Evaluation of the source term prediction capability of RASTEP against integral response code calculations

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Abstract: RASTEP (Rapid Source Term Prediction), developed by Vysus Group in cooperation with the Swedish Radiation Safety Authority, SSM, is an emergency preparedness tool that uses a probabilistic approach to provide decision support in nuclear emergency situations, with the aim of being robust to lack of or uncertain information.
This work examines the possibility of developing a comprehensive validation scheme for RASTEP models by using cross-validation against integral response code results. Three transients were studied for a generic PWR plant: Large Break LOCA, Interfacing System LOCA and Station Black-Out. For each transient, the RASTEP software was fed with data extracted from integral response code results representing the same transient on a similar reactor model. In each case, the time of onset of release of radionuclides is compared with the time when the simulated case is detected by RASTEP as the most probable release category. The results of this work show that for each of the three transients RASTEP predicts the correct release category ahead of time, even when there is a certain level of information lost due to defective sensors or information missed by the user.

Keywords: BBN, NPP, RASTEP, LBLOCA, ISLOCA, SBO, V&V, source term.

1. INTRODUCTION

To make appropriate decisions in emergency response, there is a need to make predictions of the potential consequences of an ongoing event.

RASTEP, developed by Vysus Group in cooperation with the Swedish Radiation Safety Authority, SSM, is an emergency preparedness tool that uses a probabilistic approach to provide decision support in nuclear emergency situations, with the aim of being robust to lack of or uncertain information. The software is based on Bayesian Belief Networks (BBNs) which can link variables that may take different states given the user's observation input. These variables are interconnected across the network, using conditional probabilities to capture an online diagnosis of the plant.

Making a prediction of the source term in a nuclear power plant (NPP) severe accident is important for any emergency response team. RASTEP provides a real time likelihood assessment over a set of possible source terms. This feature is important for the early diagnosis of incidents or accidents and to support decision-making to mitigate the consequences of severe accidents.

1.1 RASTEP: From sensor data to accident scenario prediction

RASTEP allows to calculate release categories probabilities and link them to source terms.
To use RASTEP for a specific NPP, the plant and its systems must first be modeled, as well as the functional relationship between them. For that, RASTEP uses a BBN, providing a representation of the systems of the NPP and to capture their inter-dependence.

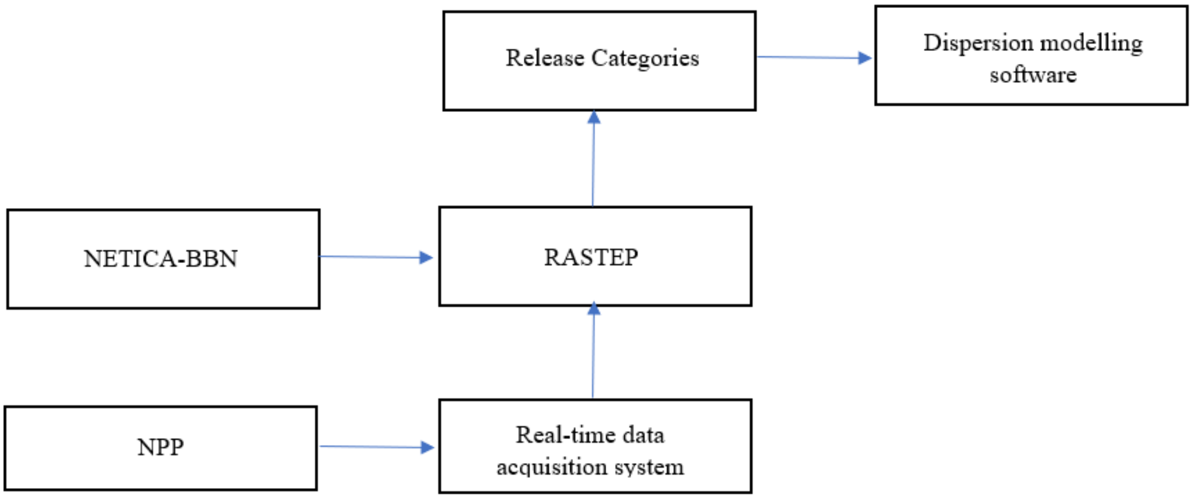
The underlying BBN network is modelled in NETICA (NorSys Software Corporation). Once the model is designed, a validation and verification scheme (V&V) is needed to assess the accuracy of RASTEP predictions.

This report focuses specifically on this step, analysing RASTEP predictions against integral response code calculations performed on a similar reactor model.

