

ASTEC extension to other reactor types than Generation II PWR

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SUMMARY

The initial IRSN-GRS requirements for the development of the ASTEC European integral code covered all present and future PWR, VVER and BWR. After 4 years and a half of SARNET activities, the ASTEC V1 code is fully applicable to all severe accident scenarios, at power operation, in Gen.II PWR and VVER. Partners performed substantial code assessment tasks on both VVER-440 and 1000 with good results on validation (such as PACTEL) and benchmarking with other codes on plant sequences.

In the frame of preparation of the new series of ASTEC V2 versions that will account for the needs of evolution expressed by the SARNET partners, the extension to Gen.III PWR is under way. The 1st version V2.0, planned for march 09, will be applicable to the EPR, in particular its external core-catcher and to new PWR designs with In-Vessel Melt Retention (IVMR), the latter thanks to CEA work. IVS calculations on VVER-440/V213 showed the ASTEC V1 capabilities to evaluate the IVMR possibilities: these capabilities will be improved in the future through a coupled analysis of in-vessel corium and cavity cooling circuit.

For BWR applications, the KTH ranking of needs of model adaptations was reviewed by GRS. These adaptations mainly concern the Reactor Cooling System (RCS) thermal-hydraulics (internal pumps, drier, separator...) and the core degradation (canisters, control rod guide tubes, penetrations in lower head...). For core degradation, the scoping application test on the CORA-18 experiment using the current ASTEC V1 models (for adaptation to absorber blades and canisters) showed promising results. The GRS benchmark with the COCOSYS GRS code showed the applicability of the CPA containment module. For ex-vessel phenomena, the only missing model is the formation of a debris bed during corium slump from lower head into a flooded cavity after lower head failure and its coolability/erosion behaviour with core concrete interaction under flooded conditions. All other current V1 models can be used, sometimes with minor adaptations or further need of validation, as for iodine behaviour in containment at temperatures up to 1000K and the possible decomposition of metal iodides to gaseous iodine at these high temperatures in atmosphere. Detailed specifications will be written soon to prepare the development and validation task planned in SARNET2 with the same partners.

For CANDU reactors applications, the priority concerns the core degradation due to the peculiarity of the vessel geometry (horizontal core, pressure tubes, calandria...). The exploratory plant applications have shown that all other current models can be used but tests are still necessary to conclude on models of PHT (Primary Heat Transfer) thermal-hydraulics. INR work takes benefit of the AECL knowledge of MAAP4-CANDU models. A significant benefit is foreseen from the BARC (India) work on core degradation models in the frame of IRSN-BARC bilateral collaboration.

A. INTRODUCTION

The initial IRSN-GRS requirements for the development of the ASTEC European integral code covered all present and future PWR, VVER and BWR. At time of SARNET's end, the ASTEC V1 code is fully applicable to all severe accident (SA) scenarios, at power operation, in Gen.II PWR and VVER. For the latter, several partners assessed the code results either through validation on VVER-specific experiments such as PACTEL or CORA-W2 [1] or through benchmarks with other codes on plant sequences for both VVER-440 and 1000. In general, good results were obtained as shown in the chapter B below written by VUJE.

In the frame of the new series of ASTEC V2 versions that will account for the needs of evolution expressed by the SARNET partners, the extension to Gen.III PWR or VVER is under way. In a first priority, ASTEC V2 should simulate the EPR (European or Evolutionary Pressurised Reactor) and the other advanced PWRs that include the concept of In-Vessel Melt Retention (IVMR). The status of these investigations is summarized in the chapter C, respectively written by IRSN for EPR and by CEA for IVMR.

Then, in accordance with the European Commission requirements in the Nov.07 call for proposals [2], efforts will focus on the adaptation to the BWR and CANDU reactors. The ASTEC V2 general requirements [3] already accounted for these reactor types. During the SARNET lifetime, several partners already investigated and ranked the corresponding model adaptations. Besides, exploratory code calculations were performed in order to show the adaptation feasibility. The chapters D (by GRS, IKE and KTH) and E (by INR) respectively present the status and perspectives of ASTEC model adaptations to BWR and CANDU.

B. ASTEC ADAPTATION TO GEN.II VVER

B.1 VVER-440

All efforts focused onto the V213 design not taking into account the older V230 design. The work was coordinated by activities of the Slovak-Hungarian ASTEC Users' Group (SHAUG) applying the ASTEC VVER-440 benchmarking matrix. Gradually upgraded, the matrix covers the benchmarking activities of the following partners: VUJE, IVS and UJD from Slovakia and VEIKI, BUTE from Hungary. The cooperation and confrontation of severe accidents results led to building of a coherent group of severe accident scenarios calculated with the latest version of the ASTEC code. The matrix covers all typical severe accident scenarios including large LOCA, medium LOCA, small LOCA, station blackout and primary to secondary circuit LOCA. Most of the scenarios were calculated using all the code modules. Some effort is still needed to successfully achieve the calculation of the primary-to-secondary leakage scenario, as there are still instabilities with the volumes involved in the break flow mass balance.

