Integrated Deterministic-Probabilistic Safety Assessment Methodologies

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IDPSA (Integrated Deterministic-Probabilistic Safety Assessment) is a family of methods which use tightly coupled probabilistic and deterministic approaches to address respective sources of uncertainties, enabling Risk informed decision making in a consistent manner. The starting point of the IDPSA framework is that safety justification must be based on the coupling of deterministic (consequences) and probabilistic (frequency) considerations to address the mutual interactions between stochastic disturbances (e.g. failures of the equipment, human actions, stochastic physical phenomena) and deterministic response of the plant (i.e. transients).
This paper gives a general overview of some IDPSA methods as well as some possible applications to PWR safety analyses.

IDPSA (Metodologías Integradas de Análisis Determinista-Probabilista de Seguridad) es un conjunto de métodos que utilizan métodos probabilistas y deterministas estrechamente acoplados para abordar las respectivas fuentes de incertidumbre, permitiendo la toma de decisiones Informada por el Riesgo de forma consistente. El punto de inicio del marco IDPSA es que la justificación de seguridad debe estar basada en el acoplamiento entre consideraciones deterministas (consecuencias) y probabilistas (frecuencia) para abordar la interacción mutua entre perturbaciones estocásticas (como por ejemplo fallos de los equipos, acciones humanas, fenómenos físicos estocásticos) y la respuesta determinista de la planta (como por ejemplo los transitorios).
Este artículo da una visión general de algunos métodos IDSPA así como posibles aplicaciones al análisis de seguridad de los PWR.

INTRODUCTION

IDPSA (Integrated Deterministic-Probabilistic Safety Assessment) is a family of methods which use tightly coupled probabilistic and deterministic approaches to address respective sources of uncertainties, enabling Risk informed decision making in a consistent manner. The starting point of this framework is that safety justification must be based on the coupling of deterministic (consequences) and probabilistic (frequency) considerations to address the mutual interactions between stochastic disturbances (e.g. failures of the equipment, human actions, stochastic physical phenomena) and deterministic response of the plant (i.e. transients). Thus it can be considered as a complementary to PSA and DSA approaches that intends to help in:

- Resolving time dependent interactions between physical phenomena, equipment failures, control logic, operator actions in analysis of complex scenarios;
- Identification and characterization of a-priori unknown vulnerable scenarios (sleeping threats) of the overall system;
- Consistent treatment of different sources of uncertainties; and
- Reduction of reliance on expert judgment and assumptions about complex time dependencies and scenarios.

Such type of methods would allow then to

- realistically quantifying safety margins with uncertainty estimation;
- identifying possible incompleteness, over- or false- conservatism in existing PSA and DSA models;
- increasing transparency and robustness of risk-informed decision making;
- improving of plant safety and operation, by assessing the risk of an accidental sequence at an early state of its development; and
- potentially reducing the cost of safety analysis due to larger involvement of computers in what they can do better: multi-parameter, combinatorial exploration of the plant scenarios space.

Past developments that would fit into IDPSA framework can be sorted into two major families:

1) Dynamic Event Tree (DET), also called Discrete Dynamic Event Tree (DDET):
   - DYLAM (JRC-Ispra) and DETAM (MIT).
   - ISA (Integrated Safety Analysis, MOSI-CSN), developed since mid-90's from previous developments, in collaboration with UPM and Ekergy Software. It includes DENDROS (DET unfolding engine), SIMPROC (Procedures Simulator) and set of TH/SA codes.
   - ADS-IDAC (U. Maryland, sponsored by NRC) using RELAP5; mostly focused in human reli-

ability analysis. More recently PSI, in collaboration with Swiss regulator, has applied ADS-IDAC coupled with TRACE.
- ADAPT (Accident of Dynamic Accident Progression Trees; OSU with support from SNL) using MELCOR. Focused on PSA level 1 and level 2.
- MCDET (GRS). It combines DET generation, MELCOR simulation and Monte Carlo (MC) sampling (uncertain parameters and times), to obtain CDF with confidence intervals.

2) Approaches which do not use event tree formalism for the exploration of the event space:
- Genetic Algorithm (GA) - IDPSA, developed in collaboration between KTH (Royal Institute of Technology, Sweden) and Moscow Power Engineering Institute. It takes benefit of global optimum search (GA) algorithms for identification of vulnerable sequences in the space of time dependent plant events. GA-IDPSA is coupled to RELAP5 and MELCOR for analysis of PSA level 1 and level 2 scenarios.

Since 2011 the IDPSA network, which includes over 20 European and US organizations, provides a platform for regular exchange of information and experience between IDPSA research, developers and potential users.

The goal of this work is to provide and discuss examples of IDPSA methodologies applications and identify some needs for future pending developments.

EXPERIENCE OF APPLICATION OF GA-IDPSA

GA-IDPSA approach

Recent development of the GA-IDPSA approach enables computationally efficient exploration of the uncertainty space which includes both scenario (aleatory) and modeling (epistemic) uncertainty.

