The Reduction of Radiological Consequences of design basis and extension Accidents: re-assessment of calculations and main outcomes of the R2CA project

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ABSTRACT

The 4-year H2020 R2CA (Reduction of Radiological Consequences of design basis and extension Accidents) project was intended to propose guidelines to harmonize the safety analysis methods through the development of generic methodologies for best estimate evaluations of the radiological consequences (RC). It was dedicated to Design Basis Accidents (DBA) and accidents in the Design Extension Condition domain without significant fuel melting (DEC-A). The project addresses a broad scope of LWR designs (BWR, EPR, PWR, VVER) from Gen II, III and III+ focusing on bounding scenarios of Loss Of Coolant Accidents (LOCA) and Steam Generator Tube Rupture (SGTR) transients.
The work performed within the project intends to reduce some of the conservatisms currently used in safety studies and/or licensing calculations, which are necessary to help a better quantification and consideration of the potential changes to come in the operation conditions (e.g. increase of fuel burn-up, usage of ATF). In doing so, some potential higher risks exhibited by knowledge improvements (e.g. clad embrittlement through secondary hydriding, increased fission product releases from high-burn-up or MOx fuels, etc.) were also highlighted. Finally, it was beneficial to the development of innovative measures or tools to prevent these accidents (i.e. algorithms, AI based tools, etc.) as well as to strengthen their management.
This article summarizes the main outcomes of the R2CA project presenting the most relevant improvements in simulation tools and calculation methodologies therein. LOCA and SGTR calculation results in DBA and DEC-A domains are discussed. In addition, some insights are given on tools/methods that could be further used to increase the NPP safety based on exploratory innovative work performed within the project. Finally, in conclusion, preliminary guidelines are given for a better estimation of the RC of the considered scenarios.

KEYWORDS
LOCA, SGTR, DBA, DEC-A, Radiological Consequences

1. INTRODUCTION

The 4-year R2CA project (Reduction of Radiological Consequences of design basis and extension Accidents) launched within the NUGENIA network and funded by the European Commission was dedicated to the improvements of methodologies for assessing the environmental radiological source terms of accidents under DBA and DEC-A conditions. It covered two main types of accidental scenarios (LOCA and SGTR identified as the most penalizing ones from PWR level 2 Probabilistic Safety Assessments) and a wide range of LWR concepts (PWR, VVER, BWR). Its main goals were:

- To provide more realistic evaluations of the environmental radiological source terms of these NPPs, then of their associated safety margins under these accidental conditions through improvements in models, simulation tools and calculation chains,
- To increase the NPP safety levels through optimizations of their emergency operating procedures and the development of innovative methods (based on artificial intelligence) for anticipated diagnosis/prognosis of specific accidental situations.

The reason for the project - to focus on DBA and DEC-A conditions mainly raised from the Fukushima Daichii Accident (FDA). In fact, the substantial Research & Development (R&D) efforts made in modelling and source term evaluations of Severe Accidents (SA) after FDA, led to reduce the risks associated to all main categories of SA conditions and consequently also highlighted the overly conservative risk assessments of DBA. These latter, covering uncertainties, unknown or poorly understood phenomena, prevent from anticipating NPP responses for other configurations (i.e. other types of fuel (ATF), fuel with increased burn-up or Pu content…). It also prevents from extending with confidence such evaluations to other NPP concepts such as Small Modular Reactor (SMR) for which realistic radiological source term assessments will be even more crucial due to their potential proximity to homes and population. Furthermore, FDA also highlighted the need to further increase the existing and future NPP safety by including in their design accidental scenarios slightly more severe than DBA and for which specific safety provisions must be implemented to prevent from a further evolution into SA (DEC-A scenarios).
The R2CA project represented a major opportunity for improving models and simulation tools all along the calculation chain used when assessing the radiological source term. It concerned, fuel rod thermomechanics and fission products (FP) releases up to FP physicochemical behaviour within the third containment barrier (i.e. containment vessel or failed Steam Generator (SG). In this article the main modelling improvements and outcomes of the environmental radiological source term re-evaluations are reported, as well as the main results gained from innovative work performed regarding accident management and prevention (i.e. EOP optimization, development of a neural network-based tool for defective fuel rod diagnosis).

2. RE-ASSESSMENTS OF ENVIRONMENAL SOURCE TERM FOR DBA & DEC-A

The approach used for LOCA and SGTR analyses was identical. After each partner has identified transients of interest (depending on reactor type, safety studies etc...), two series of calculations were carried out using the same transients: the initial one generally using current practices in their own countries for DBA and DEC-A studies, and the final one taking advantage of the improvements made during the project. The benefits due to improvements in calculation chains, from core to environmental FP release modelling, were analyzed and highlighted through the environment source term and radiological consequence evaluations.
The first set of reactor calculations was used to identify the needs in terms of model, simulation tool and/or calculation chain improvements to be made to minimize conservatisms as much as possible and avoid the decoupling factors currently used, for Design Basis Accidents, in safety studies. A specific database was then developed (incl. more than 200 tests at different scales and NPP measurements) [1-].

