

WORKSHOP ON EVALUATION OF UNCERTAINTIES IN RELATION TO SEVERE ACCIDENTS AND LEVEL 2 PROBABILISTIC SAFETY ANALYSIS

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SENSITIVITY STUDIES OF MAIN UNCERTAIN CORE DEGRADATION PARAMETERS ON SEVERE ACCIDENT CONSEQUENCES

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Abstract

The analysis of the main uncertain core degradation parameters impact on the consequences of a severe accident has been performed. This investigation has been performed to assess severe accident codes weaknesses and to identify the potential needs of separate effect tests to reduce uncertainties regarding the more sensitive core degradation parameters. On this purpose, a sensitivity study of the hydrogen production (total amount and flow rate), of fission product (FP) release and of the corium composition, to some degradation modelling uncertainties, has been realised with ICARE2 and ASTEC codes, taking into account a particular accidental sequence (loss of feed water) for a French 900MWe pressurized water reactor. The results of this study indicate that, depending on the different set of parameters retained in the modelling, the mass of hydrogen produced can range from 320 to 580 kg, whereas the flow rate can vary between 0,1 to 0,4 kg.s⁻¹. Moreover, the FP release increases significantly when the amount of dissolved fuel increases. Furthermore, the uncertainty regarding the Molybdenum release (regarding also the other semi-volatile FP) induces a variation of the iodine gaseous release and of the caesium airborne from the primary circuit break. Finally, the corium composition (ratio between oxides and metal) could vary from a factor 3.

1. Introduction

Evaluating the consequences of hypothetical severe accidents in pressurised water reactors produces variable results that depend on the modelling and calculation code used (ICARE2, ASTEC, MELCOR...), as well as the input parameters applied in the modelling [1] [2] (regarding for example TMI-2 accidental sequence) and [3]. The study discussed in this document aims at evaluating the impact of uncertain parameters related to phenomena occurring during the reactor core degradation phase upon the evaluation of the hydrogen production (kinetics and total quantity produced), the fission product releases and the composition of corium in the vessel bottom.

The research carried out does not intend to compile a comprehensive inventory of uncertain parameters on core degradation, but estimate the predictive gain when simulating a severe accident that could improve knowledge of certain phenomena through additional separate effect tests programmes for example.

The study was restricted to only one accidental sequence that was simulated using the calculation codes ICARE2 [4] and ASTEC V1 [5].

2. Study approach

The transient chosen for study is provoked by a feed water loss of the steam generators, combined with the unavailability of the safety injection. The chronology of the main events is described below (indicative times capable of varying depending on the calculation code):

- At 0 second: failure of the steam generator feed water system,
- At 28 seconds: automatic reactor shutdown after the “low level alarm” of the steam generators has been triggered,
- Approximately 2,500 seconds later: the Pressurizer Operated Relief Valves (PORV; “SEBIM” in French nomenclature) are opened by the operators,
- At about 3,800 seconds: the fuel rods begin to be uncovered (beginning of cladding oxidation),
- About 200 seconds later: the accumulators begin to discharge (the primary system pressure is now less than 42 bar) provoking a core reflood and stopping the core from heating,
- At approximately 7,900 seconds: the accumulators are isolated (the primary system pressure is less than 15 bar),
- At about 8,500 seconds: the core is uncovered a second time, with core heating and oxidation starting up again

Based on the ICARE2 and ASTEC V1 codes and within the scope of evaluating phenomena capable of occurring during this sequence, different modelling systems and related sets of parameters were chosen, distributed in terms of the limits of their range of uncertainties. Such choices were made based on:

- Analysis of test results (as Phebus FP program) simulating core degradation,
- Analysis of results from studies on accidental reactor transients,
- Appraisal of phenomena backed by engineering expertise.

The study involved two phases. The first phase was carried out using the stand-alone version of the ICARE2 code (with results obtained using the CATHARE code serving as thermal hydraulics boundary conditions), which made it possible to analyse the impact of variations in the chosen parameters upon the hydrogen production and fission product releases.