1.2 RASTEP: Example of the information flow

Figure 1 depicts an example of the information flow between the NPP's sensors and a source term dispersion modelling software. In the setup currently developed for SSM, information from the NPP can, during an accident, be sent and constantly acquired by a real time acquisition system. This system will feed RASTEP with plant observations, which, based on the BBN model of the reactor will calculate conditional probabilities on a pre-defined set of release categories. Typical source terms of these release categories will be used by the dispersion modelling software to determine, based on the scenarios and the climate conditions, the dose as a function of the position and time.

Figure 1: Information flow from the NPP to a dispersion modelling software



An important question that arises when analyzing the information flow is how RASTEP would behave in case of loss of information. This situation could take place in case there is either, a defective signal transmission system from the NPP, or a loss of information due to the consequences of a severe accident.

1.3 BBN modelling of nuclear power plants

In this work, a generic PWR model was used. The BBN network for this model contains numerous nodes and connections. Each of the nodes can be either deterministic or probabilistic. For instance, the former represents logical relations such as:

"if neither offsite power nor diesel generators are available, then no AC power is available"

Technically, a deterministic node is a probabilistic node with probabilities ranging from 0 to 100 % in its conditional probabilistic table.

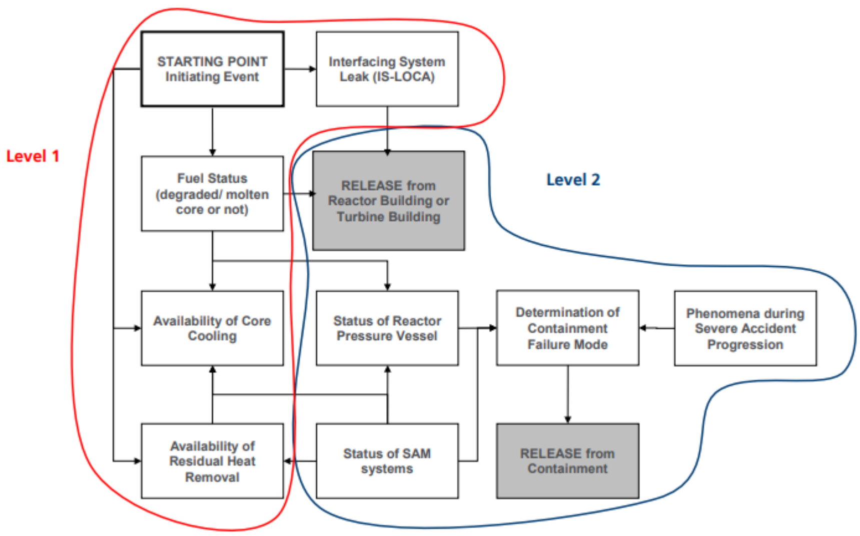
On the other hand, the later is based on a probabilistic safety assessment (PSA) or its underlying data, and it represents probabilistic relations such as:

"The probability of a large break LOCA is P"

To improve the readability and maintainability of the BBN, the nodes are grouped into subnetworks, representing important systems or collections of systems in the plant.

The basis used for BBN modelling is the first and second level of the probabilistic safety analysis (PSA). In Figure 2, a generic block structure of a BBN model for a NPP is presented.

Figure 2 General model structure – BWR example [1]



A satisfactory BBN model would allow to predict potential incidents or accidents in the NPP modelled by connecting observations of the ongoing situation to high likelihoods of relevant release categories. A perfectly matching BBN model of the NPP should allow to fulfill two criteria for any transient scenario studied: to predict the same release category as simulated by the integral deterministic codes, and to predict it ahead of the onset of release time. This might not be achievable due to inaccuracies such as issues in the nodes (not optimal conditional probability tables) or issues in the interconnection between the nodes.

The criteria used to evaluate whether the V&V process is successful or not must consider the fact that RASTEP can only predict release categories that have been included in the model. Furthermore, the identification must be provided ahead of time. Being that considered, the requirements to consider a V&V process as satisfactory are the following:

Requirements for transient scenarios included in the BBN model:

- The BBN model must predict the correct release category as the first or second most probable before it occurs in the deterministic calculation.
- In case it is predicted as the most probable, the second most probable source term must be relevant in view of the events described by the deterministic calculation, otherwise its predicted probability must be at least one order of magnitude lower than the most probable release category.

Ideal requirements for transients involving multiple release paths (for instance, a transient leading to a breach of the containment walls, followed to a basemat melt-through):

- Regardless of the relative position, both source terms involving the largest amount of radioactive release must be predicted by RASTEP as the two most probable. Furthermore, the prediction must be ahead on time.