The purpose of the event space exploration in GA-IDPSA is to identify a vulnerability in the plant safety system. Vulnerable scenario is defined as a sequence of initiating events, component failures, plant control and safety systems operations that can lead to a failure of a safety barrier (e.g. fuel cladding, pressure vessel, containment etc.). This problem is formulated in GA-IDPSA as a global optimum search problem for a fitness function, Y=F(U). Where U is a vector of uncertain parameters which determines the search space, and Y is the fitness function defined by critical values of system parameters (e.g. cladding temperature, containment pressure, magnitude of release, etc.) which determine performance of engineered safety barriers (cladding, reactor pressure vessel, containment, etc.).

There are two major classes of tasks for the GA-IDPSA analysis:

i) identification of a "worst case" scenario with most severe violation of safety limits, which corresponds to the global maximum of the fitness function; and

ii) identification of "failure domains" in the plant scenario and modeling uncertainty space where fitness function exceeds a threshold associated with respective safety limit.

The difference among them is in the way how fitness function is determined. For solving the first task it can be original safety-critical parameter (e.g. such as containment pressure, or amount of released radioactivity). For the second task the fitness function is modified such that it reaches its maximum at the boundary of the failure domain.

Main advantage of GA-IDPSA is that instead of "brush" search, GA uses "clues" provided by the previous response of the fitness function to the variation of uncertain parameters in order to make intelligent selection of the new points in the uncertainty space and thus significantly reduce computational efforts necessary for identification of the failure domains. GA-IDPSA can handle search in ~102 dimensional uncertainty space using realistic plant models and affordable computational resources.

A vulnerable scenario can be characterized by both (i) frequency of events which lead to such scenario (aleatory uncertainty), and by (ii) probability (expressed in terms of degree of confidence, or beliefs) of failure in the given scenario which is determined by the modeling (epistemic) uncertainty. GA sees no difference between such parameters in the process of exploration of the uncertainty space. Nevertheless, frequency and probability characteristics of the identified failure domains can be quantified in the aftermath. GA pertains to the class of stochastic methods and it can provide a way for estimation of the probabilities P(Y > Ytarget) based on the points generated in the process of the exploration of the uncertainty space. It is a stochastic biased method, because points in the uncertainty space are biased towards Y > Ytarget. It is instructive to note that, unlike in standard biasing approach in the MC method, GA uses adaptive biasing.

Example of GA-IDPSA application

Analyses presented in this section were carried out using a realistic model of the WWER-1000/V320 NPP implemented in the RELAP5 code. The effects of activation of high pressure injection system which consists of low capacity (JDH) and high capacity (JDN) systems are considered.

Case 1: Most dangerous size of the break in LOCA

A hypothetical transient accident scenario with failure of 4 hydro accumulators (HA) and all 4 pumps of JND system is considered for 9000 second

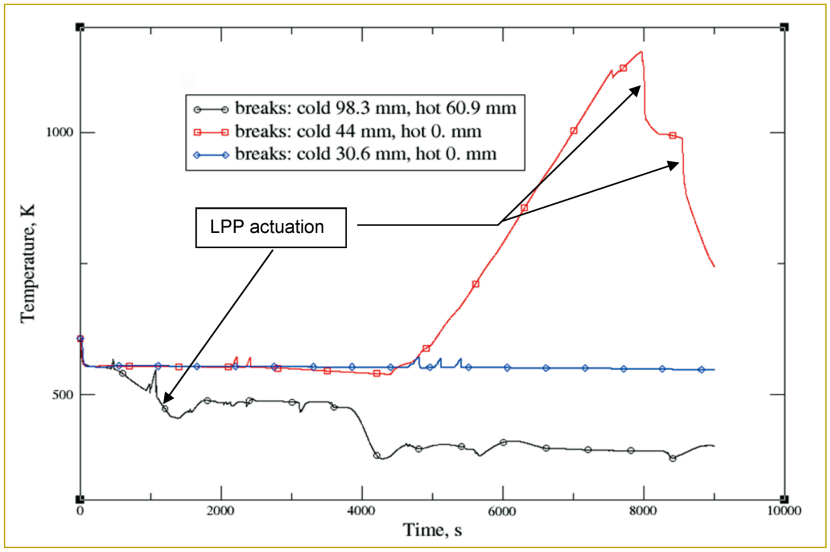

Figure 1. Dependence of fuel cladding temperature for hot channel from time of accident (Case 1).

after the pipe break. Diameters of the break in cold and hot legs were varied from 0 to 123 mm. The peak fuel cladding temperature is used as a fitness function in the GA search. Total computational time was 153.6 hours on a cluster with 21 CPUs.

GA-IDPSA can correctly identify that the medium size break LOCA is more dangerous. It also allows finding the most dangerous sizes of the breaks. Specifically the "worst case", determined by the highest PCT, is identified for cold leg break sizes in the range of 44 – 46 mm in diameter and for the hot leg break sizes in the range of 0 – 0.2 mm in diameter. Larger or smaller sizes of the break are much less dangerous in terms of resulting peak cladding temperature. This finding is illustrated in Figure 1.

