Method implemented by the IRSN for the evaluation of uncertainties in level 2 PSA. Some examples.

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Abstract

During the last few years, the IRSN has developed a level 2 PSA for French 900 MWe PWRs. The last version of the study has been completed in 2003. This version is going to be updated to take into account hydrogen recombiners installation and modifications envisaged for third decennial visit of these plants.
The objective of this paper is to comment the methodological approach of the IRSN for uncertainties assessment in level 2 PSA, with examples for accident progression event tree and releases assessment.

1. Introduction

During the last few years, the IRSN has developed a level 2 PSA for French 900 MWe PWR. The last version of the study has been completed at the end of 2003 and an updated version is in progress.

The realization of the study provides to the IRSN a framework for discussing many issues of severe accident research activity. Results are used to support analysis of severe accident safety issues. Moreover, the knowledge on severe accident scenarios, coming from studies undertaken in support of level 2 PSA, is useful to help the experts involved into the emergency response analysis teams [3].

General information on level 2 PSA approach at the IRSN and on the uncertainties assessment has been provided in reference [1,2]. This paper is a complement of these references and comments the application of the followed method for uncertainties evaluation in the accident progression event tree (APET) and for releases assessment.

2. General objectives

A level 2 PSA of 900 MW PWRs is developed by the IRSN with the following objectives:

- to contribute to reactors safety level assessment,
- to estimate the benefits of accident management procedures and guides to reactor safety,
- to provide more quantitative judgment elements about the advantages of any modifications to reactor design or operation,
- to acquire quantitative knowledge for emergency management teams and tools,
- to help in the definition of research and development programs in the severe accidents field

Learning from the detailed studies achieved for 900 MWe PWRs level 2 PSA are also extended to other French plants (1300 or 1400 MWe PWRs).

The study progresses according to the following steps:

- a preliminary internal version (1.0), based on the IRSN level 1 PSA published in 1990 has been completed in 2000,
- a version 1.1, achieved in 2003, is a revision of the preliminary version and integrates several improvements (detailed physical calculations of containment failure, assessment of uncertainties of radioactive releases …),
- a version 2, based on the updated level 1 PSA, which includes shutdown states of reactor, is planned for beginning of 2006,
- a version 3, based on the updated level 1 PSA (according to envisaged plants modifications at the third decennial visit) is planned for end 2006.

In France, “reference PSAs” have to be provided by utilities. In 2004, results of EDF level PSA 2 have been examined by the IRSN in the framework of safety review before third decennial visit of 900 MWe PWRs.

From version 2, the study takes into account hydrogen recombiners, which are going to be installed by the French Utility.

Specifications of a level 2 PSA for French 1300 MWe PWR series that could be available before the third decennial visit of these reactors are in preparation at the IRSN.

3. General comment on uncertainties assessment

The general methodology followed by the IRSN has been initially based on those developed in the NUREG-1150 study, and includes four main steps: (1) binning of level 1 sequences into Plant Damage States (PDS) according to interface variables, (2) representation of important severe accident events in a Accident Progression Event Tree (APET), (3) binning of level 2 sequences into Release Categories and (4) assessment of radioactive releases into the environment for each release category.
Some specificities of the study have been described in reference [1] (detailed interface, quantification of uncertainties in APET, a specific model for containment leakage through reactor building penetrations, radioactive releases calculation model, quantification software KANT …).

Before comments on quantification of uncertainties for physical phenomena in APET and release calculations, the Table 1 gives a short overview of uncertainties treatment in the 900 MWe level 2 PSA. This list of uncertainties is mainly issued from exchanges in the SARNET PSA2 working group [4].

Table 1 – Treatment of uncertainties in the IRSN level 2 PSA

Type of uncertainties	Approach
Level 1 uncertainties (initiating event, component reliability, human reliability)	These uncertainties are available in level 1 PSA but, up to now, have not been considered relevant for decision-making. They are not propagated in level 2 PSA.
Uncertainties (in sense of approximation) due to the binning of the level 1 sequences in Plant Damage State (PDS) in interface and definition of a representative system transient for each PDS	An effort is made to have a detailed interface (> 100 PDS) in order to minimize uncertainties due to the binning of level 1 sequences. As many system transient calculations as possible have been carried out. Nevertheless, there are obviously approximations in the definition of these transients associated to a PDS. For example, level 1 PSA does not provide the failure instant of component nor precisely break size for pipes. Conservative assumptions have to be made here. Uncertainties due to thermal hydraulics phenomena are not taken into account because they are supposed to be weak enough.
Uncertainties on the probabilities and instants of stochastic events (human actions, CHRS repairing or failure, Safety injection repairing and failure …)	Human actions are represented with a specific model that takes into account available delay for operators, difficulties of the scenario and severe accident management guide. The level 2 PSA APET model is static, and all combinations of situation cannot be evaluated. Effort is made to generate and to evaluate as many situations as possible. Nevertheless, quantification of uncertainties cannot be done here. Only a probabilistic dynamic method could solve this issue.
Uncertainties on physical phenomena assessment in APET (lack of knowledge, approximation in modeling)	A particular effort has been made in the study for each severe accident phenomenon. This point is described in §4
Uncertainties (in sense of approximation) related to the binning of level 2 sequences in Release Categories	Effort is made to generate precisely defined release categories (> 1000). Delay before release and kinetics of release are supposed to be an important issue for applications of PSA2 results to acquire quantitative knowledge for emergency management teams and tools
Uncertainties (lack of knowledge) for release assessment	A simplified model for release calculation that allows uncertainties assessment on key parameters has been developed.

