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Simulation of Iodine Spiking phenomenon during Design Extended Condition type A conditions at VVER-1000 and Generic PWR

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ABSTRACT

After the Fukushima Daichi accident in 2011 design extension conditions (DEC) came into focus. Most regulatory regimes now require that design extension conditions are considered in the design of new reactors, while existing reactors should try back fitting safety enhancements to be able to deal with design extension conditions as far as reasonably achievable. DEC are divided in categories A and B, where DEC-A scenarios are characterized by multiple failures of safety systems but with the reactor core remaining intact.

A phenomenon that plays a significant role in DEC-A scenarios involving a containment bypass is Iodine Spiking (IS). In this paper we analyse a PRImary to SEcondary leaking accident (PRISE) with the assumption of a stuck open atmospheric relief valve at the first opening at two different reactor designs (VVER-1000, Generic PWR). For the thermal hydraulic best estimate simulation we use RELAP5-3D, for the transport of iodine we use the Relap5-3D radionuclide transport model. Additionally, to analyse the Iodine Spiking (IS) phenomenon for the examined accident scenario, we created an empirical IS model which is based on the existing NRC IS model to estimate how much iodine is released to the primary system from the fuel.

The results of this study show that the Relap5-3D radionuclide transport model is capable of providing a bounding analysis on the iodine released to the environment. The analysis also shows that good results can be achieved for the analysed scenarios using an empirical model to estimate the iodine released to the primary system. The analysis also shows that during said scenario significant quantities of radioactive fission products are released into the environment.

1. Introduction

Iodine Spiking (IS) is a phenomenon that can occur when fission products enter the coolant due to minor damage to the fuel cladding. A rapid shutdown of a reactor that might be actuated in emergency situations leads to a reduction in temperature and pressure of the reactor coolant system. This changes the aggregate state of the coolant around the fuel rods and releases of iodine increase significantly. This phenomenon is known as an iodine spike (Hózer and Vajda, 2001). Under normal conditions, the radionuclides remain contained in the Primary System (PS) of a Pressurized Water Reactor (PWR) and gradually decay over time. I_{131} for instance has a half-life of about 8 days. Therefore for a long time this phenomenon has been considered to a lesser extent in the licencing safety analyses of Nuclear Power Plants (NPP) and received increased attention only in the 2000 s.

The Fukushima Daichi accident in 2011 highlighted that the consideration of Design Base Accidents (DBAs) is not sufficient in licencing procedures. It became apparent that additionally scenarios involving multiple failures of safety systems, beyond the single failure criterion and repair case assumption, and even scenarios involving core damage should be considered in the design and looked at during safety analysis of a nuclear power plants. Such scenarios are termed design extension conditions A or B respectively (IAEA, 2016).

IS is an important phenomenon at DEC-A accidents as they include primary to secondary leak accidents which constitute a containment bypass. In such accidents iodine released to the primary system can be transported directly to the secondary system and from there into the environment.

According to the NRC Standard Review Plan Section 15.6.3 (US NRC, 2003) for the safety analysis report (SAR) it must be assumed that, in the event of a transient, the release rate of iodine increases by a factor of 500 in comparison to normal operation. An NRC study of Adams and Atwood analysed 168 iodine measurements at 26 plants before and during iodine spiking events (Adams and Atwood, 1991). The average iodine concentration during the steady state is at $1.81 \times$

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10^2 Bq/g (coolant). According to the mentioned approach above, an average iodine spike of 9.15×10^5 Bq/g would be expected. However, in reality the measured IS values were at 2.8×10^4 Bq/g (1.06×10^{13} Bq/h). This shows that the approach in the NRC Standard Review Plan is very conservative. In the literature there are various approaches to calculate IS (deterministic and empirical). Adams and Atwood (1991) built an empirical model based on their findings. Lewis et al. (1997) developed the most prominent deterministic IS model. However, according to a study by Hózer empirical models should be preferred as the uncertainties of deterministic models are very high and of certain processes within the reactor no data can be obtained (Hózer and Vajda, 2001).

In the present study we focus as well on empirical IS modelling. The aim of this work is to further improve the model created by Adams and Atwood for the NRC by considering the current position in the fuel cycle as proxy variable for fuel rod damage (Section 3.3.4). To test our new approach we simulate the IS phenomenon for two different reactor types (VVER-1000 and Generic PWR) under specific DEC-A conditions, where fission products are transferred to the secondary side and further the environment due to a containment bypass using Relap5-3D as thermal hydraulic system code and the Relap5-3D radionuclide transport model.

