Comparative Analysis of Generic RASTEP Models: VVER-1200 and Westinghouse 3-Loop PWR Reactors

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Abstract: The Rapid Source Term Prediction (RASTEP) methodology has been developed by Vysus Sweden AB, mainly supported by the Swedish Radiation Safety Authority and the EU Horizon-2020 FASTNET project. RASTEP is a decision support tool to be applied in nuclear emergency situations, by answering questions on the observed status of an affected plant, with the aim of ranking pre-calculated accident sequence types and their associated source terms by likelihood, given available current observations. RASTEP provides emergency response organizations with quick, independent and easy-to-use decision support based on best available knowledge, thereby increasing the capability of acting independently of the affected plant and nation as well as facilitating the provision of advice and information to other organizations and the public.
Throughout the development of RASTEP software, Vysus Group have developed several generic-type RASTEP models for different reactor designs, such as the Westinghouse 3-loop PWR, VVER 440/213, Nordic-type BWR, M310 PWR, and recently, a generic-type VVER1200 RASTEP model, in cooperation with VATESI (State Nuclear Power Safety Inspectorate of Lithuania). The VVER1200 model contains, amongst others, safety systems such as a core catcher, passive hydrogen recombiners and passive residual heat removal with steam generators and passive containment heat removal systems. This paper presents a comparative analysis of the generic Westinghouse 3-loop PWR and the generic VVER-1200 RASTEP models, focusing on predictive capability of these models, against the most typical severe accident scenarios (in terms of likelihood and in terms of consequences). The ability of the models to predict the correct accident scenario and mode of radioactive release and associated time is considered as the main figure of merit.

Keywords: Emergency preparedness, Source Term, RASTEP, Bayesian Belief Network.

1. INTRODUCTION

With the necessity of efficient tools for the prediction and diagnosis of radiological source term in the event of severe accident scenarios at a nuclear power plant the implementation such tools need to improve the preparedness of accident response and provide information for off-site emergency management.

The RASTEP (RApid Source TErm Prediction) is such a decision support software tool that is based on integration of inputs from probabilistic (PSA) and deterministic (DSA) safety analyses, Figure 1. RASTEP encapsulates both cause and effect for complex cases with a large number of variables and where certain data is incomplete or have high uncertainty.

The RASTEP tool is based on Bayesian Belief Networks (BBNs) that represents probabilistic and deterministic relations among observations, events, and process variables. The BBN model connects known data and expert judgements with observations of the ongoing situations and maps the outcome to pre-calculated scenarios (from database provided by DSA). Due to the nature of BBNs, the RASTEP tool can always provide a best estimate of the ongoing situation, based upon available data (provided by PSA), information, and expert judgements already built into the plant model.

Using the RASTEP tool the user answers a series of pre-formulated questions on specific parameters relating to the affected plant. As circumstances develop, new or updated information on the specific system parameters can be entered into the tool using the interface. RASTEP applies immediately the updated data in the model, with a result of a new diagnosis of the overall state of the affected plant and the potential end state (including the source term), thereby supporting decision-making at national and local authorities, see Figure 1.

Bayesian Belief Networks (BBN) represent an established method of modelling uncertain relations among random variables and capturing the relationships between these variables using Bayes' theorem. The BBN approach is to take prior beliefs at the outset and when information on the progression of an event becomes available, modify, and update those beliefs.

