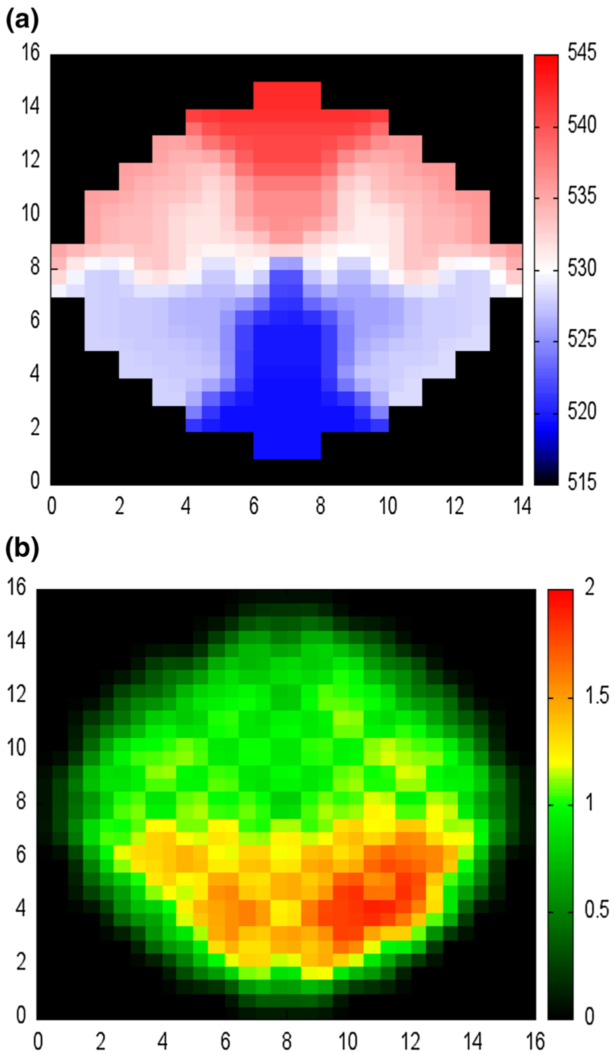

Fig. 10 Temperature (K) unbalance at reactor core inlet (**a**) and core relative radial power distribution—i.e., local-to-average power ratio, weighted on height (**b**) at 130 s. The lower and upper half of each figure represents respectively the SG1 and SG2 sides of reactor vessel (CLs at corners and HL at center)

Referring to this solution, 1D system codes, such as previous versions of Relap5, solve the full set of 1D fluid equations in the main flow direction and apply simplified momentum equations in the transverse direction to account for cross-flow between parallel channels. This approximation is sufficient for application where cross-flows are modest due to high resistance to transverse velocity but may be not adequate where asymmetric TH perturbation occurs in the reactor core or in vessel regions where transverse flow are relevant, such as the downcomer and lower/upper plenums. Moreover, several cross-junctions may be used attempting to simulate 3D core behavior by pursuing such approach, and this promotes uncertainty propagation. Where applicable, analysts tend to overcome this drawback by not adopting or limiting cross-junctions between the 1D pipes simulating the core channels in

