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Uncertainty calculation of thermal hydraulic simulation of DEC-A accident scenario at VVER-1000 NPP using CIAU method

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ABSTRACT

All thermal hydraulic best estimate simulations of transient scenarios at Nuclear Power Plants (NPPs) are subject to uncertainties. One method to evaluate those uncertainties is implemented by the "Code with the capability of Internal Assessment of Uncertainty" (CIAU-Code), which calculates the uncertainty for the parameters primary system mass, primary system pressure and hot rod temperature by accuracy extrapolation.

The present work now presents an extended use of the CIAU method, by deriving the uncertainty of the parameter iodine-131 release to the environment. For the analysis, a steam generator hot header break with the assumption of a stuck open safety relief valve (BRU-A) at the first opening at the VVER-1000 reactor was selected. Such transient constitutes a design extension condition A (DEC-A), characterized by multiple failures of safety systems but with the reactor core remaining intact. The thermal hydraulic best estimate simulation was conducted with RELAP5-3D. Additionally, the Iodine Spiking (IS) phenomenon for the examined accident scenario was simulated using two different empirical IS models (NRC IS Model and BOKU University IS Model). To allow the application of the CIAU method for the IS calculation a theoretical derivation is provided.

The results show that for the selected DEC-A scenario a high uncertainty range has to be expected. The analysis allows a conservative estimation of the risk expected in the event of the examined transient sequence.

1. Introduction

Thermal hydraulic simulations of transients at Nuclear Power Plants (NPPs) are always subject to uncertainties (IAEA, 2008). The uncertainty has many causes, e.g. the implementation of mathematical models, which can only reflect reality to a certain extent and have to resort to simplifications in particular cases. For this reason an uncertainty analysis has to be conducted, whenever a best estimate method is applied for licensing purposes of NPP (D'Auria, 2019). This approach is widely known as Best Estimate Plus Uncertainty (BEPU). To determine the uncertainty of thermal hydraulic transient simulations, several methods have been developed including the GRS method (Glaeser, 2008), the "Uncertainty Methodology based on Accuracy Extrapolation" UMAE (D'Auria et al., 1995) and the CIAU (Code with capability of Internal Assessment of Uncertainty) of the University of Pisa. The methods differ fundamentally in their approach. For the GRS method a large number of thermal hydraulic simulation-runs are performed in which certain input parameters are randomly changed. The number of needed runs depends on the desired confidence and is determined by Wilks Formula (Wilks (1942) and Lee et al. (2014) for an example of an application). The result is a spectrum of possible outcomes for the analysed scenario. In contrast, the CIAU method requires only one (best estimate) transient simulation and afterwards, the results are compared with the weighted average of a set of experimental data of NPPs during transient scenarios. Therefore, in the analysis only combinations of parameters are considered that have actually already occurred in reality. For this study it was decided to apply the CIAU approach which has been used extensively in the literature (Petruzzi and D'auria, 2005; Leung et al., 2009; Del Nevo et al., 2007).

The Fukushima Daichi accident in 2011 highlighted that the consideration of Design Base Accidents (DBAs) is not sufficient in licensing procedures. The additional analysis of Design Extension Conditions (DEC) is very important, see IAEA (2016), Requirement 20. It is essential to consider the behaviour of a reactor in a DEC situation in order to develop strategies for successful accident mitigation.

This work is conducted within the Euratom project "Reduction of Radiological Consequences of design basis and design extension Accidents" (R2CA). R2CA aims at the assessments of radiological consequences (RC) of design basis accidents (DBA) and design extension conditions (DEC-A) reactor accidental situations.

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A phenomenon that plays a significant role in DEC-A scenarios (i.e. DEC scenarios that are beyond DBA but do not involve core damage) is Iodine Spiking (IS). The present analysis considers a SG hot header break with multiple failures at a VVER-1000 reactor. The thermohydraulics of such an event has been simulated several times before (Berezhnyi et al., 2023; Andreeva et al., 2015). As the examined transient scenario contains a containment bypass, radioactive iodine is transported directly into the environment. Therefore, it is highly relevant to consider fission product transport in the analysis. In the literature there are various approaches to calculate IS (physical and empirical).

According to a study by Hózer and Vajda (2001) 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.

