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Optimisation of accident management measures to reduce iodine releases during SGTR

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A R T I C L E I N F O

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A B S T R A C T

Steam generator tube rupture (SGTR) accidents create a bypass of the containment of a pressurised water reactor (PWR) and can therefore result in the release of primary system coolant to the atmosphere via the steam relief or safety valves. In general, primary system coolant will transport radionuclides such as iodine-131. Accident management strategies for SGTR accidents therefore aim to reduce releases to the environment while ensuring core cooling.

The Downhill Simplex algorithm is used in this paper to optimise the timing of accident management measures during a SGTR accident. The secondary system steam relief and safety valves (SRV) are assumed to fail in the stuck open position at the first opening. Depressurisation of the primary system by opening the pressuriser pilot operated relief valve (PORV) and keeping the primary pressure low by shutting down two of the three trains of the high pressure injection system (HPIS) is assumed as the accident management procedure. The success of the measures is evaluated by a Relap5-3D simulation, which calculates the thermal hydraulic behaviour of the system. One of the key parameters used to assess success is the amount of iodine-131 released into the environment. The algorithm varies the timing of a set of three operator actions — opening the PORV and shutdown of HPIS trains one and two. In addition to iodine release, two other parameters are evaluated — reactor core coolant level and primary system pressure. Three normalisation functions are used to convert these parameters into a single target value, which is low when the core is covered and both primary system pressure and iodine release are low. The simplex algorithm then modifies the timing of operator actions to achieve a local minimum of the target value.

The results show that the Downhill Simplex algorithm can be used to optimise the timing of operator actions. Although timing cannot be directly implemented in Accident Management Procedures (AMPs), it is important to be aware of time sensitivity when designing AMPs. In addition, the algorithm can be adapted to optimise design parameters such as valve sizes, hydro accumulator nominal pressure levels.

The work was performed within the EURATOM R2CA project.

1. Introduction

The design of a nuclear power plant (NPP) requires consideration of various plant conditions. These include normal operating conditions, anticipated operating events and accident conditions. The accident conditions are further divided into design basis accidents (DBA) and design extension conditions (DEC). The operating conditions mainly provide input to the design basis of the process equipment for normal operation and for the control, limitation and reactor trip systems. The accident conditions provide the basis for the design basis of the safety systems and safety features for DBAs and DECs (IAEA, 2016c). DBAs consist of accident conditions for which a plant is designed according to established design criteria and conservative methodology and for which releases of radioactive material are kept within acceptable limits. In DBAs, accident management (AM) measures based on operator actions are not normally required in the short term. However, as an event progresses over time, operator actions may become necessary. Generally DECs are more severe, but less likely, accidents that go beyond the design basis.[1] Therefore, the IAEA defines DEC as "... accident

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[1] This does not exclude that there are DEC scenarios without radiological consequences.

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conditions that are not considered for design basis accidents, but that are considered in the design process for the facility in accordance with best estimate methodology, and for which releases of radioactive material are kept within acceptable limits" (IAEA, 2016b). Design extension conditions were added to the safety analysis of nuclear power plants after it became evident (e.g., due to the accidents at Three Mile Island and Fukushima) that design-based accidents do not cover all plant states. As a result, features were added to the designs to deal with multiple system failures to prevent core melt ("DEC-A") and to mitigate the consequences of core melt scenarios ("DEC-B"). For DEC-A, greater reliance on accident management measures and operator actions is allowed than for DBA (IAEA, 2016c).

Various computer-based accident management and source term prediction tools are used by operators and national emergency teams to improve the response to an accident scenario. These tools either evaluate existing management strategies or analyse transients to establish accident management strategies. Many recent publications on this topic favour artificial intelligence (AI) (Chung, 2021; Mena et al., 2022; Vicente-Valdez et al., 2021; Lee et al., 2020). Although AI has many advantages (e.g. for live monitoring and AM support), it also has a significant disadvantage - a relatively large and representative database is required to train the AI algorithm. Especially in the field of nuclear energy, and even more so in the field of accident management, open available data is scarce. This paper presents a computational optimisation approach based on the Downhill Simplex method that can be applied in accident management to optimise both accident management measures and design parameters of safety-relevant systems.

The general approach was introduced by Muellner et al. (2007) and applied to a station blackout in a VVER1000 reactor by Cherubini et al. (2008). In this paper, the approach is applied to the Steam Generator Tube Rupture (SGTR) for a generic pressurised water reactor. A steam generator tube rupture creates a bypass for radionuclides from the primary to the secondary system, thus removing the containment as a safety barrier. If we further assume that in addition to the standard SGTR, the Safety Relief Valve (SRV) of the affected steam generator is stuck open and one of the emergency cooling systems is not working due to technical failure, we are dealing with a DEC-A of multiple system failures. In this work, we assume a DEC-A SGTR scenario where the core structures remain intact. Therefore, it is sufficient to use a thermal hydraulic system code to simulate the plant. However, the combination of SGTR and the stuck open SRV creates a release path for radionuclides to the environment.

Accident management measures are aimed at mitigating an accident and reestablish stable conditions and avoid severe consequences by preventing or at least reducing the release of radionuclides to the environment at a nuclear power plant (IAEA, 2019). Hence two types of accident management guidance documents were developed to ensure the safety of NPPs (IAEA, 2016a, 2008; NRC, 1982):

- Emergency Operating Procedures (EOPs), typically for design basis accidents with the aim of preventing core degradation, and
- and Severe Accident Management Guidelines (SAMGs), typically for severe accidents with the aim of stabilising core degradation that has already occurred.

These procedures and guidelines are intended to assist the main control room operators in the event of situations beyond operational procedures (accidents). As this paper focuses on a SGTR scenario without core degradation, SAMGs are not of major importance here.

EOPs and communication (protocols) (Kim et al., 2010) form the basis of a functioning safety culture and ensure that accident management functions properly in the event of an accident at a nuclear power plant. SGTR scenarios are an important subset of design basis accident EOPs (Callow, 1988; Gregoric et al., 1990; Ishigami and Kobayashi, 1993; Izquierdo-Rocha and Sánchez-Perea, 1994; Parzer et al., 1995; Auvinen et al., 2005; Kozmenkov et al., 2020). Therefore, this paper is based on the importance of both EOP development and SGTR research, as we show how numerical optimisation can be applied in testing and could even be used to improve and develop EOPs and accident management strategies. Thus, our scope is to show what is possible rather than to claim improvement of a specific accident management strategy for a specific reactor type.

