Dynamic and Classical PRA: a BWR SBO Case Comparison

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As part of the Light-Water Sustainability Program (LWRS), the purpose of the Risk Informed Safety Margin Characterization (RISMC) Pathway research and development is to support plant decisions for risk-informed margin management with the aim to improve economics, reliability, and sustain the safety of current NPPs. In this paper, we describe the RISMC analysis process illustrating how mechanistic (i.e., dynamic system simulators) and probabilistic (stochastic sampling strategies) approaches are combined in a dynamic PRA fashion in order to estimate safety margins. We use the scenario of a "station blackout" (SBO) wherein offsite power and onsite power are lost, thereby causing a challenge to plant safety systems. We describe the RISMC approach, illustrate the station blackout modeling, and compare this with traditional risk analysis modeling for this type of accident scenario. In the RISMC approach the dataset obtained consists of set of simulation runs (performed by using codes such as RELAP5/3D) where timing and ordering of events is changed accordingly to the stochastic sampling strategy adopted. On the other side, classical PRA methods, which are based on event-tree (FT) and fault-tree (FT) structures, generate minimal cut sets and probability values associated to each ET branch. The comparison of the classical and RISMC approaches is performed not only in terms of overall core damage probability but also considering statistical differences in the actual sequence of events. The outcome of this comparison analysis shows similarities and dissimilarities between the approaches but also highlights the greater amount of information that can be generated by using the RISMC approach.

I. INTRODUCTION

The Risk-Informed Safety Margin Characterization (RISMC) Pathway develops and delivers approaches to manage safety margins [1,2]. This important information supports the nuclear power plant owner/operator decision-making associated with near and long-term operation. The RISMC approach can optimize plant safety and performance by incorporating a novel interaction between probabilistic risk simulation and mechanistic codes for plant-level physics. The new functionality allows the risk simulation module to serve as a "scenario generator" that feeds information to the mechanistic codes. The effort fits with the goals of the RISMC Pathway, which are twofold: 1) develop and demonstrate a risk-assessment method coupled to safety margin quantification and 2) create an advanced RISMC Toolkit which would enable users to have a more accurate representation of nuclear power plant safety margins and its associated influences on operations and economics.

In general terms, a "margin" is usually characterized in one of two ways: a deterministic margin, typically defined by the ratio (or, alternatively, the difference) of a capacity (i.e., strength) over the load, and a probabilistic margin, defined by the probability that the load exceeds the capacity.

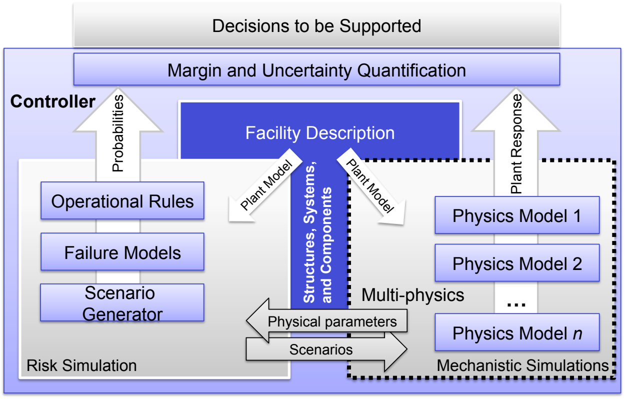

Fig. 1. The approach used to support RISMC analysis.

A probabilistic safety margin is a numerical value quantifying the probability that a safety metric (e.g., for an important process observable such as clad temperature) will be exceeded under accident scenario conditions.

The RISMC Pathway uses the probabilistic margin approach to quantify impacts to reliability and safety. As part of the quantification, we use both probabilistic (via risk simulation) and mechanistic (via physics models) approaches, as represented in Fig. 1. Safety margin and uncertainty quantification rely on plant physics (e.g., thermal-hydraulics and reactor kinetics) coupled with probabilistic risk simulation. The coupling takes place through the interchange of physical parameters (e.g., node pressure) and operational or accident scenarios.

In this paper we describe a comparative analysis between RISMC approach with traditional risk analysis modeling for a BWR SBO accident scenario.

Traditional PRA methods are based on logic structures such as Event-Trees (ETs) and Fault-Trees (FTs) [3]. These static types of models mimic system response in an inductive and deductive way respectively, yet are restrictive in the ways they can represent spatial and temporal constructs.

An example of ET-FT structure is shown in Fig. 2 for a simplified Station Black-Out (SBO) Initiating Event (IE). The ET structure shows how system success (i.e., outcome OK) can be achieved after a SBO accident scenario when either AC power is recovered or firewater (FW) is available. When neither of these two conditions is met, a core damage (CD) condition is reached. This logic progression is shown in the ET structure of Fig. 2.

FTs are used to build logical event relationships between basic events (typically representing component failures) that affect branching conditions in the ET. In Fig. 2 the two simplified FTs for AC and FW recovery are shown. For the first case, either Diesel Generators (DGs) or offsite Power Grid (PG) are sufficient conditions to recover AC power. In order to recover FW capabilities, the system needs to be depressurized (ADS) and FW outlet has to be aligned to the reactor vessel.

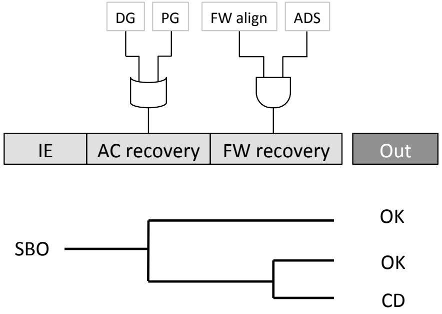

Fig. 2. Example of ET-FT structure for a BWR SBO IE.

Note that the structure shown in Fig. 2 follows a precise logic that has been defined a priori by the user, i.e., the sequences of events in the ET are fixed and not interchangeable (in other words, they are part of a static model represented by a simple Boolean logic expression).

As indicated in the historical accident in the nuclear industry, the timing of occurrence of such events can play a major role in the accident evolution. This timing information is not implicitly considered in an ET-FT structure shown in Fig. 2; it is in fact only loosely considered in the definition of the basic events, e.g., DG recovery within 4 hours.