For this reason we decided to use empirical IS models within our analysis. However, it is undisputed that due to the complexity of the phenomenon, empirical IS models are subject to high uncertainties as well. Until now the uncertainties of IS models were not addressed in the literature. Therefore, the aim of this work is to determine the uncertainties stemming from the thermal-hydraulic calculation of a DEC-A accident with containment bypass and to derive the expected uncertainty regarding the transported iodine quantity to the environment. We derive the primary pressure, primary mass and hot rod temperature uncertainty using CIAU methodology and, via the Gaussian law of error propagation, we use those quantities to derive the uncertainty of the iodine release to the environment.

2. Examined NPP design - 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) with atmospheric relief valves (BRU_A). A high-capacity pressurizer (PRZ) is connected to one loop and contains two safety valves and a relief valve for overpressure protection. For some VVER 1000 units it is possible to operate with enrichments between 4.4%–4.95% and a burnup up to 65 MWd/kgU. In case of emergency, the following safety systems are available: High Pressure Injection System 3 × 100% (HPIS), Low Pressure Injection System 3 × 100% (LPIS), passive hydro - ACCumulator 4 × 33% (ACCs), Emergency Feedwater System 3 × 100% (EFW) and Containment Spray System. One operational system that might be used to stop accident progression in DEC scenarios is the auxiliary feedwater system. Furthermore, one aim of this analysis is to examine the capability of the make-up system to contribute to stopping accident progression in DEC-A scenarios.

3. Methodology

The following section presents the methods that were used to obtain the results. The thermal hydraulic system code and the nodalisation approach are introduced, followed by the model used to model the iodine released to the primary system and finally the method to derive the uncertainty.

3.1. Transient simulation

The analysis was performed using RELAP5-3D which 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 (INL, 2012). The VVER 1000 NPP nodalization is based on a nodalisation that was developed at the University of Pisa, see e.g. Melikhov et al. (2006) and was adapted according to our requirements. An overall view of the nodalization, suitable for the identification of nodes is provided in Figs. 1 and 2.

3.2. Iodine Spiking model

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 further leads to the fragmentation of the Uraniumdioxide fuel pellets. 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 two different empirical IS model were applied for this study. First, 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 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.

$$2.63\,E10\ \mathrm{Bq/h} * \mathrm{MW}(e) = Activity_{\mathrm{IS}} \tag{1}$$

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

For the average power mentioned in Adams and Atwood, this would mean an IS of 1.05 E13 Bq/h.

However, for this work 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:

$$0.286\mathrm{Ci/h} * P_{\mathrm{electric}} + \text{time in fuel cycle} * 0.750 = Activity_{\mathrm{IS}} \tag{2}$$

$Activity_{\mathrm{IS}}$ = Activity of Iodine Spike [Ci]
P_{electric} = Electrical Power [MW]
Time in fuel cycle = days in fuel cycle [days]

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.

3.3. Uncertainty calculation of thermal hydraulic simulations using the CIAU method

The CIAU method was developed by the University of Pisa. It provides an uncertainty calculation for specific thermal hydraulic parameters at transient simulations. The approach requires only one (best estimate) transient simulation and afterwards the results are compared

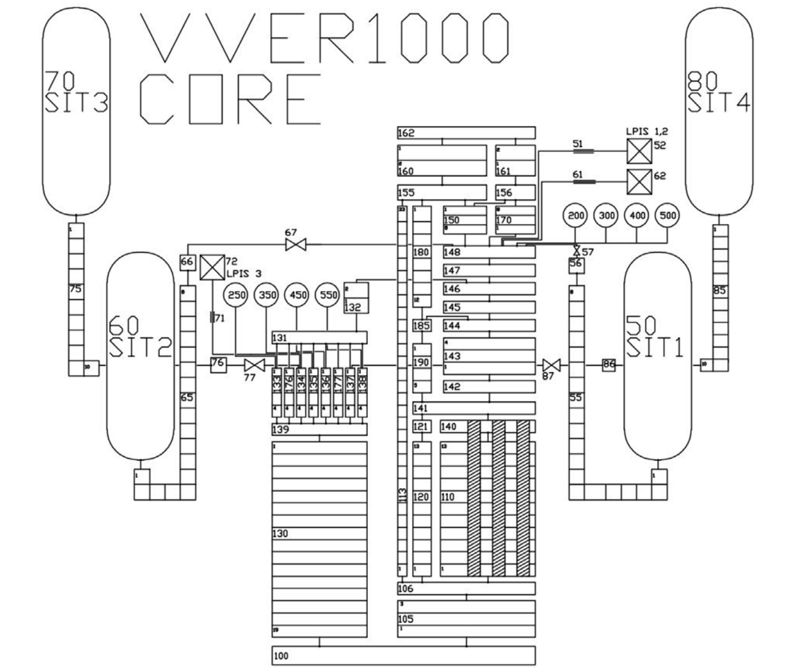

Fig. 1. Nodalization of VVER-1000 core segment and ACCs (Institute of Safety and Risk Sciences, BOKU Vienna, 2015).

