Severe accident risk assessment for NPPs
Software tools and methodologies for level 2 PSA development available at IRSN

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IRSN, acting as technical support organization of the French Nuclear Safety Authority (ASN), develops its own level 2 probabilistic safety assessments (L2 PSAs) for the French Nuclear Power Plants (NPPs), with high ambitions in the quality of the severe accident progression modelling in the L2 PSAs. RiskSpectrum PSA and KANT are used respectively for the L1 and L2 PSA event trees. OIPK is used to develop and maintain easily the interface between L1 PSA and L2 PSA. The radioactive releases in environment and their consequences are calculated with MER and MERCoR.

KANT, OIPK, MER and MERCOR have been developed by IRSN engineers and researchers to meet the requirements in the objectives assigned to IRSN L2 PSA. The complete chaining of these tools (including RiskSpectrum) allows the analysis of each L1 PSA accident sequence (or minimal cut set) in all release categories of L2 PSA.

I. INTRODUCTION

IRSN, acting as technical support organization of the French Nuclear Safety Authority (ASN), has developed its own level 2 probabilistic safety assessments (L2 PSAs) for the French Nuclear Power Plants (NPPs) for several years. They are used for the safety review activities and for internal uses at IRSN. To carry out these L2 PSAs, IRSN engineers and researchers have designed a set of software tools: ASTEC to calculate accident progression scenarios, KANT for the L2 PSA events trees, OIPK for the L1-L2 PSA interface and MER to calculate the radioactive releases in environment. RiskSpectrum, which is developed by LRC, is used for the L1 PSA event trees. The following objectives have been assigned to IRSN for its L2 PSAs: to limit the importance of expert judgments, to promote the use of severe accident knowledge coming from simulation tools, to be as realistic as possible and to address epistemic and stochastic uncertainties. Furthermore, the choices in modelling shall avoid cliff-edge effects (for instance, by using slopes instead of thresholds) and include uncertain events such as the start of a system during the accident further to its repair.

The Fig. 1 summarizes the chain of tools that is applied, and the paper introduces their main capabilities.

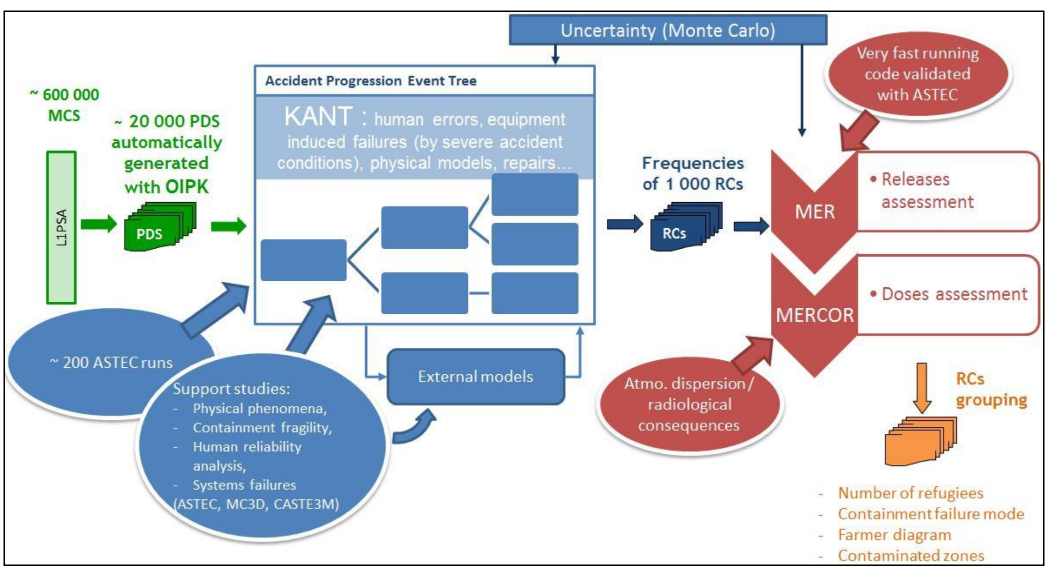

Fig. 1. Software tools for level 2 PSA development available at IRSN

II. SOFTWARE FOR THE L1-L2 PSA INTERFACE

II.A. Issues and challenges

IRSN uses RiskSpectrum PSA software for the L1 PSA, which is appropriate for the systems availability modelling, and KANT for L2 PSA, which is more adapted for physical models implementation. Consequently, the only link between L1 PSA and L2 PSA events trees is the list of Plant Damage States (PDSs). The PDSs are the equivalent in L2 PSA to initiating events in L1 PSA, i.e. they are the initiating events in the L2 PSA event trees but they include much more information. Accident sequences (minimal cut sets) from L1 PSA are grouped together into PDSs in such manner that all the accidents within a given PDS can be later treated in the same way in the L2 PSA events trees. The grouped sequences must have similar characteristics such as accident timelines, potential for loads generation on the containment or availability of systems for severe accident management.