4. Quantification of physical phenomena with uncertainties in APET

4.1. Method

The different physical phenomena that might occur during a severe accident are explicitly represented in the APET.

They have been organized in « physical models » so that :

- each physical model represents a set of physical phenomena tightly coupled because of feedback processes or temporal dependencies,
- two separate physical models are linked by a limited number of variables which can be transmitted by the APET.

Application of these principles has led to define the following physical models :

- accident progression before core degradation (BCD model)
- accident progression during core degradation (DCD model)
- induced SGTR in case of core melting with a pressurized RCS (I SGTR model)
- hydrogen combustion during core degradation (H2 model)
- advanced core degradation (ACD model)

- in-vessel steam explosion and mechanical consequences (IVE model)
- ex-vessel steam explosion and mechanical consequences (EVE model)
- direct containment heating (DCH model)
- accident progression after vessel rupture (melt-corium concrete interaction) (MCCI model)
- combustion during MCCI (H2 –CO model)
- containment mechanical behavior (CMB model).

The concatenation of the different physical models done in the APET, as illustrated in Figure 1, represents the accident progression from the physical point of view. APET has also been separated in 4 phases: before core degradation, during core degradation, vessel rupture, after vessel rupture.

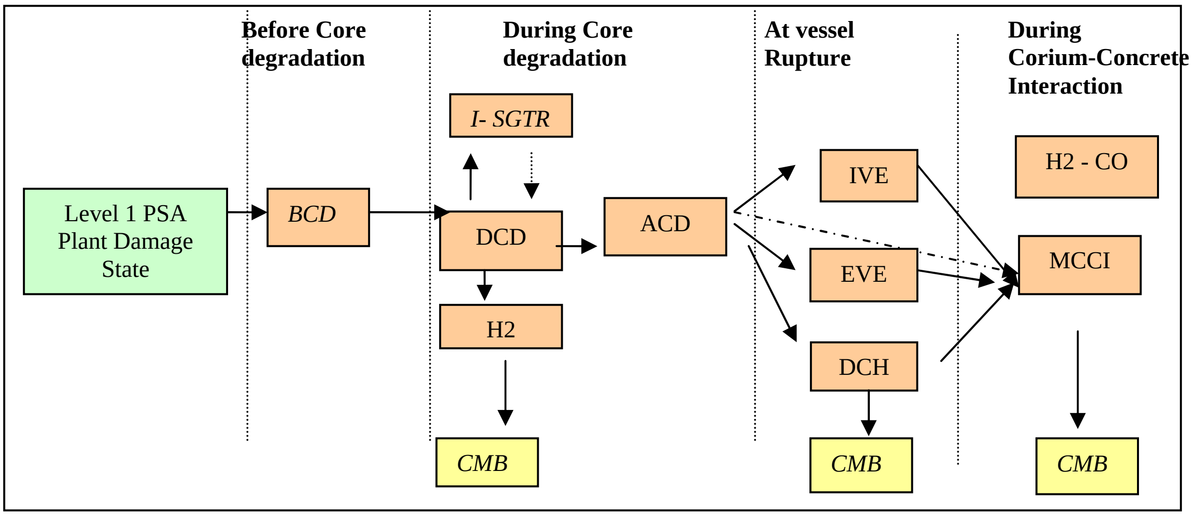

Figure 1 - Physical models of the APET

In the IRSN approach, construction of the APET physical models is based, as far as possible, on results obtained by validated codes calculations. Expert's judgments are used for result interpretation and when direct code calculations are not possible.
The Figure 2 presents the codes used for each APET physical model.