Both these issues (fixed logic structure, lack of timing considerations) preclude the ability to fully analyze possible accident evolution trajectories and, thus, also the possibility to evaluate importance of basic events in the overall CD probability. This is the reason why the RISMC Pathway is employing state-of-the-art simulation based methodologies to evaluate accident evolution and the risk associated with these scenarios.

These issues are particularly relevant for the RISMC project where it is needed to evaluate the impact of plant changes such as power uprates and life extension on existing NPPs. From an ET-FT logic point of view, both power uprate and life extensions are not modeled, which further shows the limitations of these kinds of methodologies for design and operational considerations.

II. RISMC APPROACH

As mentioned in the previous section, accident progression is modeled by directly using system simulator codes. A simulator code is, per se, a tool that can be represented as:

$$\frac{\partial \boldsymbol{\theta}(t)}{\partial t} = \boldsymbol{\mathcal{H}}(\boldsymbol{\theta}, \boldsymbol{p}, \boldsymbol{s}, t) \tag{1}$$

where:

- $\boldsymbol{\theta} = \boldsymbol{\theta}(t)$ represents the status of the system as function of time t, i.e., $\boldsymbol{\theta}(t)$ represents a single simulation
- $\boldsymbol{\mathcal{H}}$ is the actual simulator code that describes how $\boldsymbol{\theta}$ evolves in time
- $\boldsymbol{p}$ is the set of parameters internal to the simulator code (e.g., pipe friction coefficients, pump flow rate, reactor power)
- $\boldsymbol{s} = \boldsymbol{s}(t)$ represents the status of components and systems of the simulator (e.g., status of emergency core cooling system, AC system)

By using the RISMC approach, the PRA is performed by following these steps:

1. Associate a probabilistic distribution function (pdf) to the set of parameters $\boldsymbol{p}$ and $\boldsymbol{s}$ (e.g., timing of events and clad fail temperature)
2. Perform sampling of the pdfs defined in Step 1[1]
3. Perform a simulation run given the $\boldsymbol{p}$ and $\boldsymbol{s}$ sampled in Step 2

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[1] The sampling associated to the vector of parameters $\boldsymbol{p}$ is usually defined as uncertainty quantification while sampling the timing of events $\boldsymbol{s}$ is usually called PRA. In our applications, we include in the definition of PRA the sampling of both $\boldsymbol{p}$ and $\boldsymbol{s}$.

4. Repeat Steps 2 and 3 N times and evaluate user defined stochastic parameters such CD probability (P_{CD}) as the ratio between the number of simulations that lead to CD divided by N (the total number of simulations).

Steps 2, 3 and 4 are performed by the RAVEN statistical framework which that allows the user to perform statistical analysis. By statistical analysis we include:

- Codes sampling: either stochastic (e.g., Monte-Carlo [4,5] and Latin Hypercube Sampling (LHS) [6]) or deterministic (e.g., Dynamic Event Tree [7,8])
- Generation of Reduced Order Models (ROMs) [9]
- Post-processing of the sampled data and generation of statistical parameters (e.g., mean, variance, covariance matrix)

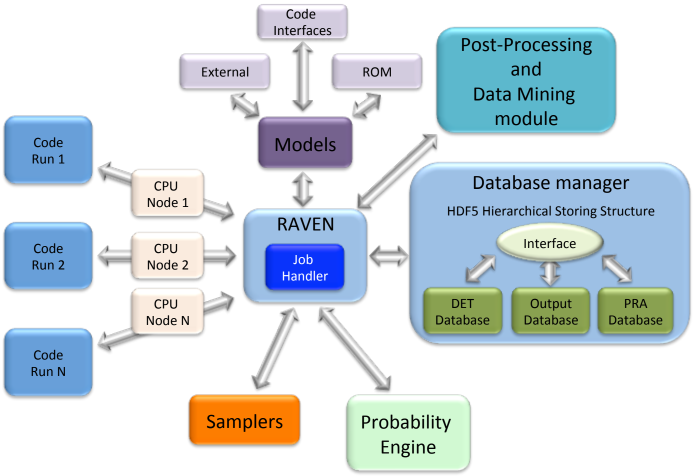

Fig. 3. Scheme of RAVEN statistical framework components.

Figure 3 shows a general overview of the elements that comprise the RAVEN statistical framework:

- Model: represents the pipeline between input and output space. It comprises both codes (e.g., RELAP-7) and also Reduced Order Models (ROMs)
- Sampler: is the driver for any specific sampling strategy (e.g., Monte-Carlo, LHS, DET)
- Database: the data storing entity
- Post-processing module: module that performs statistical analyses and visualizes results

III. RISMC BWR SBO DATA

The accident scenario under consideration is a LOOP IE followed by loss of the diesel generators (DGs), i.e., a SBO IE (see Fig. 4) [10,11]. At time $t = 0$ the following events occur: LOOP condition occurs due to external events (i.e., power grid related), LOOP alarm triggers the following actions:

- Operators successfully scram the reactor
- Emergency DGs successfully start
- Core decay heat is removed from the reactor vessel
- DC systems (i.e., batteries) are functional

At a certain point, due to internal failure, the set of DGs fails and SBO condition is met. Thus, the removal of the decay heat is impeded. Reactor operators start the SBO emergency operating procedures and perform RPV pressure and level control along with containment monitoring.

As part of the scenario, plant operators start recovery operations to bring back on-line the DGs while the recovery of the power grid is underway by the grid owner emergency staff. However, due to the limited life of the battery system and depending on the use of DC power, battery power can deplete. When this happens, all remaining control systems are offline causing the reactor core to heat until clad failure temperature is reached, i.e., core damage CD.

If DC power is still available and one of three specific conditions are reached, then the reactor operators activate the ADS system in order to depressurize the reactor.

As an emergency action, when reactor pressure is below 100 psi, plant staff can connect the firewater system in order to cool the core and maintain an adequate water level. However, this task may be difficult to complete since the physical connection between the firewater system and the reactor vessel inlet has to be made manually.

When AC power is recovered, through successful re-start/repair of DGs or off-site power, reactor core cooling can be restored.