Case 2: Influence of the timing of the operator action

The same plant configuration as in the Case 1 is considered in this case. Uncertainty space was formed by 58 variable input parameters in the RELAP5 model. Specifically we consider variations of the following parameters of the model: cold and hot legs breaks area, possibility of successful of one of the elements of the JND system and failure of 3 others, variation of the time delay for actuation of the JDH system, timing of manual actuation by the operator of fast pressure reduction system to the turbine condenser (TBV), and timing of possible manual activation by operator of the JDH injection system into the pressurizer. Size of the breaks varied simultaneously and independently on hot and cold legs was from 0 mm to 89.1 mm. In addition, the following epistemic uncertainties in modeling were considered: discharge coefficients of Henry-Fauske model for cold and hot breaks; heat transfer coefficients from heat structures to the coolant in the core and steam generators. The ranges for variations of the parameters are based on the earlier analysis. The peak fuel cladding temperature was used to guide GA in exploration of the uncertainty space.

A domain in the parametric space was identified where the fuel cladding temperature is higher than 1300 K and sometimes close to 1400 K. In total 943 cases were calculated and among them 7 cases reach peak cladding temperature above 1300K. Calculations were carried out on a cluster with 72 CPUs. Computational time was 54.1 hours. Worst case scenario is shown in Figure 2.

Analysis of the parameters in the most interesting transients with highest peak temperature suggests that:

i. There is a strong interaction between the components of the uncertainty space in their influence on the peak cladding temperature.
ii. Several parameters have a trivial influence on cladding temperature. The worst case can be achieved when e.g. working JND pump is located on the leg with the break. It is instructive to note that one should avoid such trivial attractors by imposing appropriate restrictions on the search process, etc.
iii. Influence of several components are not obvious a priory. Namely the influence of the combination of the cold and hot legs breaks sizes; negative influence of the JDH system injection into the pressurizer during the time window of about 0-1800 sec. before maximum temperature peak is achieved, etc.

Operator action plays crucial role in this scenario. Manual injection of the JDH in the pressurizer affects primary system pressure and changes the heat transfer direction between primary and secondary side in the steam generator. This delays activation of low pressure HA (Figure 3), since only after about 1000 seconds the pressure becomes low enough for its activation. Such adverse effect of the manual activation of high pressure injection into pressurizer is observed only within certain time window, and not identified "a priori".

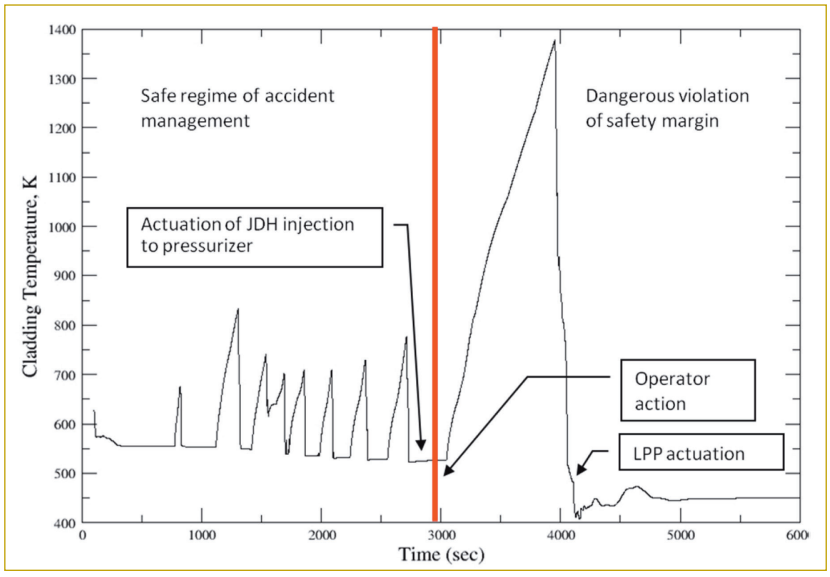

Figure 2. Time history of the fuel cladding temperature for the hot channel in the worst case scenario (Case 2).

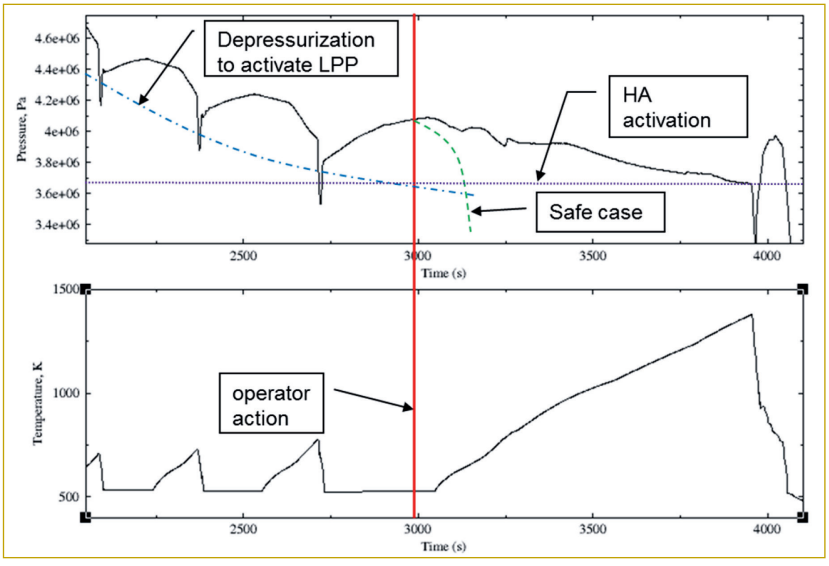

Figure 3. Transient pressure and temperature in the primary system for the "worst case" scenario in Case 2.