In the second phase, the sets of parameters leading to the minimum and maximum consequences in calculations performed by ICARE2 were then chosen for calculations performed with the ASTEC V1 code. This made it possible to model the chemistry and transport of fission product up to the primary system break, as well as the relocation of corium in the vessel bottom and the vessel failure.

In the first phase of the study performed with the ICARE2 code, the parameters having been assessed in sensitivity studies are related to the modelling of the:

- residual power during the transient,
- oxidation kinetics of the Zircaloy cladding, (from [6] and [7])
- protective effect of the zirconia layer,
- dissolution limit of the fuel and oxidised cladding by liquid Zircaloy,
- oxidation of the U-O-Zr mixture,
- criteria concerning the loss of oxidised cladding integrity, ([8]),
- flow velocity of materials during relocation (standard option and alternative value given from Phebus FP experiment analysis),
- fuel and cladding relocation; function of the solidus and liquidus temperatures (when the fuel begins to melt, its liquid fraction is transferred in the cladding, with this liquid fraction then able to flow down). The analyses of the Phebus experiments outlined that large uncertainties remained on fuel rod relocation temperature even if it has been demonstrated that values are largely below the pure UO_2 melting temperature [9].

Table 1 illustrates the different options under consideration in this study for each of the parameters, with a detailed description provided in reference [10].

Following validation work performed on the ICARE2 code [11], a preliminary calculation qualified as a “standard case” was first carried out using the recommended default options.

Figure 1. a) Variability of oxidation kinetics correlation (from [3]) and b) solubility limit of uranium in the (U, Zr, O) mixture for the UO_2 /liquid zircaloy interaction