The choice of the set of stochastic parameters to consider in the analysis was based on the preliminary PRA model results obtained for a typical BWR SBO case. For all basic events (e.g., DG fail to run) we have considered the following sensitivity indexes common to classical PRA: the Fussell-Vesely and Birnbaum importance and a typical ET structure for a LOOP-SBO [10].

The probabilistic modeling of the possible human interventions was done by looking at the SPAR-H [12] model from a generic BWR PRA. In this respect, we have identified three actions:

- Manual activation of the automatic depressurization system: operator manually depressurizes the reactor by activation of the automatic depressurization system

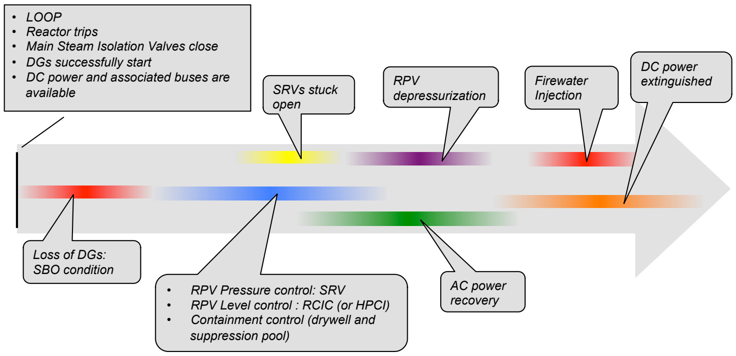

Fig. 4. BWR SBO simulated data: sequence/timing of events.

- Extended ECCS operation: operators may extend RCIC/HPCI and SRVs control even after the batteries have been depleted. This action actually summarizes two events: manual control of RCIC/HPCI by acting on the steam inlet valve of the turbine and obtain DC power availability through spare batteries
- Firewater injection availability time (measured after depressurization has been activated)

SPAR-H characterizes each operator action through eight parameters – for this study we focused on two important factors [10]:

- Stress/stressors level
- Task complexity

These two parameters are used to compute the probability that such action will happen or not; these probability values are then inserted into the ETs that contain these events. However, from a simulation point of view we are not seeking if an action is performed but rather when such action is performed. Thus, we need a probability distribution function that defines the probability that such action will occur as function of time.

Since modeling of human actions is often performed using lognormal distributions [10], we chose these distributions where its characteristic parameters (i.e., μ and σ) are dependent on the two factors listed above (Stress/stressors level and Task complexity). We used Table 1 [10] to convert the three possible values of the two factors into numerical values for μ and σ. Note that it is here assumed that human actions are performed correctly.

Table 1. Correspondence table between complexity and stress/stressor level and time values.

Complexity	μ (min)
High	45
Moderate	15
Nominal	5

Stress/stressors	σ (min)
Extreme	30
High	15
Nominal	5

A summary of the stochastic parameters and their associated distributions is shown in Table 2.

The stochastic analysis for the BWR SBO test case has been performed using the code RAVEN [13] that is being developed by INL. Originally, RAVEN was designed to control the code RELAP-7, but its capabilities have been extended to include also stochastic analysis methodologies such as Monte-Carlo [5] and Dynamic Event Tree algorithms [8].

In addition, RAVEN has been coupled to RELAP5-3D [14] and RELAP-7 [15] in order to perform multiple RELAP runs (through LHS sampling). To evaluate the impact of the uncertain parameters summarized in Table 2 on the simulation outcome, we performed an extensive LHS analysis that consisted of generating 20,000 runs.

Table 2. List of stochastic parameters and their distribution*.

Stochastic variable	Distribution	Distribution parameters
DGs fail time of (h)	Exponential	$\lambda = 1.09$ E-3
DGs rec. time (h)	Weibull	$\alpha = 0.745$, $\beta = 6.14$
Battery life (h)	Triangular	(4, 5, 6)
SRV 1 failure	Binomial	8.56 E-4
SRV 2 failure	Binomial	8.56 E-4
PG rec. time (h)	Lognormal	$\mu = 0.793$, $\sigma = 1.982$
Clad Fail temp. (F)	Triangular	(1800, 2200, 2600)
HPCI fail time (h)	Exponential	$\lambda = 4.4$E-3
RCIC fail time (h)	Exponential	$\lambda = 4.4$E-3
FW avail. time (h)	Lognormal	$\mu = 0.75$, $\sigma = 0.5$
Ext. ECCS oper. (h)	Lognormal	$\mu = 0.75$, $\sigma = 0.5$
Man. ADS act. (h)	Lognormal	$\mu = 0.083$, $\sigma = 0.25$

* Human related stochastic parameters are in italics.

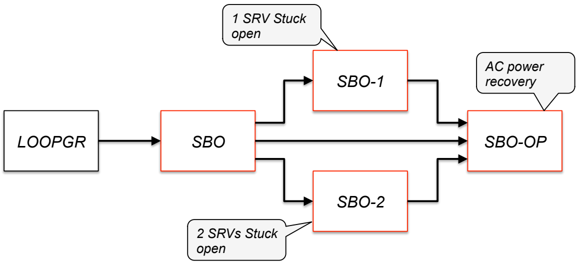

Fig. 5. ET structure for the BWR SBO model contained in SAPHIRE.

IV. CLASSICAL PRA BWR SBO DATA

In traditional PRA, the BWR SBO case studied is modeled with the following ETs [16] (see Fig. 5) that are linked together with the transferring feature in SAPHIRE software [17]:

- LOOP: Loss of Offsite Power

- SBO: Station Black Out
- SBO-1: SBO with 1 SRV stuck open
- SBO-2: SBO with 2 or more SRVs stuck open
- SBO-OP: AC recovered ET

There are actually four LOOP ETs based on the cause or location of the LOOP event occurred: LOOP-GR (grid related), LOOP-PC (plant centered), LOOP-SC (switchyard centered), and LOOPWR (weather related). The four trees have identical structure and top events except the initiators. LOOPGR is used as the representative LOOP ET in this analysis.