2. PROPOSED VALIDATION AND VERIFICATION PROCEDURE

The flowchart that represents the V&V procedure is represented in Figure 3.

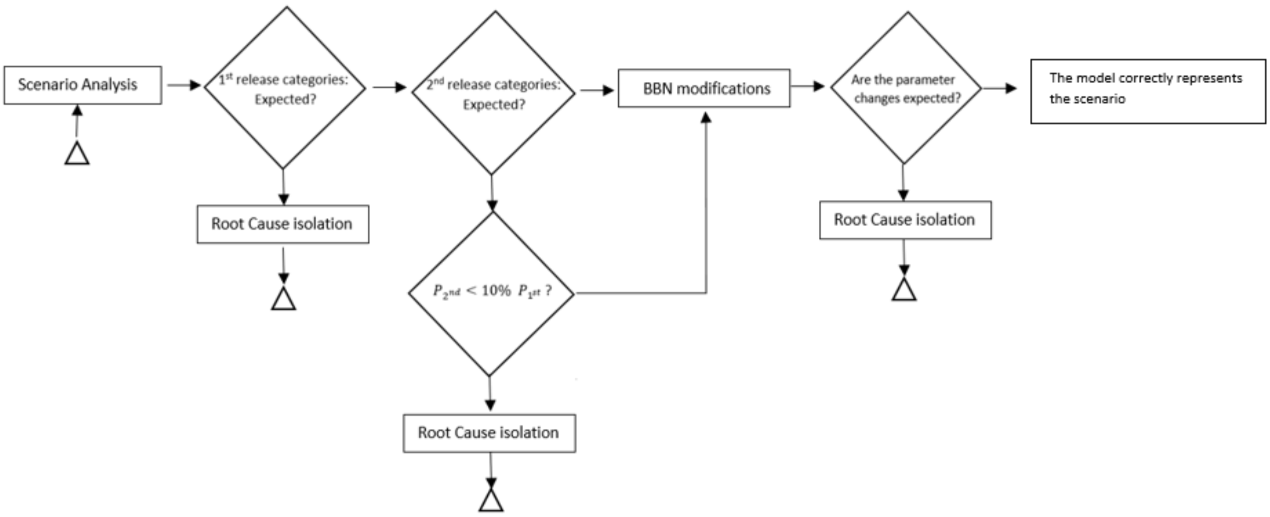
The first block is 'Scenario Analysis' which consists in representing the scenario by selecting states in the BBN model based on observations of process variables. The objective is to reproduce the release categories as the ones from a specific accident calculated by the integral plant response code.

The second block corresponds to the analysis of the results obtained for each scenario. The main question to answer is whether the most probable release categories match with the results obtained by the deterministic scenario in question. Otherwise, the flow continues to the root cause identification.

The third block focuses on the analysis of the second most probable release category. If the corresponding results do not match with the real categories and the model predicts them with a probability larger than one tenth of the most probable release categories, then the flowchart requires to isolate the root cause. Otherwise, it leads to the fourth block.

The fourth block is called "BBN modifications". It consists of studying how a change in the state on one subnetwork impacts on the others. In case the parameters' changes obtained are as expected, it is possible to conclude that the model correctly represents the scenario.

Figure 3: Process used for V&V



3. TRANSIENT SCENARIOS

3.1 General procedure to analyze a transient scenario

To proceed with the V&V process, the release categories predicted by the BBN model are compared with the correct ones as calculated by an integral response code, in this case MAAP v5.04.

As the results of the deterministic simulations are time dependent, the process parameters time evolutions (like core exit temperature, or primary pressure trend) were discretized in time intervals, each of them describing a specific state of the transient, to match the corresponding BBN model node states.

Each state is characterized by constant values or trends on the most critical parameters of the reactor, such as: primary pressure trend, coolant temperature or gamma radiation levels in the containment. This means that when an important parameter shows a change in its value or its trend, a new state is created.

