IRSN Level 2 PSA for the French 1300 Mwe PWRs Series: Some New Features in the Updated Version

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Abstract: IRSN (as TSO of the French Nuclear Safety Authority) has been developing level 2 PSAs for the French NPPs fleet for many years. This paper presents an overview of new features introduced in the level 2 PSA for the French 1300 MWe PWR series. These features are related to interface between level 1 and level 2 PSA, human reliability assessment, reactor containment fragility under severe accident loading (temperature and pressure), ex-vessel accident modelling, and metrics for the accident radiological consequences.

Keywords: level 2 PSA, level 1-level 2 interface, severe accident, containment fragility, human risk assessment, corium ex-vessel cooling, risk metrics

1. INTRODUCTION

Since the end of 90s, IRSN (Technical Safety Organization for the French Nuclear Safety Authority - ASN) has developed level 2 probabilistic safety assessment (L2 PSAs) for the French PWRs. These PSAs provide a risk ranking by combining the severe accident scenario frequencies with the radiological consequences they induce. IRSN PSAs are independent from those developed by the French utility (EDF) and are used to support the IRSN safety review activities performed for ASN.

IRSN PWR L2 PSAs are based on the so called "separated approach [1]" using two probabilistic software: Risk-Spectrum® for the L1 PSA and KANT for the L2 PSA. KANT is a probabilistic event trees software developed by IRSN that proposes functionalities for the modelling of the accident progression after core damage. KANT has been designed to integrate accident simulation results as far as possible (especially from the ASTEC severe accident integral code [2]). It enables the development of simplified (fast running) physical models (meta-models), and the propagation of uncertainties (Monte Carlo algorithm). All relevant plant parameters (time, pressures, temperatures…), as well as equipment status parameters, are transmitted through the Accident Progression Event Tree (APET) for a precise description of the Nuclear Power Plant (NPP). KANT allows parallel Monte Carlo runs on Windows or Linux computing servers, resulting in significantly shorter computational time. Moreover, a very fast-running code, called MER, has been developed to assess amplitude and kinetics of release for each release category (RC) generated by the APET. IRSN has also developed the MERCOR software to assess the radiological impact of release for one standard meteorological condition (see [3] and [4] for more details).

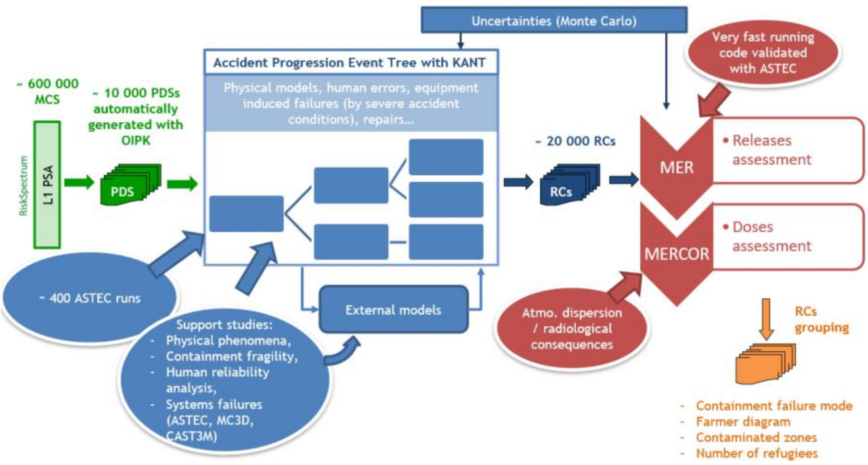

Figure 1. General Structure of IRSN L2 PSA

In order to limit expert judgment in the APET, more than 400 ASTEC accident simulations are performed for the in-vessel core melt, and around 300 for the ex-vessel accident progression (Molten Core-Concrete Interaction phase (MCCI)). These simulations are periodically updated according to the state of the art. APET quantification generates around one million of severe accident scenarios that are grouped in RCs for release and consequence calculations (around 20,000) and then gathered for the summary report. At the end, all information from L1 PSA and L2 PSA is available in the results including uncertainties quantification (for containment failure mode frequencies and source term evaluation). Figure 1 above gives the general structure of IRSN L2 PSA.

This paper presents an overview of some new features introduced in the L2 PSA for the French 1300 MWe PWR series, which relate to interface between L1 and L2 PSA, human reliability assessment, containment fragility under severe accident loading (temperature and pressure), ex-vessel accident modelling and metrics for radiological consequences.

2. INTERFACE BETWEEN L1 & L2 PSA

Since the L2 PSA is the prolongation of the L1 PSA's sequences after the beginning of core degradation, an interface between the L1 PSA and the L2 PSA is required to transfer all information needed for level 2 [1]. In the first versions of the IRSN PSAs, L1-L2 PSA interface was a mostly manual process, resulting in significant resources allocation. As presented in the reference [5], to cope with such a difficulty, a new interfacing approach has been developed.