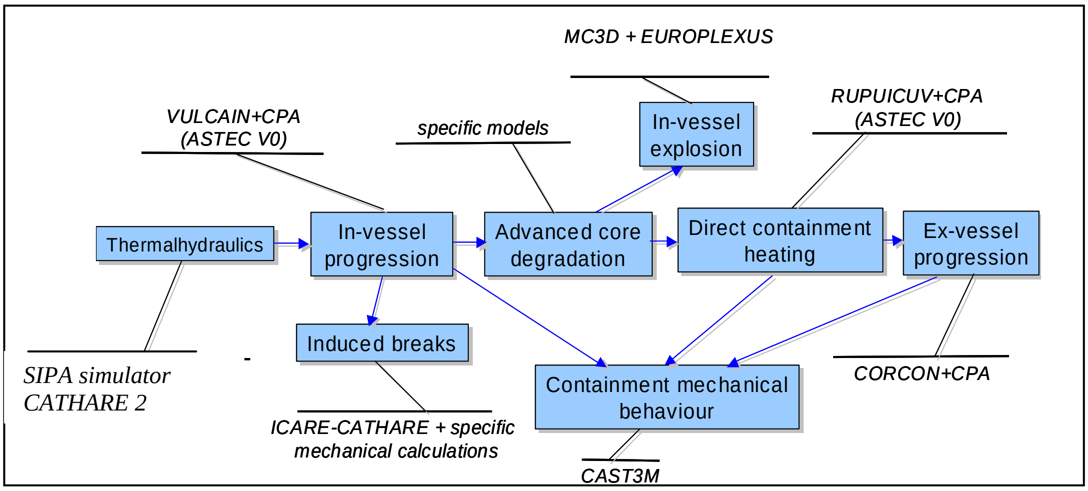

Figure 2 – Codes used in 900 MWe PWR level 2 PSA

In the APET, a physical model is a function that links upstream variables and downstream variables (Figure 3). Two categories of upstream variables are distinguished:

- state variables that describes the state of plant (systems availability and physical states of reactor),
- uncertain variables associated to a distribution law that are random choose by a Monte-Carlo pulling ; experts are free to define these uncertain variables that can concern different aspect, like uncertainties on the physical state of reactor or confidence on severe accident codes.

Downstream variables describe the physical state of reactor after the considered phenomena.

Such a physical model is supposed to give a relevant evaluation of a physical phenomena consequences for all scenario calculated by the APET. For the construction of these physical models, 2 methods are used :

- response surfaces,
- tables of results.

The first method is applied when scenarios effects and discontinuities are not too important. Response surfaces are mathematical functions that calculate downstream variables (*results*) in function of upstream variables (*state or uncertain variables*). Experimental design technique is used to minimize the number of calculations. Development of the physical models follows three steps [2] :

- choice and hierarchy of upstream variables by experts,
- elaboration of one response surface for each downstream variable,
- checking of the response surfaces accuracy.

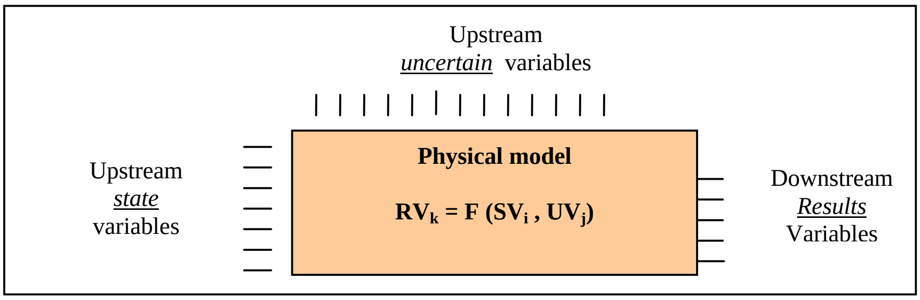

Figure 3 – Schematic view of an APET physical model

In the second method, the physical model is elaborated as a table: each line corresponds to one severe accident code calculation. First columns of the table allow identification of calculation and last columns proposed results of code calculation. In some cases, a distribution law is associated to results if uncertainties have to be taken into account. With that method, an identification process has to be implemented for choosing the best line (the more relevant calculation). This approach is supposed to be less consistent for uncertainties assessment than previous one.

Table 2 – Table of result

Transient number	**Identification variables values**						**DCD downstream (results) variables values**						

42 Examples of physical model for APET

As said previously, physical models introduced in the APET are represented either by a grid approach, or by a response surface approach, as shown on the following table:

Table 3 – Application of methodology in APET

Grid of results	Response surface
Core degradation	Advanced core degradation
In-vessel steam-explosion	Ex-vessel steam explosion
Melt-core-concrete interaction	Direct containment heating
Containment leakage	Mechanical behavior of containment building

Ex-vessel steam explosion, direct containment heating and mechanical behaviour of containment building have been described in [2]. Examples of application for core degradation and melt-core-concrete interaction are given below.