The main challenge of the PDS definition is to transmit from L1 PSA enough details to insure the representativeness of the L2 PSA sequences, and then to get a modelling that can be easily managed. The next section summarizes the solution developed by IRSN.

II.B. Methodology and tools for the interface

The first step in the interface consists in the identification of systems, human actions and Instrumentation & Control systems (I&C) that can have an impact on accidental sequence progression after the beginning of fuel (in core or spent fuel pool) damage. Even if a L2 PSA only focuses on plant behaviour after fuel damage, the NPP accident scenario coming from L1 PSA, which is simulated with ASTEC, starts at the initiating event. That is why PDSs are also used to define some representative accident scenarios (cf. part III) and must also include information on the accident progression (thermal hydraulic phase) before fuel damage.

Systems traced in the interface are the front line systems that may have a direct impact on RCP pressure / temperature / water level, core reactivity, containment temperature, pressure or isolation. The supporting systems (electrical distribution for instance) are not considered in the interface, but their failures appear through the supporting systems availability. All human actions (from main control room or local) that influence the accident progression are also integrated in the PDS definition. The I&C indications can influence strongly the operators' and crisis teams' decisions during a severe accident and their availability is also transmitted to L2 PSA through the PDS definition.

Each PDS is defined by a set of interface variables attributes. These interface variables are defined in order to provide all needed information from L1 PSA to L2 PSA as described above. For example, Table I presents the interface variable dedicated to the High Pressure Safety Injection (HPSI) system (French 900 MWe PWR) and its different values that combine system availability, system reparability and human action status (success or failure).

TABLE I. Attributes for the interface variable High Pressure Safety Injection (HPSI) of the 900 MWe PWR L2 PSA

HPSI values	HPSI description
1	Available and started by the operators
2	Started by the operators and available until the switch in recirculation mode
3	Available but the operators have failed to start it (human error)
4	Available until the switch in recirculation mode but the operators have failed to start it (human error)
5	Not available

The second step is to verify that all the L2 PSA systems and their failure modes are modelled in the L1 PSA fault event trees and to complete the L1 PSA modelling where needed (fault trees and flags events are added to the L1 PSA Risk Spectrum modelling in order to create the information requested by the interface variables).

The third, and last step, consists in defining the PDS list with their frequency. This operation is automated thanks to the IRSN software named OIPK and described in Ref. 1. This tool filters all the L1 PSA Minimal Cut Sets (MCSs) and creates automatically the PDS list with their frequencies. With this approach, it is very easy to add a new interface variable and to modify an attribute of a given interface variable values. The level of details in the interface can be adapted if need be (e.g. for the safety injection, to distinguish the failure to start from the failure to switch in recirculation mode). OIPK can also edit all the L1 PSA MCSs included in one PDS or linked to a given value of an interface variable. It allows the full understanding of the PDSs, an accurate verification of the interface and the possibility to perform specific analysis (typically, identifying and assessing the main L1 MCS/events contributing to a risk dominant accident of the L2 PSA).

III. REPRESENTATIVE ACCIDENT SCENARIOS

III.A. Issues and challenges

Simulations of a list of representative accident scenarios shall provide to the L2 PSA event tree all the information related to the physical progression of the accident. They clearly constitute a key issue for the quality of a L2 PSA: if they are not correctly defined or performed, the L2 PSA may have "a weak physical sense" (Ref. 2) and some risks might be ignored, underestimated or overestimated. Consequently, each L2 PSA developed at IRSN is supported by a large and varied set of ASTEC accident simulations (typically between 100 and 300). This significant effort aims at ensuring the representativeness of the L2 PSA and limiting as much as possible approximation in the modelling. These ASTEC simulations are defined with the PDS list and are stopped at the vessel failure (the ex-vessel phase is studied in decoupled studies). Nevertheless, the PDS attributes contain useless information for ASTEC simulation (for example, the status of the containment isolation system, which is only considered in KANT and MER). Therefore and luckily, different PDS may be associated to the same ASTEC simulation. Thus, it appears necessary to have a strong and efficient bridge between PDSs and definition of ASTEC simulated sequences.