This approach is based on the introduction of "flag events" (basic events with a probability of one) into the L1 PSA minimal cut sets (MCSs), in order to transfer information related to front lines systems status (needed for severe accident management) and human actions, without altering their frequencies. This is realized thanks to the creation of a "prolongation fault tree" in the L1 PSA model, which is connected to core damage sequences and is used to specify the state of systems considered in L1 PSA sequences and required for L2 PSA (for example: availability/status of the safety injection system, maneuverability of primary circuit relief valves…). Additional branches are also created in this tree, to model systems that are not modelled in the L1 PSA but whose availability is needed during the severe accident progression (for example: containment isolation system, severe accident instrumentation in the main control room, reactor containment venting, …). This "adapted" L1 PSA modelling provides the same MCS than the original L1 PSA modelling, but with additional information (represented by the flag events) related to L2 PSA modelling.

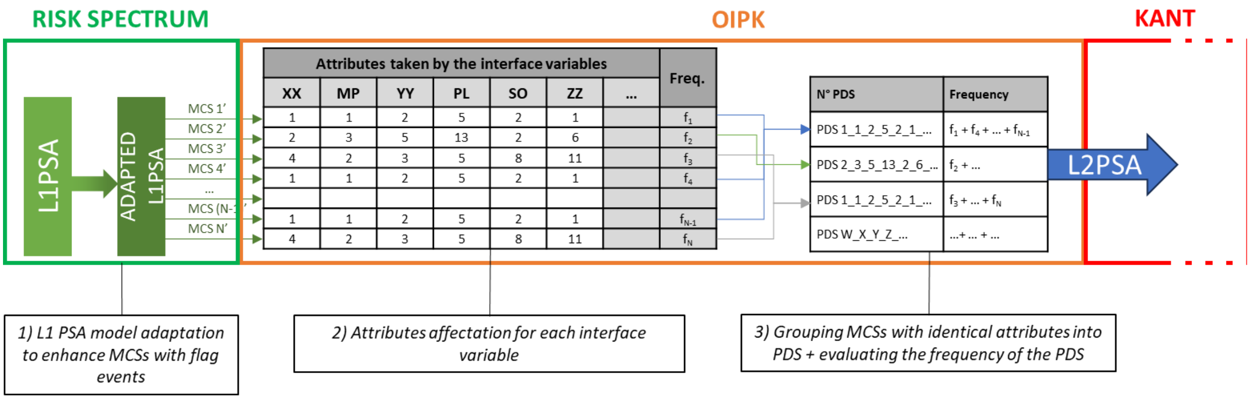

Figure 2. Main Steps of L1/L2 PSA Interface

Afterwards, the MCSs are filtered to automatically build the different plant damage states (PDSs) of the L1-L2 PSA interface using a dedicated tool, called OIPK. This is made by defining "interface variables" (which represent systems or initiating events) that can take several values called "attribute". For example, interface variable "AS" describe the status of the containment spray system and can take values from 1 (if the system is totally available in direct injection and recirculation mode) to 8 (if the system is totally unavailable, whatever the causes of the failure). The values between 1 and 8 describe intermediate possible situations (spray system only available for direct injection and/or with only 1 train out of 2, etc.). In the L2 PSA for the French 1300 MWe PWRs series, around 70 "interface variables" have been defined. The filtering operation leads to affect an attribute for each of the interface variables. This is done thanks to the flag events of the adapted modelling. This operation is described in the Figure 2.

The automatic PDS generation allows implementing a very detailed L1-L2 PSA interface easy to update. By applying this approach to the latest versions of L1 and L2 PSA for the French 1300 MWe PWRs series, around 200,000 MCS and 10,000 PDS were generated. These PDSs are then used as the input of L2 PSA.

3. HUMAN RELIABILITY ASSESSMENT

Similarly to equipment failures, the failure of human actions is considered in the modelling of L2 PSA. The Severe Accident Management Guideline (SAMG), for the French NPPs, requires two types of human actions:

- “Immediate actions”: group of actions that can be performed immediately by the control room crew, without previous concertation because they can unequivocally reduce the consequences of the accident. If these actions are carried out too late, they can become ineffective or lead to significant radiological releases into the environment,
- “Delayed actions”: implementation of delayed actions requires a risk analysis to draw the benefits and the disadvantages of the considered actions. So, an expertise by the nuclear crisis organization is necessary before implementation by the control room crew.