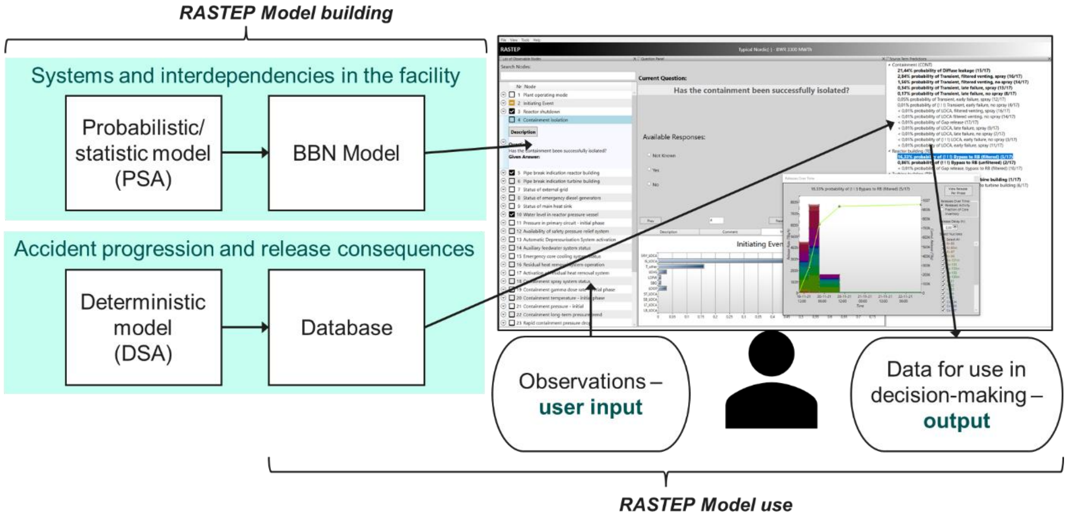

Figure 1. RASTEP Overview [1].

The BBN model for a specific reactor type is based on a priori information of the nuclear power plant PSA analysis, of first and second level, and is developed in the BBN software Netica (NorSys Software Corporation). The network consists of a connection of numerous nodes, which represent variables in the model and links plant observations to possible initiating events, damage states of barriers and systems as well as various accident phenomena through conditional probabilities. The BBN model is structured by grouping the nodes in subnetworks, each representing the main elements or systems in the plant logic, in an order that reflects the progression of the accident [1].

In this paper two generic models are compared: a generic Westinghouse 3-loop PWR (W3 PWR) and the model of a generic VVER1200 reactor. The main figure of merit is the time between the correct prediction and the time of release of I-131 to the environment, considering the importance of the nuclide in the release. This paper also serves as validation of the new generic model of the VVER1200 reactor against integral code calculations. For validation calculations and a description of the generic W3 PWR model see [2].

2. GAP ANALYSIS OF THE GENERIC W3 PWR AND VVER1200 MODELS

In terms of RASTEP modelling of the initiating events groups, main safety systems and their groupings into safety functions and associated subnetworks in RASTEP, the VVER1200 and W3 PWR reactor designs are quite similar. However, there are differences, especially when looking at the Gen III+ safety features added to the VVER1200. See Figure 2 for a general diagram of the VVER1200 safety systems.

The main differences between the reactor designs and the effects of these differences on the RASTEP models were identified in [4] and the differences relevant to the scope of this paper are listed below:

- The VVER1200 design does not have a filtered containment venting system, and thus no possibility of filtered venting of the containment as in the W3 PWR.
- The VVER1200 includes a core catcher.
    - Due to presence of the core catcher the risk of ex-vessel steam explosion and basemat melt through is eliminated. In case of failed water injection into the core catcher/core catcher compartment, the likelihood of basemat melt through is deemed to be very low, due to presence of crucible materials in the core catcher and inversion of molten metals.
- The VVER1200 includes passive residual heat removal via steam generators (SGs) and passive containment heat removal systems.

- The Refueling Water Storage Tank (RWST) is located inside the containment in the VVER1200 reactor design, and it serves as the containment sump, and the primary water source for the containment spray (JMN), high pressure safety injection (JND) and low-pressure safety injection (JNG-1) systems.
    - The high-pressure safety injection (JND), low pressure safety injection (JNG-1) and containment spray (JMN) systems have residual heat removal system heat exchangers in the suction lines of these systems.
- The spent fuel, service/internal parts (revision pit) pools are located inside the containment in the VVER1200 reactor design. The water from the revision pit serves as a water source for the core catcher and ex-vessel debris coolability.