2. Examined NPP designs

2.1. VVER-1000

The VVER 1000/320 is a Russian light water cooled and moderated reactor. It is rated at a thermal power of 3000 MW and an electrical power of 1000 MW. 31 units of this reactor design are currently in operation (Rosatom, 2001). The primary coolant system (360.9 m^3) consists of a RPV (Reactor Pressure Vessel) with four primary loops, four MCPs (Main Circulation Pumps) and four horizontal SGs (Steam Generator). The four steam lines, each separable with a fast acting main steam isolation valve, each are equipped with atmospheric relief valves (BRU-A) and two safety valves. A high-capacity pressurizer (PRZ) is connected to one loop and contains two safety valves and a relief valve for overpressure protection. The VVER 1000 is operated with enrichments of 4.4%–4.95% and a burnup up to 65 MWd/kgU. In case of emergency, the following safety systems are available for this plant: High Pressure Injection System (HPIS), Low Pressure Injection System (LPIS), passive hydro - ACCumulator (ACCs), containment spray system and emergency feed water system. Three redundant Emergency Diesel Generators (EDG), each 100% sufficient, power the safety systems in case of a loss of off-site power.

2.2. Generic PWR

The Generic PWR is a light water cooled and moderated reactor. It is rated at a thermal power of 3750 MW and an electrical power of 1450 MW. The primary coolant system consists of a RPV (Reactor Pressure Vessel) with four primary loops, four MCPs (Main Circulation Pumps) and four vertical SGs (Steam Generator) with power operated relief valves. A high-capacity pressurizer (PRZ) is connected to one loop and contains one safety valve and a relief valve for overpressure protection. The Generic PWR is operated with enrichments up to 4.3% and a burnup up to 60 MWd/kgU. In case of emergency, the following safety systems are available for this plant: HPIS, LPIS, hydro - accumulators, containment spray system and emergency feed water system. Four EDGs are available. Each EDG can sustain 50% of the reactor's cooling capacity.

3. Methodology

3.1. Transient simulation using RELAP5-3D

RELAP5-3D is developed and maintained at the Idaho National Laboratory (INL) for the United States Department of Energy (US DOE). This code is a successor of RELAP5/MOD3 and is primarily used for the analysis of potential accidents and transients in water-cooled nuclear power plants and for the analysis of advanced reactor systems (Idaho National Laboratory, 2012b).

An Eulerian radionuclide transport model is applied to simulate the transport of radioactive or fertile nuclides in the reactor coolant systems. In connection with the nuclear detector model, this model can be applied to describe the response of the control and safety systems to the existence of radioactive species in the coolant systems. The radionuclide species may be transported by either the liquid or vapour/gas phases. It is possible to create a radioactive specie by either neutron absorption in a fertile specie or by injection into the coolant system using general tables or control variables (Idaho National Laboratory, 2012b). The concentrations of radionuclide species are assumed to be sufficiently dilute that the following assumptions are valid:

- The fluid properties (liquid or vapour/gas) are not affected by the presence of radionuclide substances.
- Energy absorbed by the transporting phase from the decay of radionuclide species is negligible.
- The radionuclide species are well mixed with the transporting phase so that they are transported at the phase velocity (Idaho National Laboratory, 2012a).

For the iodine transport simulation the amount of iodine which is released in the core segment has to be calculated via an IS model (Section 3.3).

3.2. Nodalization of thermohydraulic models for VVER-1000 and gen. PWR

The nodalization for both reactor designs has been obtained from specific supporting documents (D'Auria et al., 2002) and was adapted according to our requirements. An overall view of the nodalization, suitable for the identification of nodes is provided in Figs. 1–4. In Fig. 1 the components 250, 350, 450 and 550 are the positions where coolant of the 4 different loops enters the core segment. After the downcomer (component 130), the coolant is channelled around the fuel rods (component 110) to heat up. This is also the location were I131 is released at the simulation. Eventually the coolant is released into the 4 loops at the components 200, 300, 400 and 500. Furthermore, the figure shows the 4 hydroaccumulator that are capeable of injecting borated water into the core segment (components 50, 60, 70 and 80). Fig. 2 displays one loop of the reactor. PS and SS are shown. The hot leg of the loop starts at component 200. The components 213, 215, 217, 219, 221 and 223 are the heat exchanger that connects PS and SS. The hot header break connects component 224 with component 645 at the SS. The loop is concluded at the end of the cold leg at component 250. At the SS (under normal conditions) the steam is channelled through the main steam line at component 680. The atmoshperic releave valves, where radioactive coolant can reach the environment, as the valve is stuck in open position, is component 690.

The nodalisation for the generic PWR is very detailed, as each loop is modelled seperately. Fig. 4 shows the nodalisation of the loop containing the PRZ. Furthermore, the SG with the defect atmospheric releave valve is visible (component 433). At this location it is possible that I131 can reach the environment during the transient. The reactor core is divided in five sections and the downcomer in five. This allows the simulation of assymetric accidents and transients. The core nodalisation of the generic PWR is depicted in Fig. 3.