Core degradation APET model

This model has to describe accident progression from beginning of core degradation to appearance of a corium flow in the lower head. Transients are calculated with SIPA simulator (with CATHARE 2) from initiating event to beginning of core uncovery. They are calculated with ASTEC V0.4 after core dewatering.

For core degradation progression, strong scenario effects and discontinuities have to be taken into account (valve opening, RCS cooling with steam generator, accumulators discharge, water injection into RCS…). In concrete terms, the model is directly linked to interface variables in the APET, which are all discrete variables. Preliminary studies have showed that the physical calculation results cannot be accurately described by response surface methodology. A grid of results has been then used according to the two following stages:

- for each considered scenario (depending on systems availability, human actions, residual power…), the APET has to choose a representative transient ; the choice is done according to the identification variables values by a selection tree;
- ASTEC calculation results used for accident progression evaluation are then extracted from the grid for the selected transient.

One thermal-hydraulic (SIPA) transient is defined for each PDS. Two ASTEC transients (with and without severe accident management actions) are then defined. Some situations are supposed to be very close to each other for the core degradation phase and finally, 117 ASTEC transients have been defined for 130 PDS.

This high number of transients (in comparison with the international practice) is an attempt to reduce the uncertainty (lack of precision) due to the choice of a representative transient for each PDS (and consequently for all PSA level 1 associated sequences). Nevertheless, it is obvious that this number of transient calculations is low in comparison with the number of situations that can be generated by the APET (by modeling of severe accident management actions for example). No quantification of uncertainties has been performed at this stage.

Results of simulation of core degradation process remain poorly validated (at reactor scale) and it is a necessity for level 2 PSA to assess uncertainties on results. Many physical information are extracted from ASTEC calculations and are used in APET . Examples are provided in table 4.

Table 4 – APET Core degradation model – Example of variables extracted from ASTEC calculations

Residual power at beginning of core degradation	Vessel temperature in upper plenum
Delay before beginning core dewatering	Water temperature in lower head
Moment of total core dewatering	Sump temperature at vessel rupture
Moment of application of severe accident guide	Oxidation fraction of zirconium at vessel rupture
Moment of clad rupture	Melted core composition before corium flow
Moments of corium flow towards lower head	Melted core composition after corium flow
Moment of vessel rupture	Moment of core flooding
Moment of 5% core melt	Pressure at flooding
Average primary pressure	Mass of melted core at flooding moment
Primary pressure at corium flow	Available water mass in accumulators
Primary pressure at vessel rupture	Accumulator pressure
Containment pressure at vessel rupture	Minimum flow for evacuation of residual power by evaporation
Fraction of melt core at corium flow toward lower head	Maximum possible hydrogen combustion peak during core degradation
Mass of corium flow	Burnt hydrogen mass in case of ignition by recombiners
Mass of melted core at vessel rupture	First moment of possible ignition by recombiners
Water mass in lower head	Hydrogen mass in containment at vessel rupture if no combustion has occurred
	Hydrogen burned mass at first possible ignition by recombiners

Experts could propose uncertainties evaluation for all these variables. To maintain a quite simple model, uncertainties are only assigned to results which are supposed to have a major impact on safety issues. For example :

<u>A/ The hydrogen mass in containment at vessel rupture</u>

This value is a safety issue because over pressurization in case of direct containment heating (DCH) could damage containment if enough hydrogen was present in containment

Uncertainties are supposed to be high because :

- hydrogen in-vessel generation is a complex phenomenon,
- combustions can occur (but not certainly) before vessel rupture,
- burnt hydrogen mass at each combustion before vessel rupture cannot be quantified,
- recombiners efficiency depends on hydrogen distribution in containment.

For these reasons, hydrogen mass in containment at vessel rupture is deduced with the following assumptions:

- the mean value corresponds to the ASTEC value with combustion at the time of first ignition by recombiners,
- lower boundary: null if the mixture (vapor-hydrogen) is flammable at least once during degradation of core and half of the mean value if otherwise,
- upper boundary: evaluated according to the containment atmosphere composition; corresponds to the hydrogen mass necessary to reach the limit of recombiners ignition criteria (criteria are based on H2PAR, KALI H2 and AECL experiments).

B/ The total mass of relocated corium in lower head

This value is a safety issue because of its influence on the prediction on the vessel rupture time, the corium mass that could be dispersed in containment in case of DCH, the available mass for an ex-vessel steam explosion and indirectly the available mass of corium for the MCCI phase. This value concerns both the risk of short-term loss of containment and the stop of accident progression during MCCI phase.

In ASTEC V0.4, the calculated relocated mass strongly depends on the numerical meshing.

This is why, in the APET, the relocated mass is sampled between two ASTEC results : the mass of relocated corium, and the total melt core mass. Experts considered an exponential distribution between these boundaries.

The sampled value is then transmitted to advanced core degradation model and direct containment heating model.