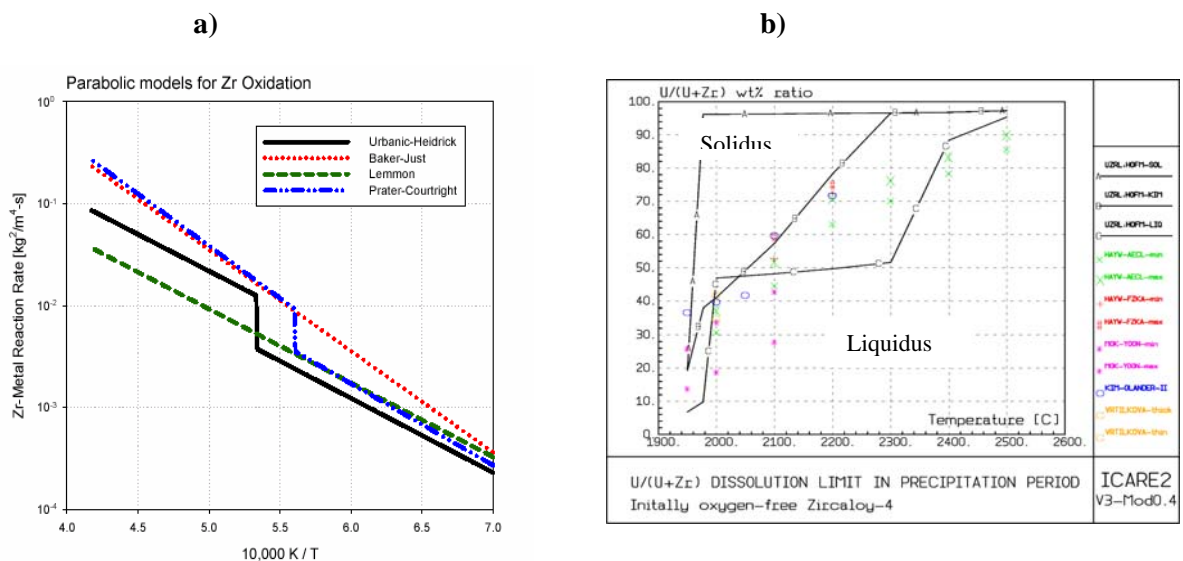


Table 1. Different studied parameters – Different possible options

Possible options	1 – base case	2	3
Residual power	Decrease of values versus time given by a specific table (in the hard ware)	values computed by the code versus fission product release	
Oxidation kinetics for Zry-steam reaction (in the namely ZROX module) Fig 1. a)	Urbanick-Heidrick Correlation with zirconia layer growth	Prater- Courtright correlation with zirconia layer growth	Previous Correlation with oxygen mass gain
Zirconia layer protection effet	NO	YES (specific ICARE2 code option PROTECTI)	
Solubility limit of uranium in the (U, Zr, O) mixture (UZRL module)	Liquidus temperature (fig.1 b)	Solidus temperature	
Oxidation of mixture	NO	YES	
Cladding Oxide Shell criteria (from [8])	$T > 2260 \text{ K}$ and $e_{\text{ZrO}_2} < 160 \mu\text{m}$ or $T > 2280 \text{ K}$ and $e_{\text{ZrO}_2} < 200 \mu\text{m}$ or $T > 2340 \text{ K}$ and $e_{\text{ZrO}_2} < 220 \mu\text{m}$ or $T > 2380 \text{ K}$ and $e_{\text{ZrO}_2} < 240 \mu\text{m}$ or $T > 2450 \text{ K}$ and $e_{\text{ZrO}_2} < 300 \mu\text{m}$	$T_{\text{clad}} > 2600 \text{ K}$ if $e_{\text{ZrO}_2} < 250 \mu\text{m}$ or $T_{\text{clad}} > 2700 \text{ K}$	
Candling velocity	1 cm.s^{-1}	60 cm.s^{-1}	
Relocation temperature for UO_2 and ZrO_2	$T_{\text{solidus}} = 2550\text{K}$ $T_{\text{liquidus}} = 2650\text{K}$	$T_{\text{solidus}} = 2800\text{K}$ $T_{\text{liquidus}} = 2850\text{K}$	

The details of this study are given in the document [10]. The following section gives the mains results as regard hydrogen generation, fission product release and corium composition in the lower head vessel during a severe accident sequence.

3. Impacts on hydrogen generation

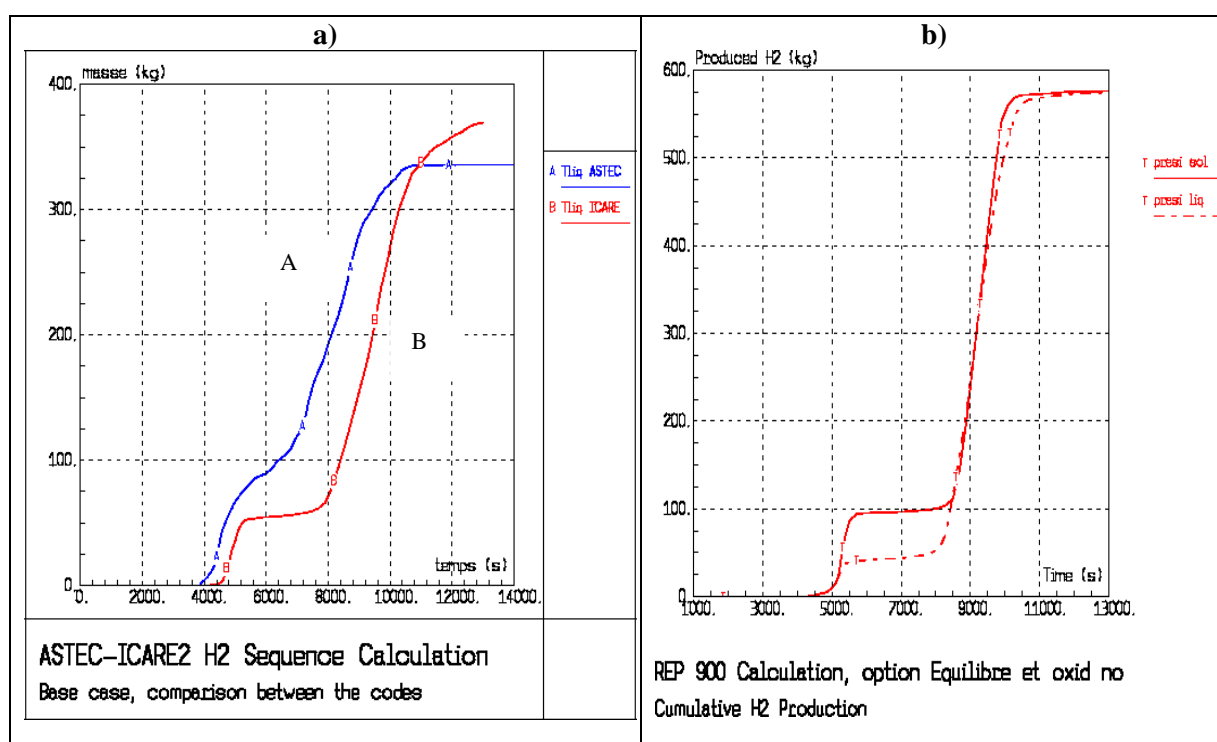
Generally speaking, the results of the first phase of the study obtained using the ICARE2 code revealed that a significant variability in the amount of hydrogen produced during the first core uncovering (before the water injection from the accumulators) could be observed depending on the option chosen for the dissolution limit of UO_2 by liquid Zircaloy¹; during this transient phase, the hydrogen production can vary by as much as 100% in certain cases. However, the total quantity of hydrogen produced at 11,000 seconds² is only slightly affected by the dissolution limit applied to calculations.