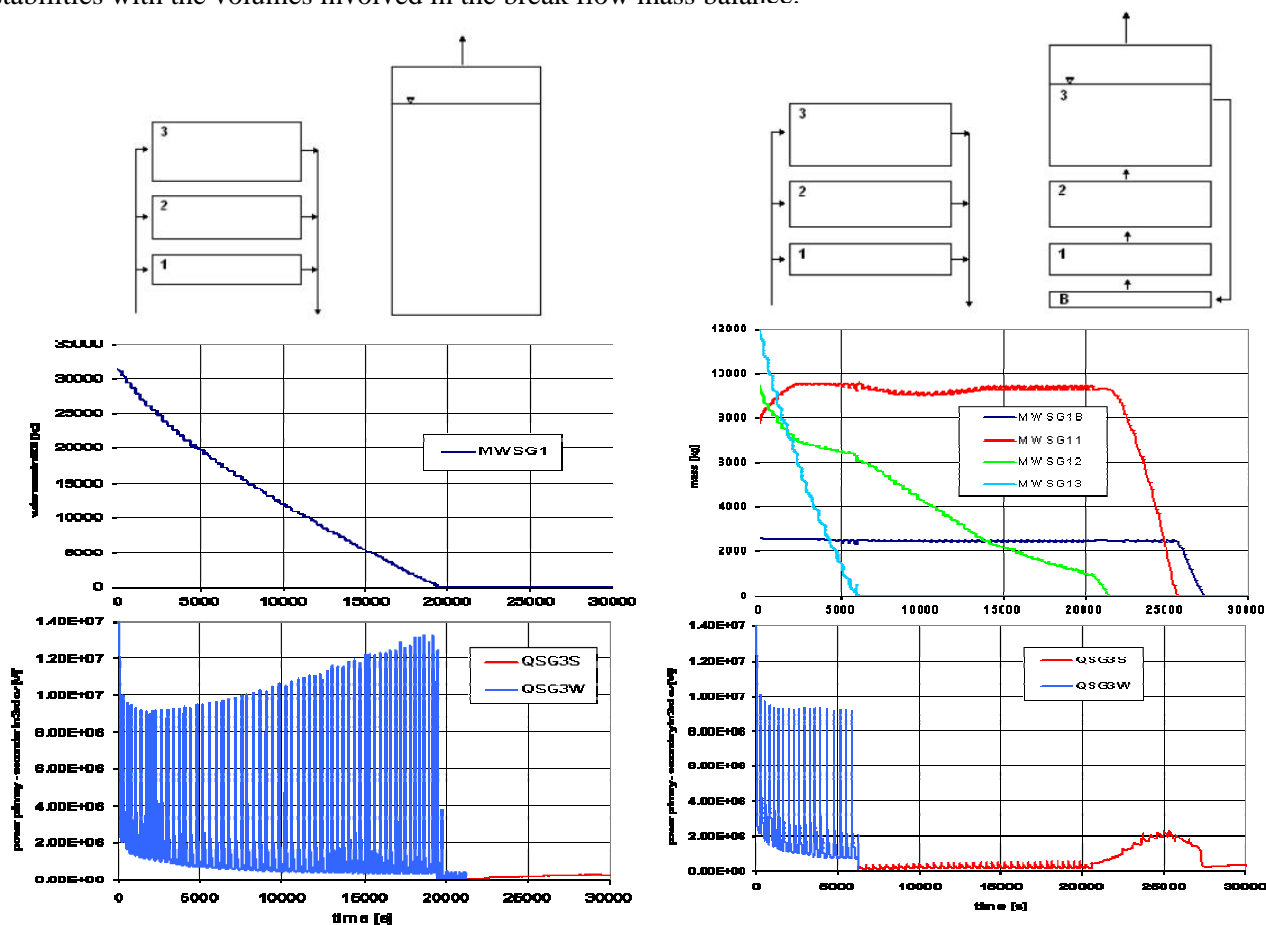


Figure 1: Influence of the VVER-440 SG secondary side meshing (top schemes: simplified on the left and detailed on the right) on SG secondary water inventory and primary-secondary heat exchanges

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The continuous development of the input decks led to intensive collaboration with the development team of the ASTEC code. The development of the horizontal SG nodalisation, which was thoroughly analysed, showed possibilities of improvement as the components tend not to avoid uncertainties with more detailed nodalisation. The development of horizontal SG nodalisation in VUJE focused on advanced modelling of heat transfer in the SG tube bundle with higher detail of modelling. The uncertainties were observed during the analysis of increased number of primary circuit nodes modelling the tube bundle (Figure 1). The solution leading to elimination of uncertainties led to the increase of volumes for modelling of the secondary side of steam generator tube bundle. As shown in Figure 1, heat transfer to liquid secondary coolant continues even if the coolant level at the secondary side drops below the bottom elevation of primary circuit tube bundle nodalisation volumes. Heat transfer to liquid phase ceases after the level drops.

Regarding the VVER-specific validation, one of the main thermo-hydraulic experiment facilities for the purpose of validation is the PACTEL facility. The PACTEL facility is a scaled-down model of Loviisa NPP (VVER-440/V213). IVS validated the CESAR module CESAR against 2 PACTEL experiments [1], reaching an overall good agreement (except for the pressurizer level). Further validation by IVS focused onto the Paks accident - destruction of fuel in a fuel cleaning tank. The capability of ASTEC code in the modelling of fragmentation of fuel, as a result of cold water ingress into the tank after the problem with cooling was noticed, was underlined. Two calculations were performed: one with CESAR stand alone, up to the start of fuel heat-up, the other one with a coupled calculation (CESAR, DIVA, ELSA and SOPHAEROS modules) focusing onto the scenario period since the fuel temperature exceeds 450°C, but using more simple nodalisation. The system behaviour was well predicted by ASTEC and the comparison with calculation with other calculations of the same scenario showed good correlation.

B.2 VVER-1000

VVER-1000 calculations were performed by various companies from the countries possessing this reactor design, including UJV from Czech Republic, TUS and the FoBAUs group from Bulgaria. Analyses were performed also in Germany under the cooperation of Technical University of Dresden, GRS and FZR.

The analyses of TUS focused onto sensitivity studies of small break LOCA scenarios with emphasis on hydrogen production, core degradation and degraded core configuration. A similar sensitivity study evaluated various scenarios with different configurations of passive injection systems (hydro-accumulators).

UJV analyses focused onto MCCI behaviour in the VVER-1000 reactor cavity. The comparison between MEDICIS module and MELCOR 1.8.5 led to a generally good agreement.

The FoBAUs Group (INRNE and EI - Sofia) successfully calculated a Station Blackout scenario for a Bulgarian VVER-1000 unit with fission products (FP) retention analysis.

The Station Blackout scenario was intensively studied in German consortium of companies and the results were compared with ATHLET code focusing onto primary circuit pressurisation and depressurisation during in-vessel phase and operation of steam generator safety valves during the early phase of station blackout scenario. Reasonable results of the comparison were achieved with recommendation for future use of the ASTEC code.

Out of the SARNET circle, but presented in the ASTEC Users Club, several ASTEC calculations by RRC-KI (Russia) have confirmed the code capability to perform a full scope analysis of the VVER-1000 reactor unit: large break LOCA, Steam Generator Tube Rupture (the latter showing good results of comparison with ICARE/CATHARE code).

B.3 Conclusions on ASTEC applicability to VVER

ASTEC showed a very good capability to incorporate features of the VVER-440 and VVER-1000 reactors during its development in the SARNET project. The version ASTEC V1.3 is applicable for analyses of a wide spectrum of severe accident scenarios. Yet there is need for further development, especially in the modelling of the reactor core. The new magma ICARE2 model for corium relocation will improve the simulation of the 2-part VVER-440 core and of the fuel assembly canisters.

C. ASTEC ADAPTATION TO GEN.III PWR

C.1 EPR

The EPR presents an increased safety level through reinforced protections in case of an hypothetical core melt accident. Besides the dispositions taken to reduce the probability of core melt (increased use of

redundant systems, diversification of equipments relevant for core cooling, etc.), several dispositions are taken to reduce the SA consequences:

- Dedicated pressure relief device to depressurize the primary circuit and avoid high pressure vessel failure or steam generator tubes failure;
- Passive autocatalytic recombiners to reduce the risk associated with the H₂ presence in the containment;
- Core catcher to stabilize melt corium in the reactor containment (Figure 2);
- Dispositions to avoid accumulation of important amount of water in the reactor pit and avoid the risk associated with a fuel coolant interaction;
- Double concrete wall containment with internal metallic liner to increase the containment tightness;
- Containment “two-room concept”: two parts, connected by doors, rupture discs and flaps, one relatively small containing all the primary circuit components and a large accessible part.