Before 2020, IRSN modelized human reliability based on SAMG, which lists the actions to do quickly after the core melt occurs. After 2020, IRSN has used the available procedures in the main control room (MCR) that provide detailed instructions and specify the equipment to be used. These procedures have enabled IRSN to enhance human reliability modelling by meticulously studying the execution time required for each “immediate action” outlined in the SAMG. Consequently, the success or failure probability of these actions is quantified more accurately and realistically with a precise estimation of related delays and timing. That allows to determine more accurately the time available to rectify an action execution failure. This point is crucial in the assessment of human actions. Knowing the timing and duration of an action allows to assess whether MCR crew and field agents (FA) have enough time for a new attempt if the action has previously failed. The following subchapters present the input data used to conduct the study, the parameters that have a significant impact on the results, as well as the results and conclusions obtained.

3.1 Input Data

<u>Guidelines</u>: core melting (at 1100°C) serves as a criterion for entry into the SAMG. Based on this guideline and associated procedures, IRSN identifies the immediate actions to be performed and their succession.

<u>Inspections:</u> during inspections at several NPPs, IRSN used stopwatches to assess:

- the time required to carry out procedures, including simple tests, communication, checking, etc.;
- travel times for FA from the MCR to the relevant location, which vary depending on the action's location and whether the building is subject to a radiological environment;
- the duration of operations (such as actions from the MRC, valve opening/closing in premises, etc.).

Thus, IRSN was able to build a database on the durations of carrying out elementary actions.

3.2 Significant Parameters

The following significant parameters are considered:

- <u>Success or failure of actions achievable in the main control room</u>: if an action initiated from the MCR fails, then the SAMG instructions provide for them to be carried out locally, on-site.
- <u>Number of FA available when an action must be performed locally</u>: IRSN considers 4 or 6 FA knowing that the number of FA has a significant impact on the time required to perform immediate actions.
- <u>Situation with or without total loss of power supplies</u>: the total loss of power supplies can cause accessibility difficulties in the premises, particularly due to insufficient lighting.

3.3 Results

IRSN estimated the time required to complete each immediate action of the SAMG. Time cumulation provides the moment at which each of these actions can be implemented (see Table 1).

Table 1. Chronological Succession of Immediate Actions from the Moment of Core Meltdown

Immediate actions	**Successful action from MCR**	**Successful local actions (with 4 FA) after failure from MCR**	**Total loss of electrical power Successful local actions with 4 FA**	**Total loss of electrical power Successful local actions with 6 FA**
	Time since the start of core melt (minutes)			
Entry into the SAMG	25	25	25	25
Shutdown of reactor coolant pumps	26	36	N/A	N/A
Opening of the primary circuit relief valves for depressurization	28	53	60	60
Local containment isolation	92	129	203	173
Safety injection system startup	64	99	N/A	N/A.
Inhibition of the containment spray system	89	114	271	198
Preheating of the filtered containment venting system	106	221	437	353
Refilling the steam generator tank (feedwater)	137	137	374	260

During the initial stages of a severe accident, a key urgent action is to depressurize the primary circuit with relief valves. This action mitigates the risks of Steam Generator Tube Rupture (SGTR) or Direct Containment Heating (DCH). Opening the relief valves is subject to a time constraint: it must occur within 2 hours after core damage due to equipment qualification requirements. Through analysis of execution times, IRSN has identified that operators have a time window of 1 hour and 7 minutes to make a second attempt to locally open the relief valves, if the action failed previously (from the MCR, and then from the electric room). This action (from the electric room) requires 25 minutes. Therefore, operators have a margin of 42 minutes to make this new attempt. Previous quantifications (less precise) used envelope values or were based on engineering judgment. Therefore, they were sometimes quite pessimistic.

4. REACTOR BUILDING CONTAINMENT FRAGILITY UNDER SEVERE ACCIDENT CONDITIONS

4.1 Preamble

For the previous IRSN L2 PSA, fragility curves were derived to obtain the reactor containment failure (loss of tightness) probability in function of the containment pressure. IRSN has recently developed a new methodology to consider the temperature effects as well in the fragility.
Large-scale civil engineering structures, especially within the nuclear industry, undergo simultaneous and different loads of chemical, thermal, hydric, and mechanical nature. This has an impact on the evolution of their performance over time; this is commonly referred to as “ageing” under normal service conditions [6]. On the other hand, these structures are expected to ensure, to some extent, an acceptable functional performance in accidental conditions. For L2 PSA, our interest is geared towards the probabilistic quantification of the functional performance of such structures. Our aim is to establish an objective estimation of fragility curves for several accidental conditions accounting for the inherent spatial and temporal variations of material and structural properties. Hereafter, we assume that a verified and valid physical model able to simulate the structural behavior is used, that the properties of uncertain parameters (expressed in terms of probability

density functions) are known with a sufficient confidence and that the severe accident conditions are known as well. From there, the IRSN methodology follows the scheme shown in Figure 3.