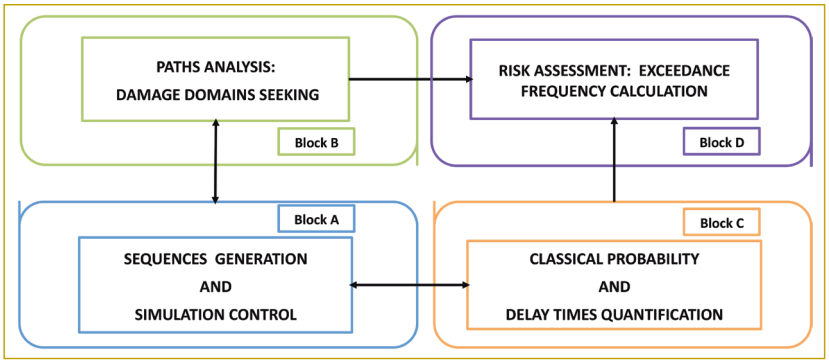

Figure 4. ISA methodology general diagram.

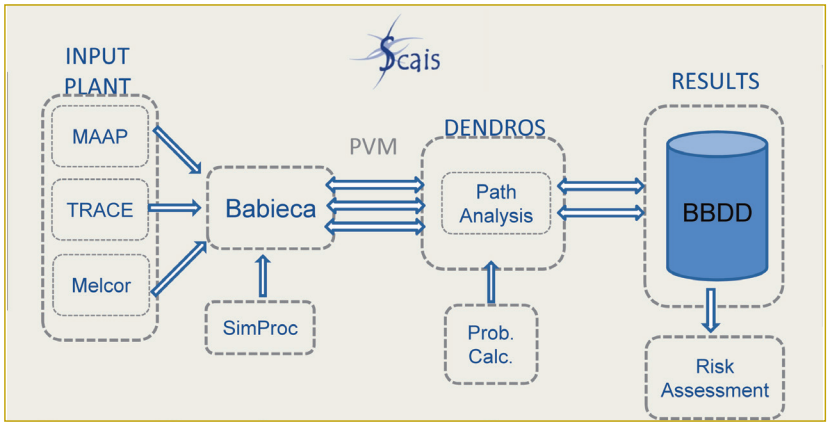

Figure 5. SCAIS components.

APPLICATIONS OF ISA METHODOLOGY

Integrated Safety Assessment (ISA) methodology has been proposed by the Modeling and Simulation (MOSI) of the Spanish Nuclear Safety Council (CSN). It is an adequate method to perform uncertainty analysis, especially suited for those sequences where some events occur at uncertain times. Apart from a theoretical approach, the basis of the method, application of ISA requires a set of computational tools. A suitable software package called SCAIS (Simulation Code System for ISA) implements the above referred method.

The numerical result of this methodology consists of the damage exceedance frequency (DEF) for the sequences stemmed from an initiating event. This is done along with the delineation of the dynamic event tree and the identification of the damage domain (DD) of the sequences that contribute to the total DEF. The DD is defined the region of the space of uncertain parameters of interest that results in damage. The UPM group has applied extensively this methodology in several projects. This paper illustrates the main results of application of ISA methodology to Steam Generator Tube Rupture (SGTR) and Station Blackout (SBO) sequences.

In previous analyses, ISA has been successfully applied to Loss of Component Cooling System, Upper Head SBLOCA, Lower Head SBLOCA, Cold and Hot Leg LOCA, hydrogen concentration inside containment, Total loss of Feedwater System and RCP trips safety issues.

This methodology consists of the following components (see general block-diagram in Figure 4) that parallel corresponding blocks in SCAIS (Figure 5):

- Block A, the sequence generation module that performs the simulation of dynamic event trees, with DENDROS-BABIECA-MAAP, Figure 5. This provides the candidate sequences with non-trivial DD (success or damage for all conditions) that will be analyzed in detail in the Path Analysis module (Block B).
- Block B, the paths analysis module that simulates repeatedly each sequence of interest obtained in Block A with different values of uncertain parameters and/or time delays (human actions or stochastic phenomena). Each such simulation, called a path, can end either in a success or damage state. The region of parameter and/or time delays values that lead to damage paths is the DD of the sequence.
- Block C, the probability and delay times quantification module that provides the necessary information to calculate in Block D (Risk Assessment) the probabilities and the contribution to DEF of each sequence of interest.
- Block D, the risk assessment module that calculates the DEF by integrating on the DD region the probability distributions obtained in Block B (Path Analysis module).