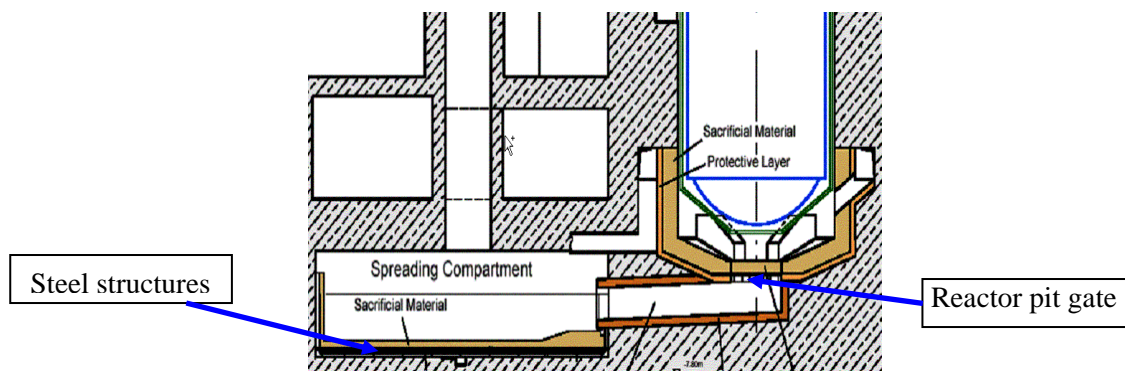


Figure 2 : Scheme of EPR core-catcher

All above features can already be modelled with the ASTEC V1.3 code except the core-catcher. Some years ago IRSN performed separate analyses using ASTEC modules such as MEDICIS and CPA [4]. Based on the outcomes of this first analysis, detailed model specifications have been recently released and the corresponding developments are ongoing:

- Molten-Corium-Concrete-Interaction (MCCI) in the cavity with MEDICIS, accounting for all radiative exchanges between the present materials (corium, remnants of vessel lower head, cavity walls). The account for possible melt-through of concrete upper cavity walls will also be possible, with account for successive corium slumps into the cavity. The absorbing nature of the gas medium will be simulated in a parametric way. The radiative heat flux from corium will be transferred to the remaining vessel walls for a better evaluation of continuation of heat-up of vessel walls after the first rupture.
- Erosion of the cavity substrate, and corium pouring kinetics from the cavity towards the core-catcher. A simple and fast-running model is today available, based on a combined use of a Bernoulli flow approach and of it can be easily implemented.
- Corium spreading along the transfer hall and the spreading chamber. Two levels of approaches will be adopted: in a first step a simplified analytical correlation, corresponding for cases where the corium characteristics lead to a relatively fast and complete spreading; and in a second step a more detailed model like the LAVA GRS code for cases where spreading may be slow and where spreading must be simulated in more details. If the corium characteristics lead to incomplete spreading, such situation that would challenge the EPR design is clearly out of the ASTEC scope: the code should stop and the user should be alerted via a flag that he should rely on separate calculations with dedicated codes.
- MCCI in the spreading chamber with upward cooling by water injection. Here MEDICIS models are already applicable, involving its upward water cooling models. For the cooling lines under the concrete in the chamber basemat, a simplified analytical model for evaluation of heat fluxes transmitted to the steel structures will be implemented.
- In the ASTEC V2.0 version, only sequential MCCI in different volumes will be represented: MCCI in the spreading chamber will start while MCCI in cavity, involving a possible remaining corium mass, will stop. This assumption corresponds to an almost total corium transfer from a volume to another one. Later on, for achieving a more general capability, the code should be able to deal with MCCI occurring simultaneously in several different volumes. This will be also useful for a limited corium transfer from a

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volume to another one, for instance after radial erosion of cavity walls and corium flow into any neighbouring volume.

For simulation of other corium catchers in new reactor designs such as advanced VVER that would possibly imply interactions between corium and sacrificial concrete or ceramics, no efforts are planned at mean term. Let us only note that the high flexibility of the ASTEC Material Data Bank (MDB) would allow to easily account for new materials.

Note that corium spreading may also occur in other reactors than EPR, for instance some types of VVER-1000 reactors where corium may spread in compartments or containment zones. The latter holds also true for some German reactor types. As these zones are not specifically dedicated to corium spreading, care will have to be taken when applying models developed for EPR.

Finally, the new models of 2D in-core relocation of molten corium and debris, issued from the ICARE2 IRSN mechanistic code for core degradation [5], will allow a much more realistic of corium relocation through the EPR heavy reflector around the core and the core lower structures (even if the ASTEC V1 models can already simulate it).

C.2 Modelling the In-Vessel Melt Retention

For achieving the In-Vessel Melt Retention (IVMR) strategy in new reactor designs, the corium retention is achieved by an External Reactor Vessel Cooling (ERVC) system associated to the flooding of the reactor pit. This concept has been adopted for advanced PWRs such as AP1000 and APR1400 (in combination with in-vessel core catcher) as well as for older VVER-440/V213 reactors. The success of IVMR depends on the thermal margins: the difference between the thermal loading imposed by the melt in natural convections on the inside and the limits of coolability (due to boiling crisis) on the outside of Reactor Pressure Vessel. The avoidance of heat-transfer crisis on external vessel surface and thus the effective cooling for above-mentioned media and higher-powered advanced designs is assured via coolant flow streamlining alongside the curved vessel surface (installation of flow baffles) and via optimising of ERVC design.

The IVMR analyses are currently performed using imposed boundary conditions on the external reactor surface. These boundary conditions are generally described using a constant heat transfer coefficient and coolant temperature. However, in practical ERVC loop arrangement, this quantity depends on the external condition of the two-phase flow in natural circulation. Furthermore, flow instabilities could be expected to occur. This may result in significant changes of heat transfer on external vessel surface.

The description and the model of the ERVC loop are necessary to get the local two-phase flow conditions, e.g. pressure, void fraction and flow rate.

A first status of ASTEC V1 possibilities was presented by CEA and IVS in ERMAR-2007 [6]. A coupling between the ASTEC/DIVA module for the in-vessel behaviour and the ASTEC/CESAR module for flow conditions in the external cooling circuit has been performed. The interface between DIVA and CESAR required a development because a boundary condition for one of the two modules cannot be linked explicitly to the result of the other module. The external part of each cell of the external ring of the vessel is part of the two code models. From the fluid heat transfer coefficients, the vessel wall temperature and the fluid temperature, CESAR computes the heat flux exchanged between the vessel and the fluid. This heat flux which corresponds to the external vessel flux, allows to DIVA to calculate the wall temperature.

Some additional constraints are linked to the description of the components in the two codes force the model of the lower head vessel description in DIVA and the external channel description in CESAR:

- To avoid averaging of the heat flux between the corium layer and the lower head cells, an axial refined mesh is necessary in the zone where the heat flux is maximal. In case of a thin upper metal layer, the axial meshing must be adapted around the elevation of this layer. The use of the DIVA automatic meshing of the lower head should thus not be advised.
- With the IVMR strategy, the residual thickness of the vessel wall in case of severe accident can be very low compared to its initial value. The meshing along the radial direction must be set to its maximum.
- The maximum cell number of the vessel in the lower head in DIVA is about 90. Thus, in ASTEC V1 the meshing of the vessel is fixed and set to 5 rings in the radial direction and 17 rings in the axial direction. In ASTEC V2 the number of vessel meshes will no more be limited.
- Each vessel DIVA cell can be connected to only one cell of a CESAR volume. The corresponding pipe in the CESAR model must then be described with 17 cells along the lower head. The geometrical

properties (elevation, hydraulic diameter, volume, exchange area,...) of each cell of this typical pipe must also be calculated from the DIVA description of the lower head.

- The calculations of the geometrical data belonging to both codes are performed using the SIGAL environment. In addition to these common data, a description of the other part of the external vessel cooling circuit must be provided by the user.