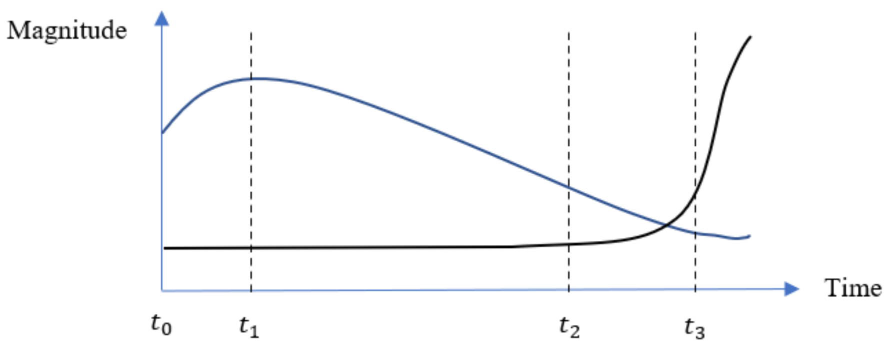
A simplified example is shown in Figure 4 to clarify how this process is performed. This figure illustrates a hypothetical scenario during which the radiation levels in the containment (black curve) and the primary pressure (blue curve) undergo changes in their magnitude levels.

In the first time-interval $(t_1 - t_0)$, the important parameters remain monotonous, as the primary pressure increases and the radiation level in the containment remains constant. Thus, this time interval will constitute the first time-step to be evaluated in the BBN model.

In the second time-interval $(t_2 - t_1)$, the primary pressure shows a change in its trend. This constitutes the second time-step to be considered for the BBN model.

Although the primary pressure trend does not change after t_2, at this specific time the radiation level on the containment surpasses a certain threshold. Therefore, at that time a new time-step for the BBN model starts. This last time-step ends at t_3, when the primary pressure changes its trend from decreasing to stationary. Therefore, this hypothetical scenario would be analyzed in the BBN model by three time-steps, each of them with constant trends on the critical parameters of the plant.

Figure 4: Representation of the time evolution of critical parameters



The information loss in accidental scenarios, out of, for instance sensor malfunction, must be considered. The process of losing sensor data is modelled as a random event. Furthermore, it is known that changed states in certain nodes in the BBN network can lead to relatively large changes in the source term prediction. These nodes are called influential nodes.

These two aspects have been considered in this work. A sensitivity analysis was performed to analyze the most influential nodes regarding the containment source term. This analysis calculates the probability distributions on the release categories before the observation on the node "*I*" is introduced ($p_{i,b}$), and the same distribution after the mentioned observation is introduced ($p_{i,a}$). Afterwards, the change in the probability distribution is calculated as

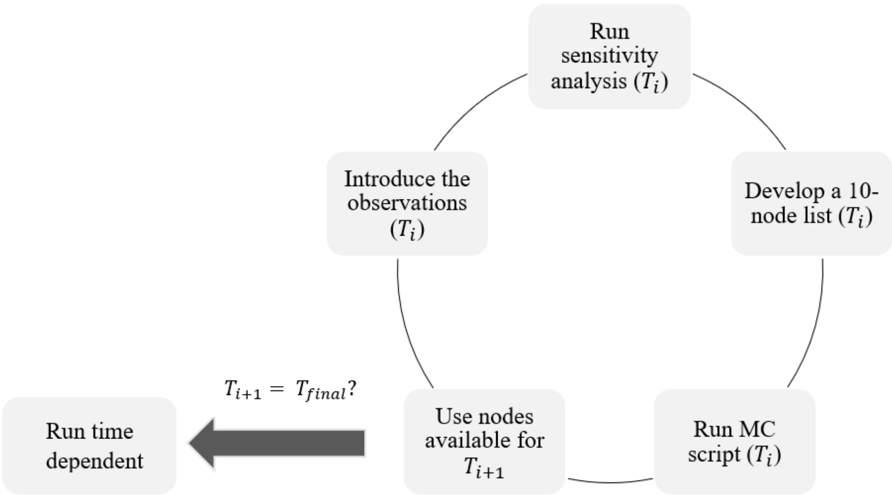
$$\Delta p = |p_{i,a} - p_{i,b}| \quad (1)$$

Those nodes that maximize the last expression are the most influential nodes. The outcome of the sensitivity analysis is a set of the 10 most influential nodes in the BBN network. The sensitivity analysis is applied to each scenario analyzed to select the ten most influential nodes.

To simulate the information loss, a sampling procedure was used. First, a random number (*R*) is generated from a uniform probability distribution and is later compared with a threshold level "*p*". If *R* is larger than *p*, then the observation corresponding to one of the ten nodes pre-selected by the sensitivity analysis to be disregarded, i.e., simulated to be lost. To that purpose, a second random number is sampled from a uniform probability distribution. Once an observation is lost, it stays lost throughout the simulation. The general procedure is illustrated in Figure 5. The procedure starts with introducing all the observations online at every timestep T_i on the BBN network. At the very beginning of the transient, all observations are online.