In our case, the operator (team) is trying to reduce the release of radionuclides to the environment by reducing the pressure on the primary side. At the same time, it is necessary to ensure the removal of decay heat from the core to the final heat sink. The core should be in long-term cooling conditions to avoid core degradation. To achieve this, various measures are available to the operator, such as operational safety systems (e.g. certain emergency cooling systems or a power-operated pressure relief valve). However, these measures have different effects on the reactor system. Consequently, there are different approaches for the optimisation of accident mitigation measures (Bang et al., 2022; Park et al., 2021; Zhang et al., 2008).

We use the Downhill Simplex algorithm (Nelder and Mead, 1965) – a method derived from linear programming – to show how a simple but robust optimisation algorithm can be used to test and determine the timing of operator contingency actions in the event of an SGTR with multiple system failures for a generic PWR.

2. Methodology

This work applies a method to optimise the timing of operator actions in case of a steam generator tube rupture in combination with a stuck-open safety relieve valve. In the following section we describe the general scenario and the mathematical algorithm used for optimisation.

2.1. The considered reactor system

Our analysis considers a SGTR in a generic pressurised light-water reactor (PWR). The considered PWR is rated at a thermal power of 3.750 MW_{th} and an electrical power of 1.300 MW_e. The reactor coolant system (RCS) consists of four loops, each loop consisting of hot leg (HL), steam generator (SG), loop seal, a main circulation pump (MCP) and a cold leg (CL). On the secondary side each SG is fed feed water by a feed water line, steam is supplied to the turbine by a steam line. The steam lines are joint in a collector. Each steam line is equipped with a main steam isolation valve (MSIV) to isolate the steam line. Steam Safety- and Relief Valves (SRV) are located on the steam lines between SG and MSIV in each line. A high-capacity pressuriser (PRZ) containing safety and relief valves (PORV) is connected via surge line to loop two. The emergency core cooling system (ECCS) consists of four trains of high pressure injection system (HPIS), eight hydro accumulators (ACCs) and the low pressure injection system (LPIS). The LPIS serves also as residual heat removal circuit. The considered NPP follows in general a 4 x 50% safety approach, within DBA two safety systems are considered to be unavailable due to single failure and repair case.

2.2. Assumed scenario and AM measures

Additional failures are assumed which moves the accident scenario in the DEC-A space. The scenario considers a SGTR of a single U-tube as initiating event. Low pressure in the primary system and activity in the steam line triggers reactor scram and closure of all four steam line main isolation valves. This leads to an increase in secondary system (SS) pressure up the opening of the SRV. It is assumed that the SRV of the affect loop remains stuck open after the first opening. It is further assumed that all trains of the LPIS are unavailable. Once the pressure in the primary system reaches the set point of the HPIS, the coolant lost from the PS to the SS is equalised by the HPIS and the pressure of the PS kept close to the shut-off head of the HPIS. In consequence significant amounts of primary coolant are released by the SRV to the environment.

The AM-Strategy therefore aims to limit the releases of primary coolant to the secondary system and further to the environment. There are several actions that are taken by the operator after 30 min: Cool down of the primary system by depressurisation of the intact steam generators at a rate of 100, depressurisation of the primary system by opening the PORV valve, shut off of all four trains of HPIS and reconnecting the make-up and let-down system instead. The timing of three operator actions is optimised (set of accident management actions):

- Time of switching off HPIS train 1,
- time of switching off HPIS train 2 and
- time and opening the PORV.

We always assume a cooldown of the primary by the secondary side after 30 min. We optimise against a baseline scenario, in which we just shut down the last two available trains of HPIS and do not open the PORV. For the optimisation, a target evaluation value was introduced that measures success or failure of the applied AM strategy. Three parameters have been selected and combined into a function that gives an indication of the condition of the plant. These parameters are chosen based on expert judgement and probabilistic cut-off criteria and explained in the following section. Then, the Nelder–Mead or Downhill Simplex method which is used for optimising the so created function is introduced.

2.3. Parameters to describe the plant status

The Downhill Simplex method described below is a method to find the minimum (or maximum) of a certain function under boundary conditions. The method requires that the function maps $\mathbb{R}^n \rightarrow \mathbb{R}$. In the considered case, the functions maps the points in time of three different interventions of the operator into one value. This value should be a measure for the success of the operator intervention. Success in the case of nuclear facilities is to ensure subcriticality, core cooling and avoid or at least minimise radioactive releases into the environment. Due to the complexity of a nuclear power plant, there is not a readily available function to do that. Three parameters are chosen to describe the status of the nuclear power plant (set of evaluation parameters):

- the iodine release after the accident,
- the primary pressure and
- the coolant level in the reactor core.

These three independent parameters are combined into one function in a way that the value of the function is a good measure of the success of the operator intervention: it will map $\mathbb{R}^3 \rightarrow \mathbb{R}$. To account for the different order of magnitudes and units of the parameter, each parameter is first normalised to only yield values between 0 and 1. This is done as follows:

2.3.1. Primary pressure

The Reactor Pressure Vessel (RPV) is an essential part of the defence-in-depth concept of nuclear power plant. Under normal conditions, the pressure is 15.8 MPa (Zimmerl et al., 2021). The function that we used to map the primary pressure P_{prim} to the interval [0,1] is given by

$$P_{prim}^{norm} = 1 - \frac{1 + exp(-a * b)}{1 + exp(a(P_{prim} - b))} \tag{1}$$

and shown in Fig. 1.

The unitless constants[2] $a = 1$ and $b = 6$ are chosen to mirror the following characteristics: The high pressure injection system as a part of the Emergency Core Cooling System (ECCS) can only perform up to a pressure of the coolant system of 10.5 MPa. Approaching this threshold, less and less coolant can be injected. On the other side, the lower the pressure, the easier long-term cooling of the reactor will be. Furthermore, in case accident progression cannot be avoided, low primary system pressure reduces the risk of direct containment heating.

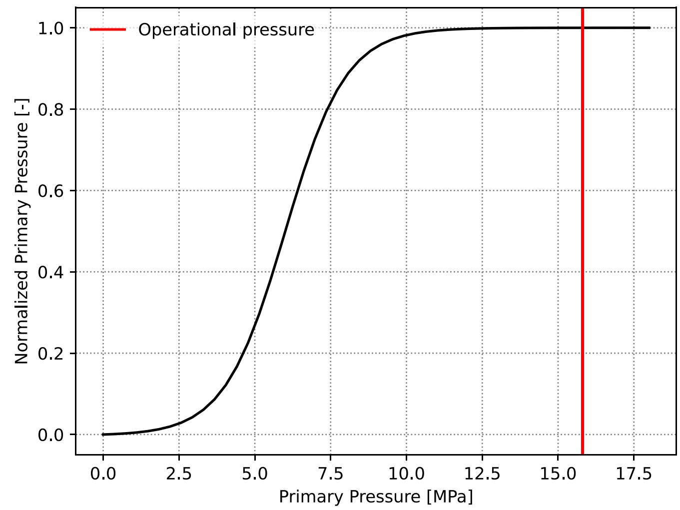

Fig. 1. Mapping of the pressure in the primary system to the interval [0,1] where 0 is a good state.