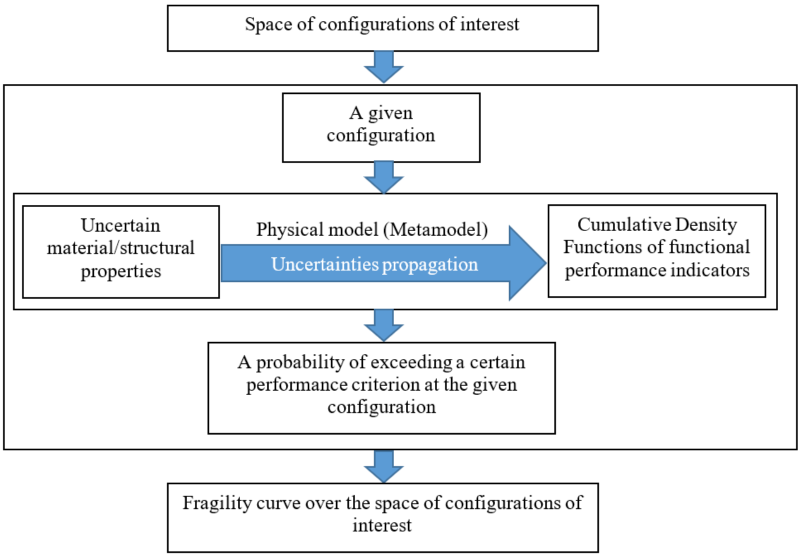

Figure 3. A General Scheme For Fragility Curves Generation

As the number of inputs is important, the use of Monte Carlo methods is recommended to cover all the domain of interest. But the detailed physical models, generally used to simulate large-scale civil engineering structure behavior (typically finite element modelling), implies high computation time, especially if non-linear material behavior is considered. Therefore, it is recommended to construct a metamodel based on a limited number of detailed calculations to obtain an explicit approximation of the large-scale structure response over the domain of interest (by selecting adapted fitting mathematical functions).

4.2 Thermo-Mechanical Modelling of the Concrete Wall's Behavior and Leakage Prediction

In the IRSN approach, the modelling accounts for ageing phenomena in normal conditions (early age behavior of concrete, creep and shrinkage, thermal strain, etc.) before considering (by numerical simulation) a severe accident leading to an increase in temperature and pressure inside the containment. This increase can cause damage of concrete in the thickness of the containment wall and ultimately reduces the containment capacity of the nuclear building (degradation of the tightness function). In the present study, numerical calculations are carried out (with Cast3m software [7]) for double-walled containment buildings without steel liner (inner wall of interest is depicted in Figure 4 – the outer wall is not studied here). The long-term behavior of concrete is calculated according to Eurocode 2 [8] (all formulas have been adapted to in situ measurements to improve the representativeness of the physical model) and is carried out for a predicted service life of 60 years. Accidental conditions (high temperature and pressure) are applied once this service duration is reached. Thermal calculations are carried out in a transient regime using a complete 3D model with around 500 000 finite elements (20-node solids). In addition, mechanical calculations are carried out according to a fictitious cracking formalism (Ottosen, 1977) using a complete 3D model with around 275 000 finite elements (8-node solids). The reinforcement bars and prestressing cables are modeled by around 310 000 linear segments with a perfect steel-concrete bond (segments with 2 nodes).

The accidental temperatures and pressures are set according to prior calculations using ASTEC. They evolve with time but also with their spatial position. For this study around 400 accidental scenarios have been calculated with ASTEC leading to various thermal transients and pressure over time. From these 400 scenarios, only few dozen are considered for the thermomechanical and nonlinear analyses.

The mechanical calculations are conducted on the basis of the strain superposition principle where the total strain is the sum of the thermal strain, the endogenous shrinkage, the drying shrinkage, the basic creep and the drying creep (According to appendix B of Eurocode 2 [8]). The fitting parameters of the formulae of each strain term are identified based on available experimental data at small scale and at full scale (long term monitoring data provided by the NPPs operators). Damage in concrete is described using the fictitious cracking model OTTOSEN available in Cast3m. Leakage quantification of a simulated containment structure in accident conditions remains a complex and difficult task ([9]).

