European Commission | Horizon 2020 European Union funding for Research & Innovation

REDUCTION OF RADIOLOGICAL ACCIDENT CONSEQUENCES

Action	Research and Innovation Action NFRP-2018-1
Grant Agreement #	847656
Project name	**R**eduction of **R**adiological **C**onsequences of design basis and design extension **A**ccidents
Project Acronym	R2CA
Project start date	01.09.2019
Deliverable #	D5.1
Title	Final Report on innovative diagnosis tools and devices
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Version	01
Related WP	WP5 INNOV
Related Task	T5.1. Pro and cons of innovative devices and management approaches (NINE)
Lead organization	NINE
Submission date	15.11.2023
Dissemination level	PU

This project has received funding from the Euratom research and training programme 2014-2018 under the grant agreement n° 847656

History

Date	Submitted by	Reviewed by	Version (Notes)
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Abstract

The main objective of task 5.1 is to identify potential innovative tools/devices and management approaches that could be beneficial for the reduction of radiological consequences of LOCA & SGTR transients for both DBA and DEC-A conditions. It mainly concerns:

- Optimization of tools for already developed procedures
- Use of advanced instrumentation to help the diagnosis
- Use of on-line neural network as a guide for operators by learning about the transient prognosis

The main aspects covered by this task are:

- A bibliographical review of available diagnosis methods (including neural networks) for accident diagnosis and management in a context of emergency or limited amount of available time
- Identification of innovative devices and approaches, which provide enhanced insight into the accident progression of DECs and, as such, assist in accident management procedures. The influence of such innovative but also of standardized measures during the course of the accident with regard to radiological consequences is to be investigated by taking some examples
- Based on the status of common and innovative tools and approaches, considerations can be made for the optimisation of accident management. For this purpose, representative transients for LOCA and SGTR are selected. Corresponding common AM procedures and EOPs are considered. In a final step, possible optimisations of the AM procedures (e.g., regarding timings, order of measures) can be identified and possible benefits through innovative approaches can be depicted
- Pro & cons of innovative devices and accident management approaches, focussing on the elaboration of benefits of the optimizations of the AM procedures for the reduction of radiological consequences
- Potential innovations will be analyzed in detail, their effectiveness and capabilities will be evaluated using quantitative assessment in the framework of the methodologies promoted by the project. It is anticipated that for both categories of accidents (LOCA and SGTR), at least one corresponding innovative improvement will be considered in detail.

1. Introduction

The main objective of WP5 is to identify potential innovative tools, devices and management approaches that can help to reduce the radiological consequences of LOCA and SGTR transients for DBA and DEC-A conditions. Different organizations are participating in this Work Package. The participating institutions are: IRSN, BOKU, ARB, NINE, SSTC.

The first step in this task was an extensive literature review. The results of literature search and a selection of Accident Management (AM) approaches can be found in Chapter 2. The goal is to improve these tools if possible and to optimize them for specific application areas in already developed procedures. For example, advanced instruments could be used to improve the diagnosis of transients. Assistance by on-line neural networks should lead to supporting the operators in the analysis of transients. First, we present a general overview of Accident Management. We describe Defence in Depth (DiD) levels 3 and 4 and explain the difference between symptom-based and event-based applications (Preventive Emergency Operating Procedures EOP) of accident management. Our research relates to DEC-A and DBA transients. DEC-B is not part of our work in this case. A questionnaire we have prepared on innovative approaches and tools of accident management is being distributed to nuclear power plant operators to give us an insight into the use of these tools. Previous experience with test plants will also be included in this report.

The need for innovative devices and procedures arises from the necessity to have a better understanding of the evolution of accident:

- to be aware of the possible consequences and to evaluate the possible source term,
- to be ready to react in preventing or reducing the amount of the source terms.

While instrumentations and procedures for the control of the plant are well defined, they can be not fully useful to the operator to have a full understanding of the plant conditions during accidents. On the other side the safety instrumentations and procedures are in some cases actuated in automatic way or by the operator based on symptoms giving to the operator a not completely clear picture of the plant conditions. In addition, the safety management is mainly related to prevent or to mitigate the effects of the accident assuring the removal of the heat decay and to keep integrity of barriers avoiding the dispersion of the fission products. Only indirectly they are related to the consideration and evaluation of the possible source terms.

LOCA and SGTR have a relevant role in this framework, because in a LOCA the diffusion of the FP involves directly the last barrier constituted by the containment and in the case of SGTR (or PRISE) the bypass of the containment could take place.

The possibility to have an early and/or realistic estimation of the source term is very important to establish in additions proper safety measures. This estimation requests dedicated devices and procedures depending by the plant design. In addition, those devices must be designed taking into account their survivability in the accidental environmental conditions.

However, the concept of innovation cannot be simply accepted as it is. Innovation implies a process of qualification and acceptance by regulatory authorities.

Main aspect to be considered are:

- Requirements for devices and procedures to be updated and/or innovate to derive a realistic evaluation of the source term. This includes both the aspects related to phases before and after the fission product release from the fuel
- The data and related instrumentations to be considered to give the operator the status of the plant to evaluate the possible source term
- Kind of software tools to inform the operator about the possible source term resulting from the actions of the operator during the accidents
- Special aspects to be considered for LOCA and SGTR accident progression

On an operative point of view relevant aspects related to:

- Applications (what are accidental the conditions when innovative devices and procedures should be used)
- Capabilities (how innovative devices and procedures should be used during an accident occurrence)
- Qualification (how innovative devices and procedures functions are proved and assured)

Those main characteristics can be used as a guide to identify of the innovative devices and procedures for a the specific NPP.
The definition of innovative procedures relates to innovative devices capable to record and to supply the necessary variables to be used in the procedures. The requirements related to needed innovative devices for the development of innovative procedures can be summarized as in the following:

- Development of the analysis of the accidental (mitigative) or pre-accidental (preventive) conditions and identification of the kind of variables supplied by the devices and evaluation of the set points to actuate the protection actions. The evaluation of the possible conditions capable to generate the signal by the device, requests the V&V process related to:
  - Capability of the device to properly record the variable in the expected conditions: qualification of the device in measuring the physical phenomenon connected with the measures
  - Capability of the device to have the correct accuracy in the measurement to promptly make the system or the operator aware that a set point is reached
- Development of the correct response and actions to be performed (manually by the operator or in an automatic way) when some set points are reached. The signal generation should be, in all the expected conditions, clearly interpretable (e.g., high intensity level of the signal or avoiding contrasts with other signals) to prevent possible doubt or misunderstanding in the procedure execution.
- Development of the interface between the generated signals and the systems to be actuated in case of the automatic or the human interface in case of manual actuation. The devices adopted in the procedure should be also proven to transform the generated signals in suitable signals to be represented in the Control Room in an immediate understandable way by the operator (if necessary) and to actuate the systems.