Dynamic Event Tree for SGTR sequences

In this ISA application an analysis of Steam Generator Tube Rupture (SGTR) sequences in a in a three-loop PWR Westinghouse at full power, has been performed with SCAIS-MAAP and RADTRAD codes.

The event begins with a breach barrier between the primary Reactor Coolant System (RCS) and the secondary side of the steam generator (SG), providing a direct release path for RCS fluid to the environment via the secondary side (steam-dump, safety and relief) valves. The classical main stages of SGTR sequences are:

1. Reactor trip and Safety Injection (SI) signal.
2. Identification and Isolation of the ruptured SG.
3. Cooldown of the RCS system by means of the intact SGs.
4. Depressurization of RCS to restore inventory.
5. Terminate SI.
6. Long term cooling.

Generic headers for these are firstly obtained from SGTR event trees (ET) of PSA studies of similar NPPs. This allows to identify relevant human actions within Emergency Operating Procedures (EOPs), related to items 2 to 6 of previous list, thereafter used in DET delineation.

The study (a) considers that SCRAM, AFW and REC-HP (High Pressure Recirculation) accomplish their respective success criteria; (b) and do not cover RWST (Refill of Water Storage Tank) and RHR (Residual Heat Removal) phases. Therefore, the set of headers considered includes: H (High Pressure Injection system), I (Isolation rupture steam generator), SD (RCS cooldown at maximum rate), LD (RCS cooldown at 55 K/h), and R (SI termination). All headers have two possible statuses (success or failure) except header H that uses three possibilities: 2 trains, 1 train or 0 trains available.

In summary, twenty four operator actions, corresponding to EOPs E-3,

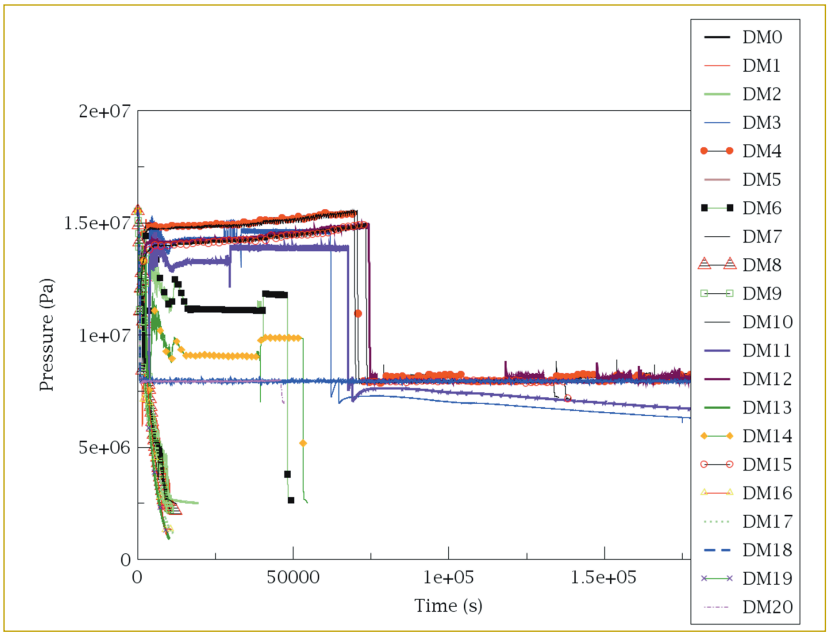

Figure 6. RCS pressure. SGTR DET.

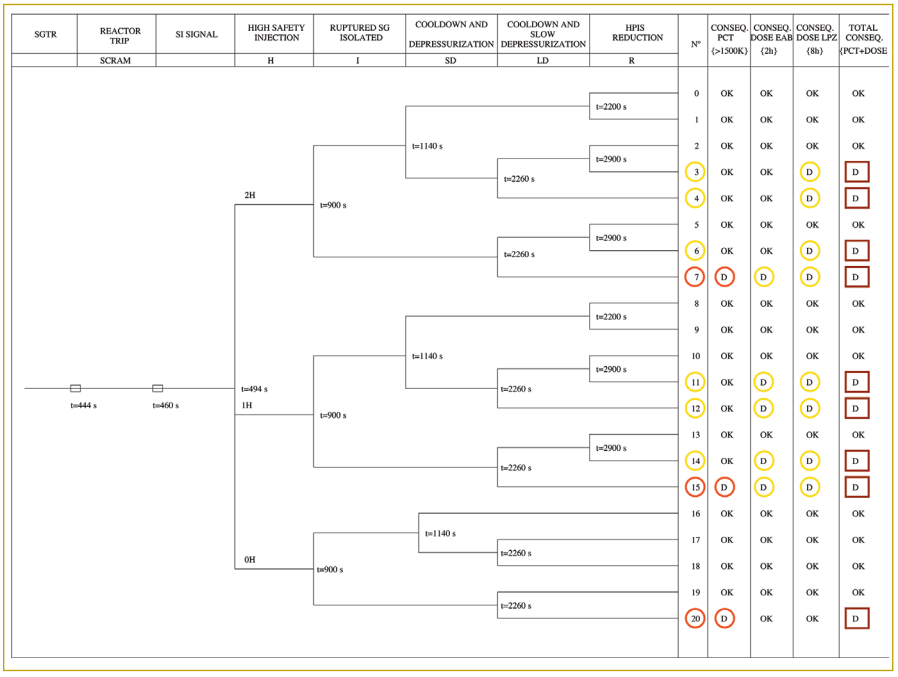

Figure 7. DET (final status includes PCT and dose criteria).