MCCI Model – Delay before basemat penetration

The aim of this model is to assess the delay before the basemat penetration (axial propagation) or complete cavity walls erosion (radial propagation) when the corium is relocated into the reactor pit.

Present model has been established with CORCON (MELCOR) code for two types of basemat concrete (siliceous or siliceous/limestone). Investigation have also been done with MEDICIS code (ASTEC V1) .

Response surface methodology has been followed. In complement, an analysis of physical models of CORCON was performed in collaboration with CEA. This analysis has confirmed the lack of knowledge on MCCI phenomena and the high level of uncertainties on CORCON predictions. To simplify the model elaboration and considering the result of the assessment of CORCON models, it has been considered that uncertainties could be assessed by the difference between results obtained with CORCON for two modeling of the corium configuration (homogeneous or stratified with metallic and oxidized layers).

The “state variables” are the residual power, the corium mass in the pit, the steel mass in corium and the non-oxidized zirconium mass. Moreover, one’s have to consider existing uncertainties on this “state” parameters, especially the mass of corium in vessel pit (which depends partly on ejected corium mass in containment at vessel rupture by DCH or ex-vessel steam explosion) and the oxidation ratio of zirconium.

Several studies have been performed to establish response surfaces. As for the core degradation model, these studies showed that response surfaces were not appropriate to describe complex and strongly non-linear phenomena as core concrete interaction, in particular for short rupture times and low corium masses. For these reasons, four grids of values (resulting from CORCON) are used in the APET, according to the concrete types (siliceous and limestone) and to the corium configuration modeling (homogeneous or stratified).

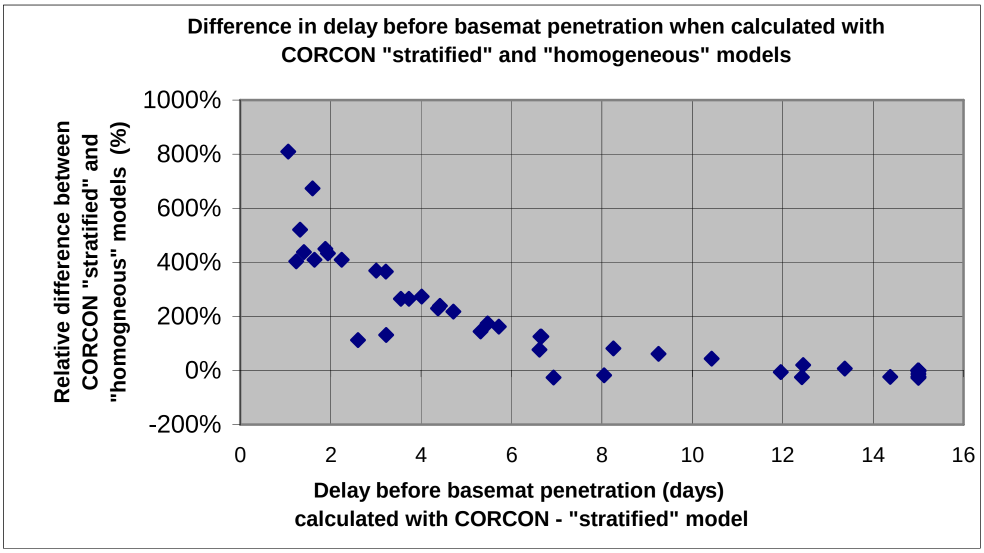

Figure 4 – Uncertainties on delay before containment failure after penetration of foundation

The Figure 4 shows the high variations for prediction of delays before foundations penetration with CORCON as a function of the model ("homogeneous" or "stratified"). Relative difference is particularly high when a short delay is predicted with the "stratified" model.

In conclusion, the application of the IRSN methodology in case of MCCI highlights the difficulties to provide a best-estimate assessment of available delay before corium penetration in foundation.

Reduction of uncertainties on MCCI calculations will remain an objective for next versions of the study.

5. Quantification of uncertainties in releases calculations

51 Methodology

One requirement for IRSN level 2 PSA was to propose best-estimated evaluation with quantification of uncertainties.
Assessment of atmospheric releases with a reliable best estimate approach is difficult because of :

- the complexity of physical behavior of fission products,
- the very large spectrum of situations from the system point of view (localization and moment of appearance of a break for example).

A second requirement, in links with the general objectives of the IRSN level 2 PSA was to define detailed Release Categories. It leads to define quite detailed variables in regard of international practice (more than 1000 release categories can be generated).

For each combination of the RC variables, a simplified model can perform a calculation for atmospheric releases. This model describes noble gas, molecular and organic iodine and aerosols behavior in containment. Uncertain parameters have been proposed to assess order of magnitude of the uncertainties on releases assessment. Examples of these parameters are presented in Table 6. This list will be extended in further steps of the study.