The next step is to analyze the sensitivity of the nodes, given the state of the BBN network from the previous step. The outcome will be a set of the ten most influential nodes. Afterwards, a Monte Carlo elimination script determines which nodes are online, and which ones are offline. Those nodes that are offline will continue in that state throughout all the following time steps, which means that the probabilities of their states are determined by the underlying conditional probability table. This procedure runs until the last time step is analyzed and simulated.

Figure 5: Representation of the general procedure.



3.2 Transients analyzed

Three transients were analyzed for a generic PWR model, namely Large Break LOCA (LBLOCA), Station Black-Out (SBO) and Interfacing System LOCA (ISLOCA).

Process data was extracted from the results of MAAP v5.04 calculations, as performed by JRC during the EU Horizon-2020 FASTNET project [2].

Furthermore, it should be kept in mind that all observations were considered online at the very beginning, but from the second time step and on, those observations selected by Monte Carlo eliminations are lost and continue in that state until the end of the simulation

3.2.1 Large Break LOCA

This scenario consists of a Large Break LOCA initiated by a 12-inch hot leg break, with the Emergency Core Cooling System (ECCS) being unavailable. However, the containment spray is operating. The SCRAM occurs 1.8 seconds after the initiating event, and the core relocation to the lower plenum occurs shortly after 1 hour of the initiating event.

The simulation performed by JRC leads to diffuse leakage (intact containment). However, the results show significant basemat erosion due to molten corium-concrete interaction (MCCI). Due to design specifics of the containment (dry cavity) and inherent uncertainties in the modeling of such phenomena in deterministic codes, it was conservatively assumed that the sequence leads to basemat melt-through within 3.5 days after initiating event. The value was adopted from the MAAP code calculation for LBLOCA with both ECCS and CSS being unavailable, since the BBN model, currently, prioritizes the (late) basemat melt-through conclusion above earlier phase and less severe release categories which distinguishes between CSS/noCSS. This however can be considered as an area for future improvement of the model.

3.2.2 Station Black-Out

The second scenario studied is SBO including the unavailability of the auxiliary feedwater turbine-driven pump as well as the unavailability of the containment spray. Furthermore, the filtered containment venting system is unavailable during the whole transient.

The core uncover occurs shortly after 2 hours of the initiating event, and a complete breach of the containment takes place around four days after initiating event.

The first release of radioactivity occurs 125 minutes after the initiating event (due to diffuse leakage from the containment), and the final release category is basemat melt – through.

3.2.3 Interfacing System LOCA

The last scenario studied is a 12-inch break outside the containment in the system interfacing with the primary system (ISLOCA). In this scenario, the ECCS is available initially, but switched off after 2.8 hours after initiating event due insufficient NPSH. Regarding the containment, it is considered that its isolation is not achieved. The SCRAM occurs 5 seconds after the initiating event, while the core relocation to the lower plenum takes place around 5 hours after the initiating event.

The release of radioactivity (through the auxiliary building) occurs 225 minutes after the initiating event. As the auxiliary building ventilation system is unavailable in this transient, the release is unfiltered, which is identified in RASTEP possible source terms as *Melt Leak AB NoFilt.*

Furthermore, the simulation performed by JRC predicts significant basemat erosion depth due to MCCI, however below the threshold for basemat melt-through. Similarly as for LBLOCA scenario (see the explanation in section 3.2.1), it was conservatively assumed that the sequence leads to basemat melt-through for the containment source term.