The time scale of the in-vessel phenomena, thus DIVA time step, is much higher than that of the outer vessel one, thus the CESAR time step. We assume that the thermal inertia of the vessel remains enough important to prevent CESAR instability due to variation of the in-vessel conditions.

Validation has been performed on the SULTAN (CEA) experiments for forced convection (Figure 3). The simulation presented here, for a vertical orientation of the heated channel with 15cm thickness submitted to 500 kW/m² heat flux, gives good results. The above validation work confirms that CESAR is stable enough for calculating the pressure loss showing particular characteristic, like the S form of pressure drop linked to the flow, in a heating channel. However, further validation based on the LIVE (FZK) and HERMES (Japan) experiments should be considered first to validate the coupling between DIVA and CESAR (heat transfer + external cooling coupling) and secondly the CESAR behaviour with a real hemispherical vessel (air injection in water). For this last task and for reactor application with free convection two phase flows, an important challenge is to compute the physical oscillations linked to this kind of flow.

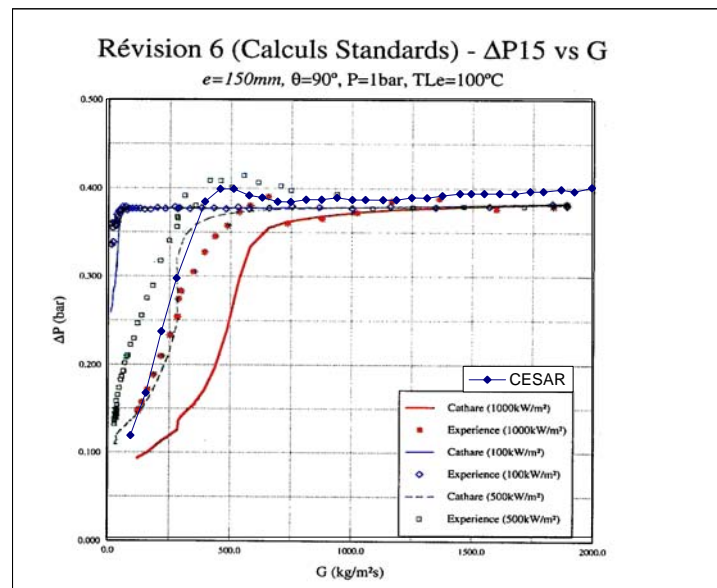


Figure 3 : SULTAN experiment – CESAR calculation comparison for pressure loss as function of flow (flux 500KW/m²).

D. ASTEC ADAPTATION TO BWR

In order to prepare the adaptation of the ASTEC code to BWR SA analysis, three partners have collaborated: KTH and GRS on overview and ranking of mode adaptations, IKE on the detailed analysis of core degradation models; KTH on the detailed analysis of ex-vessel corium models; GRS on RCS and containment models analysis.

D.1 Ranking of necessary model adaptations

A PIRT (Phenomena Identification and Ranking Table) [7] has been elaborated. Table 1 lists the main BWR specificities to be modelled, the importance of the phenomena and their ranking.

Table 1: BWR specificities for ASTEC modelling adaptation

Features	Phenomena importance	Adaptation priority
Core channels (fuel assembly, canister) and bypass (gap between assembly)	*****	+
Larger fuel inventory and different core material: impact on decay heat		+
CRGT (Control Rod Guide Tube) support of the core	*****	+
Control rod failure	***	o
Melt only relocated in the bypass before canister failure	*****	+
B ₄ C as control rod absorber	***	=
FP behaviour in the big volume of steam separators and driers	***	o
Impact of the numerous CRGTs on melt displacement, debris bed formation in LH	*****	+
Numerous penetrations in lower head and impact on lower head mode/failure timing	*****	+/o
Primary internal pumps	**	=
Inerted containment	*	no
Partially inerted/non-inerted	*****	=
Suppression pool	*****	no
Vacuum breakers	*****	no
Possibility of steam explosion → criterion	*****	+/o
Effects of CRD drive mechanisms on melt ejection	***	+/o
Flooded cavity	***	
Rupture disc venting system		no
Iodine chemistry in containment	*****	o
MCCI under water, formation of a debris bed	*****	+/o
Ex-vessel melt spreading	***	+/o
No significant difference with PWRs. Passive safety systems of new BWR designs	**	+/o
Properties of material specific to BWRs	**	=

* the number of stars indicates the ranking of priority

+ : high, o medium, +/o : medium to low, no : available, = : low priority for adaptation of models

D.2 Detailed needs for in-vessel core degradation

The review, carried out in SARNET, took into account, on the one hand, the modelling approaches and capabilities in other major SA simulation codes (MELCOR, ATHLET-CD, SCDAP/RELAP) and, on the other hand, the presently available models in ASTEC and ICARE2 (new module in ASTEC V2 series). The emphasis was on BWR-specific structures in the core, especially control blades and fuel canisters (Figure 4). The review aimed at providing detailed model requirements and specifications for the model adaptations in ASTEC and to propose concrete implementation steps.

- The present thermal-hydraulic DIVA core modelling does not deal with parallel fluid channels in the same ring, as it would be required for separate flow in fuel element and bypass. Also the possibility to open connections between initially separate flow paths (after failure of canister walls) is necessary. Such features will be available with the planned (in ASTEC V2) CESAR extension to the RPV thermal-hydraulics also in the degradation phase.
- For heat transfers with canister walls a structure (rods, plates etc.) exchanging heat with different fluid channels at its left and right side will have to be developed, which is not the DIVA case today.
- The ICARE2 new approach for radiation heat transfers, based on effective conductivities, is in principle flexible enough to account for BWR geometry with canisters etc. Such model will have to be adapted to this geometry.
- Chemical interactions: the existing model between boron carbide and steel can be applied also for absorber blades, by modelling the absorber rodlets as representative rods and taking into account the blade material through an increased clad radius. A model for oxidation of boron carbide by steam is available; corrections may be necessary concerning geometry and specific oxidation kinetics of BWR absorber material (powder). For interaction between absorber melt and canister walls, the existing model could be used (in analogy to absorber melt interaction with a guide tube) with appropriate corrections for geometry.
- For the description of the absorber blades, the adaptation of the existing control rod model seems to be feasible, requiring only moderate modifications.

- The canister walls could be modelled equivalently to a control rod guide tube, with adjusted geometry to respect actual mass, surface and wall thickness. Canister walls adjacent to control blades should be distinguished from walls not adjacent to the blades, due to the differences concerning heat transfer (mainly radiation) and material interactions .
- Relocation of absorber/steel/Zircalloy melt could be modelled like relocation along a “normal” control rod, with modifications concerning geometry.