III.B. Methodology for the definition of the representative accident scenarios

To facilitate the definition of a set of ASTEC accident simulations, a graphical interface has been implemented in OIPK. It allows an additional PDS merging through a "merging" tree formalism. Each branch of the tree has its own frequency. The "head events" considered in a tree are the interface variables. Each final branch corresponds to the definition of an ASTEC run. In the nodes of such a "merging" tree, several values of a given interface variable can be grouped together (if their values lead to the same NPP behaviour in the given context). OIPK automatically updates the frequency of each node from the L1 PSA results. A more detailed presentation of this tool is available in Ref. 1. It is then easy to identify the branches that are not significant and to adapt the grouping accordingly. This "merging" tree functionality is a flexible tool to define and to justify the set of ASTEC accident simulations based on PDS. It can also be used to reproduce some L1 PSA event trees.

In Fig. 2, a schematic example is given to illustrate the "merging" trees used to define the set of ASTEC simulations. First of all, PDS are grouped in several families according to the state of the nuclear unit at the core uncover time. Then, for each PDS family, the set of ASTEC runs is determined in accordance with a cut-off frequency and the interface variables influence on the accident progression. For instance, in Fig. 2, VAR 1 influences the definition of ASTEC calculations for the family 2 and VAR 2 influences the definition for family 1. Some different attributes can be grouped in a same ASTEC calculation (see branch "1.1 & 1.2" of family 2).

III.C. Calculation of the representative accident scenarios with ASTEC

The ASTEC integral code (Refs. 3 and 4) can model the whole reactor. Each calculation starts from the initiating event and takes into account automatisms and accident management procedures. In connection with R&D activities, many efforts are done by IRSN and around 30 partners to improve the ASTEC capabilities and to increase the quality of accident simulations (Ref. 5). In the simulations, when required systems and components are available and when there is no human error, it is assumed that the operating actions prescribed by Emergency Operating Procedures (EOPs), i.e. before core uncover, are applied with credible delays after associated physical criteria are reached. Actions, associated criteria and delays are programmed in the ASTEC datasets. Concerning human actions after core uncover, each representative sequence is calculated twice: the first time without any manual action after core degradation, and the second time with all possible Severe Accident Management Guideline (SAMG) actions after core degradation (SAMG actions are considered only if dedicated systems are available).

The analysis of the results allows the L2 PSA developer to validate the reactor status (damaged or safe core) at the end of L1 PSA sequences and control how realistic or conservative are the L1 PSA assumptions. For example, ASTEC calculations have shown that, for some scenarios with HPSI available until the switch in recirculation mode, it was possible to connect the residual heat removal system before core melt, whereas it was assumed in L1 PSA that the loss of HPSI would necessarily lead to core degradation whatever its failure mode.

For the latest version of the 900 MWe PWR L2 PSA, 140 sequences are calculated with ASTEC V2.0 at nominal power and 100 other sequences have been calculated for shutdown states. This extended set of accident simulations provides information on EOPs and SAMG efficiency, and constitutes a base of knowledge on NPP behaviour in case of an accident that can be used in other applications than L2 PSA (for example at the Technical Crisis Centre).