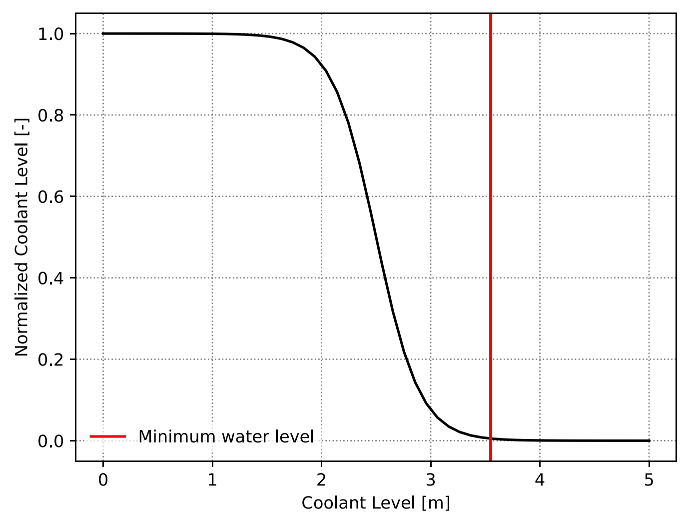

Fig. 2. Mapping of the coolant level in the primary system to the interval [0,1] where 0 is a good state.

2.3.2. Core cooling

The coolant level in the reactor vessel is important. The active area of the fuel elements should always be covered by coolant. As soon as the reactor core is exposed, the fuel elements can dry out and heat up, and even melt if core cooling cannot be restored.

In our model, the top of the active region is at 3.55 m. Consequently, we choose the following function to map the coolant level h to the interval [0,1]:

$$h^{norm} = \frac{(1 + exp(-a \cdot b))}{(1 + exp(a \cdot (h - b)))} \tag{2}$$

The mapping function is shown in Fig. 2.

The unitless constants $a = 5$ and $b = 2.5$ are chosen in a way that for a coolant level above the top of the active region, the function is (almost) zero. There is no steep step at the top of the active region, since

[2] Mathematical parameters, that define the position and shape of the curve.

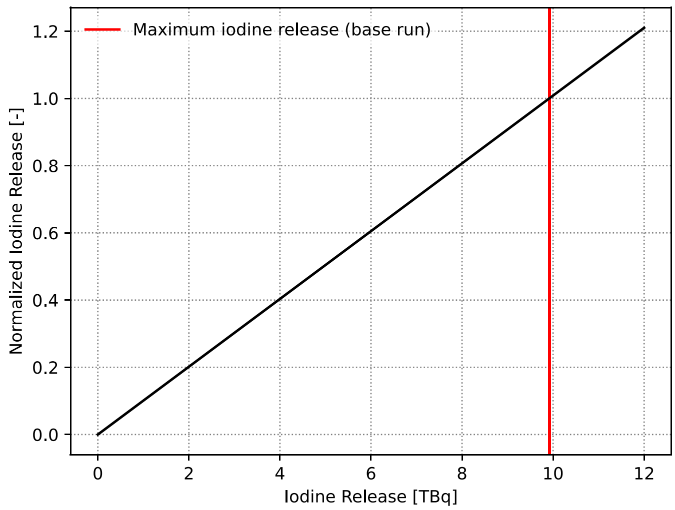

Fig. 3. Linear mapping of the iodine release to the environment to the interval [0,1] where 0 is no release at all.

water and steam are not exactly separated at a certain level. Thus, even though the collapsed water level is already below the top of the active region, certain cooling is still possible at all heights of the active region.

2.3.3. Iodine release

Rapid changes in reactor power, like the actuation of a reactor SCRAM might lead to increased values of iodine-131 in the primary coolant system, a phenomena known as iodine spiking. The amount of iodine released is oriented along the guideline of US NRC (NRC, 2003) for analysis of SGTR accidents. It is therefore assumed that the I-131 release rate to the primary system coolant increases by a factor 500 in respect to the release rate corresponding to the equilibrium concentration during normal operation. Assuming an average I-131 concentration from literature (Adams and Atwood, 1991) the I-131 released from the fuel was calculated and released into the primary system coolant over a period of 3000 s following the reactor SCRAM.

The key figure of the optimisation is the amount of I-131 that bypasses all safety barriers and is released to the environment. The release of I-131 from the baseline calculation (shut down of all four trains of HPIS after 1800 s, PORV not opened, see Fig. 11) was taken as normalisation factor. The iodine release to the environment should be as low as possible, so a linear threshold model was applied to map the iodine release iod to the environment:

$$iod^{norm} = \frac{iod}{iod_{max\_base}} \tag{3}$$

where

iod_{max_base} is the maximum iodine release of the base simulation run.

This means that lower iodine release gives a lower/better score in the objective/evaluation function (see Fig. 3).

2.4. Objective/evaluation function

The objective/evaluation function of the Simplex algorithm finally calculates the evaluation value that characterises the outcome of the run. The evaluation value is the sum of the normalised evaluation parameters (see Eq. (4) below). Hence the range for the evaluation parameter starts at 3 and goes down to 0.

$$eval^{simplex} = P_{prim}^{norm} + h^{norm} + iod^{norm} \tag{4}$$

These values are then used by the algorithm to find the local minimum.