2.1. LOCA: Modelling improvements and source term re-assessments

To calculate properly the radiological source term of LOCA DBA or DEC-A accidents the main phenomena to be considered are related to:

- Thermal-hydraulic (in core, primary vessel, coolant circuits, containment…),
- Thermo-mechanics (in fuel rods),
- FP behaviour/chemistry (in fuel, primary circuit, containment).

Initial calculations originating from a very different context and background greatly varied, then improvements were needed. The results of source term evaluations were consequently rather scattered [2-]. Some of them use rather deterministic and conservative assumptions, especially for DBA studies, and decoupled approaches between fuel studies and source term evaluation. For example, failed rod ratio in the core, directly impacting the radiological source, was sometimes assumed but it greatly varied (from 10 to 100%). Thus, the needs to evaluate this ratio more accurately according to fuel assembly (FA) characteristics (i.e., burn-up, location, internal pressure, etc.) to which they belong, through fuel rod thermomechanical calculations were clearly pointed out.
Then, although significant improvements have been made regarding transient FP release from fuel during a LOCA (mainly through the upgrading of fuel performance such as TRANSURANUS [21-] and FP behaviour codes [3-],[22-] and their coupling enhancement [4],[5-]), the main modelling improvements that have benefited radiological source term re-assessments have concerned fuel rod thermomechanics and core modelling.

2.1.1. Modelling improvements: fuel rod burst & core model

During the project, the ballooning and burst failure data of various Zr-based alloys (i.e., E110, Zr-4…) were re-assessed. It allowed the re-fitting of plastic deformation model of the FRAPTRAN tool and the reduction of its calculation uncertainties. Above all, from a thorough re-assessment of the very large existing database related to clad burst failure, several new clad burst criteria were proposed (**Figure 1**, **Figure 2**) as:

1- Simple envelopes on engineering burst hoop stress (i.e. including minimum, mean and maximum),
2- A temperature criterion depending on clad heating rate and engineering stress.

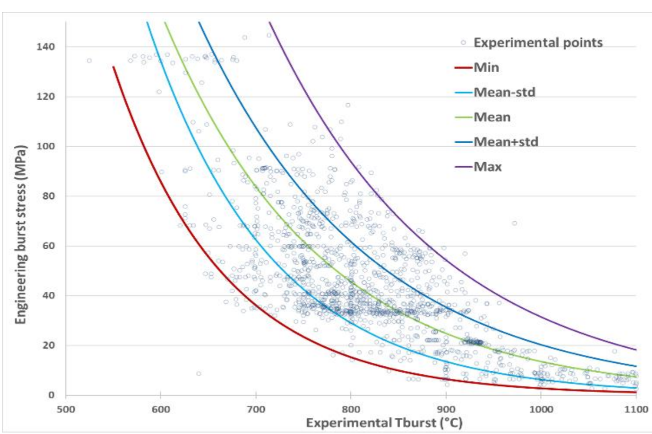

Figure 1. Engineering burst stresses vs burst temperatures: exponential models compared to experimental points from [6-]

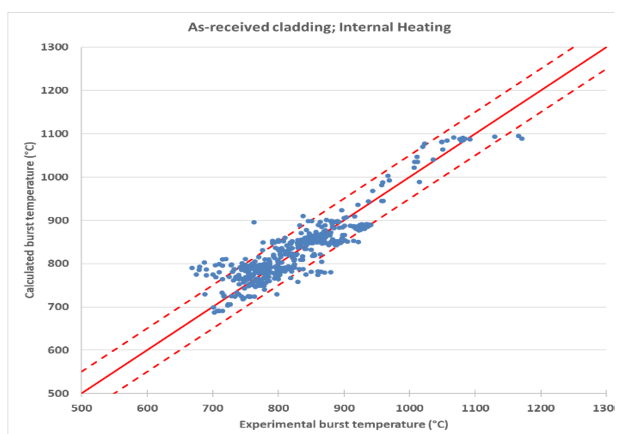

Figure 2. Calculated burst temperatures with temperature criterion vs experimental ones (dashed lines correspond to ±50°C) [6-]

All these new criteria are more specifically dedicated to clad burst occurrence prediction for radiological source term evaluation of LOCA under DBA and DEC-A conditions. They were further implemented in different simulation tools (ASTEC [10-], DRACCAR [11-], FRAPTRAN [23-]) and used in source term re-assessments. From these studies, it was recommended to test a large panel of burst criteria, as the choice of burst criteria is of first order when evaluating the fuel rod burst ratio and the uncertainties associated to these criteria remains high (±50°C for the temperature criterion).

In addition, specific 3D methodologies of core modelling were developed. These methods are the most accurate in terms of describing as best as possible the different characteristics of fuel rods in a core directly impacting their thermomechanical behaviour (i.e. rod internal pressure, burn-up, power…) and then of calculating the number of failed fuel rods. More especially, two different approaches were built both using an integral code coupling thermal-hydraulic to fuel rod thermomechanics [9-]:

1- A full RPV and 3D core model with ATHLET-CD [12-] tool simulating each fuel assembly (i.e., 193),
2- A 3D eighth core model with DRACCAR tool simulating each fuel assembly (i.e., 26).