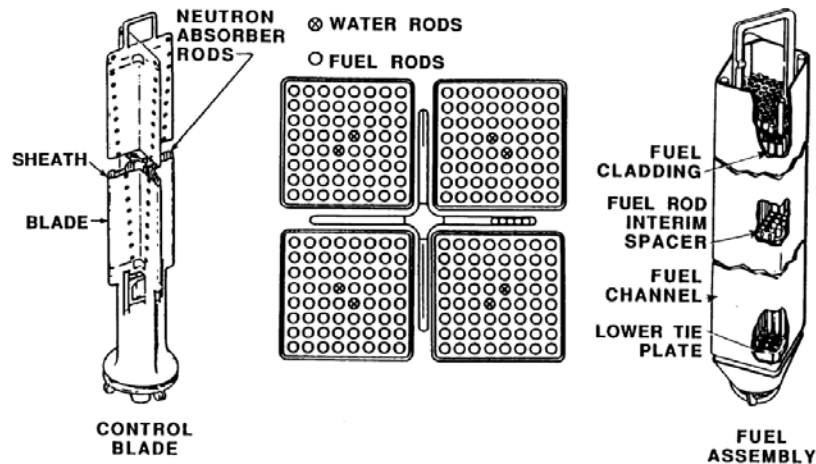


Figure 4 : Typical BWR control blade and fuel assembly

In support of the model adaptations, scoping and validation tests shall serve to identify which processes are already satisfactory modelled or which extensions are necessary and finally to check and verify the model extensions. Concerning core degradation, CORA BWR experiments are proposed, which are representative for melting of a BWR fuel element, usually with 18 rods and 1 blade ($\frac{1}{4}$ of a control cross, 9 rods on each side of the control blade). Suitable cases are CORA-28 (pre-oxidized), CORA-31 (slow heat-up), CORA-33 (dry conditions, no additional steam input) and CORA-18 (large bundle with 48 rods). Later, also American XR-experiments (Sandia) can be considered.

A first scoping calculation of CORA-18 was performed with ASTEC V1/DIVA. 5 representative rods were used to model the 48 heated and unheated fuel rods (Figure 5). The absorber blade was modelled in analogy to a control rod, with modifications to account for the specific geometry. The 11 absorber rodlets were considered as one representative absorber rod, the absorber cladding was used for lumped modelling of rodlet claddings and blade sheet. The canister wall was modelled in analogy to a control rod guide tube. Dimensions were chosen to respect the masses and surfaces. Other structures (shroud, shroud insulation, grids) also considered. Gas flow through the bundle was modelled by one channel. For radiation heat transfer, the view factors were determined for the representative structures and given by input. Material interactions between absorber rodlets, sheet and canister wall and the resulting degradation as well as oxidation of the B_4C were considered, in addition to oxidation and degradation of the fuel rods.

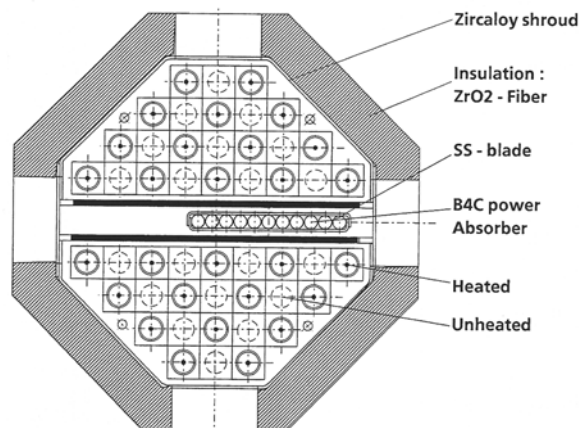


Figure 5 : Bundle layout in the experiment CORA-18

Figure 6 shows the result of the scoping calculation for bundle temperatures at different elevations. A quite good agreement is obtained, especially concerning the temperature escalation due to oxidation at around 4000s. In the upper part of the bundle, the calculated temperatures underestimate the heat-up, which explains the under-prediction by 15% of the total hydrogen production (Figure 7). It has to be taken into account that the ASTEC production rates are instantaneous values, while the measured values are obtained by flow rate measurements downstream of the test section, thus exhibiting an integrating behaviour.

In the experiment, the absorber blade and the channel box wall were destroyed between 254 and 1016 mm elevation. Between 130 and 240mm a blockage was formed between the channel box walls. The calculated bundle state at the end of the experiment in Figure 7 closely resembles this: material layers of the control blade and canister wall at the left, one heated rod in the middle and one unheated rod at the right. The blade sheet and canister walls have been dissolved and relocated between 0.2 and 1 m elevation, due to material interactions. B₄C in this region was not completely dissolved, some remnants remain in place.

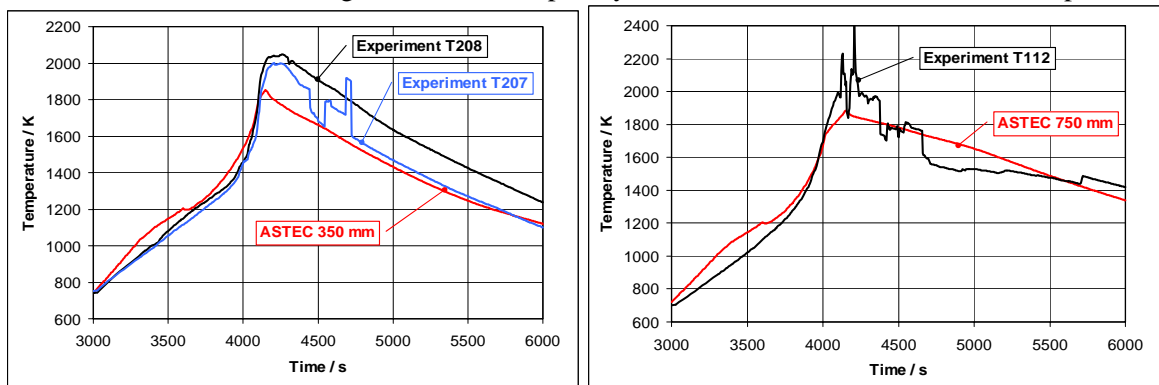


Figure 6 : Calculated/measured bundle temperatures at different elevations in CORA-18

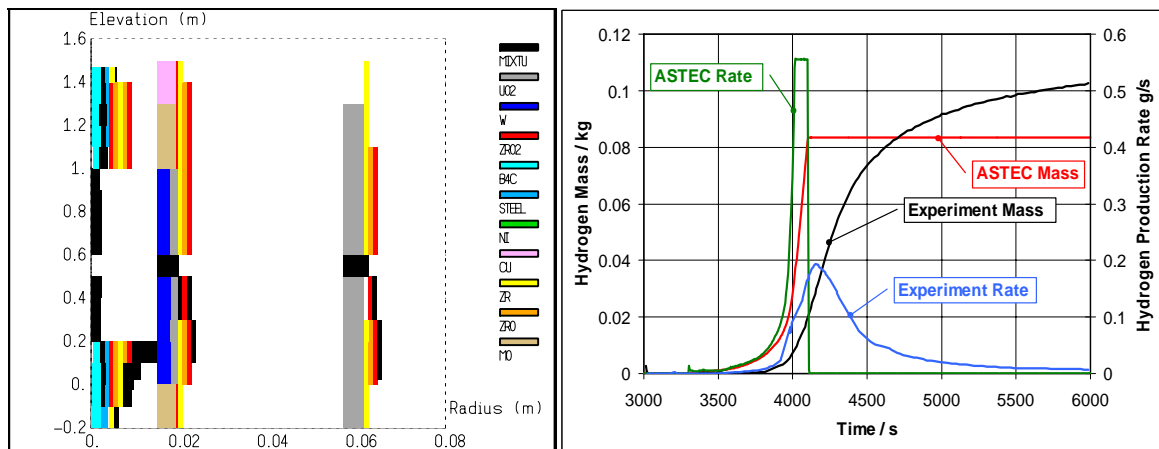


Figure 7 : CORA-18: (left) final ASTEC bundle state; (right): hydrogen production

D.3 Detailed needs for ex-vessel corium behaviour

- Molten core concrete interaction (MCCI) – dry cavity.** In principle, the MEDICIS module is able of MCCI simulation for both PWR and BWR. The only need is to take into account the influence of the specificities of BWR corium (e.g. metal-rich) and/or concrete, and check and validate the possibility of simulation of top cooling by water. Especially top cooling is a matter of further research with high priority and for the BWR plants (except maybe GE design) a flooded cavity is the normal case. Due to possible melt relocation after overflow or melt-through of barriers the possibility of parallel simulation of multiple melt pools is needed (min. 3).
- Ex-vessel melt spreading – dry/wet cavity.** So far ASTEC had no mechanistic modelling for corium spreading in the cavity (dry or under water). Simplified melt spreading models, such as the one developed by KTH, will be implemented in ASTEC V2. For a more detailed simulation the LAVA GRS code will be implemented.