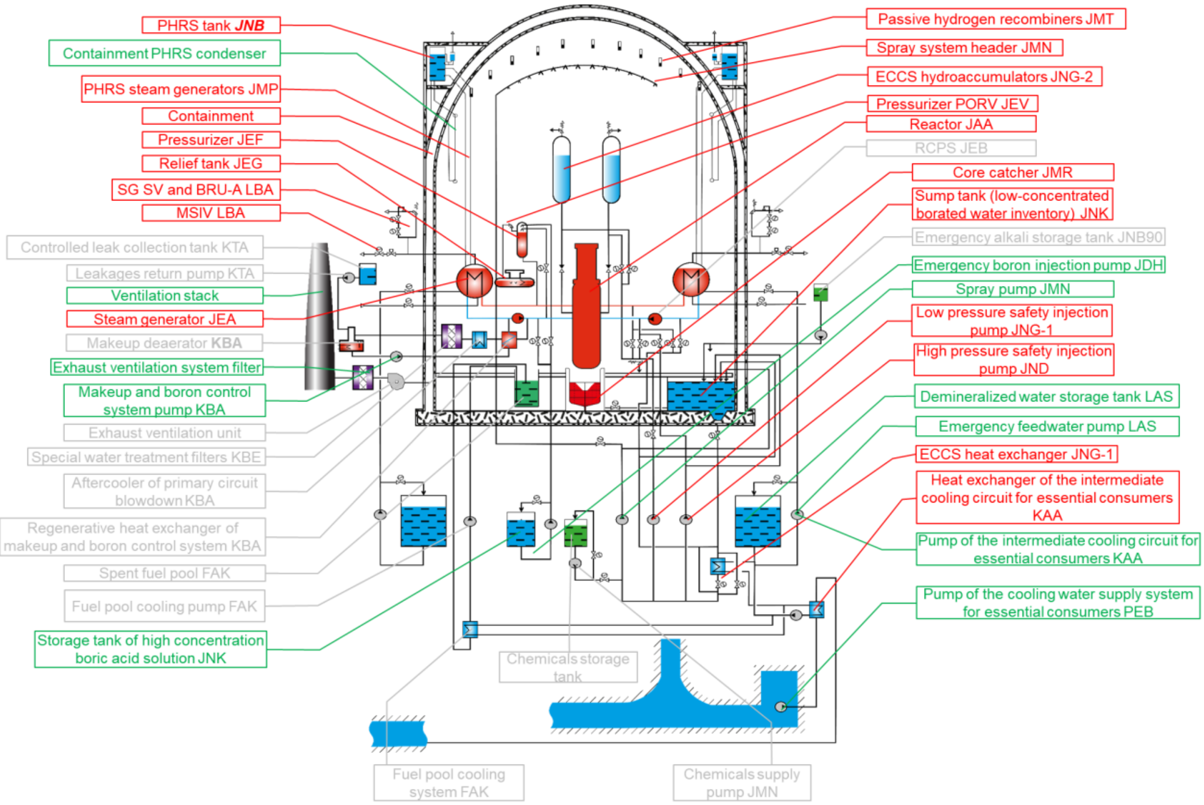

Figure 2. General Diagram of VVER1200 Safety Systems [5].

Figure 2 shows the general safety design of the VVER1200, the systems marked in red are explicitly included in the RASTEP model, the green marked systems are implicitly modelled and the grey marked systems are not included in the RASTEP model of VVER1200 [4].

Also, based on the PSA L2 results of the VVER1200, the likelihood of a large release in the case of an IS-LOCA is found to be so small [4] that this part of the model has been deactivated.

The gap analysis showed that due to the differences in the design and new safety features of the VVER1200, the number of representative accident scenarios and corresponding source terms has been reduced from 22 for the W3 PWR model to 14 for the VVER1200 model. The source terms excluded from the RASTEP model are those related to filtered containment venting, basemat melt through and containment bypass due to IS-LOCA.

3. METHODOLOGY

The new VVER1200 RASTEP model is subjected to a model validation. The results from the BBN model are compared with the data provided from integral plant response codes. The deterministic results for the transients in the VVER1200 reactor were calculated by the integral response code MELCOR.