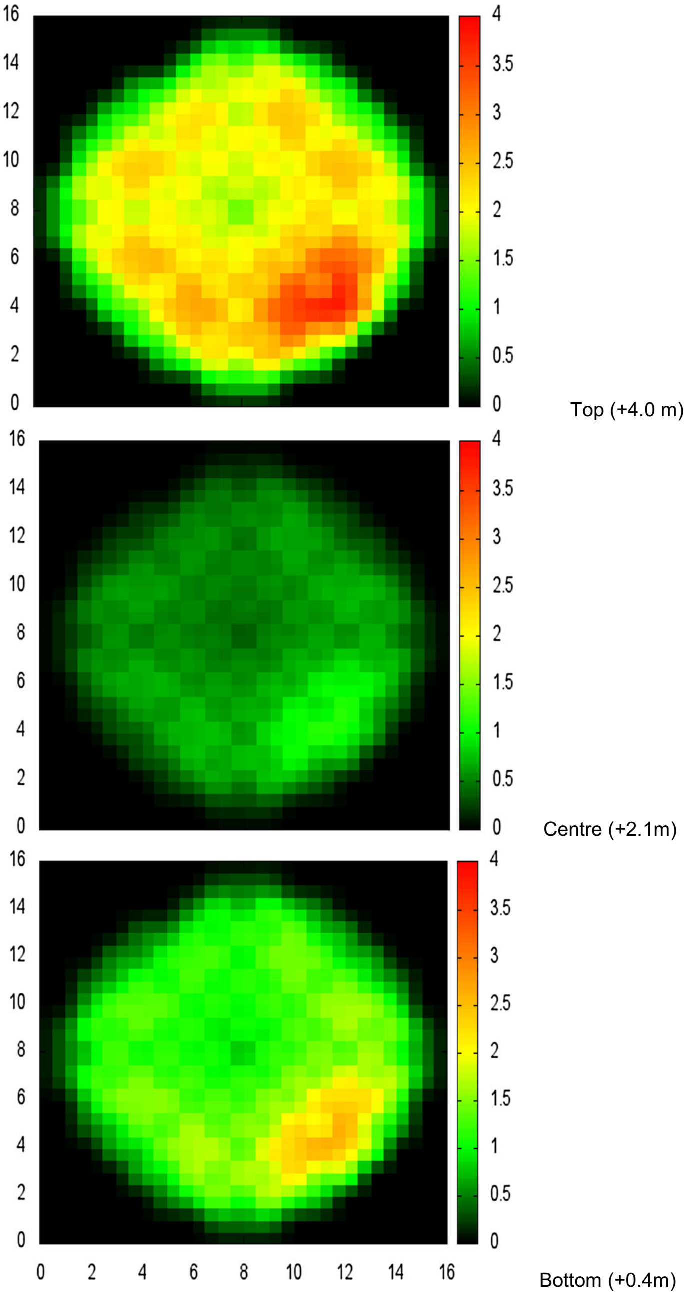

Fig. 11 Core axial power distribution (bottom, center and top) at 310 s. Stuck rod is located at $Y = 4$, $X = 12$

3D fictitious models; this approach stresses the effect of local perturbation by neglecting the beneficial effect of flow mixing but, in the end, moves from one of the main goal of 3D coupled calculation, that is to perform best-estimated calculation. Full 3D TH modeling

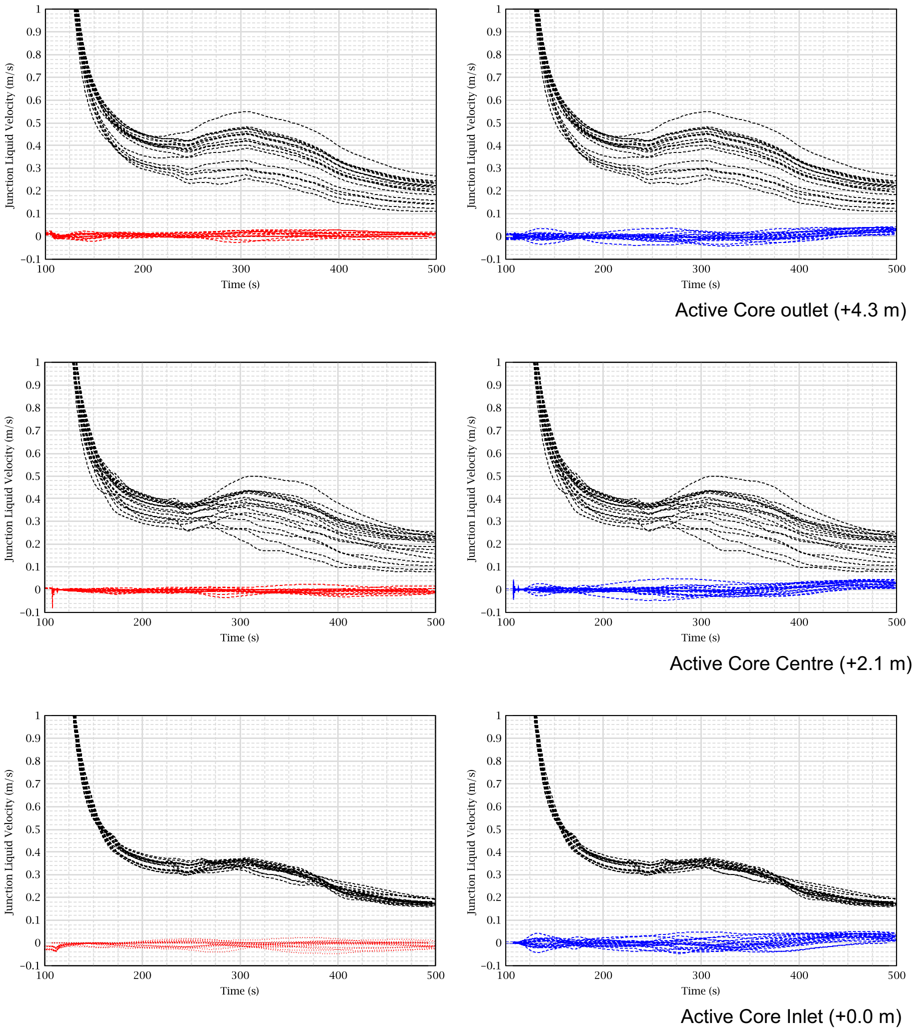

Fig. 12 Active core velocities vector components, at different axial plane: V_Z (black), V_R (red) and V_θ (blue). For each axial plane, velocities referring to the 18 active core hydraulic nodes are reported

capacity aim at overcoming these issues by providing a reliable simulation of transverse flow phenomena with few and handy 3D components, instead of combining several 1D parts; to this regard, 3D modeling in Relap5 is an innovative tool and further verification and validation is on-going to extend the code range of applicability. Since (1) the coupling of 3D NK to fictitious 3D TH core models is still a trusted option in the nuclear community and (2) the results obtained are consistent with the main assumption on which 1D cross-junctions are based (i.e., modes transverse flow compared to main direction), the next steps should be the direct comparison of the results obtained with simulations performed considering different 3D fictitious models, at increasing transverse modeling complexity (i.e., no, few or several cross-junctions between core parallel channels).