¹ Both extreme options – liquidus and solidus temperatures – were retained for each set of hypotheses.

² Beyond this moment, the results of the ICARE2 calculations obtained in a stand-alone mode were not used as corium normally flows to the bottom of the vessel, which is not possible when applying the boundary conditions provided by the CATHARE code.

Furthermore, among the different hypotheses retained for the calculations performed to take into account uncertainties and the lack of knowledge of certain degradation parameters, the minimum hydrogen production was calculated with a U-O-Zr mixture relocation velocity equivalent to 60 cm.s^{-1} . In this case, the amount of hydrogen produced at 11,000 seconds was equivalent to about 320 kg. In addition, the maximum hydrogen production was calculated by simulating the oxidation of materials during the meltdown, as well as the oxidation kinetics limited by the gain in the oxygen mass only. Based on these hypotheses, 580 kg of hydrogen was produced at about 10,000 seconds. Figure 2 illustrates the hydrogen production for both the standard case (figure 2.b– curve B) and the case where the hydrogen production is maximum (figure 2.a – continuous line).

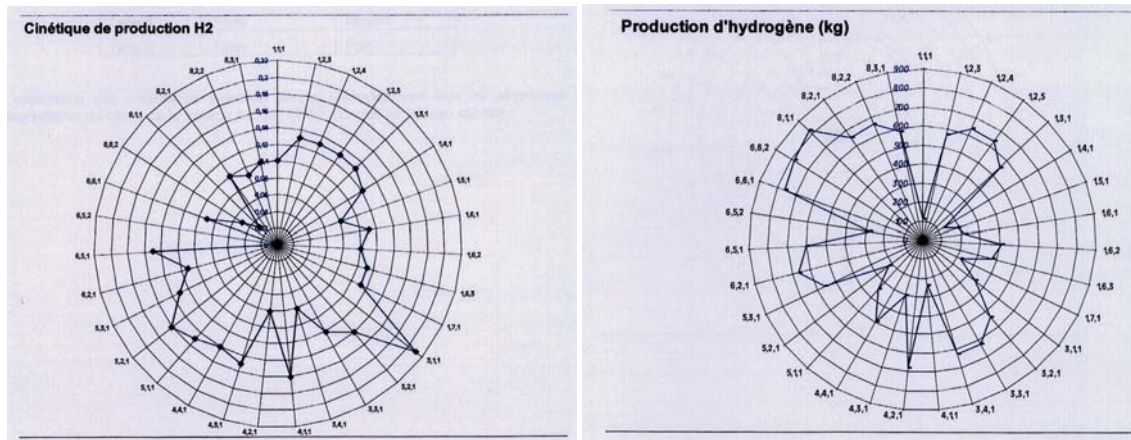
Figure 2. Evaluation of the hydrogen generation a) base case and b) with option giving maximum H2 release