4. RESULTS

4.1 Results on LBLOCA transient

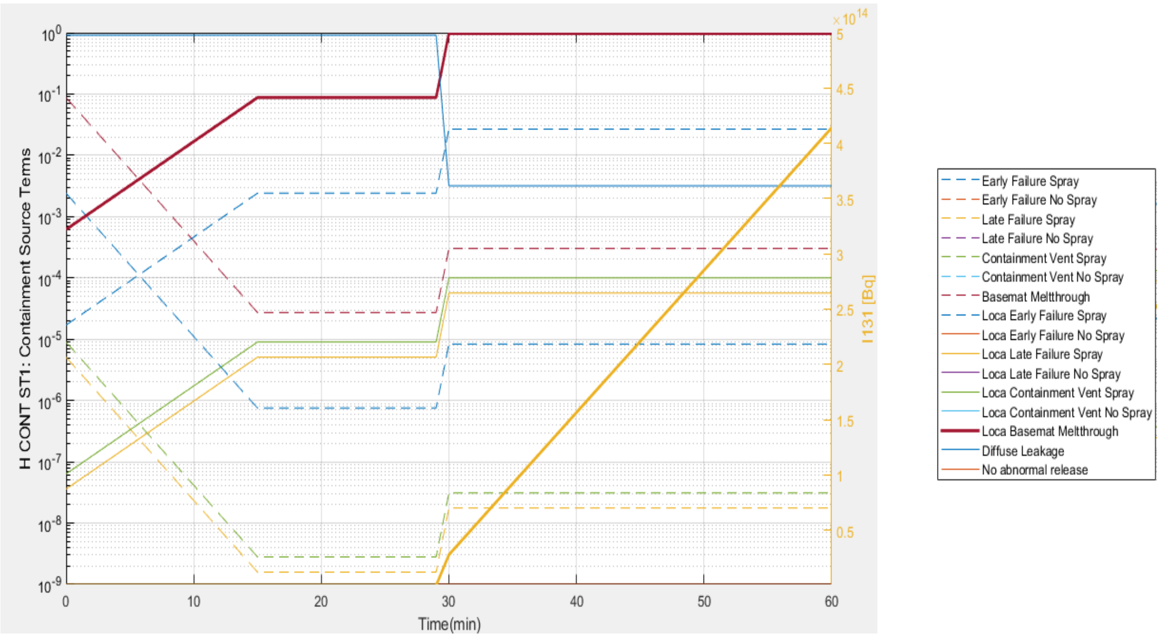
As it was explained in Section 3, the first part of the analysis consists of detecting changes in the important parameters' trends. It is important to notice that at the start of the simulation (0 minutes) there are five system states that are assumed to be known, as these states were considered as initial conditions in the transient performed by JRC. Table 1 summarizes the different time-steps detected and the corresponding parameters that trigger their selection for the LBLOCA transient. The same method was used for the remaining two transients.

Table 1: Time-steps selected for LBLOCA scenario

Time-step	Parameters
0 mins	ECCS unavailable
	Reactor depressurization system unavailable
	Filtered containment venting system unavailable
	Turbine condenser unavailable
	Ex-vessel flooding unavailable
0 - 15 mins	Steam Generator Level: Normal
	Secondary Pressure trend (Decreasing)
	Secondary Pressure (Low)
	Containment Sump (Empty)
	Primary pressure (Decreasing)
	Pressurizer level: Low
15 - 30 mins	Core exit temperature (> 1200 C)
	Steam Generator Level: Normal to Low
	Containment Sump (Full & Hot)
>30 mins	Containment Hydrogen (> 4%)
	Gamma containment readings (High)

The time evolution of the containment release categories was obtained. The results are represented in Figure 6.

Figure 6: Time evolution for the containment source term and Iodine 131 concentration (LBLOCA)



In this case, the release starts around 30 mins after the transient started. This release corresponds to unfiltered leakage from the containment, which is correctly predicted by RASTEP before it occurs (diffuse leakage). The final release mode (LOCA basemat melt-through) is also correctly predicted as the most probable release category, 3.5 days ahead on time.

4.2 Station Black-Out

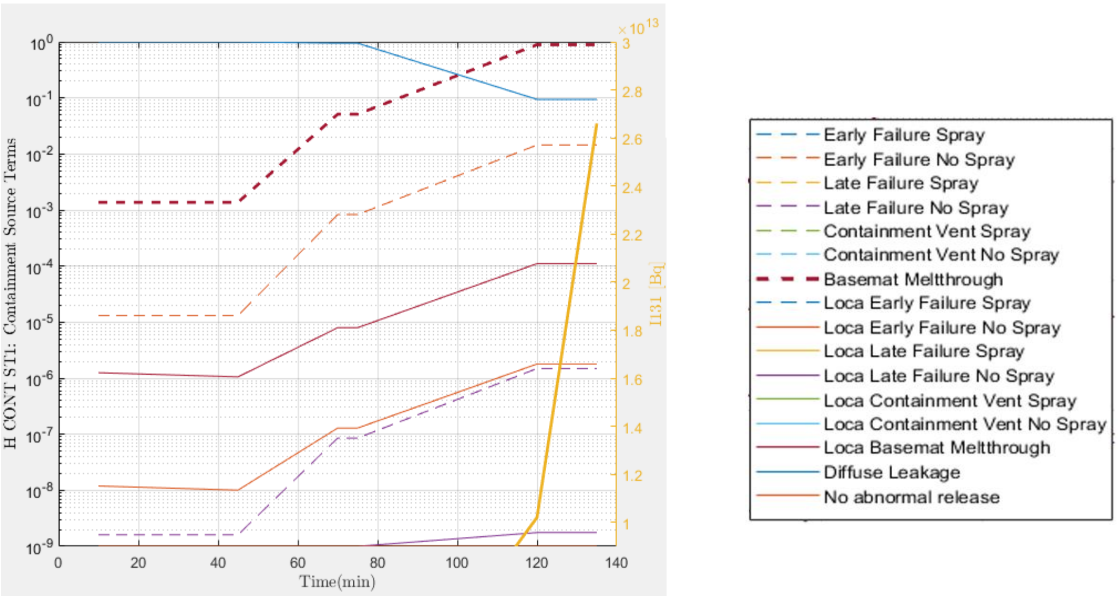
The time-steps and the corresponding parameters that trigger their selection are presented in Table 2. Similar to the procedure followed in the previous transient, the systems corresponding to t=0 mins in Table 2 were considered as unavailable from the start of the simulation.