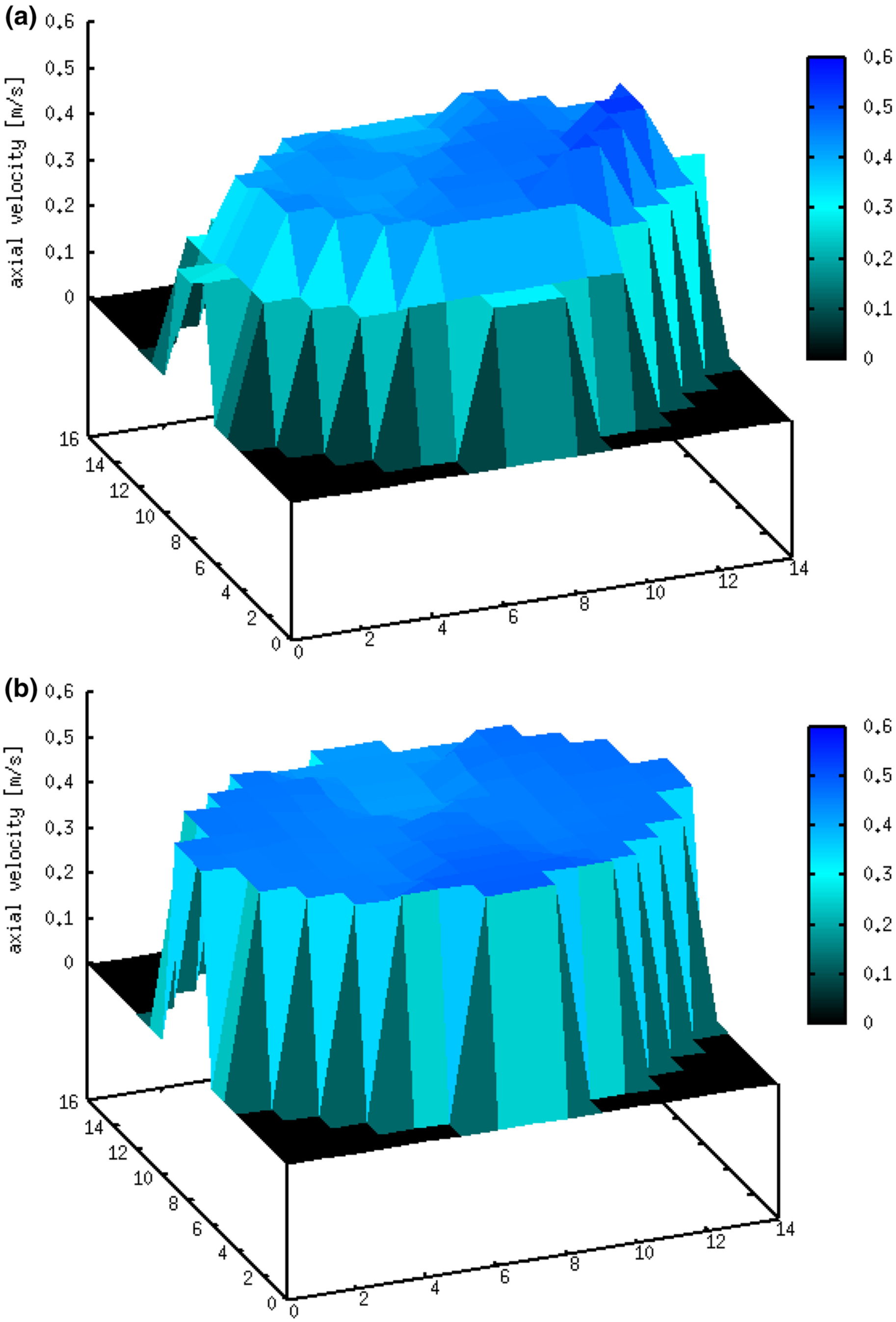

Fig. 13 Core axial velocities distribution at inlet and outlet plane at 310 s. Core orientation is the same of Fig. 11

Net of these considerations, it can be concluded that the fully integrated 3D TH and NK modeling capability makes the Relap5-3D code an advanced and interesting tool, which can deal at once and in a versatile way with the different levels of detail required in PWR safety analyses.

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