ES-3.1, ECA-3.1 and ECA-3.2, have been included in DET simulation. As in classic PSA, the sequences included in the DET reach a final state (either success or damage). In this analysis such final state depends simultaneously on **three damage criteria**: PCT limit, dose limit in Exclusion Area Boundary (EAB), and dose limit in Low Population Zone (LPZ).

PCT evaluation depends directly on SCAIS-MAAP simulations, and dose estimation has been carried out by means of using as input for RADTRAD code some MAAP results, and considering the standard meteorological conditions included in typical Safety Analysis Reports (SAR).

On one hand, PCT limit refers to ECCS acceptance criteria from 10 CFR 50 (PCT ≤ 1477 K at every moment). On the other hand, dose limits refer to TID-14844 methodology: 0.25 Sv to the whole body or 3 Sv to the thyroid, both for the critical individual for a 2-hour staying within EAB and an 8-hour staying within LPZ.

Main results emerging from DET simulation are depicted in Figures 6 and 7 and summarized in Table I. The identification of each sequence is carried out by the concatenation of header status: a header in upper case means success upon demand and lower case means failed upon demand.

Results from RADTRAD code show that the most conservative case corresponds to the dose to thyroid for the Concurrent Iodine Spike. Due to that, all value presented in columns 5 and 6, Table I correspond to ratio between estimated dose and dose limit to the thyroid, which dose limit, is 3 Sv, as previously stated. For obtained dose calculation by RADTRAD is necessary the mass flow rate evolution in tube rupture and the end of release for affected SG, columns 2-3 in table I. DET results (see, Table I and Figures 6-7) show that PCT damage end status (**red** color) is reached for sequences DM7, DM15 and DM20. In addition, dose damage end status (**orange** color) is also reached for sequences DM3, DM4, DM6, DM11, DM12 and DM14.

DET analysis shows that there are several sequences without PCT damage but with dose damage. In addition, results show that R header (SI termination) is greater contributor to dose damage and show the adequacy of the ISA methodology to study a complex problem in terms of human actions and systems actuation.

Verification of Severe Accident Management Actuations. SBO Sequences

A Station Blackout (SBO) sequence initiates by loss of both off-site and on-site AC power (Emergency Diesel Generators, EDG). As a result there is a reactor SCRAM, turbine trip and RCP trip; later on MFW pumps are tripped due to low turbine pressure. Under these conditions, only accumulators (ACC) are available, as the high-pressure and low-pressure safety injection (HPSI and LPSI) systems are unavailable as a result of the AC power loss. The containment spray and the fan coolers are also not available until AC recovery. As long as AC is not recovered on time and the seal LOCA occurs, the RCS inventory will decrease uncontrolled, and therefore the core will result uncovered.

SBO sequences including main human actions have been simulated with SCAIS-MAAP. Procedures that have to take into account are EOP E-0, ECA-0.0, E-1, ES-1.2 and FR-C.1. Severe Accident Management Guidelines (SAMG) actions, corresponding to SAG-1, SAG-2 and SAG-3, have been also considered.

Sequence	Integrated SGTR massflow (kg)	Release ending time (s)	PCT (K)	EAB (2h) Dose/Limit	LPZ (8h) Dose/Limit
DM0: 2H-I-SD-R	4.50E+04	2665	603	0.7050	0.3556
DM1: 2H-I-SD-r	1.45E+05	8162	603	0.7050	0.3500
DM2: 2H-I-sd-LD-R	5.40E+04	2914	603	0.199	0.794
DM3: 2H-I-sd-LD-r	1.40E+05	7790	603	0.9453	3.9257
DM4: 2H-I-sd-ld	1.40E+06	69560	603	0.8753	3.5643
DM5: 2H-i-LD-R	9.30E+04	10071	603	0.199	0.796
DM6: 2H-i-LD-r	1.40E+06	49714	603	0.9987	4.2593
DM7: 2H-i-ld	1.60E+06	130306	>1500	1.150	4.600
DM8: 1H-I-SD-R	4.10E+04	2378	603	0.6455	0.3262
DM9: 1H-I-SD-r	4.30E+04	2670	603	0.6454	0.3205
DM10: 1H-I-sd-LD-R	1.40E+05	8716	603	0.6454	0.3205
DM11: 1H-I-sd-LD-r	1.40E+05	9134	603	1.1502	4.5880
DM12: 1H-I-sd-ld	1.45E+06	76811	603	1.098	4.392
DM13: 1H-i-LD-R	9.10E+04	9981	603	0.8286	0.2964
DM14: 1H-i-LD-r	1.40E+06	54576	603	1.1403	4.457
DM15: 1H-i-ld	1.60E+06	137300	>1500	1.0142	3.8997
DM16: OH-I-SD	1.40E+04	996	603	0.0716	0.0354
DM17: OH-I-sd-LD	1.30E+04	995	603	0.0716	0.0354
DM18: OH-I-sd-ld	1.30E+04	995	603	0.0716	0.0354
DM19: OH-i-LD	5.20E+04	10131	603	0.2128	0.0761
DM20: OH-i-ld	1.50E+05	45982	>1500	0.2128	0.7833

Table I. Sequence information obtained from the DET.