Moreover, the results indicated that uncertain degradation parameters not only have an impact on the total quantity of hydrogen produced, but also on the hydrogen production kinetics, respectively equivalent to 0.1 kg.s^{-1} and 0.4 kg.s^{-1} for the above-mentioned calculations. Figure 2 also illustrates the hydrogen production obtained by the ASTEC V1 code (figure 2.a – curve A), which is slightly different from that obtained by the ICARE2 code in the standard case, despite their very similar modelling options. This difference seems to be mainly due to the differences observed on core water level at the beginning of accumulators discharge (perhaps due to different input decks) and to the reflooding phase modelling different between these two codes.

As regard hydrogen generation in terms of mass flow rate (figure 3.a) and total amount (figure 3.b), the results variability of these sensitivity calculations are in the same range of those obtained in the calculations of others accidental sequences [12] which have been performed in the framework of sensitivity study on the impact of H_2 auto catalytic recombiner coupons on the hydrogen risk in the containment of 900 MWe PWR (not published internal IRSN results). So, uncertainties should be reduced to increase the accuracy on the results of the Nuclear Power Plants calculations.

Figure 3. Hydrogen generation resulting calculations of others accidental sequences a) mass flow rate and b) total amount produced



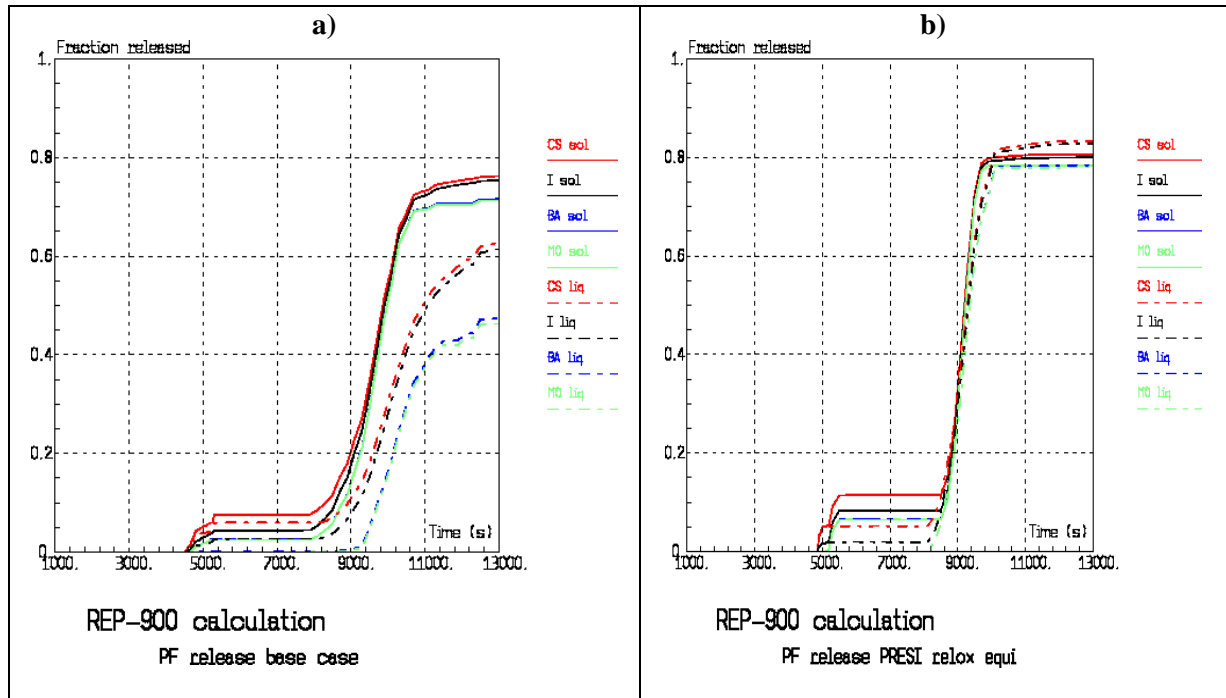
a) scale : 0 to 0.22 kg/s - steps 0.02 kg/s)

b) graph scale : 0 to 900 kg - steps 100 kg)