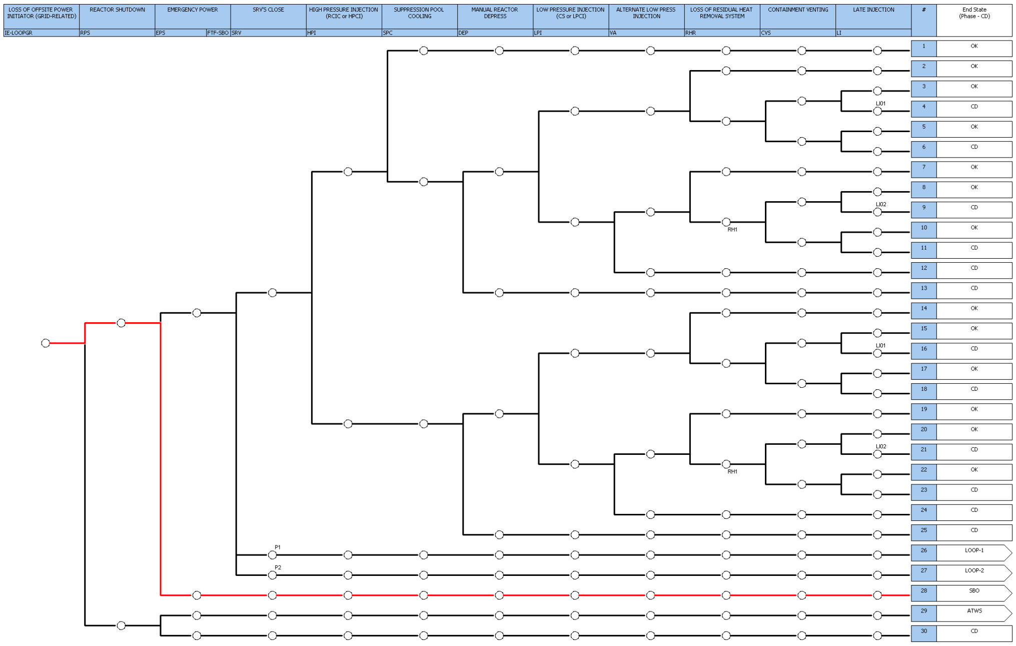

Fig. 6. ET structure for LOOP grid related; red path is characterized by the loss of DGs and leads to the SBO ET (see Fig. 5).

The LOOP-GR ET (see Fig. 6) starts with a grid related LOOP as IE followed by a branch on the success/failure of the reactor shutdown. Then the ET queries the status of emergency power (i.e., diesel generators). Success of reactor shutdown but failure of diesel generators (Sequence 28 of LOOP-GR) leads to a transfer ET: the SBO ET.

In the SBO ET (see Fig. 7) the following events are queried with a total of 36 sequences:

1. SRV(s) status: one stuck open SRV sequence (Sequence 35 of SBO) leads to another transfer ET: SBO-1. Two or more stuck open SRVs sequence (Sequence 36 of SBO) leads to the SBO-2 ET.
2. Recirculation pump seal integrity: failure of the recirculation pump (Sequence 34 of SBO) leads to the SBO-1 ET.

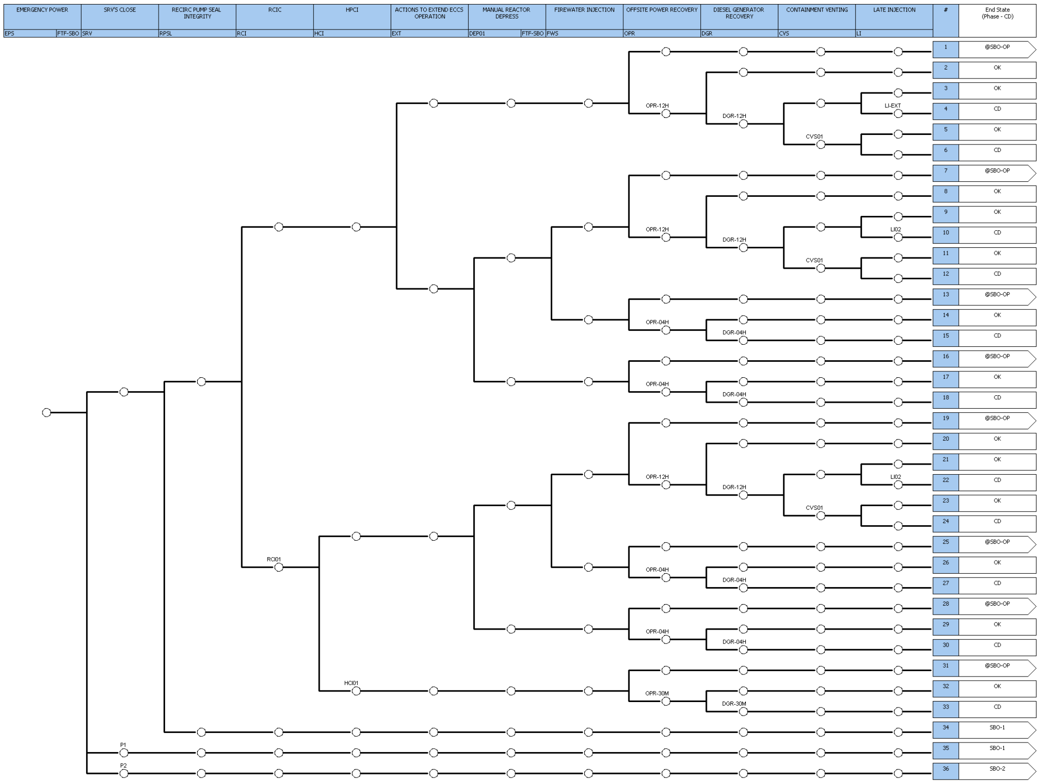

Fig. 7. ET structure for SBO.

3. RCIC availability
4. HPCI availability

5. Extended ECCS operation[2]
6. ADS activation
7. FW injection
8. Offsite power recovery
9. DG recovery
10. Containment venting
11. Late injection

In case one SRV or two or more SRVs are stuck open the following events are queried in sequence (see Fig. 8):

1. RCIC availability
2. HPCI availability
3. Offsite power recovery
4. DG recovery

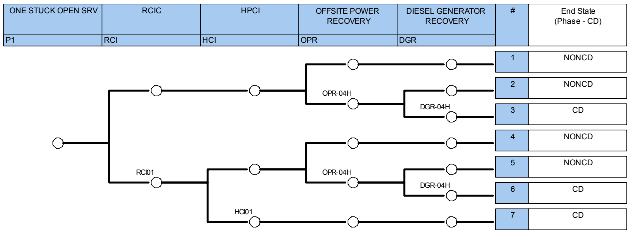

Fig. 8. ET structure for two SRVs stuck open.