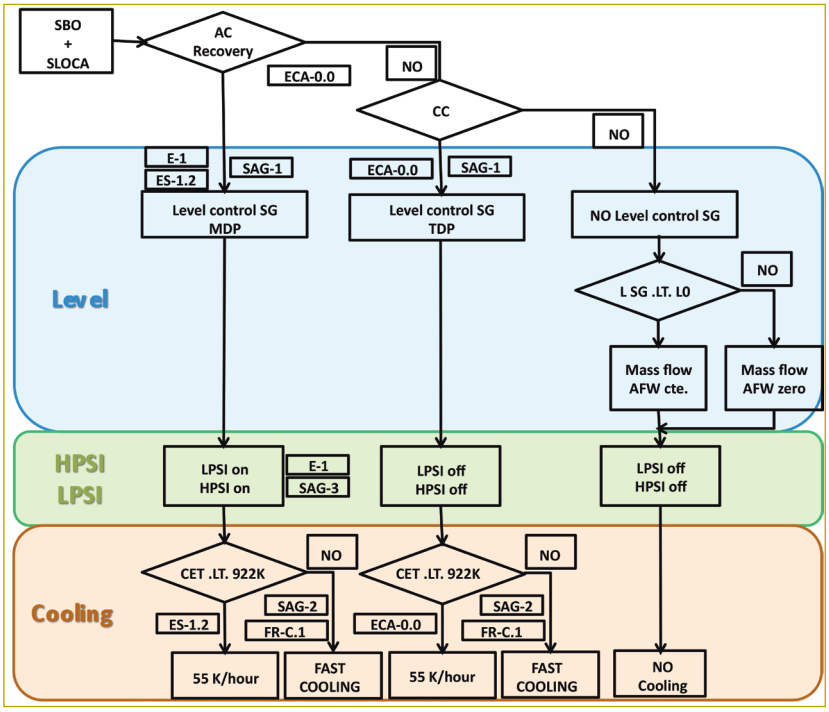

Figure 8. Main operator actions taken into account in SBO sequences with Seal LOCA.

As the Seal LOCA is likely to occur along this type of transients, the operating crew will follow EOPs corresponding for LOCA sequences, E-1 or ES-1.2, whenever the AC is recovered, changing to fast cooling phase once the Core Exit Temperature (CET) limit is exceed. This would trigger a full opening of SG PORVs, and would indicate the transition to SAMG (Figure 8).

The analysis has considered the following damage indicators:

- Core uncovery
- CET limit (CET> 922 K)
- Peak cladding temperature (PCT > 1477 K)
- Fuel relocation in lower plenum
- RPV failure

So, there are as many several damage domains for the same sequence as damage indicators. In order to show this set of damage domains in a comprehensive manner, in the sequel a color code has been defined: **green** (no core uncovery); **red** (core uncovery); **blue** (inadequate core cooling, CET>922 K); **orange** (cladding embrittlement, PCT>1477); **purple** (fuel relocation in lower plenum) and **black** (vessel failure). Only the last damage is depicted because all previous damages to the last one are assumed (i.e. if a path is marked in **orange** color it means that previously to cladding embrittlement, core uncovery and inadequate core cooling have occurred previously). This defines what is called a Multiple Damage Domain (MDD). Each point of the MDD corresponds to a simulation (called a path) of a SBO sequence with DC loss at time t0 and AC recovery at time t1.

In the study, DC is lost at uncertain time, and as, availability of AC implies availability of DC. Therefore, only sequences with t^{AC}_{REC}