2.5. Optimisation process - Downhill Simplex method

There exist several methods to find local extrema of multidimensional functions. To be applicable to the already described problem, it is mandatory that only the evaluation of the function (and not its derivative or similar) is needed. The Downhill Simplex method fulfils this requirement. It has been introduced by Nelder and Mead (1965). It is a standard linear programming technique for optimisation problems. In geometry, a simplex is the simplest possible polytope in any given dimension (e.g. point, line segment, triangle, tetraeder, and so on. The shape is defined by the number of evaluation parameters used. Several linear inequalities define a polytope as a feasible region. The simplex algorithm begins at a starting vertex and moves along the edges of the polytope until it reaches the vertex of the optimal solution. Mathematically, the problem is to find a N-dimensional local minimum or maximum $P_m = (x_1^m, x_2^m, ..x_n^m)$ of the function $f(x_1, x_2, ..x_n)$. The implementation of the algorithm is based on Muellner et al. (2007). The following steps are implemented in the algorithm:

- Step 1: construct a Simplex of N + 1 Points. This Simplex should not be chosen too small. Evaluate the objective function at each point. Identify the highest point of the simplex P_h. The remaining points are labelled P_i, so that the simplex consists now of the points $(P_i,\ P_h | i = 1, \dots, N)$.
- Step 2: Find a new point P_r by moving the highest point P_h of the simplex through the barycentre $P_b = (1/N)\Sigma P_i$ of the other points, conserving the volume of the simplex. Evaluate the objective function f at this new point P_r. Depending on the value of $f(P_r)$ the next step is:
  - $f(P_i) < f(P_r) < f(P_h)\forall i$: if the new point P_r is lower than the highest point of the simplex, but higher than the lowest point of the old simplex without the highest point, construct a new simplex by substituting the previous highest point with the new point and start from the beginning (reflection).
  - $f(P_r) < f(P_i)\forall i$: if the new point P_r is the lowest point of the simplex, construct a new point P_{rr} by moving P_r further in the same direction by a factor of two. Evaluate the function at P_{rr}. Then:
    1. $f(P_{rr}) < f(P_r)$: the direction led further to the minimum. The new simplex will be $(P_i, P_{rr}, \forall i)$. The volume of the simplex is increased (reflection and expansion).
    2. $f(P_{rr}) > f(P_r)$: moving Pr did not minimise f any further. The new simplex will be $(P_i, P_r, \forall i)$ (reflection).
  - (c) $f(P_r) > f(P_h)$: if the new point is higher than the previous highest point, a new point, $P_{rr'}$, halfway between P_h and P_b will be constructed and $f(P_{rr'})$ will be evaluated. Again, there are two possible continuations:
    1. $P_{rr'} < P_h$: the simplex already comprises the minimum and contracts. The new simplex is chosen to be $\forall i : P_i, P_{rr}$ (contraction).
    2. $P_{rr'} > P_h$: the characteristic length of the sides of the simplex is too large — f varies too much inside the simplex. The lowest point of the old simplex is chosen: $\min(\forall i : P_i, P_h)$ and all other points are moved along their sides towards the lowest point. The length of all sides is reduced by a factor of two (multiple contraction).

For the algorithm to end, an exit criterion must be defined. One possibility is that the deviation of the objective/evaluation function f $(\forall i : f(P_i) - f(P_r))$ drops below a given threshold.

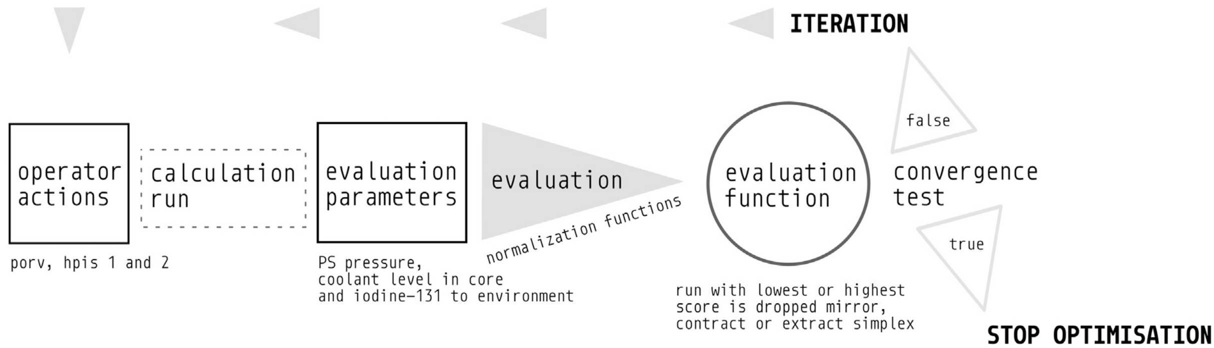

Fig. 4. Scheme of the optimisation process.

2.6. Application of the algorithm

For the problem at hand the Downhill Simplex optimisation loops over the following steps (see Fig. 4):

- Defining four sets of operator actions (opening of PORV, shutdown of HPIS 1 and 2),
- running for each of the four sets a RELAP5-3D calculation,
- extracting from the results the chosen set of evaluation parameters (PS pressure, core coolant level and I-131 release into the environment),
- application of normalisation function to calculate the target value,
- evaluation of the simplex set and if the convergence criteria is not met — construction of a new simplex

This loop iterated as long as the convergence criteria is not met.

3. Reactor model and baseline scenario

3.1. Computer code and model

For analysis of the thermal hydraulic behaviour of the reactor, the code RELAP5-3D, Version 4.0.3 is used (INL, 2012). 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. The code was validised for SGTR accidents (Callow, 1988). The nodalisation is very detailed, each loop is modelled separately, the downcomer is divided in eight sections and the reactor core in five, which allows the simulation of asymmetric accidents and transients. The core nodalisation of the generic PWR is depicted in Fig. 7, Fig. 6 shows the nodalisation of the loop containing the PRZ and Fig. 5 shows the nodalisation of the affected loop, modelling the SGTR. The emergency core cooling systems (HPIS and ACCs) as well as the SGTR break (red) and the environment are depicted in Fig. 5. One parameter to assess the value of operator actions is the release of radioactive fission products to the environment. Minor fractions in the fuel rods lead to fuel leakage to coolant and thus to enrichment of fission products in the primary coolant. For the calculation of the radionuclide transport, the RELAP5-3D radionuclide transport model is used. Roughly 300 TBq of I-131 are assumed to be released within one hour after the reactor scram into the primary system coolant and modelled as source in the Relap5-3D model.

3.2. Baseline scenario

Aim of the analysis is to optimise the timing of operator actions during a STGR accident. A baseline calculation serves as reference for all subsequent calculations. The optimisation is successful once a timing of operator actions has been found that leads to lower I-131 releases than in the baseline scenario, but without endangering core cooling and with succeeding in depressurisation of the primary system. For the set points for the actuation of the safety systems see Table 1 and for the assumed operator actions see Table 2. The accident starts with opening of the valve simulating the SGTR after 300 s steady state time.

Table 1
Overview of the set points of the relevant safety systems and trips.

Event	Criteria	Unit	Value
SCRAM signal	PRZ level or	m	<2.28
	PS pressure	MPa	<13.2
MCP-1,2,3,4 coast down	Time after SCRAM	s	+1
Closure of MSIV in steam lines	Time after SCRAM	s	+1
Turbine coast down	Time after SCRAM	s	+1
Switch from FW to EFW	Time after SCRAM	s	+20
ACCs	PS pressure	MPa	<3.0
Steam line SRV opening	Pressure	MPa	>82.9
Steam line SRV closing	Pressure	MPa	<77.0

Table 2
Availability of ECCS and overview of the actions set by the operator.