Table 2: Time-steps selected for SBO scenario

Time-step	Parameters
0 mins	Offsite power and diesels unavailable
	ECCS & injection system unavailable
	Reactor depressurization system unavailable
	Turbine condenser unavailable
	Ex-vessel flooding unavailable
	Containment spray unavailable
10 - 45 mins	Core exit temperature: normal
	Primary Pressure trend (Decreasing)
	Steam Generator Level trend: Increasing
	Containment Sump (Empty)
	Primary pressure (Low)
45 - 75 mins	Steam Generator Level trend: Steady
	Primary Pressure trend (Increasing)
	Primary pressure (Low)
75 – 125 mins	Core Exit Temperature > 600 C
	Containment Sump (Full &Hot)
>120 mins	Core Exit Temperature > 1200 C

The time evolution of the containment release mode, together with the Iodine 131 concentration, for the (SBO scenario) are displayed in Figure 7.

Figure 7: Time evolution for the containment source term and Iodine 131 concentration (SBO)



In this case, there is diffuse leakage from the containment, which starts around 100 mins after the transient was initiated. The final release mode from the containment (basemat melt-through, highlighted in dash-red line in Figure 7) is predicted as the most probable release category, predicted two days ahead.

4.3 Interfacing System LOCA

In this scenario, there is radioactive release from both, the containment and the auxiliary building.

The time-steps and the corresponding parameters that trigger their selection as well as the systems considered unavailable at the start of the simulation (t=0 mins), are presented in Table 3.

The time evolution of the auxiliary building source term is displayed in Figure 8, while for the Containment it is displayed in Figure 9.

Table 3: Time-steps selected for IS-LOCA scenario

Time-steps	Parameters
0 mins	Containment spray unavailable
	Reactor depressurization system unavailable
	Ex-vessel flooding unavailable
	Filtered containment venting system unavailable
15 - 60 mins	Core exit temperature: normal
	Primary Pressure trend (Decreasing)
	Steam Generator Level trend: Decreasing
	Containment Sump (Empty)
	Primary pressure (Low)
60 - 120 mins	Steam Generator Level trend: Steady
	Primary Pressure trend (Decreasing)
120 – 180 mins	Containment Sump (Full &Hot)
180 - 210 mins	Core Exit Temperature > 600 C
	Pressurizer Level: Low
>210 mins	Core neutron detectors out of normal range

Fig 8. Time evolution for the auxiliary building source term and Iodine 131 concentration (ISLOCA)