The uncertainties from the deterministic results are partially accounted for due to the coarse definition of the release time in RASTEP. RASTEP displays the predicted release between time steps (eg. release between 0 – 6 h after the start of the transient) instead of giving an exact time of release. A more detailed analysis of the uncertainties is left for the future.

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 predict the same release category as simulated by the integral deterministic codes, and to predict it ahead of the onset of release time [2].

Requirements for transient scenarios included in the BBN model [2]:

- 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.

3.1 Verification & Validation (V&V) Procedure

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, with MAAP v5.04 for the W3 PWR and with MELCOR for the VVER1200 reactor.

The RASTEP model is based on a static BBN, whereas the evolution of a severe accident is a dynamic process.

The verification procedure follows the steps described in [2] and listed below:

- the process parameters time evolutions (like core exit temperature, or primary pressure trend) from the deterministic simulations 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.
- When an important parameter shows a change in its value or its trend, a new state is created in the BBN model.
- Each time the RASTEP user introduces an observation, there will be a variation in the probabilities of the source terms.

Differently from the V&V procedure performed in [2], the work presented in this paper does not model the information loss during severe accident progression. This would lead to more optimistic and less sensitive results.

3.2 Transients Analyzed

This section reports the accident scenarios analyzed:

- Station Black Out,
- Large Break LOCA,
- Steam Generator Tube Rupture.

For each scenario, the time response of the two reactor types is reported, corresponding to the time discretization. For the VVER1200 reactor, the following conditions are assumed for all scenarios to ensure that the accident conditions are met:

- Main Coolant Pump stop at SCRAM time,
- Main Feedwater (FW), auxiliary feedwater (AFW) and emergency feedwater (EFW) are assumed unavailable at SCRAM time,
- Turbine isolation valve is shut down at SCRAM time,
- BRU-A batteries are assumed to last for 2 h after SCRAM,
- Emergency core cooling systems (ECCS) (HPIS, LPIS) are unavailable,
- Containment Ventilation System unavailable.

For the W3 PWR, five system states are assumed to be known at the start of the simulation, which correspond to the initial conditions of the transient analysis performed by the deterministic calculations [2].

3.2.1 Station Black Out

In this accident scenario, station blackout conditions (loss of offsite power and diesel generators) results in reactor shutdown, with all active safety systems are unavailable. Furthermore, it is assumed the passive residual heat removal system via SG (JNB) is unavailable from the beginning of the transient. The loss of active safety systems and the JNB system leads to core uncovering, heat-up and eventual degradation and relocation of the core material to the lower plenum causing subsequently the Reactor Pressure Vessel (RPV) failure and debris ejection into the core catcher. The passive containment heat removal system (JMP) limit the containment pressure increase and prevents failure of the containment due to overpressure. The final release mode in the RASTEP is Diffuse Leakage.

In the case of the W3 PWR, the Containment Spray System (CSS) is unavailable, but the Filtered Containment Venting System (FCVS) is operational. The FCVS prevents an over pressurization of the containment. The final release category is filtered containment venting.

3.2.2 Large Break LOCA

In this scenario, an unmitigated LB-LOCA is analyzed. The coolant flowing out of the primary system causes an increase in the containment temperature, pressure. The High-Pressure Safety Injection (HPSI), Low Pressure Safety Injection (LPSI), and passive containment heat removal system (JMP) are considered unavailable. The Containment Spray System (CSS) is available only for the W3 PWR case.

The break in the primary system results in rapid depressurization and depletion of water in the Reactor Pressure Vessel (RPV), leading to core uncovering, melt, relocation, and subsequent RPV failure. Pressure in the containment increases due to heat and steam generated by ex-vessel debris until containment failure occurs due to overpressurization. This corresponds to the end state "LOCA, Late Failure No Spray" in the RASTEP model.

3.2.3 Steam Generator Tube Rupture

In this scenario, the accident is initiated by a Steam Generator Tube Rupture (SGTR). Furthermore, it is assumed that the BRU-A valve in the affected steam generator will stick in open position due to water ingress from the primary system creating a release path from the primary system to the environment. The continuous discharge of water eventually depletes the coolant in the primary system uncovering the core. This leads to the eventual core melt and eventual RPV failure. This corresponds to the end state “SGTR, dry release” in the RASTEP model.