To compare the RELAP/RAVEN simulation run results with the above traditional PRA models, all of the SBO sequences (including the sequences transferred to SBO-1 and SBO-2, whether they end with CD or non CD) have to be quantified. Note that this is different from the general Level 1 PRA quantification process in which only core damage sequences are quantified. Conditional sequence probability (versus conditional core damage probability, or CCDP, in general PRA quantification) given a SBO event occurred is used as the matrix of merit for the comparison.

In order to quantify non-core damage sequences as well as core damage sequences correctly, the impact of success branch probabilities of the ET top events must be considered. Two different approaches could be used to account for the probability of success branches by using what is known in SAPHIRE as the “process flag” feature. The results of both approaches must be post-processed to provide correct sequence frequency or conditional probability [18].

In the developed event approach which uses the “W” process flag, SAPHIRE explicitly includes the success branch probability in the sequence cut sets. The ET top event is treated as a basic event for the success branch and the complement of the event is used as the branch probability. However, this approach may contain non-coherent cut sets that should be reviewed, identified, and removed from the quantification results.

In the other approach that uses the default, or blank, process flag, SAPHIRE uses a “delete term process” to prune success cut sets from the failure cut sets to generate coherent sequence cut sets. Success branch probabilities are not included in the sequence cut sets and must be manually added to be accounted for when the impact is not negligible. For example, offsite power recovery within 12 hours (OPR-12H) has a failure probability of 2.04E-2. Using the default process flag and delete term approach without accounting for its success probability (9.8E-1) may have only very small impact on the associated sequences (Sequences 1, 7, and 19 of the SBO ET) results. But for offsite power recovery within 30 minutes (OPR-30M), the failure probability is 8.63E-1 and the success probability is 1.37E-1. Without accounting for this success branch probability would increase the value of Sequence 31 of the SBO ET by 8 times.

Table 3 presents the BWR SBO PRA model quantification results. Note that using the “W” process flag or using the default process flag without adjusting the results with the success branch probabilities yields incorrect results with a total conditional probability greater than 1.0. The last column, using the default process flag and adjusting the results with the success branch probabilities, shows correct conditional probabilities for SBO sequences that will be used for the comparison in Section 4.3.

For our application scope, no failures/events occur between the LOOP and the Loss of DGs; thus we did not consider the initial ET (i.e. LOOPGR). In addition, in the RELAP5-3D simulations we did not account for failures followed after AC power recovery; hence the ET SBO-OP was not considered.

V. COMPARISON APPROACH

In order to compare the results generated by RAVEN/RELAP5-3D and traditional methods we performed the following steps:

1. Merge the ETs SBO, SBO-1 and SBO-2 into a single ET and recalculate branch probabilities (see Section V.A.)
2. Associate each of the 20,000 scenarios simulated using RELAP5-3D to a unique branch of the SBO ET built in Step 1. Perform a posteriori analysis for the scenarios that were not associated with an ET branch (see Section V.B.)
3. Identify inconsistencies between RAVEN/ RELAP5-3D and the traditional approach in terms of outcome

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[2] This functional event refers to the possibility that ECCS cooling is performed manually by the reactor operators.

(e.g., core damage CD or system OK) and probabilities (see Section VI.)

Note that a single branch of the ET might contain several RELAP5-3D simulations.

Table 3. SBO sequence quantification results for a typical BWR PRA model.

Seq.	Out	Prob. (W)[1]	Prob. (Default, Not Adj.)[2]	Prob. (Default, Adj.)[3]
1	OK	5.93E-01	1.00E+00	5.92E-01
2	OK	6.62E-03	2.04E-02	6.60E-03
3	OK	2.92E-03	9.50E-03	2.92E-03
4	CD	1.21E-03	2.31E-03	9.77E-04
5	OK	1.26E-03	2.87E-03	1.26E-03
6	CD	5.21E-04	6.98E-04	4.22E-04
7	OK	1.73E-01	2.35E-01	1.40E-01
8	OK	2.05E-03	5.10E-03	1.67E-03
9	OK	1.24E-03	2.37E-03	1.01E-03
10	CD	0.00E+00	0.00E+00	0.00E+00
11	OK	5.37E-04	7.17E-04	4.38E-04
12	CD	0.00E+00	0.00E+00	0.00E+00
13	OK	4.86E-02	8.07E-02	4.82E-02
14	OK	3.00E-03	1.38E-02	3.00E-03
15	CD	6.94E-03	9.66E-03	6.94E-03
16	OK	3.04E-02	4.34E-02	3.03E-02
17	OK	1.88E-03	7.40E-03	1.87E-03
18	CD	4.34E-03	5.17E-03	4.34E-03
19	OK	4.07E-02	6.70E-02	4.04E-02
20	OK	4.60E-04	1.40E-03	4.61E-04
21	OK	2.79E-04	6.50E-04	2.79E-04
22	CD	0.00E+00	0.00E+00	0.00E+00
23	OK	1.21E-04	1.96E-04	1.21E-04
24	CD	0.00E+00	0.00E+00	0.00E+00
25	OK	6.46E-03	1.07E-02	6.44E-03
26	OK	3.96E-04	1.81E-03	3.96E-04
27	CD	9.17E-04	1.27E-03	9.20E-04
28	OK	6.88E-03	9.78E-03	6.87E-03
29	OK	4.21E-04	1.65E-03	4.21E-04
30	CD	9.76E-04	1.15E-03	9.72E-04
31	OK	5.14E-04	4.16E-03	5.12E-04
32	OK	2.65E-04	3.59E-03	2.65E-04
33	CD	2.97E-03	3.30E-03	2.97E-03
34-1	OK	7.75E-02	1.00E-01	7.75E-02
34-2	OK	4.74E-03	1.69E-02	4.76E-03
34-3	CD	1.10E-02	1.18E-02	1.10E-02
34-4	OK	5.34E-03	6.83E-03	5.34E-03
34-5	OK	3.27E-04	1.15E-03	3.24E-04
34-6	OK	7.56E-04	8.05E-04	7.57E-04
34-7	CD	4.16E-04	4.17E-04	4.17E-04
35-1	OK	6.64E-04	8.56E-04	6.64E-04
35-2	OK	4.06E-05	1.44E-04	4.06E-05
35-3	CD	9.40E-05	1.01E-04	9.42E-05
35-4	OK	4.58E-05	5.86E-05	4.58E-05
35-5	OK	2.80E-06	9.87E-06	2.78E-06
35-6	CD	6.48E-06	6.89E-06	6.48E-06
35-7	CD	3.57E-06	3.57E-06	3.57E-06
36-1	OK	6.09E-05	1.91E-04	6.08E-05
36-2	OK	1.51E-05	1.26E-04	1.52E-05
36-3	CD	1.02E-04	1.10E-04	1.03E-04
36-4	OK	4.20E-06	1.31E-05	4.20E-06
36-5	OK	1.04E-06	8.62E-06	1.05E-06
36-6	CD	7.06E-06	7.51E-06	7.06E-06
36-7	CD	7.97E-07	7.97E-07	7.97E-07
Total		1.04E+00	1.68E+00	1.00E+00