Table 6 – Example of uncertain parameters for release assessment

Parameters
Coefficient multiplying the containment size break
Mass of noble gases emitted during core melt
Mass of aerosols emitted during core melt
Mass of volatile molecular iodine emitted in the containment during core melt
Aerosols retention coefficient inside the RCS
Aerosols retention coefficient inside the secondary system
Resuspension coefficient of the aerosols deposited on the containment walls
Coefficient characterizing the adsorption speed law of molecular iodine on the painted surfaces of the containment
Coefficient characterizing the conversion of molecular iodine into organic iodine when adsorbed on the painted surfaces of the containment
Coefficient concerning iodine separation between liquid and gaseous phases in case of a liquid leakage
Coefficient of in vessel aerosols resuspension / revolatilization
Duration of core degradation phase

52 Examples of results

Some examples, issue from the version 1.1 of the study are commented.

<u>Impact of volatile iodine behaviour uncertainties</u>

Uncertainties have been taken into account on molecular iodine emission from RCS to containment, molecular iodine adsorption on containment painted surfaces and conversion of molecular iodine into organic iodine when adsorbed, on the basis of a review of available experimental results.

Analysis of results has shown that large uncertainties on the molecular iodine releases are in large part due to uncertainties on molecular iodine adsorption on containment painted surfaces.

Two examples are provided hereafter.

The first one concerns an example of molecular iodine releases for a release category representing a severe accident without a containment break before vessel breach and with Containment Spray System (CSS) not operating. The abscissa gives the releases expressed in Bq while the ordinate indicates the number of observations as the result of a Monte-Carlo process.

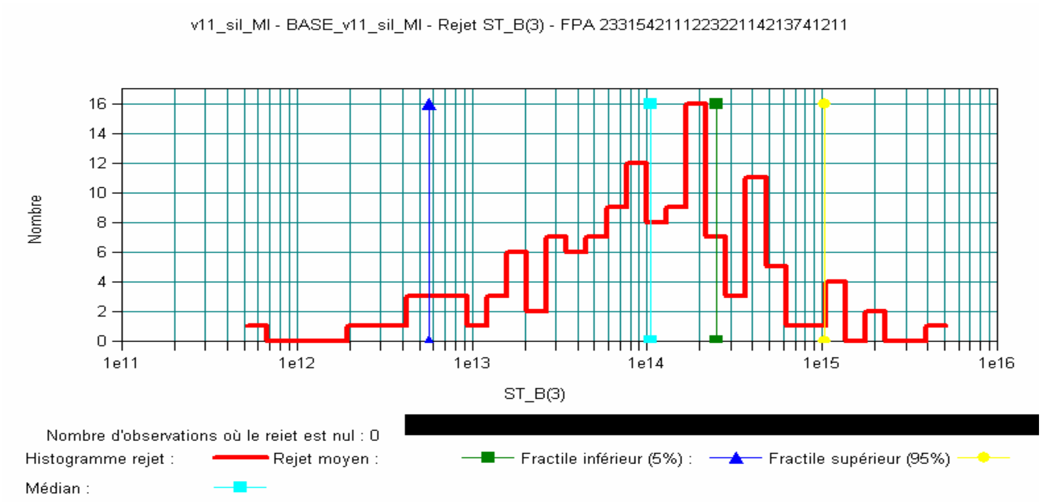

Figure 5 : Example of distribution law of molecular iodine releases before vessel breach - CSS not operating

The second example concerns molecular iodine releases for a release category nearly identical to the previous one : the values of the RC variables are exactly the same ones, except for the CSS state:

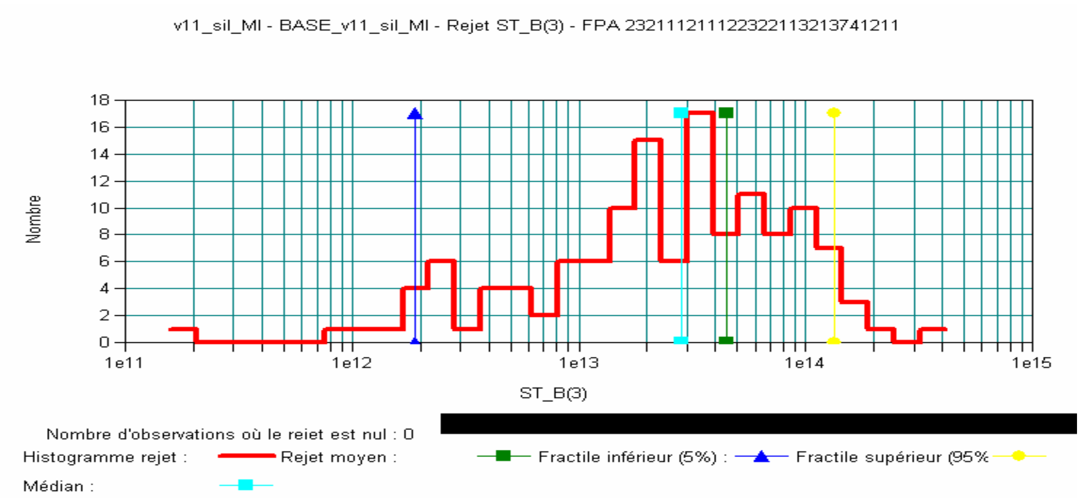

Figure 6 : Example of distribution law of molecular iodine releases before vessel breach - CSS operating

One can notice that the molecular iodine releases values spread from about 10^{11} Bq to 10^{16} Bq on figure 5 (CSS not operating) whereas these values spread from about 10^{11} Bq to 10^{15} Bq on figure 6 (CSS operating).

The explanation of the wider distribution law extent in the first case (CSS not operating) is given hereafter: when CSS is operating, the molecular iodine behaviour in the containment depends on two phenomena, molecular iodine removal by CSS and molecular iodine adsorption on containment painted surfaces, but when CSS doesn't operate, the molecular iodine behaviour in the containment depends mainly on the adsorption phenomenon, for which important uncertainties have been taken into account according to experiments results interpretation.

Impact of containment leakage rate uncertainties

Uncertainties have been taken into account on the containment leakage rate to express the loss of information regarding this rate, while binning level 2 PSA sequences into release categories. These uncertainties have a strong impact on the extent of the release distribution law of any species (not only iodine releases, but also aerosols and noble gases) except in the case of large containment bypass or large containment break, for which the leakage rate can be considered as infinite).

The Figure 7 concerns an example of noble gases releases for a release category representing a severe accident without a containment break before vessel breach. In this case, the epistemic uncertainties on noble gas behaviour are quite negligible in comparison of the loss of information due to binning (for containment leakage).

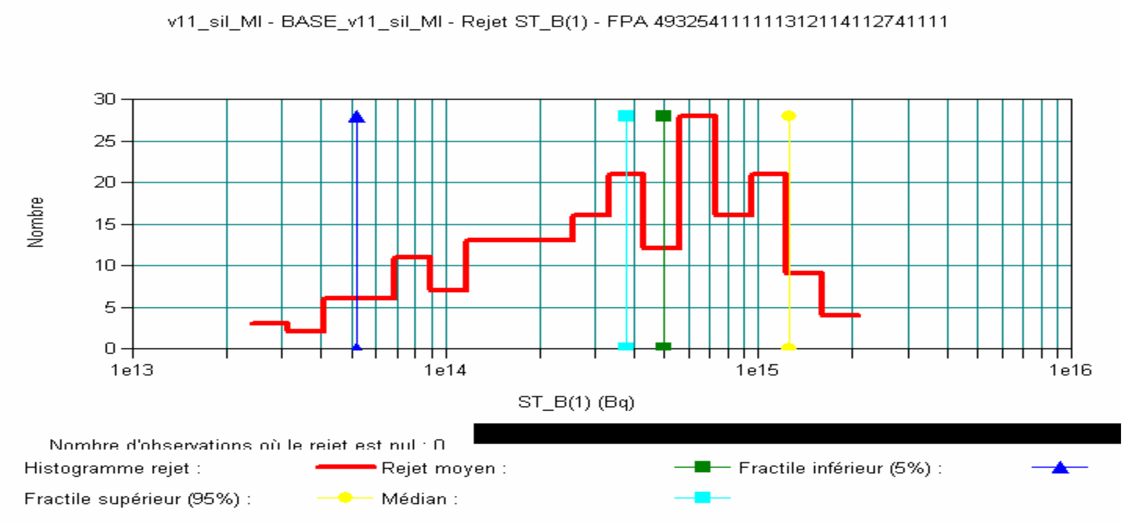

Figure 7 : Example of distribution law of noble gases releases before vessel breach

Impact of uncertainties for organic iodine release

Organic iodine has a major impact in terms of radiological short-term impact in case filtering and venting system would be opened. The figure 8 shows the very large dispersion of iodine releases in such a situation mainly due to the uncertainties on emission of volatile form in the containment and conversion of molecular iodine into organic iodine on the painted surfaces. This result explains why the IRSN considers that the improvement of knowledge on iodine behaviour remains an R&D priority.