4. RESULTS

For each transient the results from RASTEP are presented, with the source term probabilities at each time step. The graphs that are presented only contain the source terms that have a probability >1 %.

4.1. SBO Scenario

By looking at the evolution of the SBO case, as calculated with MELCOR, the transient was divided into the time steps that can be seen in

Table 1. The boundary conditions are put into RASTEP at the first time step (0 min).

Table 1. Transient Evolution for SBO Case Scenario with Observable Parameters for each Time Step.

VVER1200		**W3 PWR**	
Time steps	**Parameters**	**Time steps**	**Parameters**
0 min	- Offsite power and diesel generators unavailable - ECCS unavailable - KAA unavailable - AFW and EFW unavailable - JNB unavailable	0 min	- Offsite power and diesel generators unavailable - ECCS unavailable - Turbine condenser unavailable - AFW system unavailable - Containment spray unavailable
0-10 min	- Primary pressure: slowly decreasing - Containment gamma activity: Normal - Containment pressure: normal - Containment temperature: normal - SG water level: falling - Secondary system gamma activity: normal	15 - 30 min	- Core exit temperature: Normal - Primary water level trend: Decreasing - Primary Pressure initial: Falling or low - SG Water Level initial: Falling - SS Pressure initial: Above normal or raising - Containment Pressure initial: Normal/low - Containment Sump (Empty) - Containment Gamma activity initial: Normal
10 min - 2 h	- Containment pressure trend: Steady - Current primary pressure: High	30 - 60 min	- Primary System Pressure: High - Secondary System Pressure: High
2 h - 5 h	- Containment pressure trend: Increasing - Containment temperature: High	60 - 125 min	- Containment pressure: Increasing - Core Exit Temperature > 600°C
> 5 h	- Containment long term pressure trend: Decreasing - Current primary pressure: Low - Core exit temperature between 600°C and 1200°C	> 125 min	- Core Exit Temperature > 1200 °C - Hydrogen level in the containment > 4 % - Containment long term pressure trend: Steady

The following graphs show the change in the probability according to RASTEP for the different source terms at different time steps.

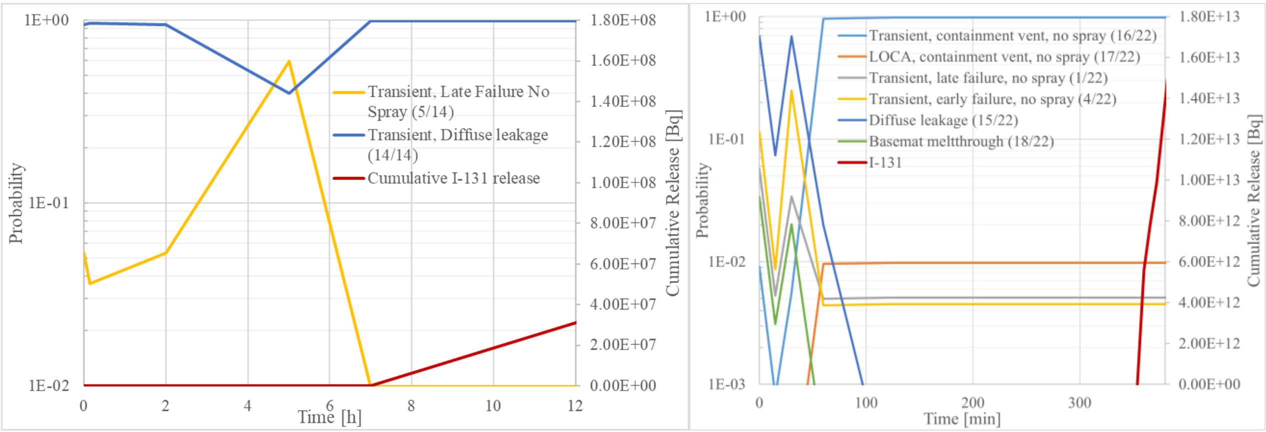

Figure 3. Time Evolution for the Source Terms and Iodine 131 Concentration for the SBO Transient. The results for the VVER model on the left and for the W3 PWR on the right.