V.A. Classical PRA ET restructuring

A modified SBO ET model (see Fig. 9) was developed for more effective comparison between the simulation results and the PRA results. The total number of sequences is reduced from 54 (see Table 3) in the original SBO ET model (including the SBO, SBO-1, and SBO-2 ETs) to 18 in the restructured ET (see Table 4).

The restructured ET queries the following top events:

1. SRV(s) status: no stuck open SRV, one stuck open SRV, or two or more stuck open SRVs.
2. High pressure injection (HPI) availability: HPI is success if either RCIC or HPCI is available.
3. Depressurization and firewater injection
4. Offsite power or DG recovery

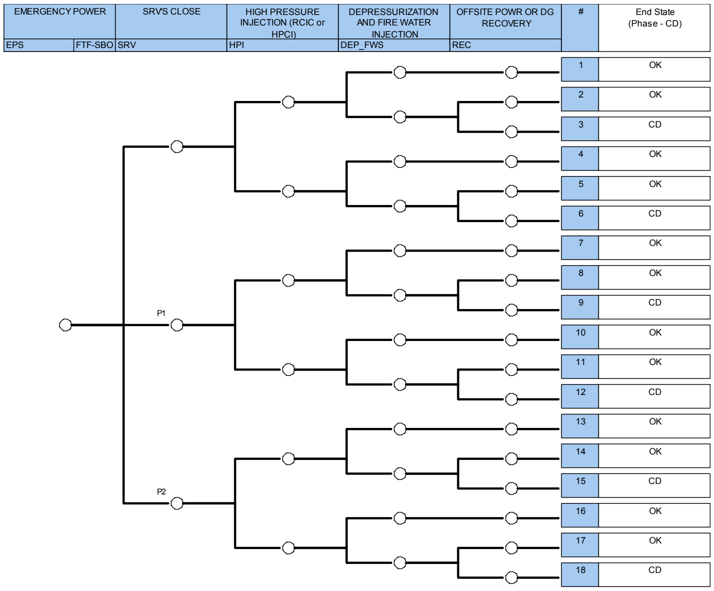

Fig. 9. Restructured SBO ET model.

Unlike the original SBO model, the simplified ET does not include the top event for recirculation pump seal integrity. Due to seal LOCA model instability, the RELAP5-3D/RAVEN simulation runs do not include the stochastic parameters related to the event and thus have no data to be compared.

Table 4 presents the 18 sequences in the simplified SBO ET, the queried system status/functionalities for each sequence, the equivalent sequence(s) in the original SBO model, as well as the end state of each sequence. For example, Sequence 1 of the simplified ET represents the scenarios in which no stuck open SRV, either HPCI or RCIC is successful, RCS depressurization and firewater injection are also successful (SRV0 * /HPI * /DEP_FWS). With the successful mitigation, there is no core damage (end state of OK). Sequences 3, 5, 7 to 12, and 19 to 24 in the original SBO model have the same characterization and are classified into the same category.

Table 4. Simplified SBO model sequences versus original SBO model sequences.

Seq.	SRV	Other Functions	Original Model Sequence	Out
1	0	/HPI * /DEP_FWS	3 + 5 + Sum(7:12) + Sum(19:24)	OK
2		/HPI * DEP_FWS * /REC	1+2+13+14+16+17+ 25+26+28+29	OK
3		/HPI * DEP_FWS * REC	4 + 6 + 15 + 18 + 27 + 30	CD
4		HPI * /DEP_FWS	n/a	OK
5		HPI * /REC	31 + 32	OK
6		HPI * REC	33	CD
7	1	/HPI * /DEP_FWS	n/a	OK
8		/HPI * DEP_FWS * /REC	35-1 + 35-2 + 35-4 + 35-5	OK
9		/HPI * DEP_FWS * REC	35-3 + 35-6	CD
10		HPI * /DEP_FWS	n/a	OK
11		HPI * DEP_FWS * /REC	n/a	OK
12		HPI * DEP_FWS * REC	35-7	CD
13	2	/HPI * /DEP_FWS	n/a	OK
14		/HPI * DEP_FWS * /REC	36-1 + 36-2 + 36-4 + 36-5	OK
15		/HPI * DEP_FWS * REC	36-3 + 36-6	CD
16		HPI * /DEP_FWS	n/a	OK
17		HPI * DEP_FWS * /REC	n/a	OK
18		HPI * DEP_FWS * REC	36-7	CD

Another example is Sequence 3 of the simplified model – this sequence also has no stuck open SRV with either HPCI or RCIC being functional. But with no RCS depressurization and/or firewater injection and without AC power recovery (neither offsite power nor diesel generators), core damage cannot be prevented (end state of CD). In the original model, the counterpart sequences are Sequences 4, 6, 15, 18, 27, and 30.