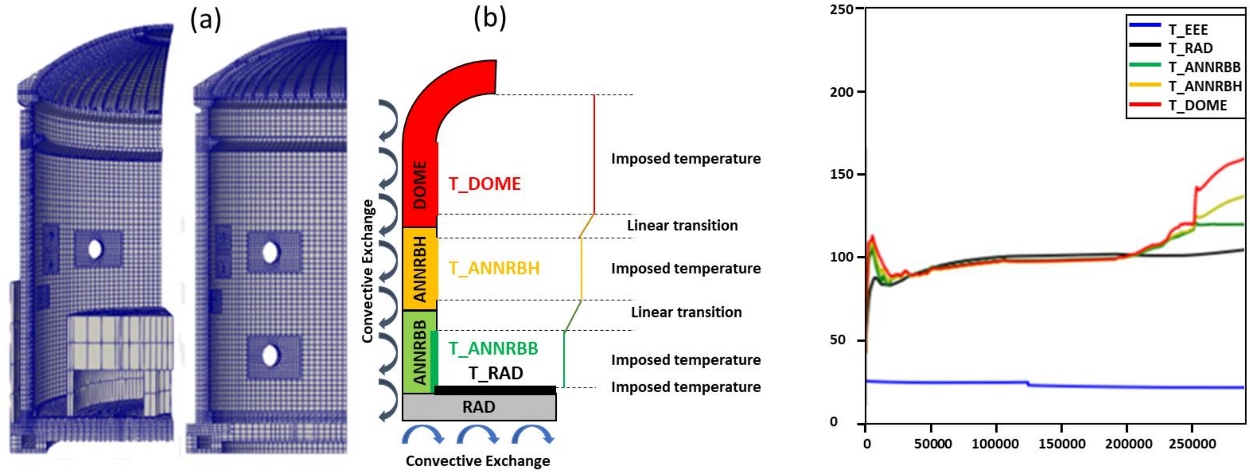

Figure 4. (a) 1/4th of the Mesh of the Internal Containment Wall of a Double-Walled Containment Building for a Relatively Coarse Mesh Intended for Mechanical Calculations and a Relatively Refined Mesh Intended for Transient Thermal Calculations (b) Example of Evolution of Temperature During Severe Accident

In a first attempt, the modelling of the correlation between mechanical damage and leakage is strictly based on measurements of dry air leakage obtained during the periodic Integrated Leak Rate Tests (ILRT). Numerical simulations provide the portion of cracked concrete and the concrete portion remaining under compressive stresses. Experimental means provide (with a low confidence level) the air flow going through existing concrete cracks and through the concrete porosities. By combining this two information, we define an analytical correlation function able to estimate the leakage rate depending on the portions of the damaged and compressed concrete. We also, as a first attempt, accept the applicability of the correlation function established for ILRT to accidental conditions.

4.3 Metamodel of Leakage Function

The principle consists of selecting few dozen of accidental scenarios that cover the physical domain of thermomechanical intensity measures (pressure and thermal gradients[1]). Those scenarios are used to simulate with Cast3m the containment building response, with a nonlinear analysis, to obtain the evolution of the portion of damaged and compressed concrete. 950 calculation points are chosen to calculate the absolute pressure and thermal gradient. We achieve then a polynomial fitting f_{poly} according to a given confidence level (ε fitting error) and by minimizing the mean squared error. Eventually, we have an expression as following:

$$\%volume_{cracked} \approx f_{poly}(P_{abs}, \Delta T) + \varepsilon \tag{1}$$

Then, we use the correlation function f_{corr} (relating the leakage rates to the portion of cracked concrete based on ILRT results), to quantify the resulting dry air leakage $Q_{dry,air}$:

$$Q_{dry,air} \approx f_{corr}(\%volume_{cracked}) \approx f_{corr}\left(f_{poly}(P_{abs}, \Delta T) + \varepsilon\right) \tag{2}$$

The considered metamodel allows for an accurate fitting with an R^2 higher than 95% and an error standard deviation of 10%. Now, this metamodel can be used at low cost for each accidental scenario instead of running full 3D finite elements analysis that might last for weeks depending on the nonlinearities level and scenario duration.

4.4 Uncertainties and Fragility Calculations

Fragility calculations consist of quantifying the conditional probability of containment failure for fixed thermomechanical intensity measures $(P_{abs}, \Delta T)$. Containment is defined by $Q_{dry,air}$ exceeding a given threshold. This conditional probability results from the existence of random uncertainties (such as the intrinsic variations in concrete properties: tensile strength, fracture energy, Young's modulus, etc.) and epistemic

[1] Difference of temperature and pressure between the intrados and extrados sides of the wall.

uncertainties (the error of fitting using the metamodel mainly). In this work, we quantify the random variation of the portion of concrete under tensile loads with values around 50%, the random variation of the air leakage rates for French PWR 1300 MWe containment buildings during ILRTs with values around 40% and the epistemic error due to meta-modelling with values around 10%. As we have now an explicit and analytical model of the dry air leakage under thermomechanical loads, the use of the crude Monte Carlo Method to propagate uncertainties is possible at low computational cost.