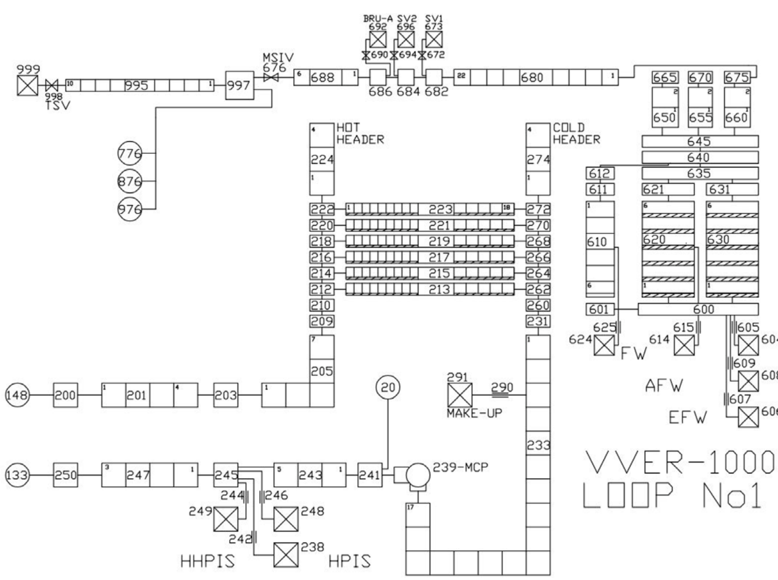

Fig. 2. Nodalization of VVER-1000 loop segment (Institute of Safety ans Risk Science, BOKU Vienna, 0000b).

with the weighted average of a set of experimental data of NPPs during transient scenarios. The main ideas of the methodology are Petruzzi and D'auria (2005):

1. Any transient scenario assumed in the reference systems can be characterized by the time and by a limited number of variables. The boundaries of variation for those variables and the time are identified.
2. The ranges of variation for those variables and the transient time are subdivided into intervals. Hypercubes result from the combination of variables intervals.
3. The NPP status is formed by the combination of one hypercube and one time interval.
4. It is assumed that uncertainty can be associated to any NPP status.

The standard CIAU procedure assesses the uncertainty of the following 3 parameters:

- Upper plenum pressure
- Primary side mass
- Cladding Temperature at 60% core height

In principle, any parameter could be included in the CIAU method if there are sufficient experimental data in the developed databank. However, for iodine transport this is not the case as at the time of publication of this paper not enough studies have been published, that address this parameter. Therefore, it was considered how the uncertainty of the iodine concentration could be derived from the basic parameters of the CIAU method. In our scenario the iodine transport to the environment occurs via a containment bypass over the secondary

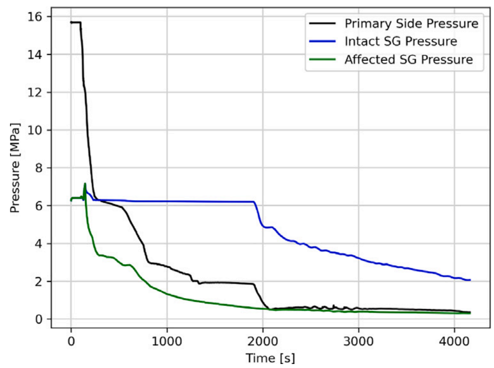

Fig. 3. Pressure on PS/SS side.

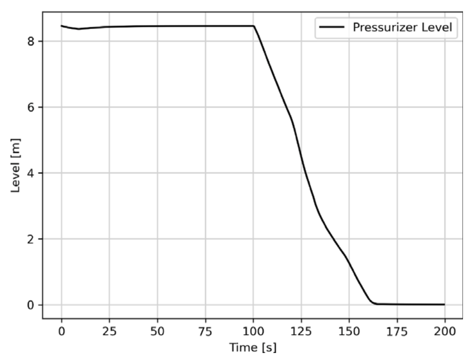

Fig. 4. Pressurizer level.

side, making the pressure difference between the steam generator and the environment majorly responsible for the transport of radioactive iodine to the environment. In two-phase systems where gas and liquid coexist, the transport of iodine is also heavily influenced by the void fraction (defined as gas volume fraction to water volume) at the point of release to the environment. Therefore, it would be reasonable to consider the uncertainty of the secondary side pressure and the uncertainty of void fraction at the relief valve as the main drivers for the uncertainty of the iodine transfer into the environment. An evaluation of the RELAP5-3D results confirmed these considerations.