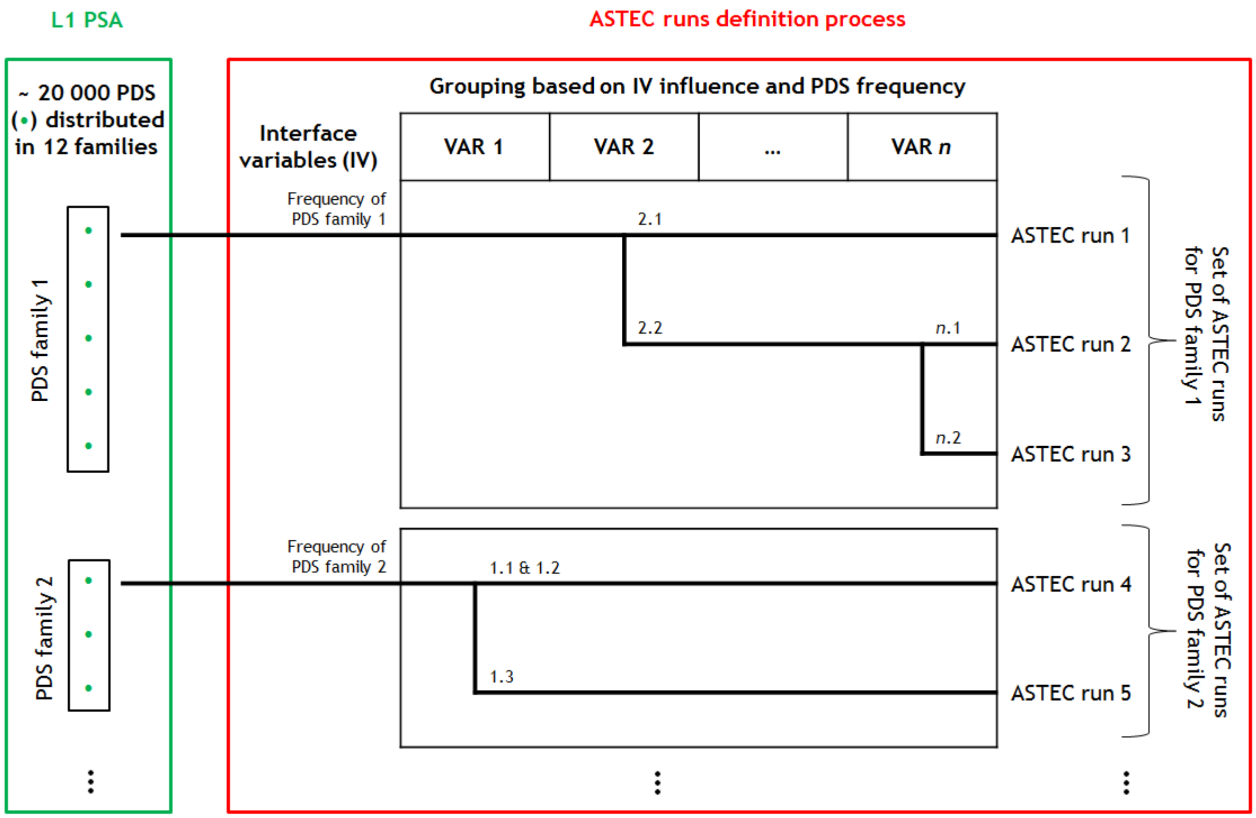

Fig. 2. Principle used by OIPK to define the ASTEC runs that support a L2 PSA

IV. CONSTRUCTION OF THE ACCIDENT PROGRESSION EVENT TREE WITH KANT

IV.A. Issues and challenges

IRSN has developed a L2 PSA event tree tool, named KANT. It must be able to:

- integrate all information from L1 PSA through the PDS list and their frequencies;
- implement representative information about accident scenarios (cf. part III);
- relate the chronology of the severe accident and integrate physical models, human errors and systems;
- model all dependencies between events (physical, systems activations and failures);
- trace the value of physical variables during the severe accident (pressure, volume, mass, etc.);
- integrate uncertainties and perform uncertainty analysis by Monte-Carlo simulation;
- take into account all the containment failure modes;
- quantify the frequency and the radiological consequences of severe accident sequences;
- perform the post-processing of the results.

The main benefit of L2 PSAs is to assess events and combinations of events which are not considered in deterministic studies and might lead to severe consequences. In this way, it is a challenge for L2 PSAs to take into account interactions between events and to model physical phenomena while integrating the state-of-the-art knowledge and uncertainties.

IV.B. Accident Progression Event Tree (APET) principles

IRSN applied the event tree formalism which is well known and allows the L2 PSA developer to represent in a simple and easily understandable way the succession of events that would occur during a severe accident. The event can be binary (e.g. system activation) or more complex (e.g. conditional probability of a containment failure given some severe accident conditions).