System/measure	Availability/actuation
HPIS	All trains available
ACC	All trains available
LPIS	All trains unavailable
PS cooldown via SS-SRV	1800 s after beginning of transient
Disconnect of HPIS, reconnect of make-up	1800 s after beginning of transient

The analysis is limited to full power conditions. The state of the emergency core cooling systems is provided in Table 2. All trains of the emergency feed water system are available. The STGR occurs at 300 s. Reactor scram and containment isolation (closure of all MSIV) is triggered by low primary system pressure and activity in the steam line. Pressure build up in the secondary system after MSIV closure leads to opening of the SS-SRV, the SRV of the affected SG is assumed to remain stuck open. Primary system pressure decreases to the set point of the HPIS system, which inhibits further decrease of PS pressure and keeps the loss of primary coolant to the secondary at a constant rate. After 30 min the operator analysed the situation and initiates a secondary side cooldown process with a cooldown rate of 100 K/h by opening the safety relief valve of the intact steam generators, disconnects all trains of HPIS and reconnects the make-up system. The measures successfully decrease the PS pressure further, basically terminating the leakage to the secondary, without endangering core cooling. Divided in phases the baseline scenario develops as follows:

Phase 1 (up to 1800 s): The initiating event is a double-sided U-tube rupture in loop 4. This steam generator tube rupture causes a leak from the primary to the secondary system (PRISE) - simulated by a valve connecting a steam generator tube of loop 4 to the secondary side of the associated SG (SGTR). The primary pressure drops and the low

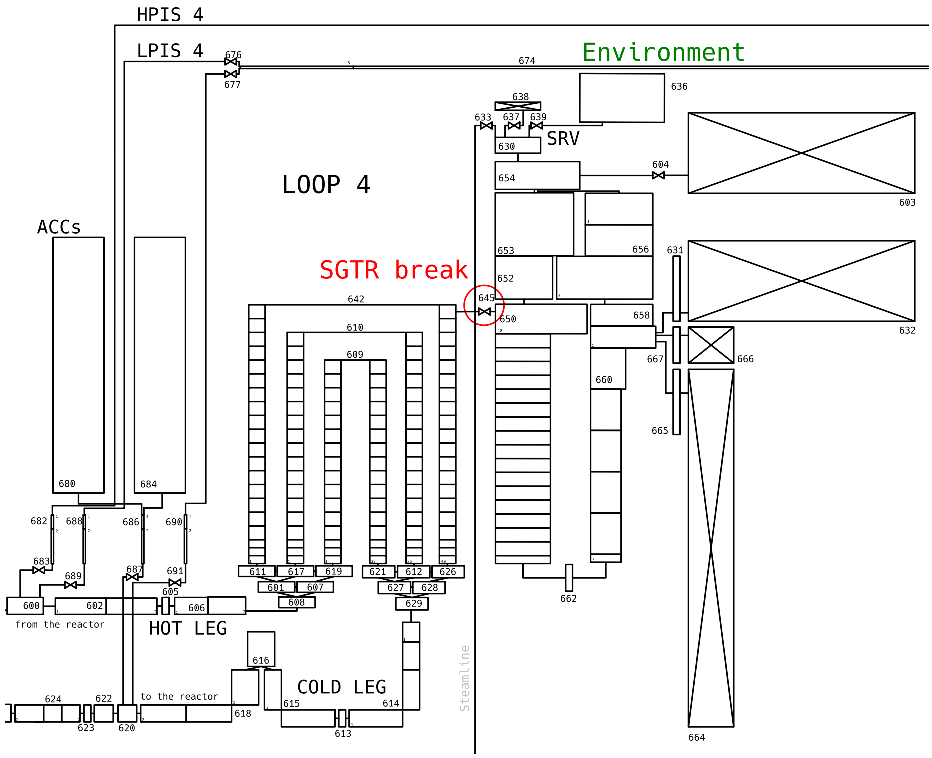

Fig. 5. Nodalisation of the affected loop 4 of the generic PWR.

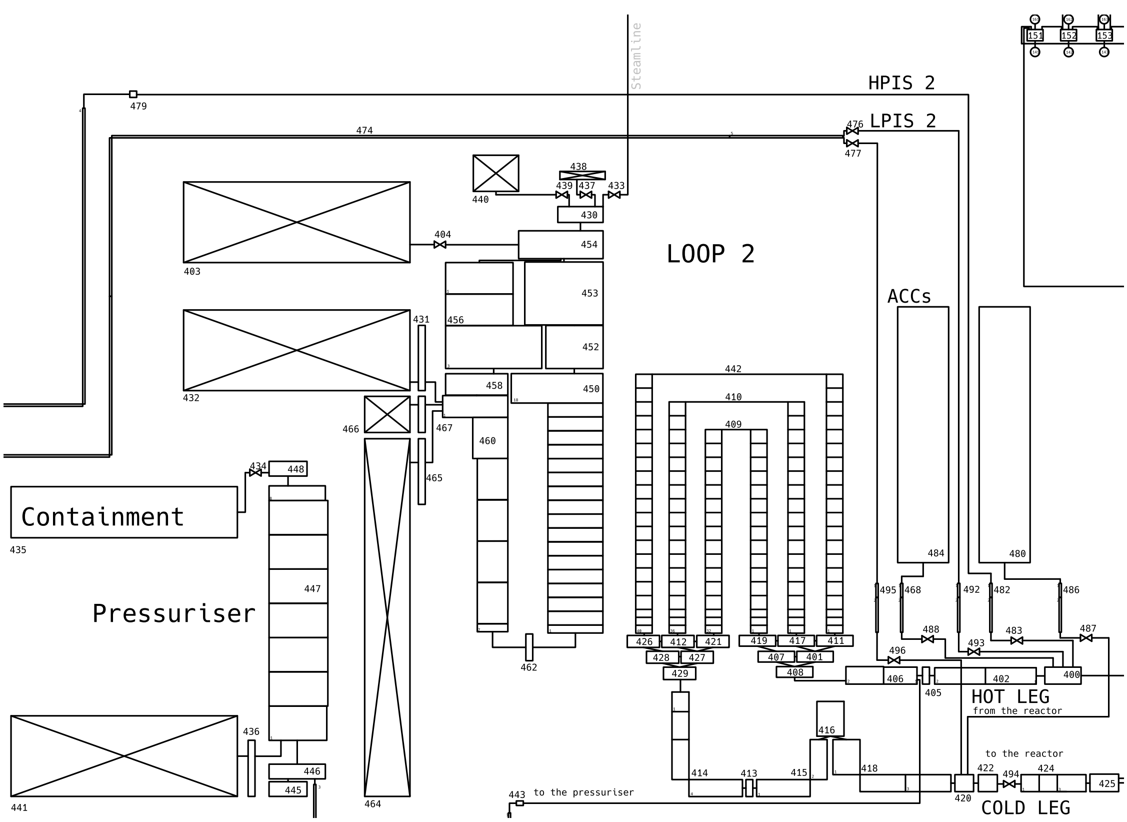

Fig. 6. Nodalisation of loop 2 including the pressuriser and the containment for the PORV.