Two air leakage thresholds have been considered: one called "DAC" correspond to the criteria to be verified during normal ILRTs (limit for normal leakage) and one called "GB" being 10 times higher than "DAC" representing a significant cumulated leakage in the containment walls. For the sake of sensitivity analysis, we also compute the fragility curves according to two methods: the first one is analytical assuming a lognormal distribution on inputs and of the fragility curve outputs and the second one is based on uncertainties propagation without any constraint on the shape of the fragility curve. Each time, results are presented considering three confidence levels: at 5%, at 50% and at 95%.

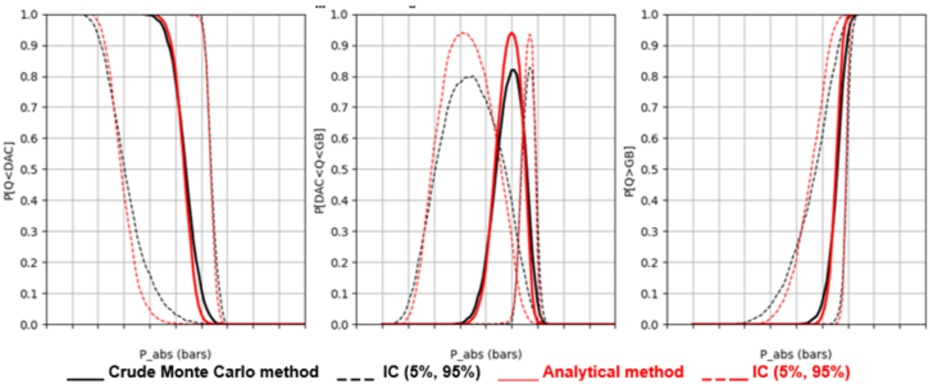

Figure 5. Fragility Curves Computed for Pressure Only Accidental Scenarios (with no Thermal Gradient)

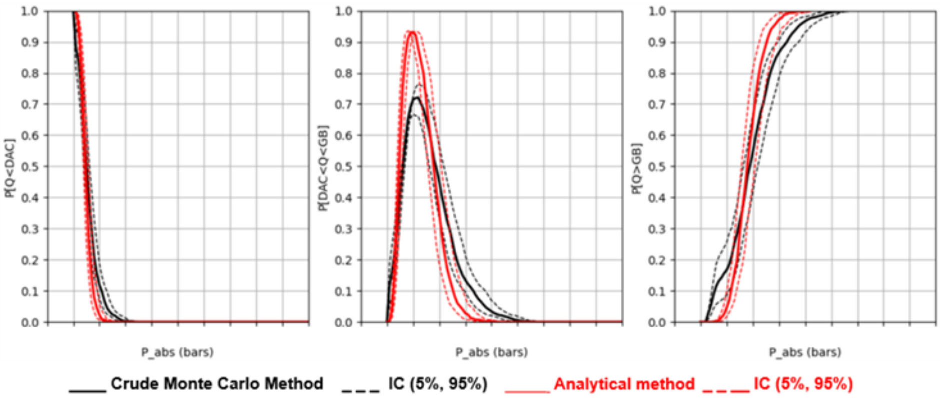

Figure 6. Fragility Curves Computed for Accidental Scenarios with an Increase of Pressure Given a Thermal Gradient of 160°C

Figures 5 and 6 show that the analytical and simplified method (assuming a lognormal fragility curve) is not always a good choice and can misestimate the probabilities of failure. The use of the crude Monte Carlo Method without any prior assumption remains the best option (no additional bias is introduced). These figures show also that the consideration of thermal effects leads to a decrease (by a factor 2 for severe thermal conditions) the mean pressure capacity of the containment wall under accidental conditions and to an increase (by a factor 5 for severe thermal conditions) the coefficient of variation of the fragility curve.

Nota : these calculations do not consider the repair operations (installation of an epoxy coating at the intrados) that are applied to the containment walls to limit the leakage rate ; they nevertheless demonstrate the importance of considering thermal effects to determine fragility curves for L2 PSA ; the methodology allows to take into account the presence of such epoxy coating by limiting the walls surface where leakage can appear.

Several improvements have been introduced in this new methodology, with the physical modelling of the concrete structure at high temperature and a more precise quantification of the leakage rate in function of the mechanical damages on the concrete. If realistic data and proper inputs are available, this global methodology allows the computation of fragility curves of a containment buildings within a reasonable time.