To deal with the necessity of strong correlation between events and to consider complex events, IRSN has developed the concept of Accident Progression Event Tree (APET) that describes the sequence of events occurring after the onset of core degradation and might lead to containment failure and radioactive releases. The APET is organized by a succession of ordered nodes (i.e. the top events of the event tree) which relates the chronology of the accident (cf. Fig. 3). Each node corresponds to a human action, a physical phenomenon or a system activation. The inputs are a list of global variables which describes the NPP state before the event and the outputs are, for each branch, an updated list of variables on the NPP state after the event and a probability. In comparison with a classic event tree (where branches are defined in advance), each node generates dynamically (during the quantification) one or several branches. The Fig. 3 provides an example.

A specific solution is applied for the modelling of the physical phenomena. A physical model is a function that links upstream variables and downstream variables where variables are calculated through a "deterministic" approach (Fig. 4). Two categories of upstream variables are distinguished: variables that describe the status of plant (systems availability and physical states of reactor) and uncertain variables associated to a distribution law that are random choose by a Monte-Carlo pulling. Downstream variables describe the physical status of the reactor after the considered phenomena. For the construction of these physical models, three methods are used: grid of results (results from deterministic sensitivity studies), response surface (applied when discontinuities are not too important) and simplified model (physical correlations).

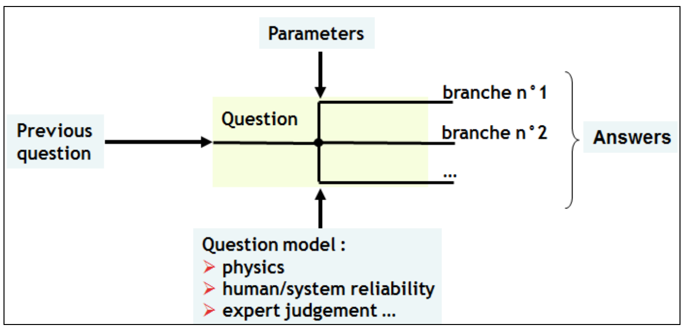

Fig. 3. APET – branches / probabilities

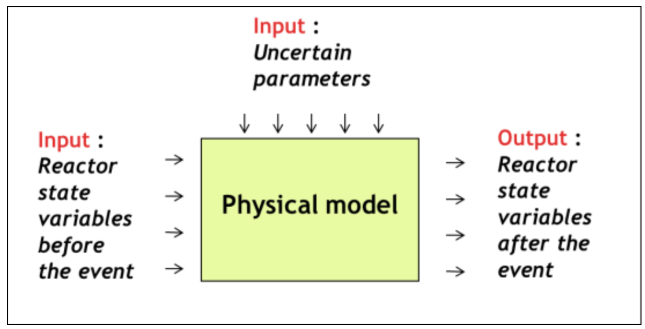

Fig. 4. Uncertainties modelling

Fig. 5 presents all the physical phenomena integrated in the APET of PWR L2 PSA. All physical models are strongly coupled: input data of a downstream model are the output data of an upstream one. For instance, in IRSN last 900 MWe PWR L2 PSA, amount of hydrogen is first initialized from results of ASTEC calculations for the corresponding representative sequences. Then, it is updated considering the following events: reflooding during core degradation (additional production of hydrogen), DCH at vessel rupture and additional hydrogen production due to Molten Corium-Concrete Interaction (MCCI) during ex-vessel phase.

At the beginning of the APET, one of the first nodes assigns a representative ASTEC accident scenario calculation (see III) to the accident progression considered in the APET. This is a complex node using a selection tree (with dedicated rules) that associates to each PDS (and SAMG actions) an appropriate ASTEC calculation. Thanks to new functionalities of KANT, the APET can also integrate ASTEC results in function of time, for instance, the time evolution of vapour and hydrogen masses in the containment.

The structure of the APET also allows the awareness of all the containment failure modes: in case of multiple containment failure modes for a given sequence, the latest failure modes are not hidden by the earlier ones (see Ref. 6).