Note that there are a few sequences in the simplified ET that have no corresponding sequences in the original model. For Sequence 4 of the simplified ET (no stuck open SRV, HPI failure, but depressurization and firewater injection are successful), the original SBO model does not credit the depressurization and firewater injection with the assumption that there is no adequate time for operator to depressurize RCS and align firewater system for injection. Sequences 7, 10, 11 (one stuck open SRV, depressurization and firewater injection success or failure) and Sequences 13, 16, and 17 (two or more stuck open SRV, depressurization and firewater injection success or failure) of the simplified ET also have no corresponding sequences in the original SBO model as the depressurization and firewater injection are not modeled for stuck open SRV ETs (see SBO-1, SBO-2) for simplification reasons.

V.B RISMC data processing

Step 3 of Section II was performed by using an ad-hoc built `PYTHON` script. Its task was to parse all 20,000 RELAP5-3D simulations and perform Step 3 by considering throughout the simulation the status of system of components queried in the BWR SBO traditional model.

For each simulation run the following are retrieved:

- SRVs status
- High pressure injection status (both RCIC and HPCI)
- FW status
- AC power status (both DG or PG)

This allows the program to uniquely match each simulation run with a single branch of the ET shown in Fig. 9. The main idea was to create a set of information that is shared between the simulation data and the ET SBO generated by SAPHIRE. Once this information is filtered from each simulation run, the script associates each scenario to a branch of the ET shown in Fig. 9.

In addition, the script generate for each branch the following information as a summary of the simulations classified into that particular branch (see Fig. 10):

- Number of scenarios classified
- Probability of all scenarios classified
- Histogram of the outcome (OK due to AC recovery, OK due to firewater availability, CD)
- Maximum temperature of the clad
- Simulation end time
- Time of DG failure
- Plot of temporal profile of selected variables
- Summary of sequencing of events

VI. COMPARISON RESULTS

After running the `PYTHON` scripts we note the following:

- Each of the 20,000 simulations were classified into a unique branch of the ET shown in Fig. 9
- The outcome of each ET branch agrees with the final state of all simulations classified into that branch.

From Table 5 we can see the probability of CD for the simulations generated by RAVEN/ RELAP5-3D is fairly similar to the value generated by traditional methods (2.00 E-2 and 1.50 E-2 respectively). Core damage probability calculated using simulation based PRA (i.e., the RISMC approach) is 23% lower than one obtained using traditional ET/FT methods.

However, we noticed that, by looking at the probabilities associated with each ET branch, some differences arise. Table 6 shows these differences for all 18 branches of Fig. 9. In particular, we noticed that the distributions associated with the recovery time of AC power (either DGs or off site grid recovery), firewater recovery and SRV failure are driving these differences.

Table 5. Comparison of CD and OK probabilities.

Methodology	*OK*	*CD*
Traditional	0.980	2.00 E-2
Simulation	0.985	1.54 E-2

Table 6. Comparison of sequences (i.e., branch) probabilities (refer to the ET of Figure 9).

Branch	*Outcome*	*Traditional*	Simulation
1	OK	0.21	0.10
2	OK	0.77	0.86
3	CD	0.017	0.010
4	OK	n/a*	0.021
5	OK	8.6E-04	0.0056
6	CD	0.0033	0.0050
7	OK	n/a**	9.9E-06
8	OK	8.2E-04	1.7E-06
9	CD	1.1E-04	2.1E-07
10	OK	n/a**	6.7E-07
11	OK	n/a**	9.7E-07
12	CD	4.0E-06	5.0E-07
13	OK	n/a**	9.5E-07
14	OK	8.9E-05	2.6E-07
15	CD	1.2E-04	1.8E-07
16	OK	n/a**	2.9E-07
17	OK	n/a**	4.3E-08
18	CD	9.6E-07	2.1E-08

Notes:
* - The original SBO model does not credit DEP_FWS due to short time window for operator actions with HPI failure.
** - For simplicity, the original SBO model does not model DEP_FWS in SRV stuck open sequences.

Note that for the comparison described in this paper, we did not include some elements of the traditional PRA that would typically be considered such as common-cause failures and time-related elements (e.g., recoveries interacting with failures in time) that would require convolution factors to adjust the PRA cut sets.

By looking at the histograms of the maximum clad temperature (see Fig. 10), we were also able to determining that for the scenarios contained in branches leading to system OK, such histograms were containing scenarios with high clad temperatures. This fact was caused by a failure of the DC system but followed by AC recovery just before reaching CD. The scenarios in which DC failure lead to high clad temperature are pictured in Fig. 10 (bottom left).

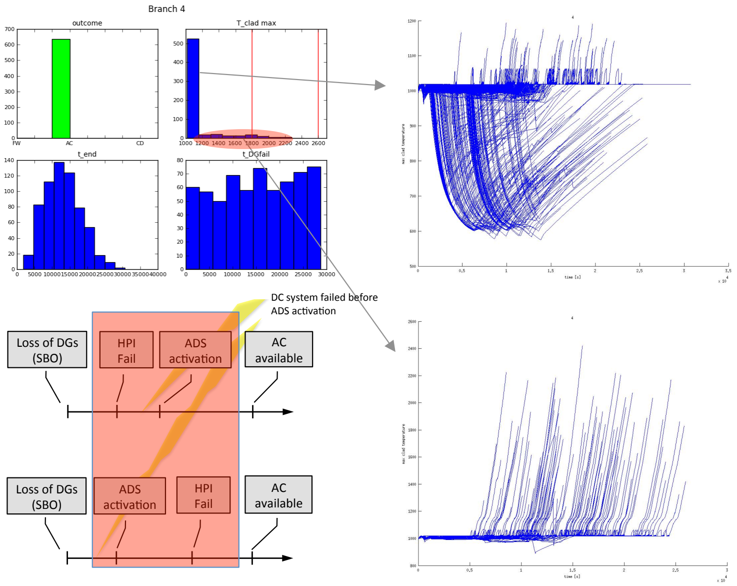

Fig.10.Effect of DC system failure on max clad temperature histogram for scenarios leading to system OK.

VII. CONCLUSIONS

In this paper we showed that a simulation based PRA analysis for a BWR SBO accident scenario generated a core damage probability value similar to the one calculated using an ET-FT methodology using SAPHIRE. At this point the following may arises from the reader:

Are the efforts required by the RISMC approach worth the results that can be obtained using state-of-practice methodologies?