While innovative devices and procedures will make more effective the actions to prevent and to mitigate consequence of accidents, some issues, based on the list of above defined aspects, can be identified:

- The simulation to check the generation of the signal by the adopted devices: this implies the requirement to include in the simulation the model for the phenomena generating the signals in the device.
- The environment condition identification and the related devices qualification. Again simulation can play a relevant role to determine the expected environmental conditions where the device is called to work. However, the necessity to test the devices in realistic conditions is an important aspect.
- The acceptance by the regulatory authority requires the demonstration of the efficacy of the new devices in accidental conditions could be not simple especially if the new devices will be incorporated in existing reactors.
- The devices should cover the early, the progression and stabilization phase of the accident. The range of variation of the physical parameters to be measured could prevent the possibility to use the same devices for all the accident phases.

- Because of the specific and more accurate measurement expected in innovative devices different kind of devices probably should be developed for preventive purposes (to avoid the occurrence of the accident itself and/or terminate the evolution of conditions leading to the accident) and for mitigative purposes (to minimize releases of radioactive material and achieving a long term stable accepted state).

2. Accident Management (AM)

2.1. Introduction

In the plant design of a nuclear power plant (NPP) the following plant states are considered (see Figure 1) (IAEA 2016a). The Operational states ("Normal Operation" (NO) and "Anticipated Operational Occurrences" (AOOs)) mainly provide input to the design basis of the process equipment for normal operation and for the control-, the limitation- and the reactor trip system. The Accident conditions ("Design Base Accidents" (DBAs) and "Design Extension Conditions" (DECs)) provide input to the design basis of safety systems (control of DBAs) and safety features for DECs (control of DECs). DBAs consist of accident conditions for which a facility is designed in accordance with established design criteria and conservative methodology, and for which releases of radioactive material are kept within acceptable limits. In DBAs, AM measures by the operator actions are normally not needed in the short term. However, as a sequence extends in time, operator actions (AM) might become necessary. Examples for a DBA scenario are the following:

- a steam line break
- a feed water line break
- a break of the cold/hot leg at the PS
- a simultaneous trip of all Main Circulation Pumps (MCP).

DEC scenarios are defined as: "Postulated accident 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 2012a) The safety features for DECs include design features for multiple system failures for core melt prevention (DEC-A) and mitigatory design features for core melt scenarios (DEC-B). For DECs, which can involve multiple failure scenarios, greater reliance on operator actions is needed (IAEA 2016a).

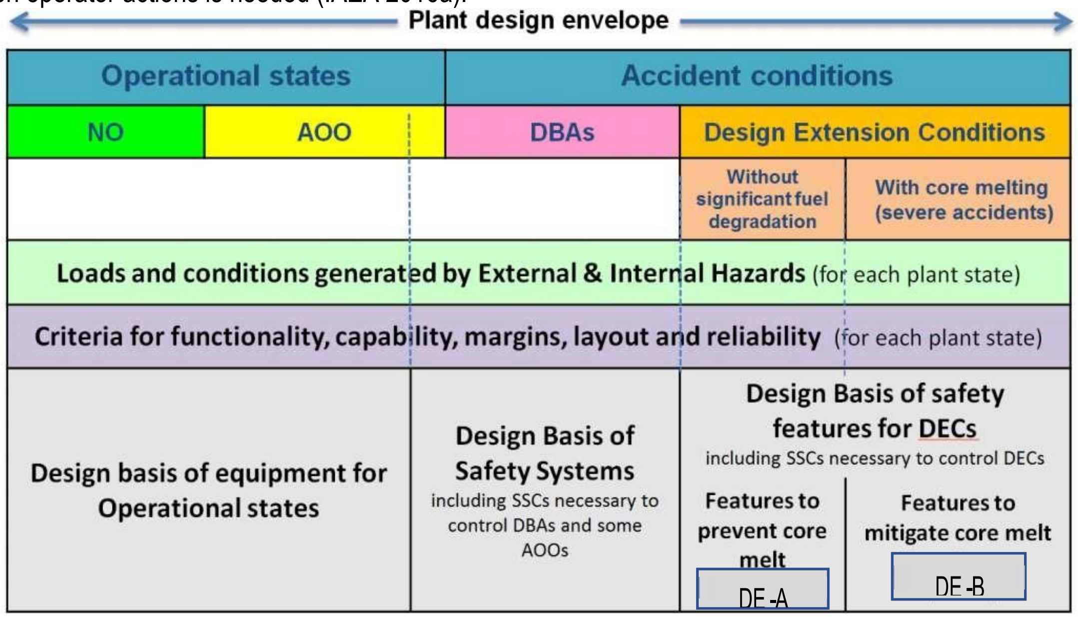


Figure 1: Plant design envelope of a NPP (IAEA 2016)

One way to classify the different phases of the plant design is the frequency of their occurrence (see Tab. 1) (IAEA 2016a).

Table 1 Expected Frequency of events in each phase of the plant design (IAEA 2016)

Plant state	Indicative expected frequency of occurrence
Normal operation	-
Anticipated operational occurrences	$> 10^{-2}$ events per year
Design basis accidents	$10^{-2} - 10^{-6}$ events per year
Design extension conditions without significant fuel degradation	$10^{-4} - 10^{-6}$ events per year
Design extension conditions with core melt	$< 10^{-6}$ events per year

The analysis of DEC-A scenarios is of relevance. This means that accident scenarios are considered that exceed DBA scenarios and address core melt prevention. However, scenarios that consider severe accidents, where the mitigation of core meltdown is necessary (DEC-B), are not addressed.

2.2. Accident Management (AM) - Measures

The definition of AM as reported in the IAEA Safety Report (IAEA 2002) states that "Accident management is the taking of a set of actions during the evolution of a beyond design basis accident:
(1) to prevent the escalation of the event into a severe accident, (2) to mitigate the consequences of a severe accident, and (3) to achieve a long term safe stable state".
As the condition of the reactor gets more critical, the use of intervening AM becomes more necessary. During the operational states (NO and AOO), AM measurements are normally not used, but for DBA and DEC scenarios the necessity arises where the operator uses specific actions to prevent further harm to the reactor or the environment. Many plants have already installed an operator support system that is related the use of symptom based EOPs. Depending on the application, the system can have different names such as the Safety Parameter Display System (SPDS), the critical function monitoring system (CFMS) or simply the operator support system. These display systems have capabilities to be helpful throughout a severe accident on the condition that the input from the plant instrumentation and computer system is available and reliable. When using this system during AM training, the availability and reliability in severe accident situation should be taken into consideration (IAEA 2002).
Since in the event of an accident an analysis of the current situation must be carried out first. AM measures are usually initiated only 30 minutes after the incident occurred. A just a selection of AM measures are (Alessandro Del Nevo et al. 2006):

-
- Opening/closing of valves at the Primary Side (PS) like the Power Operated Relief Valve (PORV) at the Pressurizer (PRZ)
- Opening/closing of valves at the Secondary Side (SS) like the Atmospheric Relief Valves (ARV) at the top of the Steam Generators (SG)
- Activation/deactivation of active systems of the Emergency Core Cooling System (ECCS) like the High-Pressure Injection System (HPIS), the Low-Pressure Injection System (LPIS) or the Make-UP System
- Activation of the Containment Spray System.