pressure in the PRZ triggers the reactor scram signal. As a conservative assumption, the reactor loses the external power supply (Callow, 1988; Gregoric et al., 1990) and the emergency power mode gets activated. Together with the activity in the steam line, the containment isolation signal is triggered, resulting in the closure of the MSIVs and coast down of the turbine.

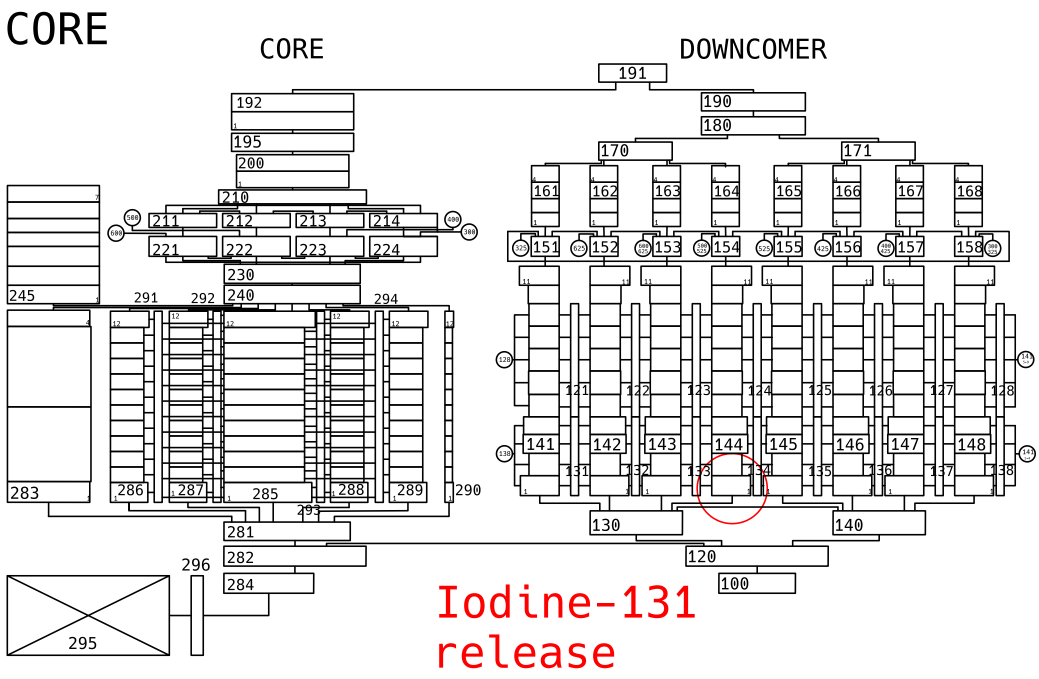

Fig. 7. Location of the iodine-131 release marked in red.

This results in a rapid pressure rise on the secondary side to the point where the SRVs open. The SRVs of loops 1–3 operate as intended, while the SRV of loop 4 (valve 639 in the affected loop) remains stuck open due to a technical failure (see Figs. 5 and 8).

Phase 2 (1800 s–10 000 s): At 1800 s (30 min), the operator starts the SS cooldown system (100 K/h), deactivates all trains of the HPIS and reconnects the make-up system. The measures successfully reduce the loss of coolant to the secondary system, but the loss of primary coolant is not fully compensated and the level in the RPV drops (see Fig. 9). At about 6300 s, the set point of the ACCs is reached and water is released to the core. This stabilises the core level at about 9 m.

Phase 3 (more than 10 000 s): After 10 000 s, the combination of the steam present in the U-tube on the primary system and the reduced pressure in the primary system effectively terminates the loss of coolant to the secondary system, from the U-tube on the primary side to the secondary side within the steam generator and finally through the safety relief valve to the environment. Fig. 10 shows the timeline of the release of iodine-131 into the environment. The peak of iodine-131 release is reached at 10 050 s. No iodine is released to the environment from 10 050 s to the end of the simulation. Fig. 11 on the other hand shows the iodine distribution in the different reactor zones and the total sum of iodine in the simulation. The iodine release to the environment is also shown in Fig. 11, but only to get an impression of the different scales of iodine in the different reactor zones. The remaining iodine retained in the water of the PS and SS is obviously much higher than the iodine released to the environment.

The subsequent decrease in activity shown in Fig. 10 is a tribute to the decay included in the simulation model.

A short summary of the scenario referencing only the main events is given below in (see Table 3).

3.3. Parameters for optimisation

Starting from the baseline scenario, the timing of certain operator actions may now be shifted by the simplex algorithm, with the aim of reducing the I-131 releases as far as possible without endangering core cooling. While train 3 and 4 of the HPIS are still shut off after 30 min, shut-off time of trains 1 and 2 is moved by the simplex algorithm. In addition, depressurisation of the primary system by opening of the PORV may also occur after 30 min. The model and the basic reference scenario are described in more detail within the framework of the EURATOM project R2CA (Zimmerl et al., 2021).

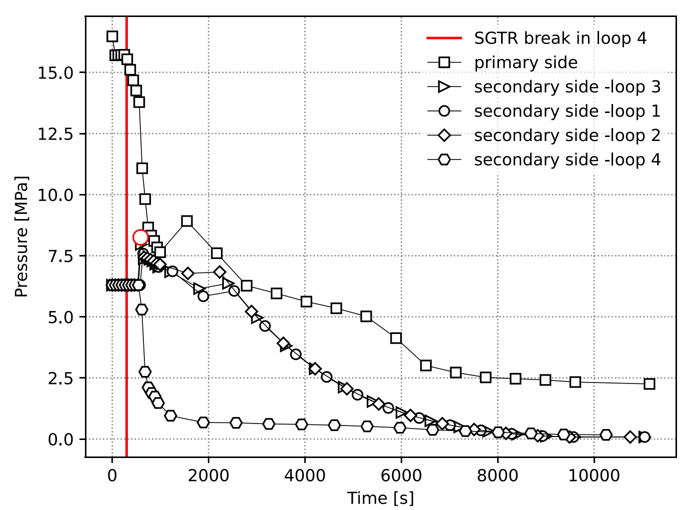

Fig. 8. Pressure on primary and secondary sides (base scenario) - SRV threshold marked as red circle.