4. Impacts on the Fission Product release

In order to integrate uncertainties regarding the release of semi-volatile fission product into calculations obtained using the ICARE2 code, changes were made to the ICARE2 code release model allowing for the release of semi-volatile fission products when the fuel is liquefied. It therefore appeared that the release of fission products – kinetics in particular – and the quantity of fuel liquefied during the transient are strongly correlated. Therefore, the hypothesis involving the possible dissolution up to the solidus temperature generally leads to an increased release both in terms of the kinetics and the integrated quantity (figure 4.a).

Figure 4. Evaluation of the fission product release with the ICARE2 code a) base case and b) with options giving maximum release



Independent of the hypothesis chosen in terms of the dissolution limit, the set of conservative hypotheses (maximum hydrogen production) used to simulate both the oxidation of materials during meltdown and the oxidation kinetics of the U-O-Zr mixture limited only by the gain in the oxygen mass and oxygen concentration (limited by the thermodynamic equilibrium) produces a fraction of released fission products equivalent to about 0.8. (figure 4.b).

Evaluation of the transport and release of fission products reaching at the break were obtained using the ASTEC V1 code [13] and revealed that the options as regard 1) the dissolution limit set at the liquidus temperature and 2) the rapid meltdown velocity of materials (60 cm/s) induced an increased release of molybdenum. When molybdenum combines with caesium and rubidium, these compounds are therefore available in lesser quantities to combine with iodine. For this reason, iodine combines with other compounds to form the organic iodide HI and gaseous caesium iodide, released through the breach; this represents a very penalising result as regard the source term.

5. Impacts on the corium composition

The results of ASTEC V1 calculations revealed a great variability in the composition of corium ; the proportion between the oxide and metallic phases was capable of varying by a factor of 3 depending on the calculation options.

Figure 5 illustrates the variations in the mass of fuel, water, metal and zirconia found at the vessel bottom during the transient (curves A, B, C and D respectively). The figure 5.a provides the results of ASTEC calculations performed with a dissolution threshold temperature equal to the liquidus temperature, whereas the results illustrated in the figure 5.b were obtained with a threshold temperature equal to the solidus temperature.

Generally speaking, taking into account the accidental sequence under investigation and the modelling assumptions applied to the ASTEC V1 calculations, the composition of the corium flowing to the vessel bottom mainly results from the degree of oxidation of the core materials and the quantity of fuel dissolved by liquid Zircaloy. The mass of relocated corium in the vessel bottom for the same transient is therefore sensitive to the fuel liquefaction temperature. This particularly applies to calculations with a fuel dissolution limit equal to the liquidus temperature of the U-O-Zr mixture seeing that most of the fuel liquefaction in this case is a result of the fuel melting rather than its dissolution

More specifically, with a low degree of oxidation in the fuel rod cladding, the relocation of a large mass of metallic materials (at low melting point) is possible before the relocation of the fuel and the oxidised cladding occurs. In such a case, the corium found in the vessel bottom is rich in metallic materials; in the opposite case, corium contains a greater proportion of oxides (UO_2 and ZrO_2) when the cladding is greatly oxidised before the relocation of materials. Furthermore, when a considerable proportion of the fuel is liquefied by dissolution (solidus temperature for dissolution limit taking into account the increased dissolution due to irradiation), corium relocates to the vessel bottom carrying a great proportion of the dissolved metal. There is no segregation between the oxide and metallic phases in this case. Corium is not depleted or enriched in metal seeing that it flows at the same time. However, when the mass of dissolved UO_2 is minor (liquidus dissolution limit), the liquid and oxide phases are separated during the relocation of corium that is rich in metal compound.