In order to be able to prevent or mitigate an accident Emergency Operating Procedures (EOP) and Severe Accident Management Procedures (SAMG) are defined at the reactor. Furthermore, the IAEA has created a safety standard called Defence in Depth (DiD), that also addresses the prevention and mitigation of DBA and DEC scenarios. Definitions are hereafter reported concerning EOP and SAMG (see Figure 2).

- EOP: set of documents describing the detailed actions to be taken by response personnel during an emergency. The main priority of EOP is to prevent fuel damage, therefore, the plant specific procedures contain instructions to operating staff for implementing preventive accident management measures for both DBA and DEC (Alessandro Del Nevo et al. 2006).
- SAMG: set of guidelines containing instructions for actions in the framework of severe accident management (SAM) where SAM is a subset of AM measures that: a) terminate core damage once it has started, b) maintain the capability of the containment as long as is possible, c) minimize on-site and off-site releases, d) return the plant to a controlled safe state (Alessandro Del Nevo et al. 2006).

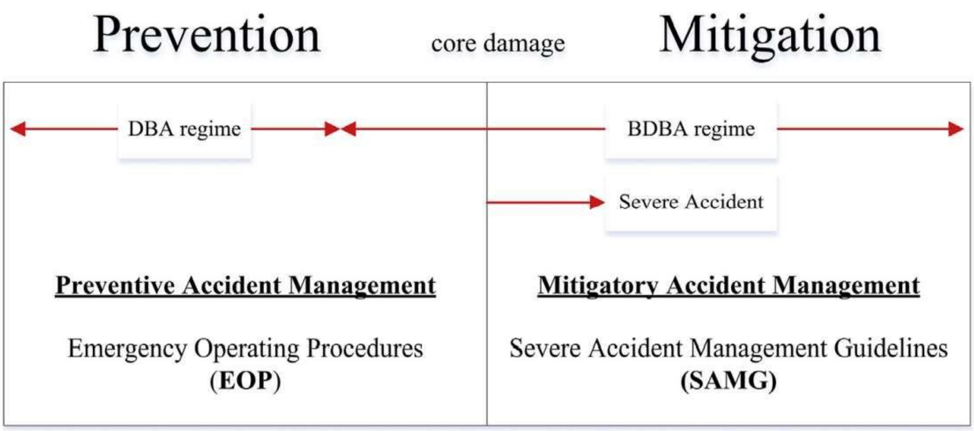


Figure 2: EOPs and SAMGs application domain in accident management (Saghafi and Ghofrani 2016)

The Defence in Depth concept was defined in SSR-2/1 as a fundamental and overarching principle of nuclear safety for preventing accidents and mitigating their consequences. It consists of 5 levels. Level 1 has a predominant preventive function and level 5 has only a mitigatory function. More detailed information about the levels is presented in Table 2. There are two approaches to integrate the DEC-A/B scenarios in the defense in depth concept. Some member states add DEC-A into level 3a and DEC-B into level 3b. In the second approach it is defined that in level 3 the postulated set of DBAs is addressed. DEC-A and DEC-B are both sorted in level 4a and level 4b respectively (IAEA 2012a).

Table 2: Defense in Depth – Levels (IAEA 2016a)

<table>
<tr><th colspan="2">Level of defence
Approach 1</th><th>Objective</th><th>Essential design means</th><th>Essential operational means</th><th>Level of defence
Approach 2</th></tr>
<tr><td colspan="2">Level 1</td><td>Prevention of abnormal operation and failures</td><td>Conservative design and high quality in construction of normal operation systems, including monitoring and control systems</td><td>Operational rules and normal operating procedures</td><td>Level 1</td></tr>
<tr><td colspan="2">Level 2</td><td>Control of abnormal operation and detection of failures</td><td>Limitation and protection systems and other surveillance features</td><td>Abnormal operating procedures/emergency operating procedures</td><td>Level 2</td></tr>
<tr><td rowspan="2">Level 3</td><td>3a</td><td>Control of design basis accidents</td><td>Engineered safety features (safety systems)</td><td>Emergency operating procedures</td><td>Level 3</td></tr>
<tr><td>3b</td><td>Control of design extension conditions to prevent core melt</td><td>Safety features for design extension conditions without core melt</td><td>Emergency operating procedures</td><td rowspan="2">4a
Level 4
4b</td></tr>
<tr><td colspan="2">Level 4</td><td>Control of design extension conditions to mitigate the consequences of severe accidents</td><td>Safety features for design extension conditions with core melt.
Technical Support Centre</td><td>Complementary emergency operating procedures/ severe accident management guidelines</td></tr>
<tr><td colspan="2">Level 5</td><td>Mitigation of radiological consequences of significant releases of radioactive materials</td><td>On-site and off-site emergency response facilities</td><td>On-site and off-site emergency plans</td><td>Level 5</td></tr>
</table>

2.3. Optimization of AM procedures with the Simplex method

The Simplex method allows the optimisation of the selection and usage of accident measurement procedures that are available for an operator during a specific transient scenario. This is particularly relevant because operator interventions mainly play a role in DEC-A (and DEC-B) accidents. In the past, this approach was applied for LOCA and SBO scenarios, among others (Nikolaus Müllner et al. 2005); (Muellner et al. 2007).
In the field of computer approaches to optimise accident management strategies a basic distinction can be made between event-base procedures and symptom-based procedures. The Simplex method can be classified as an event-based procedure (Muellner et al. 2007). This method was first developed by Nelder and Mead in 1965 to allow the minimization of a function with n variables. It compares the function values at (n+1) vertices of a general simplex, which is then continued by replacing the vertex with the maximum value by another point.
The Simplex method is capable of adapting to the local landscape and contracts on to the final minimum (Nelder and Mead 1965). According to Haulin (2014) it has several advantages over stochastic methods as it does not get trapped in local optima, quickly determines whether a set of constraints is infeasible and can therefore produce an approximation of how close to the global optimum a found solution is (Haulin 2014). In the literature this method is described as very robust and according to Kaczmarczyk (1999), the Simplex approach is very effective, if a large number of parameters are utilized. Therefore, this approach can be applied for our current accident management analysis as several parameters have to be taken in consideration. To use the Simplex method effectively, a sufficiently validated thermal-hydraulic code is necessary for the application of the approach. In past studies calculations were conducted with Relap5. In this project BOKU uses Relap5-3D, which is based on the Relap5 code and is extensively validated with experimental data (Davis 2018). Therefore, we assume that the use of the Simplex method can be justified. However, one disadvantage of the method is, that it requires a long time for the calculation. This is also the main reason this approach cannot be used as a symptom -based method.
<u>Application of the Simplex procedure</u>
To apply the Simplex method to an optimisation problem regarding accident management tools, the following four steps have to be conducted (Muellner et al. 2007):

1. Identification of AM-parameters, which define the accident management procedure (non-dimensional, if possible)
2. It is necessary to specify which safety systems are available. This could be either done via expert judgement or probabilistic approach
3. Identification of critical safety barriers/functions. Indication of the state of the safety function as parameters (if possible, non-dimensional)
4. Creation of an ideal diagram, which allows the analytical investigation of the effects of the procedure. The results are summarized e.g. using an ideal diagram of the PS pressure and the temperature

The aim of the analysis is to find a functional dependency between 1.,2. (independent variables) and 3. (dependent variable). The interaction of the components of the procedure are shown in Figure 3.