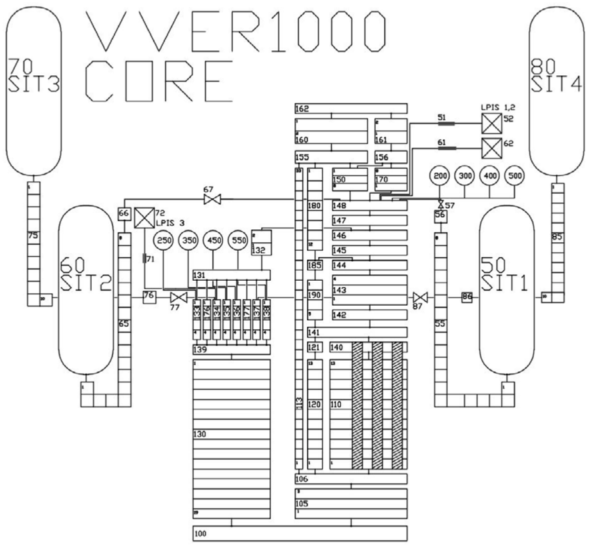

Fig. 1. Nodalization of VVER-1000 core segment (Institute of Safety and Risk Science, BOKU Vienna, 2022a).

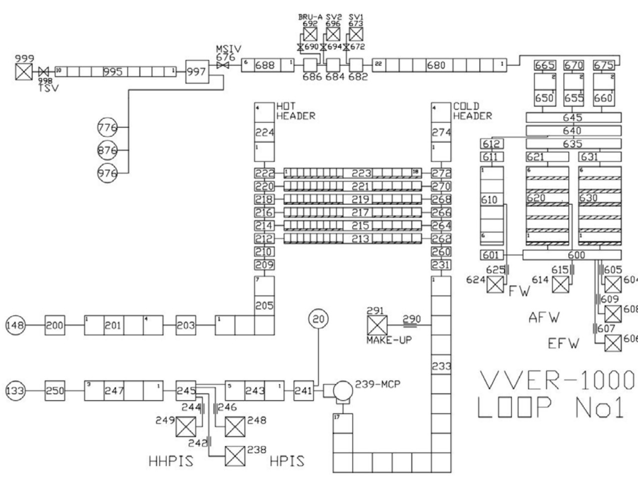

Fig. 2. Nodalization of VVER-1000 loop segment (Institute of Safety and Risk Science, BOKU Vienna, 2022b).

3.3. Iodine spiking model

3.3.1. Iodine spiking phenomenon

It is assumed that due to minor fractures in the fuel rods (FRs), fuel leakage is occurring, especially at older FRs, which further leads to a release and accumulation of fission products in the core during normal operation (Hózer and Vajda, 2001). The reduction of the reactor power results in the decrease of coolant temperature around the fuel and changes the aggregate state of the coolant. This leads to a significant increase of fission product release (Hózer and Vajda, 2001). Due to the depressurization of the primary system, transfer of isotopes from the fuel into the coolant is initiated (Lewis et al., 1990). Furthermore, the decrease in pressure allows the formation of steam near the core components, which can enter the defect fuel and increases the release of fission products due to evaporation (Eickelpasch et al., 1978).

For the transient calculations, it was necessary to be able to estimate the extent of fission product releases during the accident. For this reason a literature research about IS measurement data was conducted and two different empirical IS model were applied.

3.3.2. Data of IS events

In order to obtain sufficient data for the iodine spiking phenomenon, an extensive literature research was conducted. The literature review revealed that there has been very little publicly available data on this topic over the last 30 years. Most data regarding this subject originates from studies prior to 1990. A comprehensive study by the NRC on this topic was published in 1989 which contains 168 entries of iodine spiking events (Adams and Atwood, 1991). A second study that provided several IS entries was conducted by Lewis et al. (2017). In addition, data from smaller studies or reports from individual power plants were included in our dataset (Hózer and Vajda, 2001; Smiesko et al., 2005). Considerably more data was found for PWRs produced by the American manufacturers than from Russian VVER. For VVER 1000 reactors no data was accessible. In addition, it was possible to expand the data set with entries from the US fuel reports from the years 1983 to 1988 in order to record the position in the fuel cycle on the day of the respective event (US NRC, 1984, 1986, 1989).

3.3.3. NRC IS model

The US NRC has developed an IS model in 1989. The data bank used to build the model contained 168 iodine measurements during shut down sequences at 26 American PWRs. Using unrestricted linear modelling it was possible to derive a statistical formula that allows an elementary calculation of the expected iodine concentration during the transient. This approach was selected as the expected IS activity at a reactor with 0 MW power is 0 Bq/h. Therefore, the determination of the intercept is not necessary (Adams and Atwood, 1991). According to this formula, the iodine concentration is only dependent on the power of the reactor at the time the transient began.

$$Activity_{\mathrm{IS}} = 2.63\,E10 * Power + \epsilon \tag{1}$$

$Activity_{\mathrm{IS}}$ = Activity of Iodine Spike [Bq/h]
Power = Electrical Power of the NPP at the time when the transient starts [MW]
ϵ = *Error term*

For the average power mentioned in Adams and Atwood, this would mean an IS of 1.05E+13 Bq/h. However, it should be noted that the approach to establish this formula is rather simplistic as only one independent variable is included in the model. In Table 1 NPPs are displayed were more than one IS was measured. It is visible that for the same reactor (with the same or similar power level) considerably different results in iodine concentration can be obtained. This indicates that additional parameters are needed to thoroughly explain the phenomenon. For this reason we decided to develop a new IS model where additional explanatory variables are included.