The time of release in the SBO case is around 7 h after the beginning of the transient for the VVER1200. The model is able to predict the correct source term, “Transient, Diffuse Leakage”, about 7 h in advance, ignoring the third time step where “Transient, Late Failure No Spray” has a slightly higher probability according to the model.

For the W3 PWR the model predicts the first release as diffuse leakage. The final release mode, “Transient, containment vent, No Spray”, is predicted correctly as the most probable release category around 60 min after the beginning of the transient. The time of release of I-131 happens around 6 h after the beginning of the transient, meaning that the model is able to predict the correct source term around 5.5 h in advance.

4.2. Large Break LOCA Scenario

By looking at the time evolution of the LB-LOCA case, as calculated with MELCOR, the transient was divided into the time steps that can be seen in Table 2. The boundary conditions are put into RASTEP at the first time step (0 min).

Table 2. Transient Evolution for LB-LOCA Case Scenario with Observable Parameters for each Time Step.

VVER1200		**W3 PWR**	
Time steps	**Parameters**	**Time steps**	**Parameters**
0 min	- ECCS unavailable - KAA unavailable - JMP unavailable - Containment spray unavailable	0 min	- ECCS unavailable - Primary system depressurisation unavailable - FCVS unavailable - Turbine condenser unavailable - Containment spray available
0 - 15 min	- Status of external grid: Available - Primary pressure: Low - Pressurizer level: Low - Containment pressure: High - Containment temperature: High - Containment gamma activity: High - Secondary side gamma activity: Normal	0 - 15 min	- Primary system pressure: Low - Pressurizer level: Low - Containment pressure: High - Containment sumps: Empty - Secondary side pressure: Low - Secondary system gamma activity: Normal - SG water level: Falling
15 - 40 min	- Current primary pressure: Low - Containment long term pressure trend: Decreasing	15 - 30 min	- Current secondary side pressure: High - Containment sumps: Full & Hot - SG water level: Increasing - Core exit temperature > 600 °C
40 min - 5 h	- Core exit temperature > 1200 °C	> 30 min	- Core exit temperature > 1200 °C - Containment long-term Pressure trend: Increasing - Containment gamma activity: High - Containment H2 > 4%
> 5 h	- Containment long term pressure trend: Increasing		

The following graphs show the change in the probability according to RASTEP for the different source terms at different time steps.

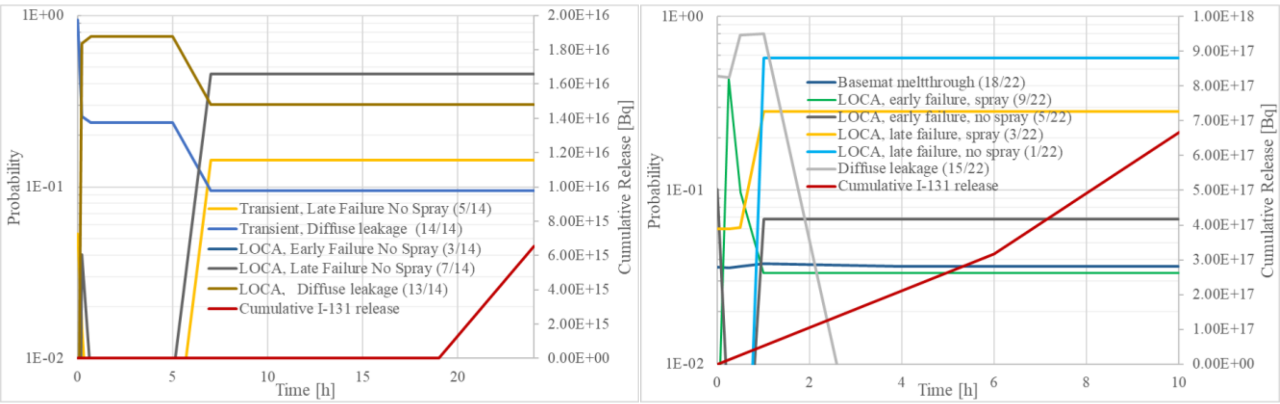

Figure 4. Time Evolution for the Source Terms and Iodine 131 Concentration for the LB-LOCA Transient. The results for the VVER on the left and for the W3 PWR on the right.