- **Fuel coolant interactions (FCI) – wet cavity**
 - Corium slump and melt jet fragmentation into a water pool. The ASTEC V1 simple model for melt fragmentation during corium slump from the core to the lower plenum may be adapted. KTH is working on the study of debris bed formation, including experiments (DEFOR) and simulation tool development (DEFORSIM). This research program could lead to build a simplified model to predict debris bed characteristics of coolability importance.
 - Steam explosion energetics. So far ASTEC has no model for steam explosion. Investigations of models in FCI mechanistic codes (MC3D, IKEMIX/IKEJET, JASMINE) could be performed in view of the potential to extract simplified models for ASTEC. For ASTEC a detailed steam explosion is up to now out of the scope of the code, but a correlation for the determination of the risk of a steam explosion would be useful.
- **Corium coolability – wet cavity.** Several codes like WABE (by IKE), DECOSIM (by KTH), ICARE/CATHARE V2 (IRSN) calculate two-phase and coolability of debris beds if the characteristics of the beds are known and the existing interfacial exchange relations (e.g. friction laws) are valid. ASTEC V1 also has a simplified model (Lipinski model) for analysis of in-vessel debris bed coolability: the improvement of this model is planned in ASTEC V2. For re-melting of melt pool formation, ICARE2 models for in-vessel melt behaviour may be adapted to ex-vessel case. This also belongs to one of the joint research topics in SARNET-2

D.4 Status of CPA applicability to containment

The ASTEC/CPA module has been assessed in priority on thermal-hydraulic BWR-specific containment phenomena. Since CPA and the GRS containment codes system COCOSYS originate from the same codes RALOC and FIPLOC, which were already applicable to BWR containments, the step forward for ASTEC should be straightforward.

Unlike PWRs, BWR containments surround only the reactor vessel and the main coolant pumps. Due to their smaller containment volume, pressure suppression systems, composed of condensation pool, blowdown pipes and vacuum breakers, have been used to retain the pressure below its limit under all operation and accident conditions. The BWR containment has two main parts: a primary compartment (drywell) and a secondary compartment (wetwell). The drywell encloses the reactor vessel and the primary coolant piping. The wetwell represents a water reservoir at atmospheric pressure and room temperature reduces pressure during any abnormal event in which steam is released either to the drywell or directly to the wetwell by the pressure relief system. When steam escapes from the coolant system, it is initially vented to the drywell where it is subsequently forced through large vent pipes submerged in the suppression pool. The steam condenses, thus mitigating a pressure increase in the containment. In cases where the condensation pool cannot keep the pressure below the design value, the containment venting can be activated or deactivated via the rupture disk to avoid the possibility of containment failure. A special filter and scrubber system is provided to retain the radioactive FPs.

A CPA input dataset for the containment of a German BWR of the series '72 was developed. A scenario of “loss of main heat sink plus loss of pressure limitation” was analysed, using as input tables for the mass and energy flow rates into the containment. Only the thermal-hydraulic behaviour including gas distribution was analysed thereby focussing on the phenomena coupled to the pressure suppression system. CPA results were compared to COCOSYS ones. Figure 8 gives a schematic view of the nodalization of the containment which was used for the ASTEC and COCOSYS calculations. The simulation of vent pipes of the pressure suppression system was done using a simplified model that assumes quasi-steady flow of steam/gas mixtures into a water pool.

The agreement is relatively good at the beginning of the calculation between both codes: the timing of the venting of the wetwell is satisfactorily calculated. Only in the long term phase, beyond ~25000 s after the first wetwell venting, the pressure built up calculated by the codes begins to slightly differ, which leads to a second venting in ASTEC calculation earlier by about 3000s. The differences can be attributed to different heat transfer models at the water pool-atmosphere boundary.

While the general trend of the drywell temperature profiles in both calculations is qualitatively quite similar, ASTEC gives always a lower temperature (Figure 9). The deviations are due to differences in the heat transfer correlations at structures. At such elevated temperatures, much higher than in PWR, additional chemical reactions with bare metallic surfaces and effects of pyrolysed cable insulation and cable fires can

even more contribute to increased temperatures. This first application has shown the applicability of the CPA thermal-hydraulic models to BWR containments.

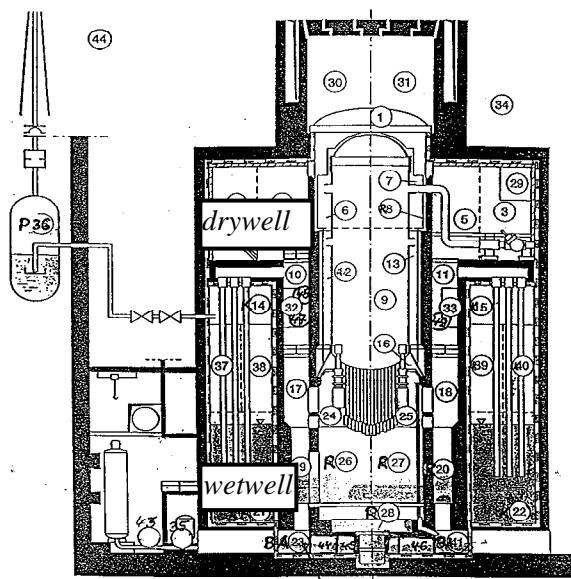


Figure 8: German BWR-72 containment zone model for ASTEC and COCOSYS

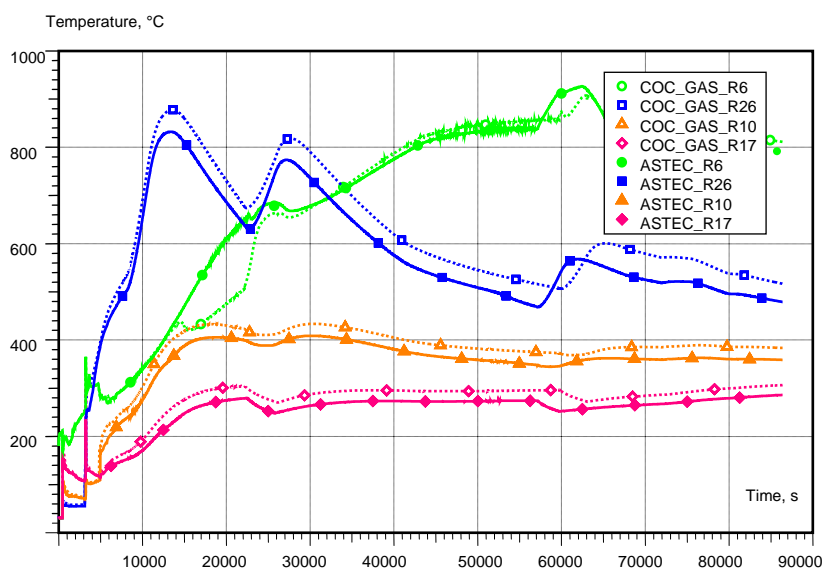


Figure 9: Temperatures in different zones of the drywell (R6, R26, R17, R10) (ASTEC results with filled symbols, COCOSYS results with opaque symbols)