3.3.4. BOKU IS model

During the Horizon 2020 R2CA - project, we improved the NRC IS model by introducing a second explanatory variable. In comparison to the NRC model not only the power is considered as explaining variable but also the current position (amount of days) of the fuel cycle. The current position in the fuel cycle is an important indicator as an IS only can take place if there are small breaks at the fuel rods. Those defects develop over time. Therefore, it can be assumed that if the reactor is further in the fuel cycle it is more likely to have defects at the fuel rods. Data regarding the fuel cycle of the different reactors was collected from the US nuclear fuel annual reports (US NRC, 1984, 1986, 1989). By using unrestricted linear modelling we derived the following formula:

$$Activity_{\mathrm{IS}} = 1.06\,E10 \; * P_{\mathrm{electric}} + 0.750 * \text{time in fuel cycle} + \epsilon \tag{2}$$

$Activity_{\mathrm{IS}}$ = Activity of Iodine Spike [Bq]

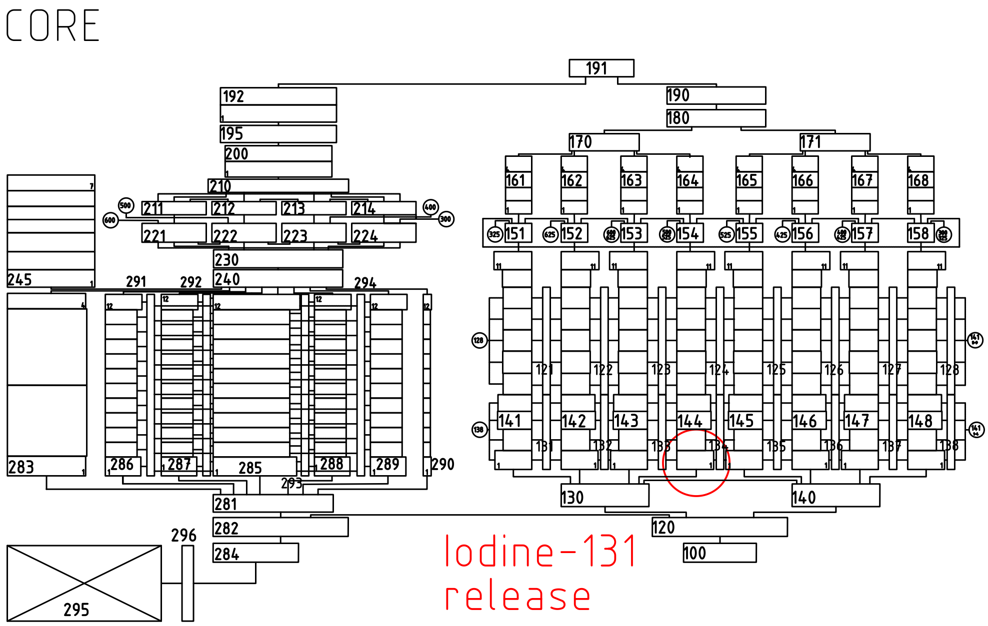

Fig. 3. Nodalization of PWR core segment (Institute of Safety and Risk Science, BOKU Vienna, 2023a).

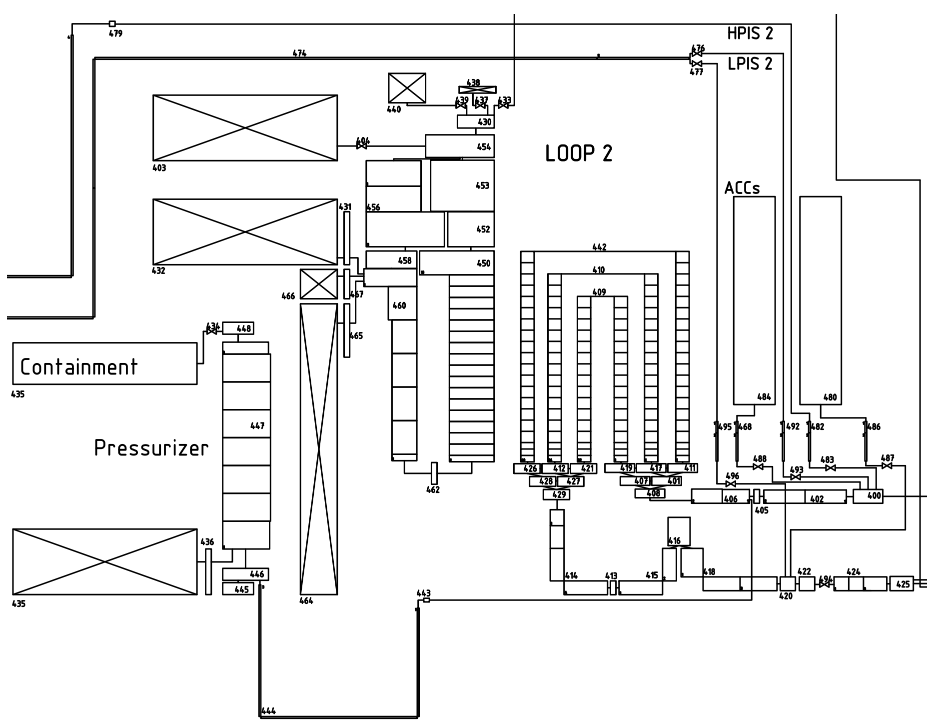

Fig. 4. Nodalization of PWR loop segment (Institute of Safety and Risk Science, BOKU Vienna, 2023b).