In the LB-LOCA case for the VVER1200 model, the time of I-131 release happens about 19 h after the beginning of the transient. The model predicts “LOCA, Diffuse leakage” and “LOCA, Early Failure No Spray” to be the most probable source term in the first five hours following the LOCA after which, it correctly predicts “LOCA, Late Failure No Spray” to be the most probable source term. The model is thus able to predict the correct source term 14 h in advance.

For the W3 PWR, the time of I-131 release happens about 76 h after the beginning of the transient. The model predicts “Diffuse leakage” to be the most probable source term during the first hour following the LOCA.

After this, the model correctly predicts "LOCA, Late Failure" to be the most probable source term, about 75 h in advance.

Note that both designs employ a double-wall containment with an annular space between them. In the MELCOR simulations of the VVER1200, it was assumed that both the inner and outer walls will fail simultaneously, while in the MAAP simulations of the W3 PWR, it was assumed that the inner and outer walls of the containment fail independently of each other. These differences, together with assumptions related to the MELCOR modeling of the core catcher, may have contributed to the differences in the results.

4.3. Steam Generator Tube Rupture Scenario

By looking at the evolution of the SGTR case, as calculated with MELCOR, the transient was divided into the time steps that can be seen in Table 3. The boundary conditions are put into RASTEP at the first time step (0 min).

Table 3. Transient Evolution for SGTR Case Scenario with Observable Parameters for each Time Step.

VVER1200		W3 PWR	
Time steps	**Parameters**	**Time steps**	**Parameters**
0 min	- ECCS unavailable - KAA unavailable - Secondary system pressure relief valves stuck open - AFW and EFW unavailable	0 min	- FCVS not available - Containment sprays available - Ex-vessel cooling unavailable - ECCS unavailable - Turbine condenser unavailable
0 - 10 min	- Primary pressure: falling - SG water level: falling - Containment pressure: normal - Containment temperature: normal - Secondary gamma activity: High - Containment gamma activity: Normal	0 - 30 min	- Pressurizer level: below normal - Secondary circuit pressure trend: Increasing - Primary Pressure Trend: Decreasing - Containment Pressure: Increasing - Secondary gamma activity: High - SG water level: falling
10 min - 5 h	- Containment long term pressure trend: Steady - Current primary pressure: Low	30 min - 1 h	- Primary Pressure: Low - Secondary Pressure: High
5 - 7 h	- Water level in faulted SG: Falling	1 - 2 h	- Primary Pressure: High
7 - 7.5 h	- Core Exit Temperature > 600 °C	2 - 3 h	- Core Exit Temperature > 600 °C - Containment long term pressure trend: Increasing
>7.5 h	- Core exit temp >1200 °C	> 3 h	- Primary Pressure: Low - Core Exit Temperature > 1200 °C

The following graphs show the change in the probability according to RASTEP for the different source terms at different time steps.