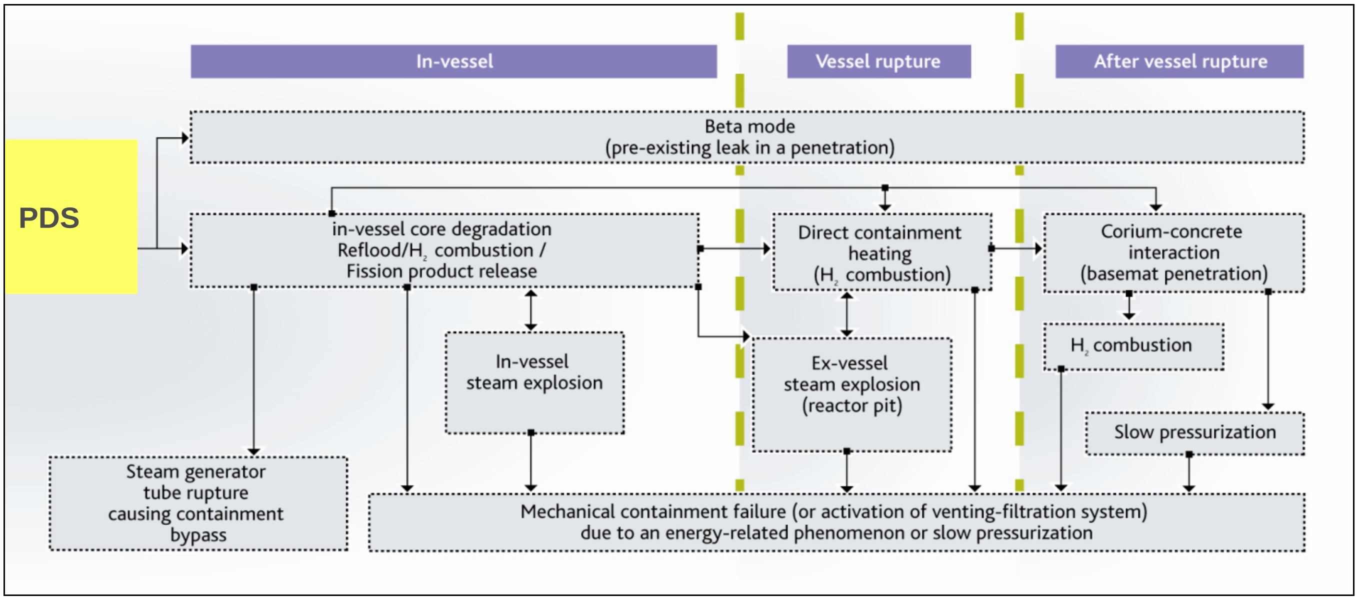

Fig. 5. Physical phenomena included in the APET

Uncertainties are assessed on the basis of a set of uncertain parameters that can concern different aspects, like uncertainties on the physical state of reactor or confidence on severe accident codes (see Fig. 4). The uncertain parameters are sampled for the Monte-Carlo simulation. In addition, thanks to the new functionality which allows to read ASTEC results in function of time, it is also possible to implement in the APET some models able to simulate random events (which are mainly containment spray repair and primary circuit make-up restoration). For example, to cover the cases of random start of Safety Injection System (SIS) during degradation phase, first an ASTEC calculation without SIS start is run. Then, in the APET, an instant of SIS start is generated randomly (Monte Carlo method). Once this instant is known, the level of core overheating is read by KANT in files extracted from the ASTEC run. Finally, the duration of the reflooding, the mass of hydrogen produced during this phase and the time of vessel rupture are evaluated on the basis of simplified physical models that includes uncertain parameters (like the oxidation power during reflooding). The global variables of the APET such as containment pressure or containment pressure peaks due to hydrogen combustion are updated consequently.

IV.C. KANT computational tool

As input for its calculation, KANT takes the existing PDS (see II) and their frequency identified by OIPK software. It provides functionalities to develop the APET structure as described above. It can be noted that the construction of the APET is automated and that a node is actually a simple computer program. The coding uses both the KANT language (logical function for system availability, simplified physical models for physical phenomenon, etc.) and external C++ functions.

The quantification of an APET with KANT leads to thousands of accident scenarios. If the user chooses to run uncertainty quantification with KANT, the latter will produce the Monte Carlo draw. The same draw is then used in tools presented in part V. The KANT software allows parallelized calculations of radioactive releases (on multi-core machines or on several workstations). The gain obtained on APET quantification delays allows the user to reduce the event tree cut-off frequency and to increase the number of Monte Carlo runs for the assessment of uncertainties. Furthermore, recent functional features also allow a friendly development of APET and an easy post-processing of the results of quantification.