4. Results

The Downhill Simplex method was tested on the SGTR scenario described. As the base simulation run shows, the peak of iodine-131 release to the environment is reached after 10 050 s. Consequently, the optimisation simulation runs were limited to 10 500 s to increase calculation efficiency and reduce time. If it is possible to reduce the iodine-131 release to the environment at 10 500 s, the optimisation is successful. In contrast to the base simulation run, not all four HPIS are stopped by the operator 30 min after the SGTR. Two of them, HPIS 3 and 4, are stopped by the operator at 30 min, but for the shutdown

Table 3
Table of events for the base scenario (all HPIS trains stop at 1800 s and the PORV is not opened manually by the operator).

Time	Event	Comment
0–300	Steady state	
300	SGTR (break opens)	Scenario start
300	Iodine-131 release in the reactor starts	Release lasts for 3000 s
581	SCRAM	Pressure in PS below 13.7 MPa
582	Emergency power mode	Due to loss of external electricity supply, Emergency diesel generators start
582	Start of MCP coast down	Due to emergency power mode
582	Containment isolation signal	Closure of all 4 MSIVs, isolation FW
582	Turbine coast down	Due to closure of all MSIVs
582	ACC valves open	SCRAM and water level in ACC tanks
589	Opening SRV SS Loop 4	Pressure in loop 4 reaches 8.29 MPa/ SRV stuck open after opening
589–590	Opening SRV SS Loop 1–3	Open and close as pressure drops again
612	MCP coast down finished	
625	HPIS valves open	P in HL < 11.0 MPA (train 1–4)
658	HPIS pump feed pressure Reached	P_{prim} < 10.5 MPa
662	HPIS pump start to inject	Pump start-up takes 37 s
808	Iodine-131 release to the Environment starts	
2100	SS cooldown system (100 K/h) starts	Operator action
2100	All 4 HPIS systems stop To inject	Operator action (MFW increases)
2100	Makeup system starts to inject	Operator action
3300	Iodine-131 release in the reactor stops	
6310	All ACCs start injection	P_{prim} < 30.0 MPa
10 050	Iodine-131 release to the Environment stops	Peak value reached/the rest of iodine-131 Remains in the cooling water of PS and SS
53 550	Last ACCs stop to inject	
100 000	Simulation stops	For the base run 100 000 s, (for the simplex runs 10 500 s)

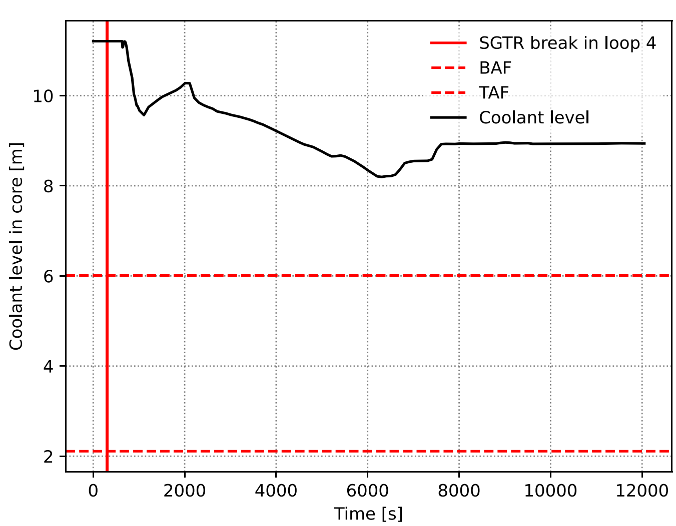

Fig. 9. Coolant level in the reactor core (base scenario) with bottom and top of active fuel indicated.

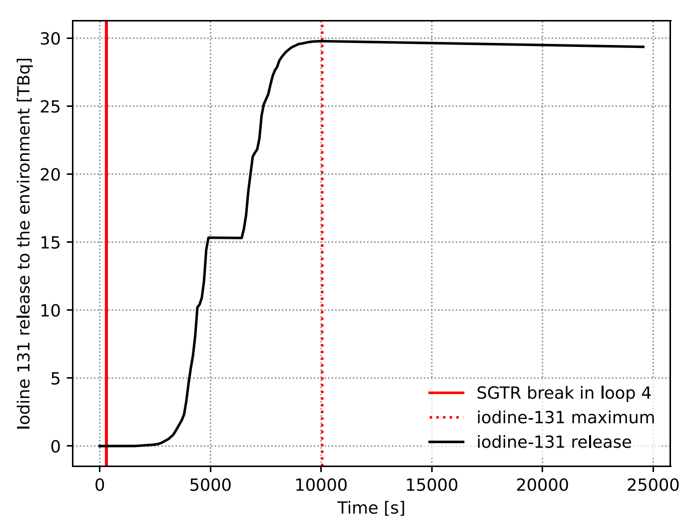

Fig. 10. Iodine release to the environment (base scenario).

of HPIS 1 and 2, the simplex algorithm searches for a set of shutdown times (in combination with opening the PORV) that reduce the release of iodine to the environment.

The speed of the optimisation convergence depends on the input starting 4 showing the times when the PORV opens and HPIS 1 and 2 stop supplying water to the reactor core. The algorithm calculates the results of these initial input runs and then proceeds using the evaluation value to approach the local minimum. In our case, the optimisation algorithm quickly converges to a defined threshold[3] of less than 10% after 4 iterations and less than 1% after 6 iterations (see Table 5). From then on, the number of iterations to reach the next step of 0.1% increases significantly (see Table 6).

The results show that the opening point of the PORV is increasing step by step until it surpasses the simulation time (10 500 s), while the point of time for shutting down HPIS 1 analogously decreases. The

[3] Deviation among the calculated values of the objective/evaluation function.