5. EX-VESSEL ACCIDENT MODELLING

A new strategy for the long-term corium stabilization in case of a severe accident will be implemented by EDF for the French 1300 MWe PWRs series in the framework of their 4th PSR. This strategy includes 4 steps:

- preventive filling of the containment sumps by water once the severe accident criteria has been reached,
- dry spreading of the corium (flowing out from the vessel breach) on an extended basemat area (the corium spreads in the reactor pit and then in a neighboring room, after ablation of a fusible concrete gate in the reactor pit),
- corium passive top-flooding through several flooding gates (they open after corium spreading thanks to a passive device),
- corium active top-flooding *via* a dedicated system that also allows to remove the heat from the containment thanks to an ultimate heat sink installed by the EDF nuclear rapid response force (FARN).

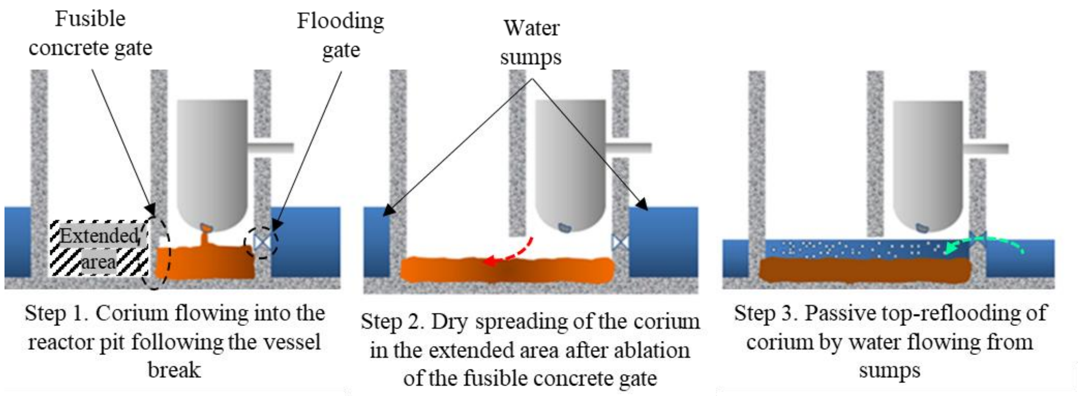

Figure 7. Description of the New Severe Accident Strategy for Ex-Vessel Phase

IRSN has therefore updated the ex-vessel phase modelling in the L2 PSA for the 1300 MWe PWRs. Physical uncertainties related to the corium behavior have firstly been studied (with around 2,000 ASTEC simulations to support the L2 PSA) when it relocates to the vessel lower plenum and then in the reactor pit. These uncertainties influence the kinetics of corium relocation in the reactor pit, the corium composition and the elevation of the vessel breach.

The ex-vessel phase L2 PSA modelling considers that the first corium flow from the reactor vessel (at vessel failure time) only includes a fraction of the corium mass relocated in the vessel and that secondary flows may occur later. The properties of each corium flow are considered for the calculation of:

- the corium spreading in the reactor pit,
- the ablation rate of the fusible concrete gate.

Once the fusible concrete gate is fully ablated and the corium is spread in the extended area, the modelling considers that water is injected in the reactor pit (passive or active injection), leading to the corium cooling. If water is present in the reactor pit when a corium flow occurs (first flow or secondary flows), the modelling considers:

- a poor spreading of the corium (possible no spreading in the extended area), and thus an increase of the basemat erosion,
- a possible ex-vessel steam explosion.

Two ASTEC corium configurations (homogeneous or evolutive; that determines the evolution of the oxidic and metallic corium phase) have been applied to calculate the basemat erosion by the corium. The configuration choice depends on the NPP concrete basemat composition (siliceous or limestone common sand concrete). The heat flux from the corium on the pit walls can be isotropic or anisotropic depending on the basemat concrete composition.

Considering these elements, the basemat failure by the corium erosion is then evaluated, thanks to the selection of MCCI ASTEC simulations (in a database with more than 300 calculations), based on:

- the occurrence of the fusible concrete gate ablation,
- the success of the water injection in the reactor pit once the fusible concrete gate has been eroded,
- the effectiveness of the evacuation of the corium decay heat (ultimate heat sink and active top-reflooding of corium),
- the corium configuration (homogeneous or evolutive) and the type of erosion (isotropic or anisotropic),
- the mass of corium in the reactor pit.

In addition, the following phenomena are considered in the APET, which may lead to the containment failure, even if the corium is properly cooled:

- ex-vessel steam explosions if secondary corium flows from the vessel occur after injection of water in the reactor pit (after fusible concrete gate erosion),
- slow containment pressurization due to the corium-basemat interaction,
- explosion of combustible gas generated during the MCCI phase.

Finally, this modelling allows to quantify the safety benefits of the new strategy for the long-term corium stabilization.