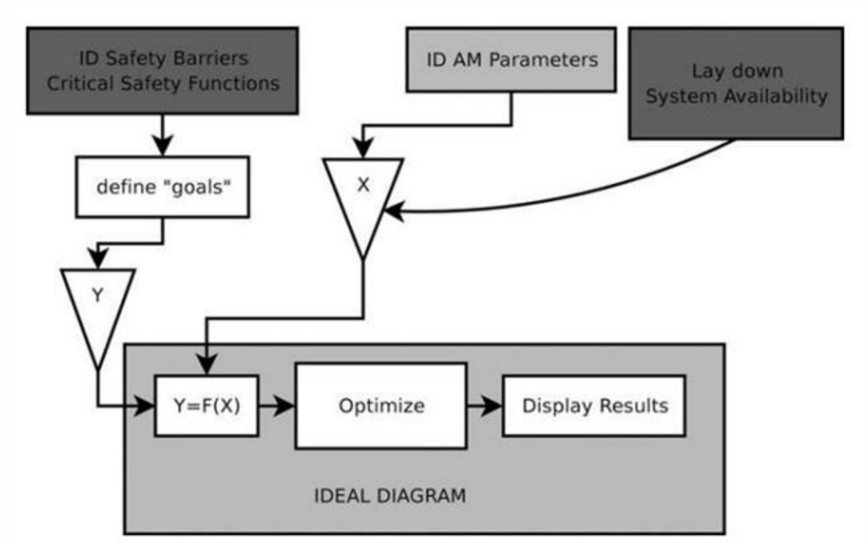


Figure 3: Schematic overview on the Simplex method

Examples for independent/dependent variables:
Dependent variables:

- Time duration until pressure / temperature / water level in core reaches certain threshold.
- Time duration until breakflow reaches certain threshold
- Time duration until iodine activity in environment reaches a certain threshold.

Independent variables:

- Active AM systems (high pressure injection system, low pressure injection system, opening of valves, etc.)
- Passive AM systems (hydro-accumulators in primary side)

Weighting of variables of the objective function
The simplex procedure allows the selection of more than one dependent variable. However, a weighting has to be applied regarding the importance of each parameter. This is conducted via expert judgement. If an optimisation model contains the following dependent variables:

1. “time until temperature in core cladding exceeds 1200 °C” and
2. “time until water level in core is below 0.5 m”,

with the main objective to search the best strategy to prevent dry out of the core, a proper weighting might be: 1. = 0.25; 2. = 0.75.
Expected effect on radiological consequences
The simplex method allows to optimise the operator’s strategy during the transient. Therefore, it is capable of selecting the optimal activation/deactivation of AM measurements to reduce radiological consequences, if the dependent variables and their weightings are chosen accordingly.
The approach was applied to mitigate the impacts of the iodine spike phenomenon on the environment. This can easily be accomplished by the simplex method. Furthermore, it is possible, to create a strategy which at the same time minimises other important parameters, like the PS pressure, as well.

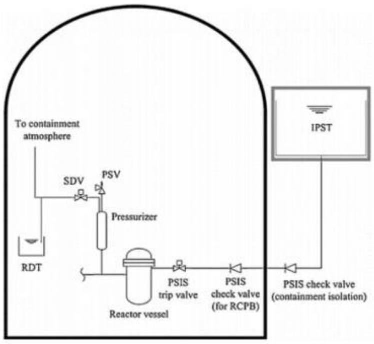
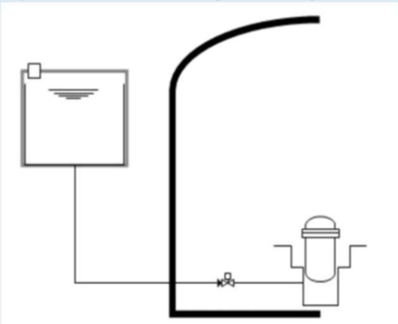
3. Devices

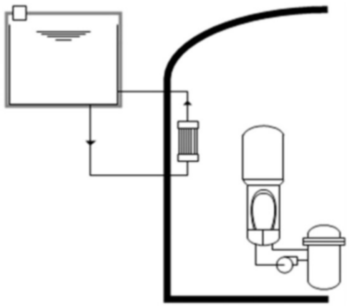
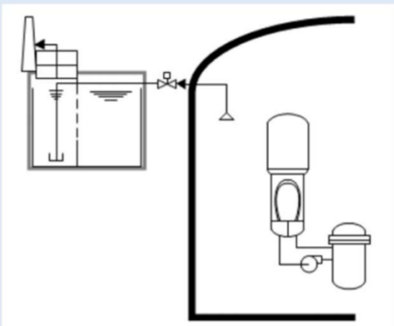
Various technical devices exist to support operation safety and accident management. The following scheme gives a general overview of AM devices and technical settings of these devices, as far as devices often can only be implemented in combination. It includes all AM devices, that serve preventive measures (DBA or DEC-A) as well as mitigative measures (BDBA or DEC-B).

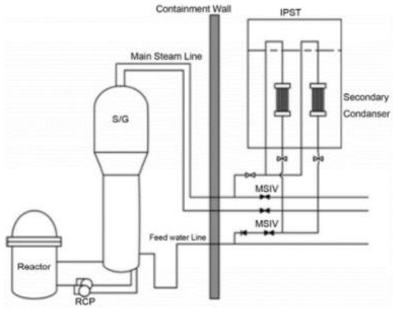
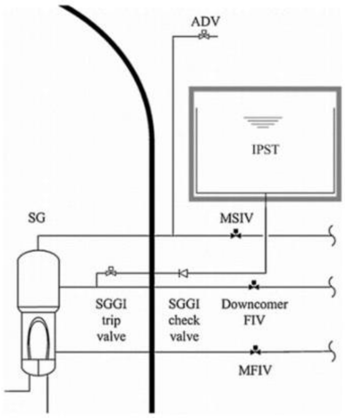
Device	Aim (DID regime)	Description	Literature sources
PAR (Passive Autocatalytic Recombiner)	binding of H_2 (preventive/ mitigative)	Metal sheets (surface consisting of ceramic washcoat with catalytic material – e.g., palladium or platinum) forming vertical flow channels For the problem of overheating due to the exothermic reaction process and innovative design changes see (Agrawal et al. 2017; Reinecke et al. 2004) For the effect of CO on the H_2 absorption of PARs see (Klauck et al. 2014) For modelling PARs in RELAP, TRACE and CORSAR see (Avdeenkov et al. 2018)	(Agrawal et al. 2017; Avdeenkov et al. 2018; Kelm et al. 2009; Klauck et al. 2014; Reinecke et al. 2004)
HTIRT (High Temperature Irradiation Resistant Thermocouples)	monitoring and evaluation (preventive/ mitigative)	With the higher temperature resistance (up to 1450 °C) of HTIRTs monitoring of temperature development of relevant system zones can be improved. On the problem of overheating CETs (Core Exit Thermocouples) see (Hashemian and Jiang 2009; Suzuki and Nakamura 2010)	(Hashemian and Jiang 2009; Palmer et al. 2020; Suzuki and Nakamura 2010)