For simulating the IS uncertainty, we conducted a propagation of uncertainty from secondary pressure uncertainty and void fraction uncertainty. As the uncertainty of the primary side pressure is the main driver for the uncertainty of the pressure at the affected steam generator, the secondary side pressure uncertainty was assumed to be equivalent to the primary pressure uncertainty calculated in the CIAU analysis. An educated estimate of 2.5% was made for the void fraction uncertainty, based on research on void fraction uncertainty in the reactor core. For both variables, delta values of 0.01 were chosen. Delta values indicate how much iodine transport is affected by errors, and are the components of the partial derivatives in the uncertainty propagation formula. Lastly, the iodine transport at the uncertainty bands was necessary. Using RELAP5-3D, the mass flow at the relief valve was simulated with the uncertainty bands as base conditions. The mass flow was then related to the iodine transport.

4. Results

4.1. Transient analysis

As initiating event a hot header break is assumed, this means a leakage (1.4% - equivalent diameter of 100 mm) from the PRImary to the SEcondary side (PRISE) of the reactor occurs in loop No. 4. It is assumed

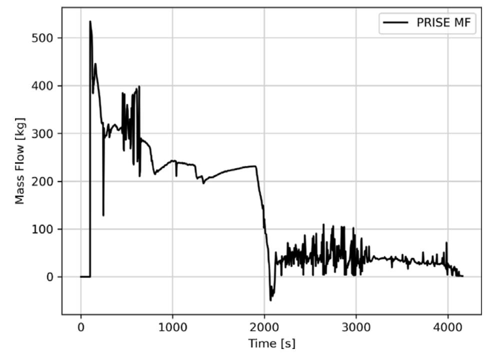

Fig. 5. Breakflow to secondary side.

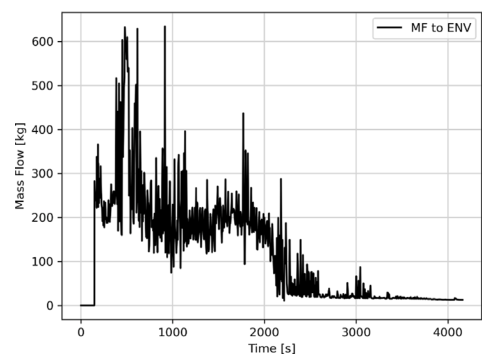

Fig. 6. MF of coolant to environment.

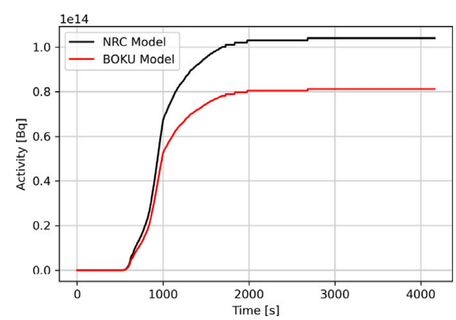

Fig. 7. I131 Pathway to environment - Model comparison.

that the BRU-K valves which should open after the secondary side pressure reaches 6.8 MPa, fail due to loosing the vacuum in the condenser. Furthermore, it is supposed that the atmospheric relief valve (BRU-A 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 assumed that the LPIS is not available, however the HPIS and the ACCs are active. Assumed operator actions are the depressurization of the primary side (60 K/h) via the BRU-A valves in the intact loops. This measurement reduces the pressure difference between PS and SS and therefore the breakflow. Additionally, the deactivation of the HPIS after 1800s/2700s, the disconnection of the ACCs after 1800s and the activation of the Make-Up system after 2700s are conducted 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). The initial conditions of the NPP are included in Table 1:

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

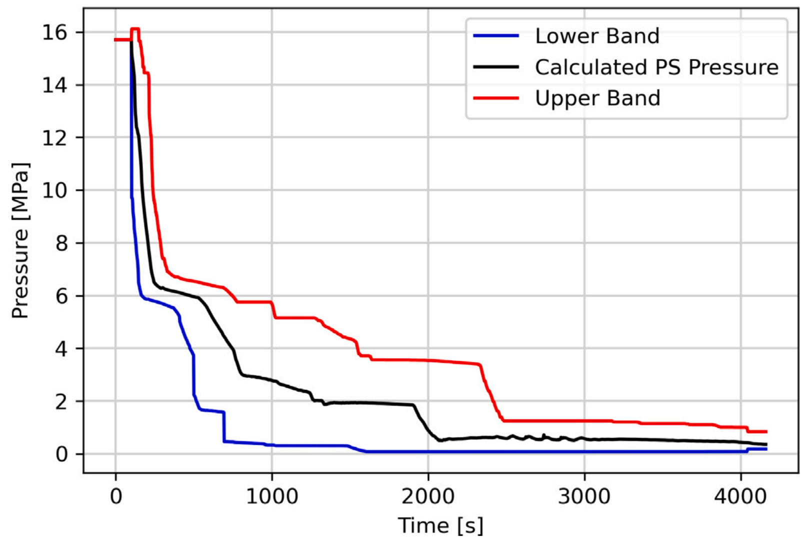

Fig. 8. Uncertainty calculation of PS - Pressure.

Table 1
Initial conditions of VVER-1000.

Parameter	Units	VVER-1000
Pressure in UP	MPa	15.6
Coolant T at UP outlet	°C	309.0
Coolant T at UP inlet	°C	277.0
Core power	MW	3000.0
Level in PRZ	m	8.5
Pressure SG1–SG4	MPa	6.3
Level SG1–SG4	m	1.7
Mass flow in one loop	tons/s	4.5
Feed water flow rate	tons/s	1.6

This simulation consists of a hot header break (PRISE) in loop No 4. The BRU-A valve 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:

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. We assume that the SCRAM signal is given when the UP pressure threshold of 13.7 MPa is reached. 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 of 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 1800s, 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 2700s, the operator activates the make-up system and the last functioning HPIS is deactivated.
6. At 4700s and a PS pressure below 0.4 MPa, the leakage of coolant to the environment is completely terminated and the simulation is stopped.

The chronology of the main events of the transient calculation is given in Table 3. Additionally Figs. 3–6 show the development of main parameters during the transient.

4.2. 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 results of both models are depicted in Figs. 7. Both models provide the I131 concentration which has to be released at the core during the transient. Since only the iodine input is changed, but all other parameters affecting the transport simulation performed with RELAP5-3D remain the same, only the magnitude of the models' results varies.

4.3. Uncertainty evaluation

The results of the CIAU analysis are included in Table 4 and are shown graphically as well (Fig. 8). For PS-pressure between 500 and 2500s the uncertainty increases. This can be explained as in this duration several safety systems of the reactor are activated which leads to additional sources for uncertainty. Table 4 depicts that the uncertainty of the calculation of the mass inventory is larger than the other parameters. Regarding the figures it can be detected that the uncertainty band changes over time, as each timestep can be in a different hypercube which has a different uncertainty due to the experimental data it contains.

4.4. Derivation of uncertainty of transported iodine to the environment

The results of the propagation of iodine transport uncertainty are depicted in Fig. 9. It shows, that the uncertainty bands are greatest between 500 and 2500s. At the peak of the iodine release to the environment the uncertainty is more than double the nominal value. The variation of the uncertainty bands can be explained by the fluctuations

Table 2
Steady state calculation.

Parameter	Units	Power plant steady state calculation	Standard deviation [%]
Core thermal power	MWth	3 000.00	±5.00
Pressure in the pressurizer	Bar	157.00	±0.01
Pressure in the steam generators	Bar	62.70	±0.30
Inlet temperature in the core	K	563.15	±1.00
Outlet temperature in the core	K	593.15	±1.00
Primary loop mass flow rate	kg/s	4 530.00	±1.00
Primary inventory	kg	240 800.00	±0.01
SG liquid mass inventory	kg	158 800.00	±0.01
Feedwater mass flow rate	kg/s	1 632.00	±4.00
Feedwater temperature	K	493.15	±0.50
Main steam line temperature	K	550.00	±1.00
Pressurizer level	m	8.45	±0.10

Table 3
Main events of the transient.

#	Event	Set points	Time after steady state [s]
1	Break opening	Time	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:		150
	L1		150
	L2		150
	L3		2100
	L4		
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:	Time (Operator action)	
	- CL-1		1800
	- CL-2		–
	- CL-3		2725
	- CL-4		1800
20	Start of operation of PS make-up system:	Time (Operator action)	2725
21	Stop of simulation:		4700

Table 4
Average deviation of uncertainty bands.