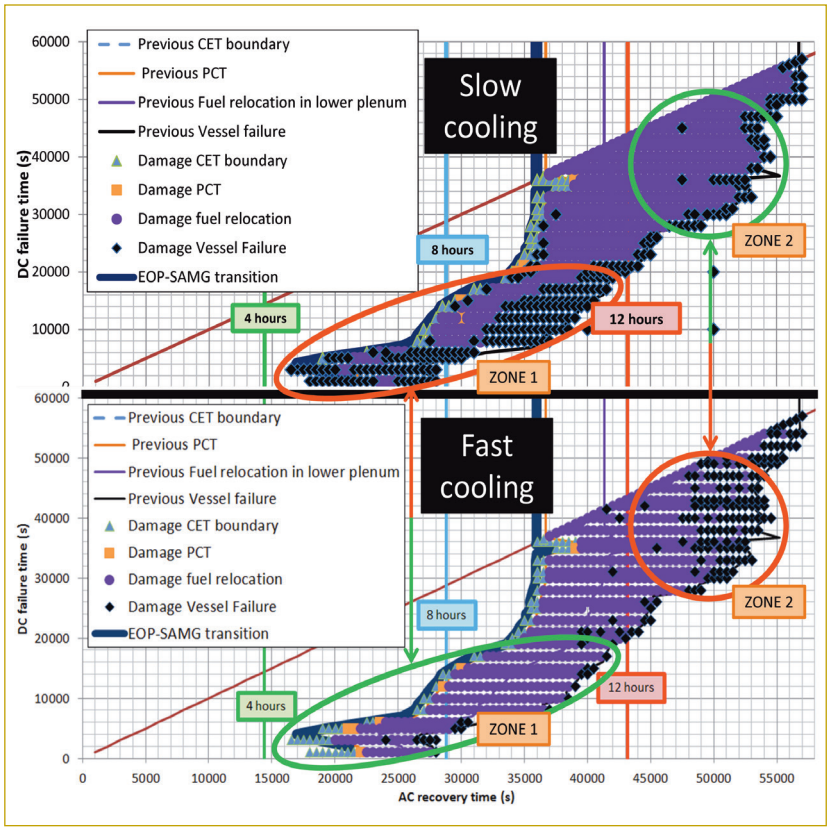

Figure 9. Comparison between MDDs with slow and fast cooling.

greater than t^{DC}_{LOST} times have been considered. The different DD are enclosed with the diagonal of $t^{DC}_{LOST} = t^{AC}_{REC}$ and previous damage curves previously calculated for different damages.

A first MDD has been obtained performing near 850 MAAP simulations and taking into account only EOPs (E-0, ECA-0.0, E-1 and ES-1.2) with slow cooling (55 K/hour) by means of SG, but not SAMG. Results show the final state of the accident for each path, since sequences of success to vessel failure, crosses through different damage conditions. Therefore, the previous damage curves indicate the initial state after AC recovery and the MDD indicates the final state of the path at the end of the simulation. Later, a second MDD has been obtained (750 simulation runs) considering fast cooling by means of SG PORVs (from SAG-2 action). Comparison among both MDD (slow and fast cooling) allows to measure the impact of the application of this accident management action, Figure 9.

The comparison of both MDD shows that there are two differentiated zones (Zone 1 and 2). Zone 1 allows concluding that if AC is recovered a few hours later than core uncovery (t< 3 hours) then fast cooling is an efficient procedure in order to avoid vessel failure. However if the AC power is recovered later (Zone 2) there are several path that arrive to a worse damage condition whenever fast cooling is applied.

DISCUSSION

Increasingly stringent safety requirements from one side, power uprates, aging and modifications of already quite complex plant safety systems from the other side put more emphasis on completeness and consistency of safety analysis. IDPSA methods have been developed to address the aleatory and epistemic uncertainties in risk analysis in a consistent manner. Examples of successful applications of the IDPSA methods and demonstrations of potential benefits are plenty in open literature, including in this work. Yet, deployment of such tools in industry and regulation is slower than it could be, given exponentially growing affordability of such computationally demanding analyses methods. We find that following problems are the main obstacles which slow down new methods enter into practice of safety analysis.

First, we recognize that huge efforts and resources have been invested into development of PSA and DSA models for existing plants. These models have been and will be used extensively in the future because in many cases they can provide an adequate risk assessment. However, when the nature of major uncertainties is not amenable to an adequate treatment with the state of the art PSA and DSA approaches, IDPSA could be used as a complementary tool in order to ensure completeness and consistency of the analysis. It is instructive to note that meaningful application of IDPSA requires significant amount of information as input. Fortunately, most of the necessary information is already available in PSA or DSA. Therefore more emphasis is required on the development of approaches for mutual exchange of information between PSA, DSA and IDPSA analysis. We believe that demonstration of such connections in the pilot applications of IDPSA will pave the way towards broader acceptance of the IDPSA methodology by PSA and DSA practitioners.

Second, in the process of exploration of the uncertainty space IDPSA can generate hundreds and thousands of transient simulations with system safety analysis codes. Interpretation and understanding of the physics and logic behind identified failures can present a formidable challenge for an expert. Therefore it is important to equip IDPSA tools with adequate data mining and classification techniques, which would help to establish lossless extraction and condensation of useful information amenable to expert evaluation. This is a critical step which should facilitate risk-informed decision making process, make it more robust and transparent.

Finally, from the application examples it can be concluded that these methodologies also allow to verify the EOP and SAMG in a natural way and to take into account all damages indicators that can contribute to the global risk of the event.

Therefore, those methodologies can be a powerful tool in the evaluation of the post-Fukushima analysis in present reactors, Generation II, and the licensing of the new reactors, both Generation III/III+ and IV.

While application of IDPSA methodology requires considerable efforts, potential benefit –compete and consistent risk assessment – is paramount for sustainable development of safe and nuclear power.