From our perspective we believe that simulation based methods are the natural extension traditional methods. An extension that aims to overcome the natural limitations of the latter ones such as: user-defined accident progression and lack system dynamic feedback into timing/sequencing of events. So now, the question presented above is replied by the following:

Are these two limitations justifiable to employ classical tools (ET-FT) for the applications targeted by the RISMC pathway?

We believe that the answer for such question is negative. Neither power uprate nor ageing are implicitly taken into account in an ET-FT based methodology. They could only be considered in the actual approximated computation of the ET branches or FT basic event probabilities without modeling their actual feedback on timing/sequencing of events.

This has been shown in this work where we employed the RISMC approach to evaluate impact, from a statistical point of view, of power uprate for a BWR SBO accident scenario. The actual power uprate was implicitly modeled in the RELAP5-3D simulator while the distributions of the uncertain parameters remained unchanged. In an ET-FT approach, a power uprate would have required a "re-computation" of the FT basic events and/or the ET branching probabilities. Such computation would involve few system simulator runs in order to asses the time needed for certain basic events to occur before CD status is reached. In the RISMC approach such computation is implicitly embedded in the sampling process of the simulation run internal parameters.

REFERENCES

[1] C. Smith, C. Rabiti, and R. Martineau, "Risk Informed Safety Margins Characterization (RISMC) Pathway Technical Program Plan", Idaho National Laboratory INL/EXT-11-22977 (2011).

[2] D. Mandelli, C. Smith, T. Riley, J. Nielsen, J. Schroeder, C. Rabiti, A. Alfonsi, J. Nielsen, R. Kinoshita, D. Maljovec, B. Wang, and V. Pascucci, "Overview of new tools to perform safety analysis: Bwr station black out test case," in *Proceedings for PSAM 12 Conference*, Honolulu (USA), 2014.

[3] U.S. NRC, "WASH 1400 - Reactor Safety Study - An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants", Division of Systems Research, Office of Nuclear Regulatory Research, U.S. Nuclear Regulatory Commission, Washington, DC, (1975).

[4] E. Zio, M. Marseguerra, J. Devooght, and P. Labeau, "A concept paper on dynamic reliability via Monte Carlo simulation," in *Mathematics and Computers in Simulation*, pp. 47–371, (1998).

[5] A. Alfonsi, C. Rabiti, D. Mandelli, J. Cogliati, and R. Kinoshita, "Raven as a tool for dynamic probabilistic risk assessment: Software overview," in *Proceeding of M&C2013 International Topical Meeting on Mathematics and Computation*, CD-ROM, American Nuclear Society, LaGrange Park, IL (2013).

[6] J. C. Helton and F. J. Davis, "Latin hypercube sampling and the propagation of uncertainty in analyses of complex systems," *Reliability Engineering & System Safety*, **81**-1 (2003).

[7] A. Amendola and G. Reina, "Dylam-1, a software package for event sequence and consequence spectrum methodology," in EUR-924, CEC-JRC. ISPRA: Commission of the European Communities (1984).

[8] A. Alfonsi, C. Rabiti, D. Mandelli, J. Cogliati, and R. Kinoshita, "RAVEN: Development of the adaptive dynamic event tree approach," Tech. Rep. INL/MIS-14-33246, Idaho National Laboratory (INL), 2014.

[9] H. S. Abdel-Khalik, Y. Bang, J. M. Hite, C. B. Kennedy, C. Wang, "Reduced Order Modeling For Nonlinear Multi-Component Models," *Interna-tional Journal on Uncertainty Quantification*, **2**-4, pp. 341-361 (2012).

[10] D. Mandelli, C. Smith, T. Riley, J. Schroeder, C. Rabiti, A. Alfonsi, J. Nielsen, D. Maljovec, B. Wang, and V. Pascucci, "Support and Modeling for the Boiling Water Reactor Station Black Out Case Study Using RELAP and RAVEN", Idaho National Laboratory report: INL/EXT-13-30203 (2013).

[11] D. Mandelli, C. Smith, C. Rabiti, A. Alfonsi, J. Cogliati, and R. Kinoshita, "New methods and tools to perform safety analysis within RISMC," in *Proceeding of American Nuclear Society (ANS)*, Washington (DC), 2013

[12] D. Gertman, H. Blackman, J. Marble, J. Byers and C. Smith, "The SPAR-H Human Reliability Analysis Method", U.S. Nuclear Regulatory Commission (2005).

[13] C. Rabiti, A. Alfonsi, D. Mandelli, J. Cogliati, R. Martinueau, C. Smith, "Deployment and Overview of RAVEN Capabilities for a Probabilistic Risk Assessment Demo for a PWR Station Blackout," Idaho National Laboratory report: INL/EXT-13-29510 (2013).

[14] RELAP5 Code Development Team, "RELAP5-3D Code Manual", Idaho National Laboratory Technical Report INEEL-EXT-98-00834 (2012).

[15] D. Anders, R. Berry, D. Gaston, R. Martineau, J. Peterson, H. Zhang, H. Zhao, L. Zou, "RELAP-7 Level 2 Milestone Report: Demonstration of a Steady State Single Phase PWR Simulation with RELAP-7," Idaho National Laboratory: INL/EXT-12-25924 (2012).

[16] D. Mandelli, C. Smith, Z. Ma, T. Riley, J. Nielsen, A. Alfonsi, C. Rabiti, and J. Cogliati, "Risk-informed safety margin characterization methods development work," Tech. Rep. INL/EXT-14-33191, Idaho National Laboratory (INL), 2014

[17] S.T. Wood, C. L. Smith, K. J. Kvarfordt, S.T. Beck, "Systems Analysis Programs for Hands-on Integrated Reliability Evaluations (SAPHIRE): Summary Manual (NUREG/CR-6952, Volume 1)", Office of Nuclear Regulatory Research U. S. Nuclear Regulatory Commission Washington (2008).

[18] Z. Ma, J. Schroeder, "Event Tree Success Branch Modeling Approaches in SAPHIRE/SPAR," *Transaction of the American Nuclear Society* **110**, pp. 349 -351 (2014).