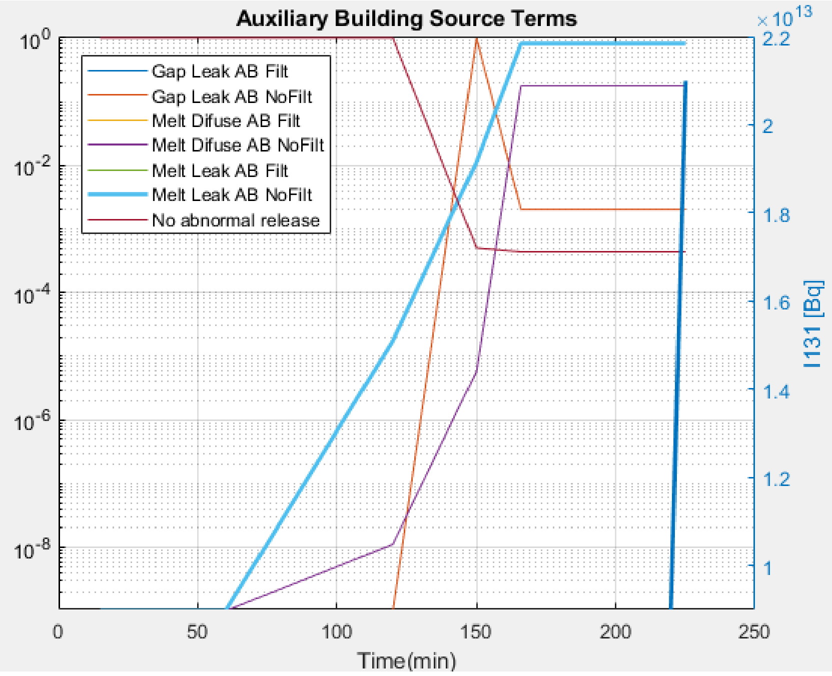
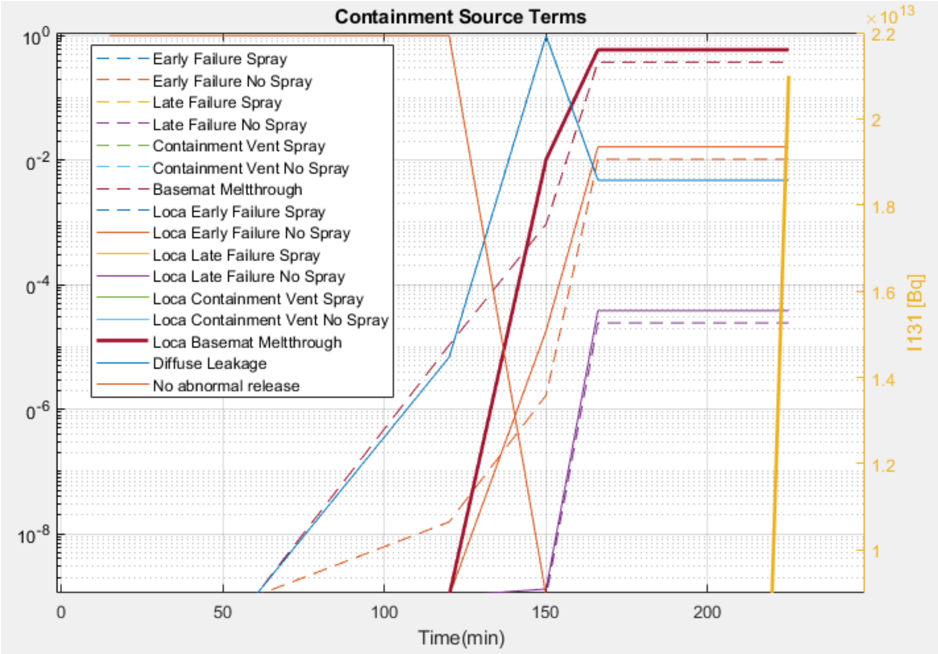


Fig 9. Time evolution for the containment source term and Iodine 131 concentration (ISLOCA)



It is possible to see from Figure 8 that the release happens at 225 minutes, while RASTEP predicts the correct release mode (unfiltered release, represented with light blue color in Figure 8) 58 minutes before the radioactive release.

For the containment, the correct containment release mode (diffuse leakage, represented by the blue line in Figure 9) is predicted 75 minutes before the radioactive release. The final release mode from the containment (LOCA basemat melt-through) is predicted by RASTEP 3.5 days ahead on time.

5. CONCLUSIONS

RASTEP was able to predict the correct release category for the three scenarios studied, either as the first or the second most probable source term. The major release term was predicted hours ahead for the ISLOCA transient, while in the case of the LBLOCA and SBO transients, the major release (basemat melt-through) was predicted days ahead.

For the LBLOCA transient, RASTEP was able to predict the first release (diffuse leakage), which occurs 30 minutes after the initiating event, as well as the final release mode (LOCA basemat melt-through), predicted as the most probable release mode, which takes place 3.5 days after the initiating event.

For the ISLOCA transient, two source terms take place, namely in the containment building and in the auxiliary building. RASTEP predicts a radioactive release from the auxiliary building 75 minutes before the actual release, and the correct source term is predicted 58 minutes ahead on time. Furthermore, RASTEP correctly predicts the containment source term.

For the SBO transient, the first release mode was predicted by RASTEP as the most probable 3.5 days ahead, and the second most probable release mode (diffuse leakage) was also predicted ahead on time.

References

[1] Di Dedda F., Olsson A., Klug Joakim, Riber Marklund A., *RASTEP – A novel tool for nuclear accident diagnosis and source term prediction based on PSA and Bayesian Belief Networks*, PSAM14, 2014.

[2] FASTNET project official webpage, FASTNET (fastnet-h2020.eu)