	Lower band (mean)	Nominal value	Upper band (mean)
Primary side pressure	0.85	1.00	1.14
Mass inventory	0.77	1.00	1.24
Cladding Temperature	0.90	1.00	1.11

in void fraction (which are greatest between 300 and 2500s). Furthermore, it is the case that during this timeframe several of the active and passive safety systems of the reactor are activated/deactivated which leads to enhanced uncertainty. In Table 5 the average uncertainty of the IS simulation is shown. The uncertainty is significantly larger as for the other parameter which were examined (Table 4). This is the case as the uncertainty of the iodine transport depends on several (thermal hydraulic) parameters that are themselves subject to uncertainty and therefore the error propagation law has to be applied.

Table 5
Average deviation of uncertainty bands for iodine.

	Lower band (mean)	Nominal value	Upper band (mean)
Primary side pressure	0.51	1.00	1.88

5. Conclusions and discussion

The thermal hydraulic simulation showed that for the examined accident scenario the installed safety systems and the anticipated operator actions are sufficient to stabilize the reactor within a reasonable timeframe. The reactor 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 break flow, which transports the radioactive fission products to the secondary side and further the environment, to a minimum.

With the CIAU analysis it was possible to mathematically show to which extend the thermal hydraulic simulation of the analysed accident

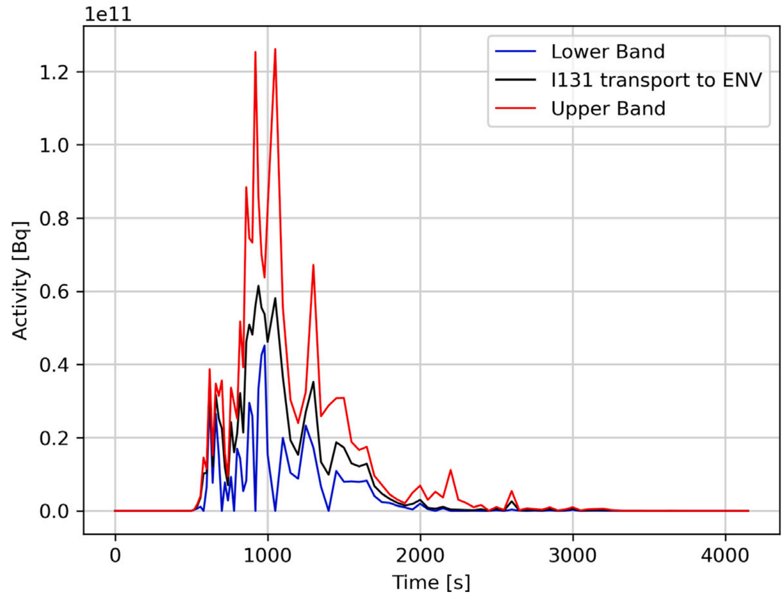

Fig. 9. NRC model: Uncertainty of I131 simulation.

scenario is subject to uncertainty. The analysed parameters have a different order of magnitude of uncertainty. Apparently the simulation power of Relap5-3D for the examined DEC-A scenario is most precise for the cladding temperature, where the uncertainty band is within 21%. However for the primary side pressure (29%) and the especially the mass inventory (47%) the uncertainty bands are more pronounced. To determine, if this phenomenon is specific to the examined accident scenario, or an overall condition, it would be of interest to apply the CIAU method for additional accident scenarios as well.

The propagation of uncertainty method, under the assumptions that the secondary side pressure uncertainty is equal to the primary side one and the void fraction in the SS is 2.5%, allowed to extend the uncertainty calculations by the CIAU analysis from the primary system to the secondary system and the environment without experimental data on iodine transport. It was shown that the transport of iodine into the environment is subject to a very pronounced uncertainty (Table 5). This is in line with the theoretical considerations in the literature regarding this topic (Hózer and Vajda, 2001). To increase the certainty of the uncertainty bands, detailed measurement of transported iodine during transient scenarios would be necessary.

This study has shown that in the case of a containment bypass scenario, significant quantities of radioactive fission products can be released into the environment. For the safety analyses of nuclear power plants it is essential to consider the uncertainty range identified in the simulation for iodine transport in order to avoid the possibility of underestimating potential risk factors.

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 Horizon Europe. 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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