Figure 6 gives a representation of the temperature map and materials distribution inside the reactor just after the lower head rupture, for both the two options as regard the solubility limit a) liquidus ($t=15,900$ seconds) and b) solidus temperature ($t=13,400$ seconds). For all the calculations performed during this study the time of the Lower Head Vessel Rupture can vary of about 1 hour (from $t=12,900$ to $17,200$ seconds) that could be important for the mitigation of the accident (time necessary to recover feed water to prevent LHVR).

Figure 5. Evaluation of the corium composition in the Lower Head Vessel with the ASTEC V1 code calculations

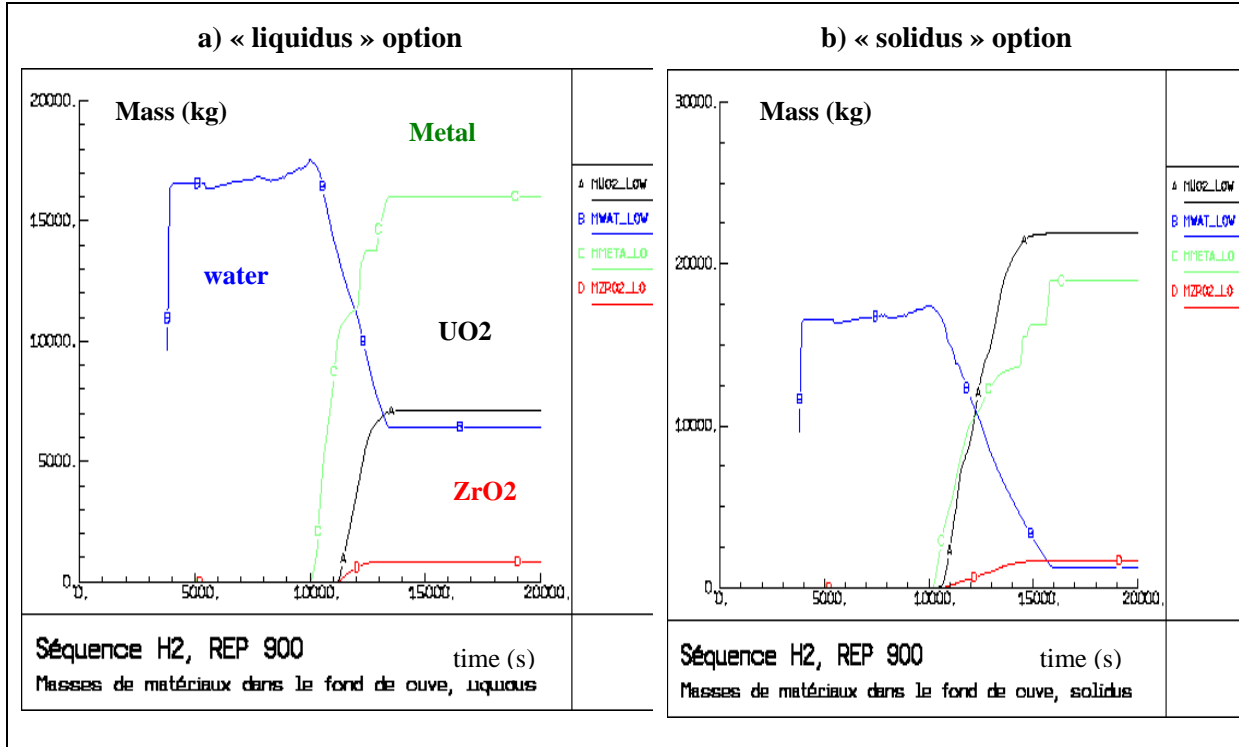
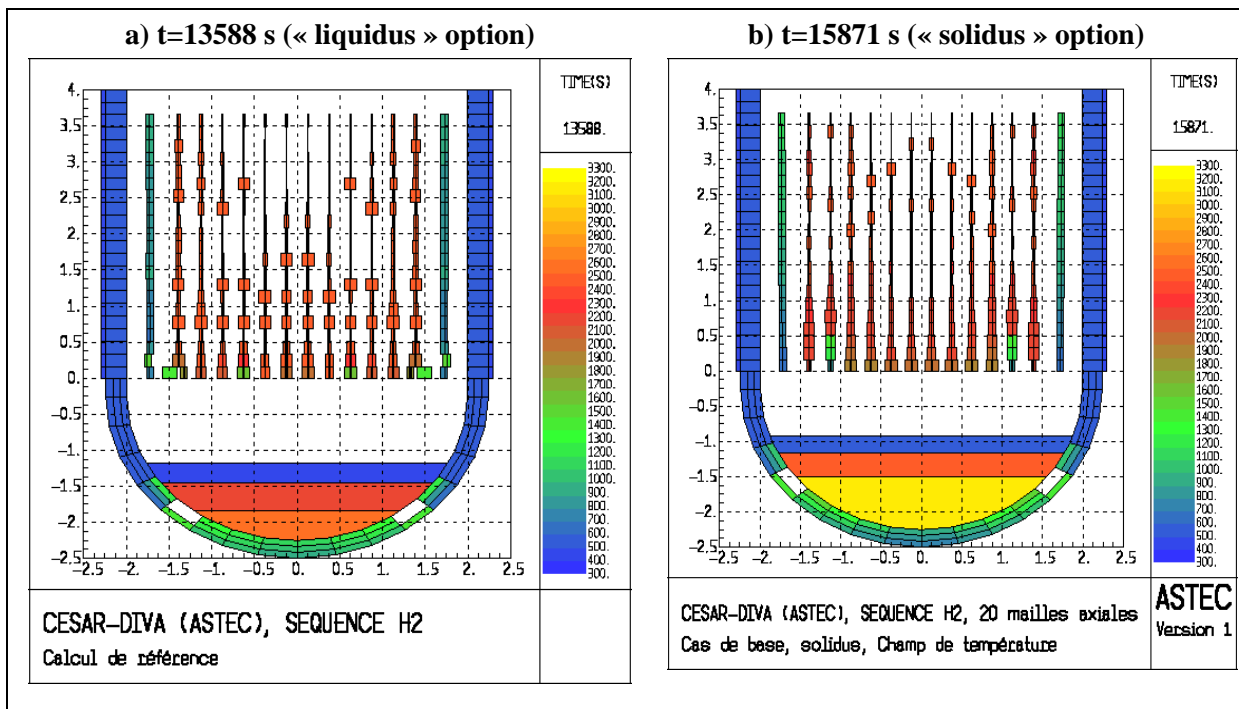