Other recent developments have also been introduced in KANT software in order to facilitate the post-processing of quantification results and the comprehension of these results. For example, (1) the list of PDSs that leads to each release category and their contribution to the RCs frequency can be calculated, (2) for each PDS, KANT can provide the list of RCs generated by this PDS and the contribution of the PDS to these RCs, (3), an output file describing all the L2 PSA scenarios generated during a quantification (PDS name, its interface variables values and frequency, the path through the APET and the name of the release category with the value of all RCs variables).

V. ASSESSMENT OF SEVERE ACCIDENT CONSEQUENCES

V.A. Issue and challenges

Characterization of consequences of severe accidents is part of state of the art L2 PSA (Ref. 2) as they constitute a key result for risks assessment. It is therefore important to perform a realistic assessment of radiological releases for the different release categories otherwise, accident scenarios with very different levels of radiological consequences would not be distinguished and some risks might be ignored, underestimated or overestimated. Assessment of consequences of severe accidents needs to take into account the chronology of radioactive releases (kinetics) and to perform with uncertainties a realistic assessment of radioactive releases. Thanks to these assessments, a ranking of the different containment failures is possible, which is an indispensable step in the L2 PSA. With the aim of ranking risks and presenting the results in an understandable way, it might also be interesting to use synthetic risk metrics such as short, mid and long term radiological consequences.

V.B. Methodological solution

Release categories (RCs) are defined at the end of the APET. A set of variables calculated through the APET (variables that enables to define the kinetics and level of release of an accident sequence) is used for the grouping of sequences into release categories. This approach allows a detailed definition of release categories (around 1 000 for the last 1300 MWe PWR L2 PSA, covering power and shutdown states). Amplitude and kinetics of releases are then calculated with a dedicated software (MER) as well as order of magnitude of radiological consequences with standard assumptions.

V.C. Computational tools

MER ("Modèle d'Evaluation des Rejets" in French) is used to assess the amplitude and the kinetics of release for each release category generated by the APET of the L2 PSA and MERCoR ("Module d'Evaluation Rapide des COnséquences Radiologiques" in French) is used to calculate the radiological impact (dose in Sievert and aerosols ground deposit versus time) of release at some few specific distances of the nuclear power plant, and for a given standard meteorological condition.

L2 PSA quantifications performed by IRSN with KANT software can generate thousands of release categories and so many calls to the calculation of release levels and radiological consequences with MER and MERCoR. Therefore, the principal objective of MER and MERCoR is to provide realistic orders of magnitude in terms of releases and radiological consequences in the environment with very short calculation times (a few second). A quick view of MER and MERCoR is presented below (a detailed presentation is available in Ref. 7 and Ref. 8). They can be directly run from KANT what facilitates and significantly optimizes the development of L2 PSA in IRSN

V.C.1. Status and description of MER software

Two versions of MER are available today. One version is adapted for French 900 MWe PWRs and another one for 1300 MWe PWRs. The adaptation of the code for any other PWR is quite easy and is being done at the moment for EPR. All assumptions introduced in MER (fraction of release for each element from the core, chemistry in the containment, etc.) take into account R&D results on fission products emission and behaviour in containment, especially the VERCORS, PHEBUS FP and ISTP programs. Uncertainty evaluations on radioactive releases are performed using a Monte-Carlo method, and sampling for uncertain parameters (containment break size, mass of release radio-elements (from fuel) at core meltdown time, etc.) with the Latin Hypercube methodology. The models introduced in MER are sometimes simplified in comparison with ASTEC ones but the consistency of both tools is checked and discussed within IRSN. The ASTEC code is used for the validation of MER. The main difference between MER and ASTEC is the fact that MER does not calculate the severe accident physical phenomena but only the consequences of these phenomena on the radio-element behaviour. For example, hydrogen combustion induces reemission of aerosol deposited in the reactor building. This reemission is calculated in MER but not the hydrogen combustion peak, which is done in the APET or with ASTEC. This explains why MER can be so fast running. The results of MER calculations are the following: environmental release (mass or activity) for each fission products group, for each phase of the accident and each release pathway; suspended activity/mass in the buildings; fission products deposit in the buildings and mass of fission products retained in the filters of ventilation systems.