FCVS (Passive Filtered Containment Venting System)	Pressure reduction and reduction of transfer of radionuclides to the environment (mainly mitigative)	Regulates the pressure inside the containment and guarantee a significant reduction of radionuclides leaving the containment. Two basic types: a) dry solid filters and b) liquid scrubber filters (e.g., submerged venturi filter see below under PFVS) (IAEA 2017) 'Innovative' FCVS measures to filter radionuclides that are currently studied on an experimental level (e.g., PASSAM-project). They are supposed to be used as pre-conditioners to filters for? standard filters: **i) Acoustic Agglomeration Systems** (Albiol et al. 2018) **ii) High Pressure Spray Agglomeration Systems** (Albiol et al. 2018) **iii) Electrostatic Precipitators** (Albiol et al. 2018) **iv) Improved Zeolites** (Albiol et al. 2018) On the interactions between FCVSs and raise in hydrogen under SA circumstances see (Ma et al. 2018)	(Albiol et al. 2018; Ali et al. 2013; Bal et al. 2018; Goel et al. 2018; IAEA 2017; Ma et al. 2018)
ATF (Accident Tolerant Fuel)	raising operation safety (preventive/ mitigative)	Alternatives to the classic UO_2-Zr system to improve operation safety under DBA and BDBA conditions. Among approaches two opposite examples were selected: a) substituting Zirconium by Si-Si composites (Deck et al. 2015) and b) improving the UO_2 compound by mixing it with UN in order to raise operation safety (Costa et al. 2020).	(Costa et al. 2020; Deck et al. 2015; IAEA 2016b)

The following devices are rather different variants of one basic device: integrated safety water tanks used to set up an integrated safety water system (combination of valves and tanks, that function passively). Most of these sets can be used to reach natural circulation.

Device	Aim (DID regime)	Description	Literature sources
PSIS (Passive Safety Injection System)	Cooling/ temperature reduction (preventive/ mitigative)	PSIS injects water directly to the reactor vessel or into the cold leg of the primary side, see Fig. 3. Three valves are necessary: reactor isolation valve, containment isolation valve and one valve to control the flow direction.  *Figure 3 Passive Safety Injection System (source (S. H. Kim et al. 2015))*	(Chang et al. 2013; IAEA 2005a, 2009, 2012b)
PIVR (Passive in-vessel retention and cavity flooding system)	Cooling/ temperature reduction (natural circulation) (mitigative)	Analogous to PSIS P-IVR uses cavity flooding to reduce the reactor vessel temperature by external cooling, see Fig. 4.  *Figure 4 Passive In-Vessel Retention Flooding (source (Chang et al. 2013))*	(Chang et al. 2013; IAEA 2005a, 2009, 2012b)

PCCS (Passive Containment Cooling System)	Cooling/ temperature and pressure reduction (natural circulation) (mainly mitigative)	PCCS uses a closed loop via the passive water tank and a condenser to reduce temperature and pressure, see Fig. 5. The valve opens at a specific pressure level and initiates natural circulation.  *Figure 5 Passive Containment Cooling System (source (Chang et al. 2013))*	(Chang et al. 2013; IAEA 2005a, 2009, 2012b)
PFVS (Passive Filtered Venting System)	Pressure reduction/ Filtering the radionuclides to reduce contamination of environment (mainly mitigative, but preventive as well)	Submerged venturi scrubber in combination with steam/moisture separator and dry fiber/ charcoal filter see Fig. 6 (Chang et al. 2013)  *Figure 6 Passive Filtered Containment Venting System (source (Chang et al. 2013))*	(Chang et al. 2013; IAEA 2005a, 2009, 2012b)

PDHR (Passive Decay Heat Removal)	Cooling/ temperature reduction (natural circulation) (preventive/ mitigative)	Two types of PSIS: a) using condensers to achieve natural circulation in the tank-SG loop, see Fig. 7.  *Figure 7 Passive Decay Heat Removal (source (Chang et al. 2013))* b) a simpler approach that functions analogously to PSIS with the difference of contributing water to the SG (secondary side), see Fig. 8.  *Figure 8 Passive Decay Heat Removal (source (S. H. Kim et al. 2015))*	(Chang et al. 2013; IAEA 2005a, 2009, 2012b; S. H. Kim et al. 2015)
PINCs (Passive In-Core Cooling System)	temperature reduction (preventive/ mitigative)	Hydraulically driven control rods (HDCR) in combination with a hybrid control rod assembly that combines heat pipes with the control rod are used to passively transport heat from the active core to a natural convection loop at the top of the reactor core. The rods are controlled by water flow and gravitational force.	(Jeong et al. 2015; I. G. Kim and Bang 2017)

A detailed description of exemplary preventive AM is given in (Hosseini et al. 2020). This article suggests the usage of two water tanks on the primary side (PSIS) and one deaerator tank on the secondary side (PDHR) for VVER1000 to avoid BDBA conditions.

4. Technical Approaches

Various computerized tools and expert systems for accident management and source term predictions are utilized by the operators and national emergency teams to improve the reaction in an accident scenario. The tools are divided in two groups:

1) Evaluation of existing AM strategies
2) Analytical tools
   a. Analysis of transient (Neuronal networks/ Fuzzy / Expert knowledge)
   b. Establishment of AM-strategy

Table 3: Types of technical approaches (Saghafi and Ghofrani 2016)

Methods	Weaknesses	Abilities
Artificial neural networks	- Not interpretable and transparent to operator (*black box* structure) - Operator out-of-the-loop problems - Difficulty in training fast accidents to feed forward neural networks - Hard training of time dependent data in recurrent neural networks	- Suitable method for pattern recognition problems - Non-linear systems modeling capability - Useful with time depended data - Quicker faultfinding by pattern recognition ability
Fuzzy logic	- Many rules and complicated structure for covering enough transients - Time consuming process of tuning membership function with genetic algorithm	- Robustness in noisy domain - Interpretable rules (white box structure) - Unlabeled accidents identification
Expert systems	- Qualitative results in accidents identification (SB-LOCA, LB-LOCA, ...) - Rules extracted from human experts with different fields and experiences	- Possibility of using rules in EOPs and SAMGs - Hierarchical structure and easy to understand reasoning process (system transparency) - Unlabeled accidents identification - Avoiding problems from keeping the operator out-of-the-loop
Hybrid methods (Neuro-Fuzzy)	- Need to combine several Neuro-fuzzy networks for nonlinear problems	- Partially interpretable (*grey box* structure) - Robustness in noisy data domain - Powerful in separation of accidents with similar parameters trend

Four different types of this approaches are described in Table 3. Furthermore, a few examples of such systems are presented in the following scheme.