Figure 6. Configuration of the reactor vessel just before or after the rupture with the ASTEC V1 code calculations



6. Main conclusion and lessons learned

The results of this study indicate that, depending on the different set of core degradation parameters retained in the modelling, the mass of hydrogen produced can range from 320 to 580 kg (almost factor of 2), whereas the flow rate can vary between 0,1 to 0,4 kg.s⁻¹ (factor of 4). The kinetics and the total amount of hydrogen generation are one of the important for H₂ autocatalytic recombiner coupons design. Moreover, the FP release increases significantly when the amount of dissolved fuel increases. Furthermore, the uncertainty regarding the Molybdenum release (regarding also the other semi-volatile PF) induces a variation of the iodine gaseous release and of the caesium airborne from the primary circuit break. Finally, the total mass of the corium in the lower head vessel at the time or just after his rupture can range from 25 to 43 10³ kg (about factor of 2) with a ratio between oxides and metal (corium composition) could vary from a factor 3. This information is the one of the most important issue as regard possible stratification that could influence the rupture of the vessel, although these phenomena (stratification and vessel rupture) are still to be improved. An other issue is the temperature of the corium which is important during the degradation processes of the accident, coolability of the molten pool within the core and, in case of crust failure, the consequence of the contact of the corium with the vessel lower head. Nevertheless, uncertainties remain on the corium overheating which is more difficult to evaluate. Moreover, this sensitivity study shows that knowledge improvement of phenomena such as the U-O-Zr mixture oxidation at high temperatures (currently the same correlation than for intact Zircaloy structures are used) and, fuel and oxidised cladding dissolution by liquid Zircaloy (more precisely the definition of the dissolution limit) could allow reducing significantly the results variability.

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