D.5 Conclusion to ASTEC applicability to BWR

The review of needs of model adaptations allowed to conclude that the main efforts must focus on core degradation but most model adaptations to BWR cores can be done through modifications of the existing ICARE2 models. The CORA scoping calculations show they provide a good basis for the envisaged BWR model extensions; the main development need concerns the extension of in-core thermal-hydraulics and the coupling with the structure models.

Concerning the other phenomena, first GRS attempts for the simulation of the RCS thermal-hydraulics with the CESAR module were performed successfully. The benchmark between CPA and COCOSYS calculations for an accident sequence in a German BWR showed the CPA ability to model the containment thermal-hydraulic phenomena. For ex-vessel simulation the most-risk relevant process is the steam explosion in case of core slump into a wet cavity: a correlation could be implemented to determine the occurrence of steam explosion. All FP models seem already applicable, except containment iodine modelling, the validity of which must be checked for high temperatures up to 1000K.

E. ASTEC ADAPTATION TO CANDU REACTORS

In order to prepare the adaptation of the ASTEC code to CANDU SA analysis, INR has performed in parallel two kinds of activities: analyses of the ASTEC models, followed by CANDU exploratory calculations; ranking the needs of adaptations and preparing detailed specifications.

E.1 Specificities of CANDU type reactors

The CANDU core consists of several hundred horizontal fuel channels placed in a large cylindrical calandria vessel surrounded by a shield tank with a large volume. This tank is penetrated by several hundred fuel channel. Each fuel channel consists of an internal pressure tube, containing the fuel and the hot pressurized D₂O primary coolant and an external calandria tube separated from the pressure tube by an insulating gas-filled annulus. Instead of the use of a large pressure vessel, a CANDU reactor uses the pressure tube concept: the fuel is loaded into the horizontal zirconium alloy pressure tubes that pass through the calandria that is filled with heavy water moderator in a low pressure circuit.

High-pressure D₂O is used as the fuel coolant. The calandria vessel contains cool low-pressure D₂O moderator that surrounds each fuel channel. The primary coolant is distributed amongst the fuel channels by common headers and individual feeder pipes. The fuel can be natural or slightly enriched UO₂. At each end of the calandria, an end-shield contains carbon steel balls and demineralised water in order to provide both radiation shielding and cooling.

A complete coolant circuit involves two fuel tubes and two circulation loops. The heavy water enters the reactor at a temperature of about 266 °C and exits at 310 °C. It passes from the reactor to a header, i.e. a junction chamber for the coolant tubes, and then to an inverted U-tube steam generator where steam is produced and carried to the turbines. The coolant then returns to the reactor, passing in the opposite direction through an adjacent fuel tube, where it is heated again before flowing to a second steam generator. The pressuriser performs the same function in the CANDU system as it does in the PWR. Although most of the heat from the fuel is carried away by the heavy water coolant, some energy is deposited in the heavy water moderator. This is removed by the moderator coolant loop. The typical moderator mass is greater than 220 ton. In postulated severe accident sequences in a CANDU reactor — as for example, a loss of coolant, plus unavailability of the emergency core-cooling system — the separately cooled moderator provides an effective heat sink. The moderator has an active heat removal capacity of about 4.4% thermal power, which is sufficient to remove decay heat shortly after shutdown. In case of simultaneous pressure tube/calandria tube rupture, rupture disks in the calandria adequately discharge the heavy water flow into the containment.

The calandria assembly is supported by the calandria vault which is filled with light water. This water plays mainly the role of radiation protection, but it can also act as a passive heat sink. In case of an accident that leads to the corium collection in the calandria bottom, the vault water provides the ex-calandria cooling. The end-shields and calandria vault cooling are provided by means of a system: if the moderator cooling fails, then this cooling system can delay melt relocation into the calandria vault. The debris will be contained within the calandria as long as it remains cooled on the outside by the shield-tank water. The light water mass in the reactor vault is about 450 tons: this large quantity involves an important reserve to delay the progression of the degrading core process during a severe accident in CANDU type reactor. The slow boil-off of the shield tank water will delay the failure of the calandria vessel for about a day.

E.2 Exploratory calculations for CANDU severe accident phenomena

Taking into account the current impossibility for the ASTEC simulation of the core degradation process, the investigations were limited at:

- Transport and deposition of FP in the primary heat system (PHT) by SOPHAEROS module;
- Thermal hydraulics and FP distributions in containment (CPA);
- Iodine phenomena in containment (CPA-IODE);
- PHT thermal-hydraulics (CESAR);
- Coupled calculation of FP transport and behaviour (SOPHAEROS-CPA-IODE).

A postulated severe accident was chosen: loss of coolant, unavailability of the emergency core-cooling system and loss of moderator system. The injection of FP into the PHT was postulated based on the FP inventory as calculated by the ORIGEN code. The release fractions were imposed taking into account literature data and physical hypotheses such as volatility class, temperature, etc. This led to a total amount of FPs released into CANDU PHT of 8.2 kg. Regarding the interval time and release profile, the following very

simple assumption was done: release between 150 and 155 s after the accident initiation, in accordance with the fuel cladding failure by temperature increase.

For PHT system, only ½ of the circuit (Figure 10) was simulated: 190 horizontal fuel channels connected to 190 horizontal out-feeders, then through vertical feeders to the outlet-header; the circuit continues from the outlet-header with a riser and then with the steam generator and a pump. After this pump, the circuit was broken; in this point the FP are transferred to containment. The containment model consists of 12 rooms and 14 junctions (Figure 10). Steam and FPs were injected from PHT (through broken pipe) in the room CV760.

The data related to the nodes definitions, temperatures and pressure conditions were chosen as possible as actual data from CANDU loss of coolant accident sequence. Temperature and pressure conditions in the time of the accident were calculated by external codes.

Based on these initial conditions and assumptions the FP deposition in the PHT and consequently the transfer to the containment trough the pipe break were calculated by SOPHAEROS module. For the point of view of the source term the most important isotopes are: Cs-134, Cs-135, Cs-137, Sr-89, Sr-90, I-129 and I-131. Taking into account the fraction of the isotope in the total mass for each element the total masses, for each important isotope, transferred to the containment, are presented in Table 2.