Table 1
Selection of NPP with multiple data entries.

NPP	Max		Min		Mean	StDev
	Activity (Bq/h)	Power (%)	Activity (Bq/h)	Power (%)	Activity (Bq/h)	Activity (Bq/h)
ANO-1	2.05E+14	100	4.74E+13	100	1.19E+14	7.96E+13
ANO-2	1.28E+13	100	5.45E+10	100	5.37E+12	3.92E+12
CalClf-1	6.14E+12	100	1.11E+10	92	3.12E+12	2.82E+12
Cook-1	7.77E+12	90	1.24E+10	90	3.36E+12	3.60E+12

Table 2
Steady state calculation.

Parameter	Units	VVER-1000	Error	Generic PWR	Error
Core thermal power	MWth	3 000.00	±5.00	3 750.00	±5.00
Pressure in the pressurizer	Bar	157.00	±0.01	155.50	±0.10
Pressure in the steam generators	Bar	62.70	±0.30	64.50	±0.50
Inlet temperature in the core	K	563.15	±1.00	563.15	±2.00
Outlet temperature in the core	K	593.15	±1.00	565.00	±2.00
Primary loop mass flow rate	kg/s	4 530.00	±1.00	4 845.00	±1.00
Primary inventory	kg	240 800.00	±0.01	240 050.00	±0.20
SG liquid mass inventory	kg	158 800.00	±0.01	222 008.00	±0.10
Feedwater mass flow rate	kg/s	1 632.00	±4.00	510.80	±5.00
Feedwater temperature	K	493.15	±0.50	491.15	±0.50
Main steam line temperature	K	550.00	±1.00	550.00	±1.00
Pressurizer level	m	8.45	±0.10	8.00	±0.30

P_{electric} = Electrical Power [MW]
Time in fuel cycle = days in fuel cycle [days]
$\epsilon = Error\ term$

The improvement of this model is that it allows a wider range of analysis as it is now possible to conduct several calculations at different points of time in the fuel cycle. Therefore, it is possible to make a more accurate prediction of the severity of an accident where iodine reaches the environment. During the R2CA project we conducted our simulation at three different positions in the fuel cycle. Those were 180 days, 365 days and 545 days. 545 days is the most conservative assumption, therefore it was applied in this research.

4. Results

4.1. Examined transient scenarios

Two scenarios were analysed. For the VVER-1000 a hot header break of 100 mm equivalent diameter was simulated in loop number 4. For the Generic PWR a Steam Generator Tube Rupture (SGTR) of a single U-tube at loop number 4 was assumed. For both reactor designs it is supposed that the atmospheric relief valve, which connects the main steam line with the atmosphere in the affected loop is stuck in an open position after the first opening. This leads to a containment by-pass scenario. Regarding the safety systems it is anticipated that the LPIS is not available, however the HPIS and the ACCs are active. Assumed operator actions are the depressurization of the secondary side (60 K/h) via the atmospheric relieve valves in the intact loops, the deactivation of the HPIS after 1800 s/2700 s, the disconnection of the ACCs after 1800 s and the activation of the Make-Up system (only at VVER-1000) after 2700 s to limit the loss of coolant through the break. The complete configurations of the safety systems are described in the framework of the Horizon 2020 project R2CA (Reduction of Radiological Consequences of design basis and extension Accidents) (Zimmerl et al., 2021).

4.2. Results of steady-state simulation with Relap5-3D

The main thermohydraulic parameters of the steady state simulation with Relap5-3D are depicted in Table 2.

4.3. VVER-1000 transient analysis

4.3.1. Main events

This simulation consists of a hot header break (PRISE) in loop No 4. The breaksize is assumed with 1.4% of the surface area, which is equivalent to a diameter of 100 mm. The atmospheric relief valve (BRU-A) in the affected loop is assumed to be stuck in open position after the first opening. Secondary side cooling via BRU-A valves is used for the depressurization of the PS. The transient is divided into the following phases:

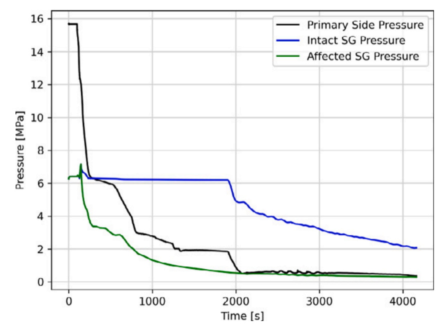

Fig. 5. Pressure on PS/SS side.