V.C.2. Status and description of MERCoR software

To introduce directly a calculation of accident consequences in terms of dose or ground deposit in the APET of L2 PSAs, IRSN has developed the MERCoR software that is able to calculate in a few seconds the impact of a release at some few specific distances of the nuclear power plant and for one standard meteorological condition. All the data provided by MERCoR help to describe the accidental sequences, but also to estimate the global risk associated with the nuclear power plant. The results of MERCoR calculations are the following: dose equivalent and thyroid dose at given points (in the axis of the plume) versus time (corresponding to total release and each pathways); deposited activities 2 weeks after the beginning of the accident, among which 137Cs deposited activity (corresponding to total release and each pathways) and data linked to emergency preparedness such as the time counter-measure levels are reached, and at a given time, the geographical extent of each countermeasure.

VI. RESULTS PRESENTATION

VI.A. General point of view

As said in Ref. 2, the frequency and magnitude of offsite releases should be considered together for the interpretation of the L2 PSA and its applications. The product of these two quantities is a good estimation of the overall risk from the plant. Moreover, the insights gained from this quantitative evaluation of radionuclide releases should be discussed together with the results of the uncertainty analysis.

VI.B. Example of the L2 PSA for French 900 MWe NPPs (IRSN modelling)

In the IRSN L2 PSA for 900 MWe PWRs, Release Categories (RCs) are defined by 37 specific parameters related to initial reactor state, Containment Failure Modes (CFMs) and accident kinetics. Due to the large number of possible values for each parameter, several thousands of RCs are generated by the APET quantification. To make the final presentation easier, those thousands of RCs are gathered into larger categories, called “Regrouped Release Categories” (R-RCs). They are defined as a function of the considered CFMs, of the phase of accident, of the delay between the initiating event and the beginning of atmospheric releases. For each sequence in the IRSN APET, the quantification of all top events is performed along the full time evolution, even if a first CFM is obtained earlier in the sequence. Therefore, many L2 PSA sequences include several CFMs.

To present the R-RCs frequencies, two methods are used at IRSN. In the first method, R-RCs are defined by the first CFM in the accident progression (classic method). In this case, the sum of all R-RC frequencies is equal to the total core damage frequency from L1-L2 PSA. In the second one, R-RCs include all L2 PSA sequences having one particular CFM, even if this CFM is not the first one to occur. In this case, the sum of all R-RCs is higher than the core damage frequency from L1-L2 PSA. With the classic method, later CFMs, although leading to important radioactive releases, may be hidden by earlier ones. Unlike the first method, the evaluation of the real frequency of late CFMs is possible with the second one. To present the R-RCs releases and doses, representative RCs are chosen in each R-RCs. Calculations, taking into account uncertainties, are performed for these representative RCs (short and long term consequences in the vicinity of the damaged plant). The R-RCs are gathered into “classes” (9 classes for the 900 MWe L2 PSA) according to the dose magnitude, and are represented on a frequency-consequence diagram or dose vs. frequency (so-called “Farmer diagram”).

Regarding the emergency measures taken in case of radioactive release, the following calculations have also been performed for each R-RC: the maximum time allowed to take these measures for a given distance to the damaged plant and the distance below which these emergency measures have to be taken for a given time after the beginning of the accident.

Regarding the long term consequences, the extent of contaminated lands has been assessed (based on Cs 137 isotope and contamination thresholds defined following the Chernobyl accident). Finally, from a given RC, it is possible to go up until the related L1 PSA MCS and to know all the physical phenomena that occurred during the cross of the APET.

VI.C. APET results presentation with uncertainties

For any type of result, a probabilistic distribution is generated and cumulative percentiles can be calculated (e.g. 5%, 50% and 95%). For a L2 PSA with quantitative assessment of uncertainties, the results are displayed using histograms, probability density functions, cumulative distribution functions and tabular formats, showing the various quartiles of the calculated uncertainties, together with the distribution mean and median estimates.

VII. CONCLUSION

The software tools and methodologies described in this paper allow IRSN to produce detailed level 2 PSA for the French 900 MWe and 1300 MWe series. Thanks to them, for each level 1 PSA accident sequence, IRSN engineers and researchers are able to perform detailed analysis of accident progression, and radiological releases and consequences. Moreover, from a given release category, it is possible to go up until the related level 1 PSA minimal cut set and to identify all the physical phenomena that occurred along the accident progression in the accident progression event tree. These L2 PSAs are now being progressively updated following NPPs upgrades by EDF or progress in research on severe accident. A L2 PSA for EPR is also under progress. These activities provide many insights for IRSN regulatory review activities of NPPs safety and severe accident management. The described methodologies and software tools can be applied on most NPPs.

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