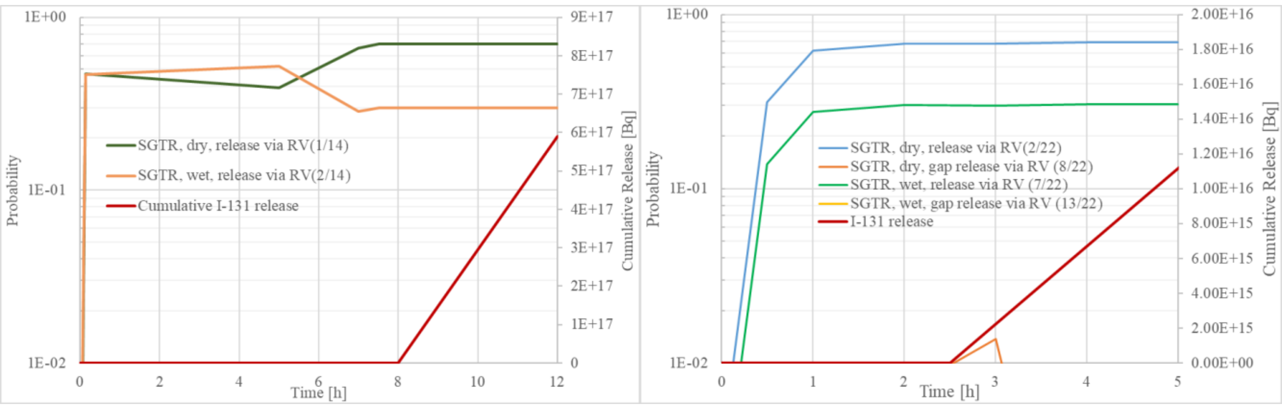

Figure 5. Time Evolution for the Source Terms and Iodine 131 Concentration for the SGTR Transient. The results for the VVER on the left and for the PWR on the right.

For the VVER1200 model, in the first time step the most probable source term is tied between SGTR dry/wet release. After the first time step, RASTEP predicts that dry release is the most probable source term. It continues to be the most probable source term until the time of I-131 release at 8 h after initiating event. Thus, the model is able to predict the correct source term almost 8 h before the release.

The model for the W3 PWR predicts the release category, SGTR dryout gap release, as the most probable release source at 60 minutes after initiating event, predicting correctly release through the secondary system. The time of I-131 release is around 2.5 h after the start of the transient.

The results for the SGTR case that are presented only contain the secondary system source terms. The RASTEP model also predicts source terms for the containment, which has been omitted from the graph.

5. CONCLUSION

For both models presented, RASTEP was able to predict the correct source term as the first or second most probable release category. The model was able to predict the radioactive release ahead of time compared to the actual release predicted by the deterministic calculations. The accuracy of the prediction for RASTEP for transients involving the primary system present source terms only for the containment source terms. Whereas, it has been observed that for the SGTR, the model predicts release categories both for the primary and secondary circuit release categories.

Comparing the two models, we can see from Figures 3, 4 and 5 that, in general, the time of release for the VVER1200 happens much later than in the W3 PWR. This is due to the differences in the design; for instance, in the case of a SBO the containment integrity is preserved in the VVER1200 by the passive containment heat removal system (JMP), such system is not present in the W3 PWR reactor design, which results in filtered venting of the containment.

Concluding, the generic VVER-1200 RASTEP model has been validated with deterministic integral severe accident analyses and is shown to quickly predict the representative source considered in the analysis. Comparing the generic RASTEP VVER-1200 model to the generic W3 PWR model, it is shown the evolution in reactor designs. Where the Gen III+ reactor can withstand the studied transients for several hours longer than the Gen II reactor.

Acknowledgements

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References

[1] Galushin S., Qin W., Yang Y. and Olsson A., Developing the RASTEP Model: A Methodological Approach and Case Study of the M310 PWR Reactor, 17th International Conference on Probabilistic Safety Assessment and Management & Asian Symposium on Risk Assessment and Management (PSAM17&ASRAM2024), Sendai International Center, Sendai, Miyagi, Japan, 7-11 October, 2024.

[2] Canepa J.F., Riber Marklund A., Galushin S. and Olsson A., Evaluation of the source term prediction capability of RASTEP against integral response code calculations. Probabilistic Safety Assessment and Management PSAM 16, Honolulu, Hawaii, June 26-July 1, 2022.

[3] Olofsson F., RASTEP Model Scope: VVER-1200/V491, NUC0500053-R01, 2023

[4] Galushin S., RASTEP Model Report: Generic VVER1200/V491, NUC0500053-R03, Release 2, 2024.

[5] Podoliakin D., Fundamentals of AES-2006 technology, Module 05: VVER (AES-2006) Safety Systems, ROSATOM Service. 2017

[6] Jokiniemi J., Steam Generator Tube Rupture Scenarios, VTT Processes, Aerosol Technology Group, Biologinkuja 7 02044 VTT, Espoo FINLAND