1. The simulation starts with the opening of the break at the hot header of loop 4, which opens up the connection between the PS and SS. The following pressure decrease in the PS leads to an increase of the PRZ heater power. After the PRZ level falls below 4.2 m, the PRZ heaters are switched off. The SCRAM signal is given when the UP pressure reaches 13.7 MPa. As a result, the simulation of the chain reaction is terminated and the decay power is modelled according to ANS-79-1 standard. The transition to the AFW is conducted and the MCPs are switched off due to the saturation margin signal in the coolant. The HPIS is activated when the PS – Pressure falls below 11 MPa. The BRU-A valve is opened after the SG pressure increases to 7.16 MPa
2. Set point for closure of the BRU-A valves in SGs is reached (6.28 MPa), but due to mechanical failure the BRU-A valve in loop 4 is stuck open. The BRU-A valves in the intact loops close properly. Full closure of MSIV of the affected loop occurs. SG 4 is full of water and PRZ is completely empty.
3. Hydro - accumulators are activated at the set point (PS pressure lower than 6 MPa).
4. At 1800 s, the AM measurements of the operator are started by initiating the secondary side depressurization system of SG 1–3 via the BRU-A valves. Two (of the three) HPIS and all ACCs are deactivated.
5. After 2700 s, the operator activates the make-up system and the last functioning HPIS is deactivated.
6. At 4700 s, the PS pressure falls below 0.4 MPa and the simulation is stopped.

The chronology of the main events of the transient calculation is given in Table 3. Additionally Figs. 5–8 show the development of main parameters during the transient.

4.4. Generic PWR transient analysis

4.4.1. Main events

The simulation consists of a Steam Generator Tube Rupture (SGTR) in loop No 4. The atmospheric releave valve in the affected loop is

Table 3
Main events of the transient.

#	Event	Set points	Time
1	Break opening	Time (Operator Action)	0
2	Start of core power reduction		1
3	Scram of reactor	UP $P < 13.7$ MPa	25
4	Switching of PRZ heaters	PRZ Level < 4.2 m	25
5	Transition FW/AFW	P in main steam line > 4.1 MPa	26
6	Turbine Valve closure	Scram + 10 s	35
7	Main steam isolation valve closure begins	Turbine Valve closure + 5 s	42
8	Start of MCP-4 coast-down	Closure of MSIV + 5 s	47
9	BRU-A valve opening in SG – 4	P in SG < 7.16 MPa	48
10	Main steam discharge valve in loop 4 closed		50
11	Start of MCP-1,2,3 coast down	Tsatt – coolant T < 10.0 °C	115
12	Start of HPIS injection in CLs:	P in PS < 11.0 MPa	115
13	Coolant reaching saturation temp. in HL at SG inlet: L1 L2 L3 L4		 150 150 150 2100
14	Pressure in the PS at 5.9 MPa		435
15	Pressure in the primary side is lower than in the secondary one (L1–L3)		185
16	Start of ACC operation:	P in PS < 6.0 MPa	420
17	Start of Cool down procedure by using the SSCS via BRU-A valves	Time (Operator Action)	1800
18	Termination of ACC water supply in PS:	Time (Operator Action)	1800
19	Termination of HPIS water supply in: - CL-1 - CL-2 - CL-3 - CL-4	Time (Operator Action)	 1800 – 2725 1800
20	Start of operation of PS make-up system:	Time (Operator Action)	2725
21	Stop of simulation:		4700

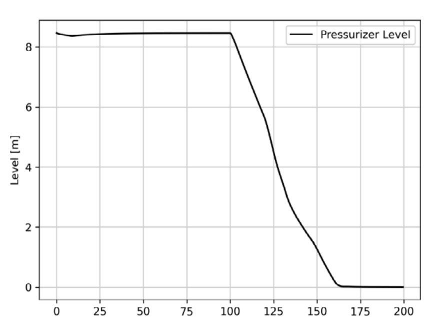

Fig. 6. Pressurizer level.

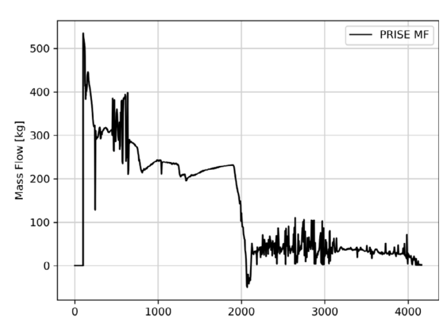

Fig. 7. Breakflow to secondary side.

assumed to be stuck in open position after the first opening. Secondary side cooling via atmospheric releave valves is used for the depressurization of the PS. The transient is divided into the following phases:

1. The initiating event of this transient simulation is a double ended break of one SGT in loop No. 4 with an equivalent break area of 600 mm^2 (2 times the tube area). This allows coolant of the PS to reach the SS (PRISE). The opening of the valve leads to a pressure decrease at the PS and a pressure increase in the affected SG (all the other 3 SGs are not affected). Consequently, the water level of the PRZ declines and the water level of SG 4 rises. Reactor Scram is initiated because the PS pressure reaches the critical value of 132.5 bar. As a result, the simulation of the chain reaction is terminated, and the decay power is modelled according to ANS-79-1 standard. As a reaction to the Scram the turbine valve and the main steam isolation valves (MSIV) are closed. The closure of the MSIV is the signal for the deactivation of the MCPs as well. Due to the pressure reduction at the PS below 11.0 MPa, the available HPIS are activated.
2. The pressure in SG 4 reaches the critical value of 82.9 bar, which results in the opening of the SG relief valves. To maximize the

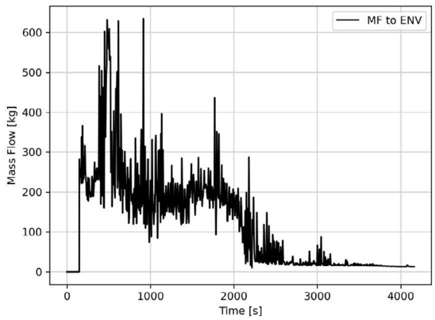

Fig. 8. MF of coolant to environment.

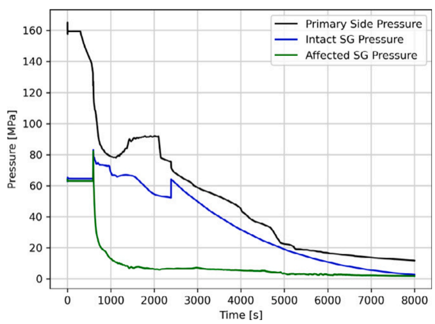

Fig. 9. Pressure on PS/SS side.

Table 4
Main events of the transient.

#	Event	Set Points	Time (s)
0	Steady State		0–300
1	Break opening	Time (Operator Action)	300
2	Start of core power reduction		301
3	Scram of reactor	UP $P < 13.7$ MPa	589
4	Relief valve in loop 4 opens and stucks open	P in SG4 > 8.25 MPa	591
5	HPIS activated	P in PS < 11.00 MPa	716
6	Secondary depressurization starts	Time (operator action)	2100
7	HPIS deactivation	Time (operator action)	2100
8	Start of ACC water supply in PS:	P in PS < 31.00 MPa	2100
9	Stop of simulation:		8000

discharge of contaminated fluid particles it is assumed that the valve in SG4 is stuck open and does not close again. Consequently, the pressure and water level of SG 4 declines. The RV of the unaffected loops open as well, but close again, after the pressure in the SGs falls below the threshold value.

3. At 2100 s the controlled cooldown of the SS is started by the operator, which gradually decreases the thermal energy output of the reactor to 20% of its initial thermal power.
   The chronology of the main events of the transient calculation is given in Table 4. Additionally Figs. 9–12 show the development of main parameters during the transient.

4.5. Results of iodine spiking simulation

During this analysis two IS models were applied, the NRC IS model and the BOKU IS model. For the BOKU model we assumed conservatively that the reactor is already running for 1.5 years as the possibility of cracks in the fuel cladding is higher with extended operation time. The models were used to calculate the iodine concentration which is released at the core during the IS sequence (Table 5). This iodine concentration is necessary as input for the RELAP5-3D fission product transport model.

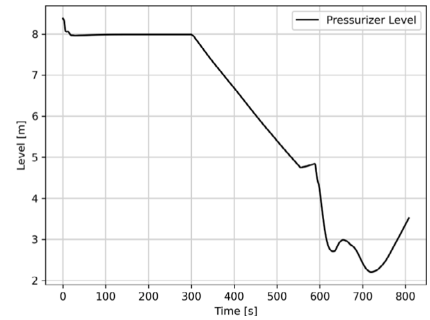

Fig. 10. Pressurizer level.

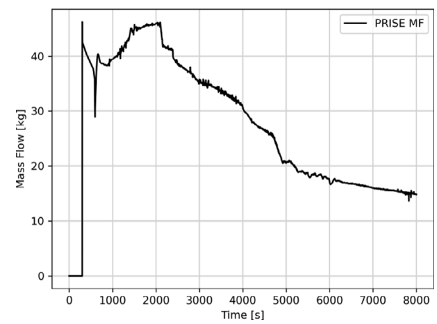

Fig. 11. Breakflow to secondary side.

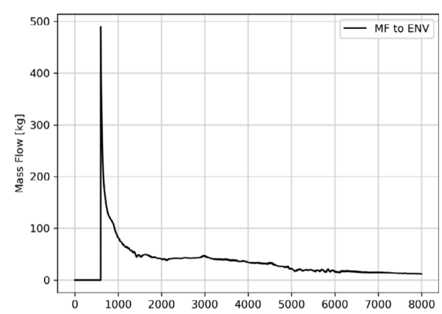

Fig. 12. Coolant MF to environment.

Table 5
Input values for I131 transport model.