Device	Aim (DID regime)	Description	Literature sources
SIMPLEX	Evaluation of AM strategies	The simplex algorithm originally developed by Dantzig and its variants are widely used to solve LP problems. Basically, starting from an initial feasible solution, the simplex algorithm tries to produce an improved solution at each iteration, maintaining feasibility until optimality is reached. Although this algorithm was developed to solve LPs, it is also used to solve linear relaxation of mixed integer problems (MIPs) in many heuristics and exact approaches. Faster simplex algorithms are beneficial, since a lot of processing time is spent on solving linear relaxation.	(Boyer et al. 2017)
ADAM	Analysis and conclusions	Analysis of selected plant data after accidents to arrive at symptom based diagnostics of potential nuclear accidents. Operates in 2 modes: 1) on-line accident monitoring/ diagnostic: Arriving parameters used to assess margins to core damage, containment failure, vent actuation and hydrogen combustion. 2) Accident Management and simulation mode: Simulation of various operational and severe accident scenarios to determine potential impact of available severe accident management strategies.	(IAEA 2005b)

ALADDIN	Classification of transients in NPPs	ALADDIN is designed to process reactor specifically formatted simulation data. Uses a combination of fuzzy clustering and recurrent ANNs. Modular ANN architecture provides "don't know" capability for recognition of an unknown event. Developed as part of the OECD Halden reactor project and is used for identification of simulated events in a 900 MW PWR. It was able to successfully identify 7 trained transients and 4 test transients with using 5 process variables.	(Darling. et al. 2015)
SEVERA	Analysis and conclusions	This application was created in the context of the EUH2020 project NARSIS. It offers support for the decision-making team during the course of an accident. The system assesses the plant damage state, predicts possible accident progressions and assesses available management actions and their consequences. A monitoring and a radioactive-release assessment module are implemented. The methodological approach primarily relies on qualitative rule-based multi-criteria models, but also includes other techniques e.g., data analysis, probabilistic safety assessment and event-tree modelling.	(Bohanec et al. 2020)
SESAM/CONRAD	Analysis and conclusions	Combined approach: these programmes were developed by IPSN (now IRSN). Sesam is an application with the capability to evaluate the status of the plant and of potential releases. Parameters such as the break size or potential fission product release within or outside the plant are quantified. In addition, CONRAD is used to simulate the evolution of the accident and asses the current and future availability of the safety systems and the functionality of safety barriers.	(IAEA 2005b)

Mars	Analysis	The application MARS uses fuzzy logic in the tracker for accident identification, break size estimation and fault location determination. Application of MARS has	(Iguchi, Yukihiro et al. 1999; Saghafi and Ghofrani 2016)

5. Machine Learning and Artificial Intelligence in the context of AM

A promising (but challenging) approach to analyze and reduce the radiological consequences of accidents (of DBA/ DEC-A transient scenarios and beyond) is machine learning (ML). Only a brief description of the prerequisites and possibilities of ML will be given here to complete the picture, because this technological field is very complex and in a continuous development.
ML can be interpreted as one of the answers of how to cope with the excessive accumulation of data in the age of digitalization. So, it can be defined "as a set of methods that can automatically detect patterns in data, and then use the uncovered patterns to predict future data, or to perform other kinds of decision making under uncertainty (such as planning how to collect more data!)" (Murphy 2012, p.1). For a detailed analysis see (Goodfellow et al. 2016; Murphy 2012).
The main types of ML are:

a) supervised
b) semi-supervised
c) unsupervised
d) and d) reinforced learning.

The differences of these learning approaches depend on the input data itself. So, the quality/ availability and quantity define which of the above-mentioned types can be applied.
In general, a ML algorithm can be summarized by the following equation:
Y (Output) = f (X (Input)) + delta (error)
Important is the relationship between Input (features), Output (labels) and Function (estimate function). In case of supervised learning the correlation between Input/ Output data is complete, so that the estimate function relies on pairs of corresponding features and labels. For the opposite case unsupervised learning due to lack of information only features are available, so that ML searches 'blindly' for patterns (labels and functions that correspond to a set of features). In the end the unsupervised learning is more complex and challenging, but generates 'new', that is additional information and knowledge. Semi-supervised learning is situated between these two opposites. Along semi-supervised learning the correlation between the feature-label pairs is not complete, so that partly features without a corresponding label are available. Reinforced learning inverts the relationship of features (input) and labels (output), so the features become the output based on a specific dataset of labels (Xu and Saleh 2021).
As nuclear power plants are large-scale and complex systems that rely on highest safety standards and therefore produce a large amount of monitoring data, it makes sense that ML algorithms are and can be applied in the field of nuclear safety and accident management as well, as ML gives the opportunity to extract specific data in order to analyze and interpret it (in short time) (Elshenawy et al. 2021).

But ML is inherently dependent on several preconditions:

a) the result of ML cannot surpass the quality of the applied input data
b) high quantity of available input data (monitoring data/ measurements from research reactors like the VVER-PSB) is necessary (quantity increases prediction accuracy)
c) ML is resource-consuming because "Machine learning problems become exceedingly difficult when the number of dimensions in the data is high. This phenomenon is known as the curse of dimensionality. Of particular concern is that the number of possible distinct configurations of a set of variables increases exponentially as the number of variables increases." (Goodfellow et al. 2016)

ML is based on automation processes and if the learning algorithm is corrupt or data are not qualified, the reliability of ML is not guaranteed and should not be applied. The application of ML is only possible if the fundamental failure mechanisms of ML are addressed and considered.
For a detailed description of possible applications of ML for accident management see (Chung 2021; Lee et al. 2021; Li et al. 2021; Mena et al. 2022; Vicente-Valdez et al. 2021).

One way to detect incidents more quickly and prevent damage could be a tighter monitoring network in the primary and secondary circuit. The tighter network of measuring points in the power plant would make it possible to react more precisely and quickly to the processes in the power plant and thus to deal with transients better. A combination of improved measurement networks with the machine learning tools could also offer significant improvements in the accident management of nuclear power plants. The main focus of accident management in VVER reactors is on depressurisation of the steam generators through the pressure relief valves BRU-A and BRU-K and pressure relief through PORV in the primary circuit (Alessandro Del Nevo et al. 2006). Since any high-pressure reactor pressure vessel failure leads to major fission product releases and possible containment damage, the main focus has to be on depressurisation. A tighter measuring network inside the primary circuit could therefore help with fighting abnormal conditions inside the reactor more quickly. A possible use case would be the implementation of redundant measuring networks in every hot and cold leg of the primary circuit. In case of severe accidents with (partial) core meltdown, it is very important to relieve the primary side to avoid a high pressure melt ejection of the corium (Mohsendokht and Jamshidi 2021). Additional measuring probes could be installed in the primary circuit i.e., at the time of the annual revisions.
As cost is also always an important consideration in the nuclear industry, innovative methods and tools that make the best use of existing instrumentation are preferable. Here, too, a combination of improved instrumentation and the use of machine learning could play an important role. A more accurate measurement network would probably incur high costs, which is why implementation without a demand from the regulatory authorities or IAEA is rather unlikely.
Some general advantages of the Artificial intelligence (AI) use in the management of a NPP are:

- AI capability to analyze each specific plant state in accident or normal operation conditions without any necessity to perform analysis by grouping or bounding similar possible conditions of the plant (or similar plants) if enough amount of data are available. The advantage is to have a more detailed and more accurate analysis of the plant not “dispersed” in some bounding cases. A more adaptive and suitable analysis is potentially possible
- AI makes possible a more accurate interpretation of the data and a more effective reaction to the conditions of the plant
- AI offers opportunities to maximize the amount and applicability of information extracted from experimental and simulation data
- AI can help manage resources in design/optimization/update of a NPP, processes that need large amounts of data and AI is becoming relevant to obtain interpretable results

On the other side the capability of the AI is not in the direction to be like the human logic and reasoning approach. The main difference is constituted by the impossibility of AI to make predictions and assumptions if not enough data are available, that is a fundamental capability of human approach. This makes less useful the use of AI in situation when a decision must be taken based on availability of insufficient or not typical data.
The AI current development is based on to the analysis of large amount of data and to recognize data patterns. However, this features associated to the continuous computer power increase, makes the AI powerful and useful. Notwithstanding the advantages, the AI capability to generate interpretations from data patterns is not directly connected to the mechanistic reality of the phenomena occurring into a NPP. Two main drawbacks are relevant in applying AI:

- There is not a real simulation (understanding) of the physic of the phenomenon. AI only makes possible to recognize important aspects and relationship between data recorded. About this aspect it is important to note that the recorded data must be comprehensive of the phenomenon, or the data could generate a misleading pattern.
- The AI interpretation is possible only for the phenomena and related conditions for which data are collected. AI predictions for different conditions not related to the available data should be considered with extreme attention because are only based on data referred to other conditions.

6. Test-Facilities Considerations

The progression of the accidents included in the DEC-A and DBA are in the framework of Thermal-hydraulic (TH) framework: the evolution of those accidents is a sequence of TH phenomena. Nowadays almost all the TH phenomena are reproduced and studied in the test facilities. Test facilities can be subdivided in Integral Test Facilities (ITF) and Separate Effect Test Facility (SETF). ITF reproduce the complete behavior of a plant and the SETF reproduce one or a group of connected TH phenomena typically taking place in a zone or a specific component of the plant.

The use of the results of these facilities are particularly relevant in the development of the TH models of the codes and in the process of code validation. Typically, in the code model development, SETF are used, because are specific. Some TH phenomena and some ITF results are also adopted to check the integration of all the models included in the code. In the validation process ITF results are used to check ability of the code to correctly reproduce the phenomena related to the behavior expected in the plant.

Concerning the validation process of the code a list of available tests has been set up related to the results obtained also indicating the intrinsic capability of the facility to reproduce the TH phenomena and the capability of the specific performed test in a facility to characterize TH phenomena.

The information is represented in some matrixes linking the capability of the facility and of the test, the kind of TH phenomena reproduced and the relevance of the test results for the different kind of plant. The experiments performed are related for the larger part to DBA and the facilities are reproducing the Reactor Coolant System (RCS).

In addition to the phenomena occurring in the RCS it is necessary to simulate and to evaluate the phenomena occurring in the containment. This is necessary because the DEC-A accidents have an evolution that typically involves RCS and the containment. Here the phenomena in the containment are for DEC-A in the field of TH. In addition, the prevention and/or the limitation of the Fission Products (FP) release in the environment is obtained by simulate the release of FP from the core to RCS, to containment, and from the containment to the environment. It is important to note that the TH phenomena occurring the containment typically are not the same of the TH phenomenal occurring in the RCS or they occurs in different modalities. The main reason is the different geometry. As an example, the main heat exchange between surfaces and coolant is typically related to rods in the core while in the containment the heat exchange could be related to the external surface of the vessel with the water injected in reactor pit.

For this reason, experiments relevant for DEC-A can be subdivided into main sections: one is related to RCS (that are the ones also used for DBA) the others are about the simulation of the containment behavior. Test facilities including both the RCS and the containment are not common. Generally, RCS and Containment phenomena are investigated in different test facilities. In addition, the kind of quantities to be recorded, and the related sensors, in the simulation of the accident are different between RCS and containment. As an example, the RCS of the test facilities generally have volumes having a dominant geometrical dimension while the containment is characterized by large rooms characterized by three dimensional volumes.

Some test facilities exist including RCS and containment models. But in generally the containment is no more than a simplified simulation of the containment and it is typically a sort of boundary conditions for the RCS, without any purpose of containment analysis.

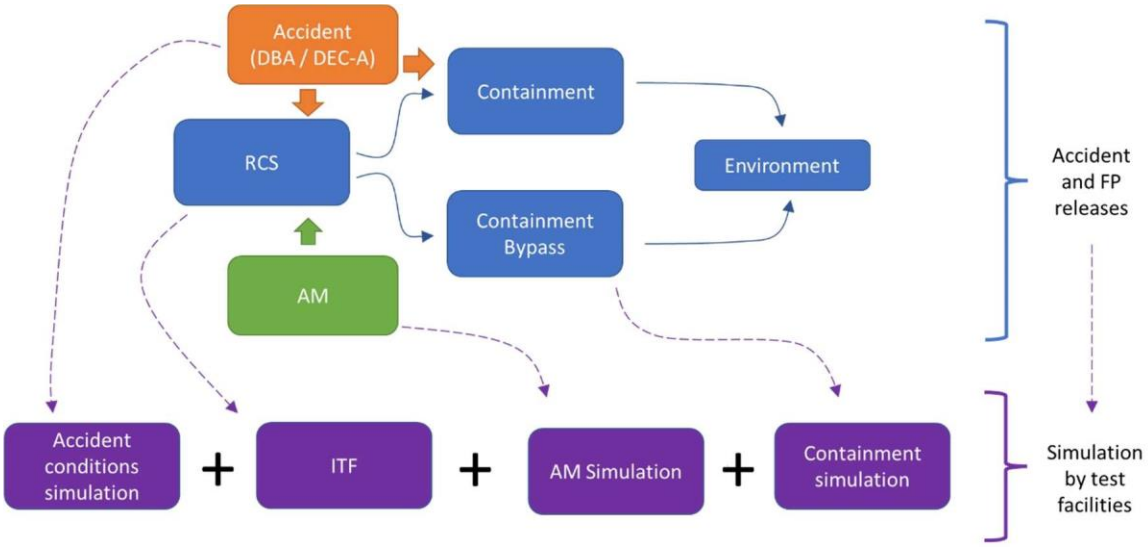


Figure 4: Scheme of the use of experimental facilities for accident and phenomena analysis

In the Figure 4 is reported the use of experimental facilities for accident and phenomena analysis. The path of the FP releases for DBA and DEC-A implies the RCS and the containment involvement. In addition, the accident management must be also considered. Therefore, the facility should have the capability to reproduce the accident conditions, the main aspect and characteristic of the plant, the capability to simulate the accident management procedures and the containment simulation. This last aspect can be neglected if the containment bypass takes place during the accident.