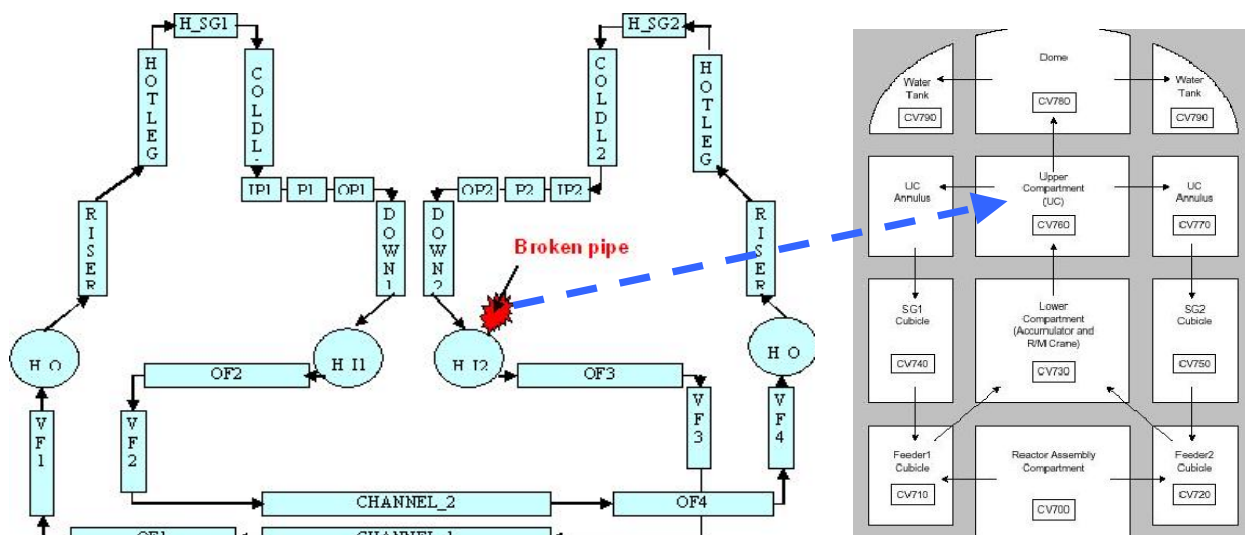


Figure 10 : ASTEC geometrical models for ½ CANDU PHT and containment

Table 2: Transferred masses for important radioactive isotopes of Cs, Sr and I to the containment

	Cs-134	Cs-135	Cs-137	Sr-89	Sr-90	I-129	I-131
Transferred mass in containment [g]	5.88	14.27	203.14	2.20	10.07	15.60	3.28
Total mass per element [g]	223.29			12.27		18.88	

We can observe that only a reduced amount of radioactive Sr and I is transferred to the containment, due to the complexity of CANDU circuit (the total length exceeds 110 m for a complete loop, large number of feeders/pipes, small diameters of pipes, etc.).

The second SOPHAEROS-CPA-IODE calculation aimed at evaluating the distribution of the radioactive isotopes in the containment rooms and for different hosts: sump water, wet surfaces, aerosols, painted walls, etc... The dousing safety system that is a powerful pressure suppression system in a CANDU reactor was taken into account (water source as the elevated tank around building dome with a capacity of 1560 m³ and a flow rate of 4500 kg/s). The dousing operation is performed by 6 spray heads each with 2 valves in series, which turn on when building pressure reaches 14 kPa and turns off if pressure falls to 7 kPa.

The calculation of the pressure evolution during the accident shows that a maximum pressure of about 167 kPa is reached rapidly, but the spray system is able to reduce the pressure at relative normal values. After consuming water from the dome tank the recirculation mode is started. The maximum pressure values are in a fairly agreement with literature results [8].

From the point of view of FP containment distributions Cesium and Strontium have a similar behaviour: the transfer from the injection region (CV760) to the other region of the containment occurred

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relatively fast (after 200-400 s the majority of Cs is transferred into CV770); in the first stage Cs is deposited mainly in water-surface (water condensed on walls) and gas-surface, but at about 440 s as an effect of the washing the content of water surface is transferred in the sump water. The distributions were calculated for all isotopes, but for simplicity only the Cesium-137 distributions 2000s after the starting of the accident are given here: 1% on water surfaces, 36% on water, 43% on gas surfaces, and 20% as aerosols.

The Iodine species distribution is calculated in details with the IODE module. It should be noticed that, after 2000s, practically all the mass of iodine is deposited in the sump water as I₂, only a small amount being deposited on other hosts or in other chemical forms.

In parallel, thermal-hydraulic CESAR stand-alone exploratory calculations were performed. Despite the complexity of the CANDU circuit (e.g. bi-directional flows, which means that fluid goes through the core twice before it gets back to where it started), CESAR seems to be applicable to CANDU PHT analysis. However, the final results of code-to-code comparison calculations, which are in progress, will offer a final conclusion.

E.3 ASTEC main needs for CANDU adaptation

The above exploratory calculations have shown that the only need of ASTEC models adaptation concerns the core degradation models. During moderator boil-off, the channels are expected to sag under heat-up and to break up to form debris. A sagging channel eventually contacts the lower channel and if the lower channel is also uncovered it is expected to sag under its own weight as well as that of the supported channel. This process would continue, as more channels are uncovered. As sagging increases, it is expected that the channel segments separate near the bundle junctions. A "suspended" debris bed is thus formed, which moves downward with the decreasing moderator level. The submerged channels will be able to support a finite number of channels after which they are also expected to fail. The loading on the submerged channels increases with the accumulation of debris from top channels, thereby leading to progressive failure of the lower channels and ultimately resulting in the collapse of the core into the moderator pool at the bottom of the calandria.

This analysis has underlined the main needs of model adaptations (for the other core phenomena, most ICARE2 models seems already applicable):

- Account for the horizontal geometry of the core channels (adaptation of CESAR),
- Heat transfers between fuel bundles, pressure tubes and moderator,
- Heat transfer between pressure tube and external moderator fluid (CESAR model),
- Oxidation (internal and external) of pressure tubes,
- Ballooning and sagging of fuel channels,
- 3D description of the debris bed after sagging of fuel channels (the 2D ICARE2 models will serve as basis for this investigation).

For thermal-hydraulics in the primary circuit, its complexity will probably need some minor adaptations of the CESAR and SOPHAEROS modules, such as for instance D₂O as coolant instead of H₂O.

All these conclusions are in total agreement with those of the BARC Indian specialists as underlined in the ERMSAR-07 conference [9].

F. CONCLUSION

After 4 years and a half of SARNET activities, the ASTEC V1 European integral code is fully applicable to all severe accident scenarios, at power operation, in Gen.II PWR and VVER. For the latter, the only small improvement will concern the more detailed modelling of VVER-440 canisters and "two-cores" configuration.

In the frame of the new series of ASTEC V2 versions under preparation, the extension to Gen.III PWR or VVER is under way, EPR in first priority but also all designs relying on In-Vessel Melt Retention. Both capabilities will be included in the 1st version V2.0 planned in march 2009.

The main efforts have now to focus on the adaptation to the BWR and CANDU reactors present in Europe, which will be important objectives in SARNET2. The ranking of these model adaptations clearly concluded for both types of reactors on the priority on core degradation processes because all exploratory calculations of other phenomena proved the ASTEC V1 global applicability. For BWR, the DIVA exploratory calculations are very promising, showing that only adaptations of existing models will be necessary. For CANDU, the INR work will be supported by the reinforcement of the IRSN-BARC (India) collaboration in the next years.

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