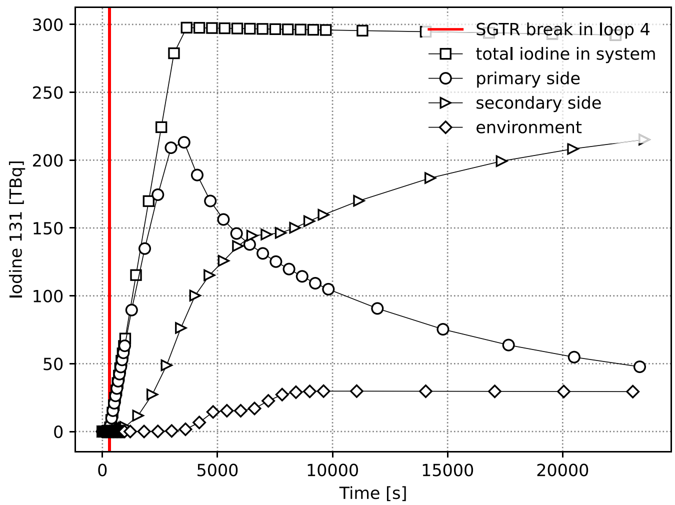

Fig. 11. Iodine-131 in the different reactor zones (base scenario).

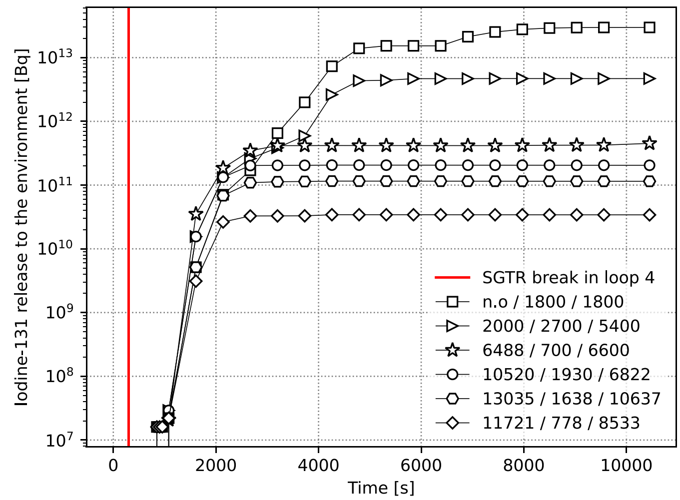

Fig. 12. Comparison of the relevant indicator parameter iodine-131 release to the environment from different Simplex iterations. The squares refer to the base scenario, where the PORV is not opened.

Table 4
Simplex results for the input starting matrix.

PORV [s]	HPIS 1 [s]	HPIS 2 [s]	Simplex value
1800	1800	3600	1.10422
2000	2700	5400	1.15759
3600	2700	3600	1.36542
7200	1800	7200	1.00402

Table 5
Simplex results for a convergence of less than 1% (reached after 6 iterations).

PORV [s]	HPIS 1 [s]	HPIS 2 [s]	Simplex value
6 296	2159	4944	1.00755
10 520	1939	6822	1.00687
6 592	1367	4400	1.00705
7 200	1800	7200	1.00402

Table 6
Simplex results for a convergence of about 0.1% (reached after 13 iterations).

PORV [s]	HPIS 1 [s]	HPIS 2 [s]	Simplex value
11 721	778	8 533	1.00115
13 510	741	10 050	1.00125
12 378	1208	9 585	1.00277
12 896	960	9 211	1.00341

running time of HPIS 2 at first increases, but then lowers again to slightly below the simulation time (see Tables 4 to 6). In respect to the reference scenario the release of I-131 can be reduced by several orders of magnitude (see Fig. 12).

To visualise the effect of the optimisation we contrasted the iodine-131 release to the environment of four different iterations (see Fig. 12). The numbers in the legend of Fig. 12 correspond to point of times of the operator actions - PORV opening/HPIS 1 shutdown/HPIS 2 shutdown. The differences between the iteration runs are so significant that they can only be depicted using logarithmic scale on the y-axis.

5. Conclusions

A word of caution: The results of a numerical optimisation should be interpreted rather as indicators. The optimisation can be used to develop and test accident management strategies and to indicate a general direction, e.g. lower releases if HPIS shutdown is staggered. Hence the results should not be taken as exact points in time, but rather as a help to test and define accident management time windows to achieve specific goals. Furthermore, the optimisation is only as good as the code and input deck running at the heart of the optimisation algorithm. Both the code and the input deck that describes the plant resemble uncertainties. These must also be taken into account.

That said, the Downhill Simplex algorithm converges – in our case to a local minimum – which shows the following in regard to the tested accident management measures:

- the PORV should be opened late or not opened in the accident — although the opening of the PORV reduces the pressure on the PS and thus reduces the leakage of contaminated water to the SS, the combination of these three parameters is complex enough to puzzle understanding. The dominant evaluation parameter is obviously the release of radionuclides to the environment, which is mainly controlled by the timing of two HPIS, but the combination of timing (amount of water) and opening/not opening the PORV (PS pressure) still complicates the outcome. If ratio of necessary amount of water supplied by the two used HPIS is not met (as in the baseline scenario), keeping the PORV closed just amplifies the negative effect and increases the release of radionuclides to the environment.
- three HPIS should be switched off at an early stage — important is to reduce the amount of cooling water injected to the possible minimum, which is necessary for cooling. Consequently, one of the two HPIS used by the operators has to be closed at an early stage and the other one late.
- the other HPIS should run almost until the simulation end (10 500 s) or should not be deactivated at all.

Thus, the results show the importance of finding the right balance between the minimum amount of water necessary to ensure cooling of the primary side and measures to reduce pressure in the primary system to reduce leakage to the secondary side. The results also show that the range of iodine release is not negligible due to the operator actions mentioned above (see Fig. 12).

We have therefore been able to show that optimisation can be used as an alternative for automated testing and evaluation of accident management scenarios. It is particularly valuable in situations where only limited data is available. In addition, this paper demonstrates the potential of this numerical algorithm for future research as a complement or alternative to AI applications.

Glossary:

ACC — Hydroaccumulators (passive emergency core cooling system)
AM — Accident Management

DBA — Design Basis Accident
DEC — Design Extension Condition
EOP — Emergency Operational Protocol
ECCS — Emergency Core Cooling Systems
HL — Hot Leg
HPIS — High Pressure Injection System
LPIS — Low Pressure Injection System
MCP — Main Circulation Pump
MF — Main Feedwater
NPP — Nuclear Power Plant
PORV — Power Operated Relief Valve
PRISE — Primary to Secondary System Leak
PS — Primary Side
PWR — Pressurised Water Reactor
RCS — Reactor Coolant System
RPV — Reactor Pressure Vessel
SCRAM — rapid emergency shutdown of a nuclear reactor
SG — Steam Generator
SGTR — Steam Generator Tube Rupture
SRV — Safety Relief Valve
SS — Secondary Side

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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