6. IMPROVEMENT OF HEALTH AND POST-ACCIDENT CONSEQUENCE ASSESSMENTS

IRSN extends its L2 PSA to L2+ PSA to characterize the radiological consequences of severe accidents (with MERCOR, see Figure 1). L2+ PSA results are presented through frequencies-consequences diagrams: for each RC, the frequency and the consequence (e.g. effective or thyroid dose, Cs137 ground deposit) are plotted. Some specific information is also used to discriminate the sequences that can be managed by the emergency offsite measures from those which cannot be. Such information is linked to emergency planning (for example, the time available before reaching some counter-measure criteria: effective dose for sheltering or evacuation, thyroid dose for iodine prophylaxis). To calculate the consequences, only one standard weather condition is considered: fixed wind direction and speed (2 $m.s^{-1}$), low diffusion, and meandering wind factor of 5. Population density around the plant is considered uniform and equal to 100 inhabitants/km^2 (mean value for the French population).

Some new health and post-accident consequence metrics have been introduced in the IRSN L2+ PSA for the 1300 MWe PWRs. The metric considered for health consequences is the number of radiation-induced thyroid cancers (lifetime), for children between 0 and 14 years old (more radio-sensitives) at the time of exposure, with an exposition during one day after the beginning the release. No protective action is considered (evacuation, sheltering, iodine prophylaxis). The number of thyroid cancers is evaluated thanks to Radrat tool [10] (developed by the National Cancer Institute, USA), by discretizing the sex (male or female) and the age (0-2 years old and 3-14 years old). These numbers of thyroid cancers are then compared to the predictable number of spontaneous (not radiation-induced) thyroid cancers. For the post-accident consequence metrics, the targeted population is the adult population of rural zones. The protection induced by the habitation is only considered. The metrics depend on the post-accident zoning described below and schematized in Figure 8:

- The “relocation zone” is defined as the zone where the population must be moved away. The criteria is an effective dose at 20 mSv/year due to external exposure during 1 year after the accident. For each dominant RC, the maximal distance and the surface of the zone, as well the induced number of radiological refugees are calculated. These parameters replace the metric “surface activity of cesium 137 in Bq/m^2” considered until now in IRSN L2+ PSAs.
- The “forbidden consumption zone” is defined as the zone where the population can still live, but in where contaminated local foodstuffs cannot be consumed. Its shape is calculated for consumption of

foodstuffs during 1 year after the accident. The criteria is 50 mSv/year for thyroid effective dose (ingestion pathway).

- A third zone is defined, inside which foodstuffs are controlled before commercialization. Three types of foodstuffs are considered (beef meat, leaf vegetables and cow milk). For each of them, the Maximum Permitted Levels (MPL) are evaluated at three different times (7 days, 1 month and 1 year).

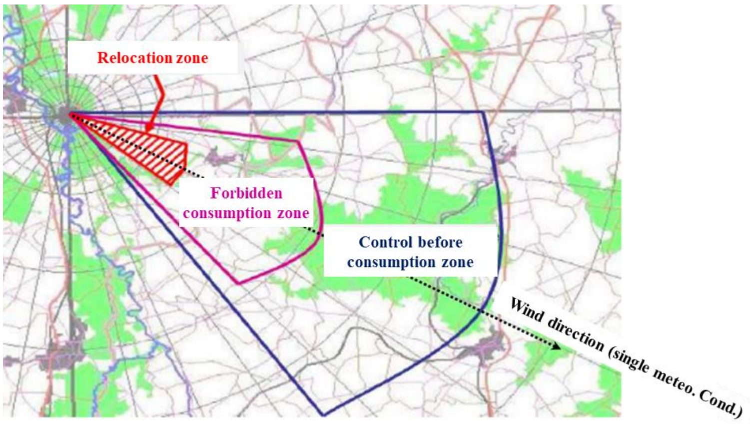

Figure 8. Schematization of Zones Related to Post-Accident Consequence Metrics

7. CONCLUSION

IRSN has introduced some new features in its L2 PSA for the French 1300 MWe PWR series, to reflect as much as possible the NPP design changes, to improve severe accident probabilistic modelling and to enhance the risk ranking. The automatic PDS generation leads to a very detailed L1-L2 PSA interface easy to update. Since this new interfacing approach is based on fault trees only, it can be implemented with most of the L1 PSA tools. The more realistic modelling of SAMG actions allows to quantify the time required for their implementation and to improve associated success/failure probabilities. The new approach to derive containment fragility curves for severe accident loading (high pressure and temperature) provides a better characterization of the containment behavior (tightness in case of an accident) while considering uncertainties. The post-Fukushima ex-vessel phase management modelling allows to quantify the safety benefits of this strategy. Finally, new consequence metrics for heath and post-accident consequences provide additional possibilities for risk rankings.

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