7. Optimization of SGTR Management procedures in VVER 1000

7.1. Assessments of the Accident Management Strategy and algorithms for SGTRs in VVERs 1000

This work was performed by ARB during the project and describes the results of their development of the Accident Management Concept (AM) for managing accidents with leakage from the primary to the secondary circuit (PRISE) for a pressurized water reactor of the VVER 1000 type.
It concerns the implementation of an automatic Algorithm for identifying and managing a PRISE accident. The results of the algorithm operation are compared in terms of accident progression and releases into environment with the final reactor calculation results of SGTR design basis accident (DBA) scenarios and type A design extension accidents (DEC-A) scenarios obtained within task 2.5 using the updated calculation schemes.
The primary to secondary leak accident is one of the most complex and specific accidents for the VVER type Reactor Unit. Design operation of Unit automatics and systems does not allow to reach a safe stable condition without actions of the plant personnel. The time of the beginning of the release of the FP into the environment is depending on the diameter of the break.
16 different kinds of scenarios were analysed scenarios for the calculation justification of the emergency management algorithm, investigating various leak diameter from 13 to 100 mm (incl. SG tube rupture and collector lift-up). It includes emergency scenarios of leakage from the primary circuit to the secondary with the imposition of failures of individual safety functions, failures of algorithm elements and the imposition of additional IEs.
An automated approach to the management of an accident with an inter-circuit flow of the coolant requires the implementation of a set of actions aimed at localizing the failed Steam Generator and transferring the power unit to a stable safe state without the activation of the secondary steam Dump Valves or with the minimum number of activation cycles (and the minimum amount of FP release) in the case of additional equipment failures.
The list of automatic actions, the sequence and time of their execution, as well as the values of the setpoints of the primary and secondary parameters are selected based on a preliminary calculation analysis of various emergency management options, including cases with equipment failures. The effectiveness of the actions and the correctness of the selected intervention time and the used setpoints are confirmed by the results of the calculation justifications.
The VVER-1000 reactor model of the ATHLET 3.2 code was used to perform the thermal hydraulic calculations. The model implements the operation logic of the power unit equipment according to the accident management algorithm, as well as the failure of the corresponding elements of the algorithm, the failure of individual safety functions and imposition of additional initiating events.

Application of the developed Algorithm makes it possible to eliminate or minimize the FP release from the primary circuit to the environment through ESG SDV, to ensure the stabilization of the emergency process and to create a sufficient reserve of time for operational personnel to perform further emergency management actions.
The results of calculation analyses demonstrate the effectiveness of accident management both during the design course of the emergency process and in cases of imposition of additional system and equipment failures, failure of individual elements of the algorithm itself, or imposition of additional IEs (Initiating Events).
More especially the algorithm ensures the fulfilment of the main and auxiliary criteria (i.e. prevention of core damage, absence of steam dump valve of the failed steam generator opening or minimum number of actuation cycles of this valve….), as confirmed by the corresponding calculation analyses, for most emergency scenarios analysed, excepted those emergency scenarios in which the failure or degradation of a certain safety function or element is postulated as part of the emergency scenario itself. In such cases indeed, some of the analysed criteria are fulfilled only partially. At the same time, the operation of the algorithm also has a positive effect since the time reserves are significantly increased for the personnel to perform the correct actions prescribed by the EOPs (Emergency Operating Procedures).

7.2. Safeguard algorithm for Steam Generator cover lift-up in VVER-1000/V320

The possibility to introduce an automatic safeguards algorithm to cope with the primary to secondary breaks similar to the one that is implemented in Gen III+ VVER NPP design at older VVER-1000/V320 NPPs is evaluated based on quantitative deterministic analysis of SG collector cover lift-up DBA. This initiating event results in containment bypass with direct release of radioactive coolant to the environment, and without timely implementation of operator recovery actions leads to a depletion of available ECCS water inventory quickly evolving into DEC B accident with severe nuclear fuel damage. Though several different designs of the algorithm are already proposed for VVER-1000/V320, they to a large extent involve normal operation systems.
The analysis of VVER-1000 automatics design has shown the large delay for prompt operator actions in case of PRISE for optimal mitigation. This was because of need of multiple manual actions on disabling interlocks and shifting setpoints. So, the main idea of the proposed algorithm is to automatize the needed prompt actions not degrading the main unit safety functions till the unit stabilization (meaning no leak to environment, stable core cooling, stable secondary pressures, and stable primary/secondary inventory).
The intent of the study is to analyse the possibility to propose algorithm which utilizes primarily the safety systems with no or minimal use of normal operation systems and/or need for operator interventions. The effectiveness of the developed algorithm being then evaluated for a most limiting case in terms of primary to secondary break size and initial reactor power, namely for the SG collector cover lift-up case at full power operation using the RELAP5 system thermal hydraulic code.
The method of automatization is targeted to:

1. Decrease of RCS pressure to SG’s one to decrease break flow

2. Shutting off the HPSI pumps

3. Keeping the RCS cooling via unaffected SGs by decreased pressure with some margin

4. Isolating the affected SG by the steam and FW lines.

This type of algorithm accounts for the EOIs actions of the unit and some actions from the existing algorithm at the plant. But it has the difference in the way that it relies on mainly safety systems, simpler logics, and shorter time window. The analysis of the accident propagation has shown that the algorithm must have some timeframes for the different actions that are dictated by TH inertia of RCS. All the actions cannot be performed simultaneously.

As the preliminary calculations for SG collector head break have revealed that the affected SG is full filling in the time frame of 250-400s depending on the loss of power assumption, the main actions of HPSI shutting off should be made before 400s and is chosen around 300s from the algorithm start. Start of algorithm was out of the scope of this work. In this analysis it was assumed that the affected SG is properly chosen by the steam line activity sensors or by the operator (SG level/activity increase). The algorithm operation analysis was performed for SG collector break, 1tube, 3tubes, 5tubes and 10tubes rupture accidents with/without loss of power assumption. It has shown stable behaviour for one affected SG. For further implementation the proper equipment failure analysis should be performed.

The analysis and verification of the algorithm efficiency was performed using via RELAP5 code boron tracing model used in the updated reactor calculations in T2.5 (where boron in the model plays role of the tracer) in which the overall approach of activity transport is based on tracing the RCS water dimensionless concentration. This allows to account for the dilution and mixing in RCS (including fresh ECCS water with tracer concentration 0) and SG volume. To demonstrate its efficiency, a comparison of the results obtained with and without algorithm operation was performed on a SG collector rupture scenario and different SGTR breaks and scenarios (i.e. with or without

Loss-Of-Power) within DBA conditions, (this scenario having been studied previously within the project in T2.5 using the same calculation tool.

For SG collector break the total activity release to environment decreases is significantly reduced (more than a 4). Nearly all release happens during first 2 min of the transient through the SG SV and cannot be avoided for such break. The case with SG collector break without Loss-Of-Power assumption results in no direct release to environment at all. Other smaller breaks (1-10 tubes rupture) release to environment also have smaller values compared to SG collector break. For the multiple SG tubes breaks (1-10 tubes) with electric power available the direct release to environment is not predicted at all with algorithm operation.

The provided analysis lacks more wide algorithm analysis with equipment failures. These analyses, as well as the possibility to optimise the algorithm setpoints and timing though a statistical approach, were beyond the scope of R2CA and will be performed outside the project.

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