	VVER-1000	Generic PWR
NRC IS model [Bq]	2.627E+14	3.809E+14
BOKU IS model [Bq]	2.497E+14	2.972E+14

The results of the iodine transport simulation are depicted for both IS models and both reactor designs in Figs. 13 and 14. Since only the iodine input is changed, but all other thermohydraulic parameters remain the same, only the magnitude of the models' results varies at the same NPP design.

The iodine concentration reaching the environment is greater in the VVER design because the rupture size between PS and SS is significantly larger and therefore more iodine is transported to the SS and into the environment. The ratio between estimated IS at BOKU and NRC model differs for both reactors, due to the inclusion of the second independent variable in the BOKU model. Therefore, the calculated I131 activity at the VVER-1000 is closer between NRC and BOKU model, than for the PWR.

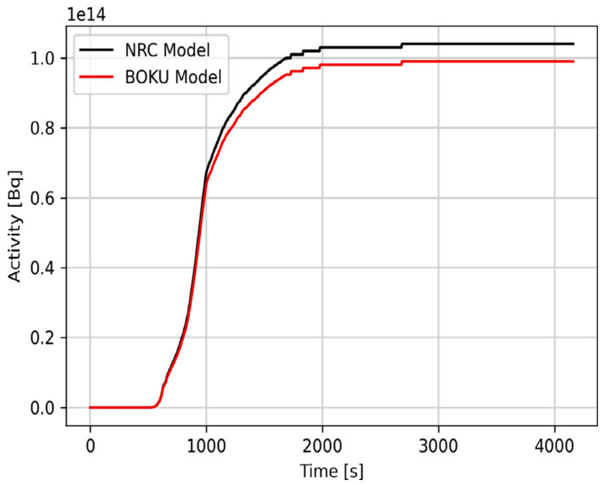

Fig. 13. VVER-1000 I131 at environment.

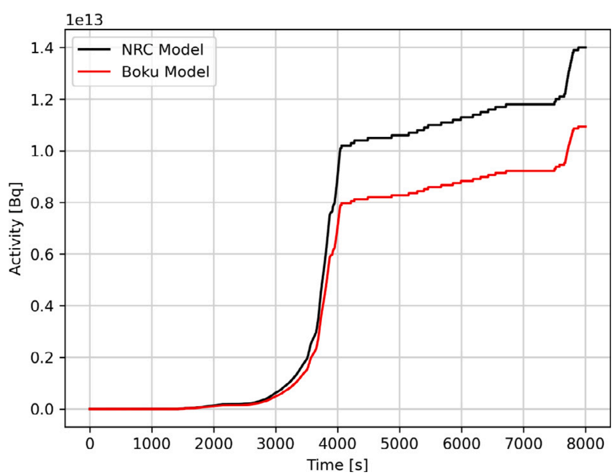

Fig. 14. PWR I131 at environment.

5. Conclusion and discussion

The thermal hydraulic simulation showed that for the examined accident scenario at both reactor designs the installed safety systems and the anticipated operator actions are sufficient to stabilize the reactor within a reasonable time frame. For both reactors the core was never threatened to run dry. After about 2000 s it was possible to align the primary side and secondary side pressure and therefore limit the breakflow, which transports the radioactive fission products to the secondary side and further the environment, to a minimum.

During the course of the analysis, it was possible to extend the existing NRC IS model by another explanatory variable. Our new model considers the time of the accident in the fuel cycle. This enhances the model's ability to more realistically determine negative impact on the environment, which is caused by the accident. If we assume that the analysed NPP is at the end of the fuel cycle our model provides similar results as the NRC model. This is especially true for the simulation of the VVER-1000 reactor. This demonstrates that our model produces realistic results. However, this circumstance can also be partly explained as we used a similar data set to estimate the model parameters.

Unfortunately, since the NRC model was created in 1989, only few new measurements have been published on this topic. Especially for the VVER design there is little publicly available literature. For the future, it would therefore be very interesting to estimate the model again using new measurement data.

Our simulations demonstrated that under the selected parameters (PRISE plus containment by-pass), radioactive iodine is released into the environment. In the VVER scenario, considerably more I131 is released into the environment than at the PWR, which is due to the increased breaksize. This shows that even within the field of DEC-A, impacts of varying severity on the environment can be observed. At the VVER, approximately 30% of the total IS was transported into the environment within the simulation time of 4700 s. The calculated iodine activity is in line with the common knowledge in the literature and confirms that for the licencing of NPP, the behaviour under DEC-A conditions must also be reviewed.

Regarding the application of empirical models to determine the IS phenomenon, it must be said that these are assumed to be subject to a certain degree of uncertainty. The uncertainty is caused by the implementation of mathematical models, which can only reflect reality to a certain extent and have to resort to simplifications in particular cases. Therefore, it is planned to analyse the uncertainty of the iodine transport in future research.

Declaration of competing interest

The authors declare the following financial interests/personal relationships which may be considered as potential competing interests: Nikolaus Muellner reports financial support was provided by EU Framework Programme for Research and Innovation Euratom. If there are other authors, they declare that they have no known competing financial interests or personal relationships that could have appeared to influence the work reported in this paper.

Data availability

The authors do not have permission to share data.

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