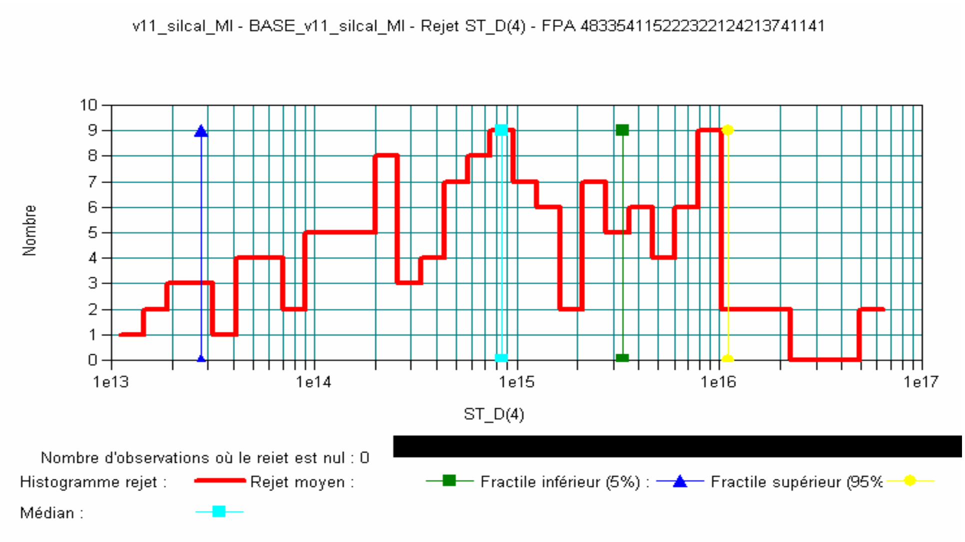

Figure 8 : Example of distribution law for organic iodine releases in case of containment venting

6. Conclusion and perspectives

The paper describes the IRSN methodology for uncertainties assessment in APET and release calculations.

Elaboration of physical models in APET is mainly based on severe accident code calculations. A great number of different scenarios has been calculated with ASTEC to limit, as far as possible, uncertainties due to the choice of a limited number of calculated transients.

In some cases, the approach allows identification of issues with a strong impact of safety issue but which are poorly predicted (hydrogen production in case of reflooding, hydrogen mass in the containment at vessel rupture time, reflooding efficiency in and ex-vessel, corium melted mass in lower plenum at vessel rupture, delay before foundation penetration in case of MCCI …).

Concerning release assessment, the presented examples show the importance of uncertainties in releases assessment for release category of level 2 PSA that may have different origins. In some cases, for example situations with opening of containment filtering and venting system during MCCI, uncertainties correspond mainly to a lack of knowledge (on organic iodine production for example).
In other cases, results of level 2 PSA for uncertainties assessment in release categories correspond both to epistemic uncertainties and binning effect (containment leak size for example), which have the same contribution to the global uncertainty. This must be taken into consideration for assessment of short-term releases (before vessel rupture for example).

The forthcoming level 2 PSA developments for uncertainties assessment (with a best-estimate approach) should have as objectives the reduction of uncertainties by improvement on knowledge on physical phenomena and also by improvement of the level 2 PSA methodology to make uncertainties due to binning of scenarios one order of magnitude below the physical one’s.

References

1 Status of IRSN level 2PSA, CSNI WG-Risk – Workshop on level 2 PSA and severe accident management, Koln, avril 2004, E. Raimond, R. Meignen, J. Dupas, D. Plet, B. Laurent, G. Nahas, B. Cirée, Rapport IRSN-DSR-19

2 Advanced modeling and response surface method for physical models of level 2 PSA event tree, CSNI WG-Risk – Workshop on level 2 PSA and severe accident management, Koln, avril 2004, E.Raimond, C.Caroli, B. Chaumont, Rapport IRSN-DSR-20

3 Technical basis for off-site emergency planning in France F. Charpin, E. Raimond, B. Chaumont – EC/JRC & OECD/NEA seminar on Emergency & Risk Zoning around Nuclear Power Plants – 26-27 April 2005 - Rapport IRSN-DSR-63

4 SARNET PSA Work package – Methodologies used for uncertainties assessment –SARNET-PSA2-P05

5 Evaluation of the β mode in the Level 2 PSA (ICONE 5 1997 - IPSN)

6 KANT : a Probabilistic Software for Level 2 PSA (PSAM-4 1998 - IPSN)

7 IPSN level 2 Probabilistic Safety Assessment of 900 MW PWRs (PSAM-5 2000- IPSN)

8 Quantification of Physical Phenomena and Assessment of Uncertainties in the IPSN level 2 PSA (PSAM-5 2000- IPSN)

9 Evaluation of the β-mode in the level-2 PSA (PSAM-5 2000- IPSN)

9 The Process of Consulting Experts on the Reliability of the Equipment Challenged in Severe Accidents as Part of a level 2 Probabilistic Safety Assessment (PSAM-5 2000- IPSN)