These most advanced 3D core modelling were more suitable to capture the 3D heterogeneities in a core due to its non-symmetrical heat-up during LOCA and to heterogeneities in fuel assembly power distribution. They are generally very challenging in terms of computational costs which limits their use to specific study and make them inappropriate for instance for uncertainty analyses requiring many simulations. To do so, it will be necessary to balance detailed core description vs computational effort and use specific calculation methods.

2.1.2. LOCA source term re-assessments

LOCA source term re-assessments benefited from three types of improvements related to:

1. The code modelling: i.e. use of more appropriate thermomechanical models for clad ballooning and burst, of new clad burst criteria…,
2. The core models: i.e. use of more detailed 3D T/H core models, or core models updated based on results from parametric analyses or from more detailed (mechanistic) fuel performance code …,
3. The calculation chains: i.e. use of detailed (mechanistic) fuel performance codes not only as a support of less-detailed tools, but also as an integral part of the chain for a better coupling of thermohydraulic and thermomechanical models at the subchannel level; use of new simulation tools in the chain….

LOCA DBA transients were analyzed by 8 partners providing 16 calculations and analyzing different kinds of reactor concepts (2 for VVER, 2 for PWR and 1 for EPR). Main improvements made between 1st and 2nd set of calculations are related to fuel rod burst (**Figure 3**). For VVER this involves the use of thermomechanical models instead of the very conservative assumptions if all fuel rods would fail during the transient. For Konvoi also the 10% failed rod ratio issued from regulatory practices were replaced by thermomechanical calculations of fuel clad behaviour. For PWR-900, improvements of both clad burst models/criteria and core modelling were provided. Finally, for EPR, in addition, new cladding creep and phase transformation laws for M5 clad materials were used for updated calculations [24-]. Finally, some partners, by using new tools in their calculation chains, improved their core modelling or the modelling of FP releases from fuel and transport up to containment. Thanks to this, the number of burst fuel rods in the core was generally reduced for all reactor concepts (especially for VVERs) except for two cases: one PWR-900 calculation where the number of failed fuel rods were similar and EPR calculation where a slightly higher number of fuel rods were calculated to fail due to the introduction of a more stringent cladding failure criterion.
In analyzed scenarios, as only containment design leakages were considered, the environmental activity was constantly increasing, due to pressure differences between containment and environment. Furthermore, as there have been no changes in thermal-hydraulics nor major modifications in the modeling of FP transfer in primary circuit (except for Konvoi) and containment vessel, the changes in burst fuel rod ratios directly influence the activity releases in the environment (**Figure 4**). That is why significant differences in environmental activities between the two sets of calculations were observed for VVER. For Konvoi, the significant reduction in

environmental releases (about 3 orders of magnitude) is mainly due to the consideration of FP decay. In the end, the relative reduction in environmental activity releases is ranging from 80 to 100% for all NPP concepts.

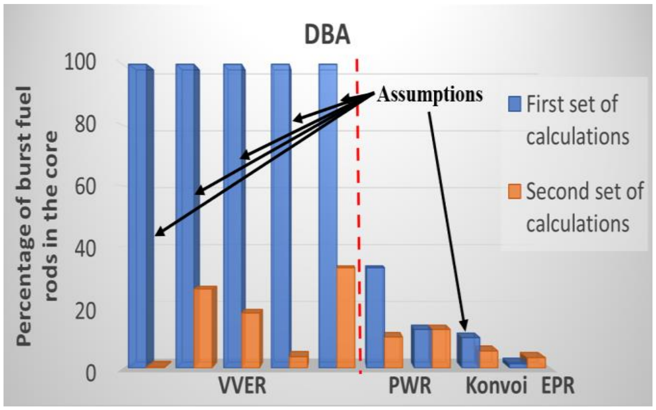

Figure 3. Rod burst ratio in LOCA DBA

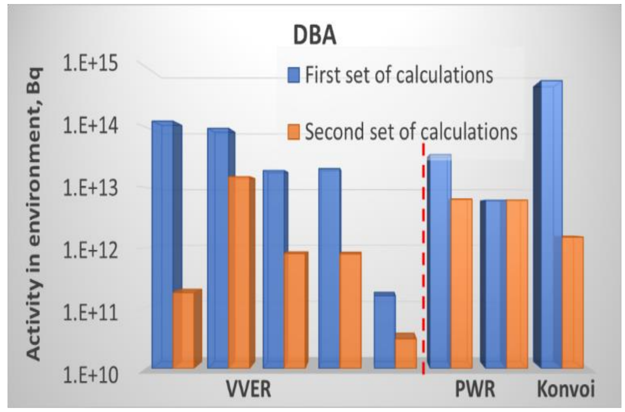

Figure 4. Activity releases in LOCA DBA

Thanks to improvements the discrepancy between released activity in environment (2 order of magnitude difference between minimum and maximum) was reduced by a factor of 2. The remaining discrepancy can be explained by differences still in initial/boundary conditions, modelling and NPP operation in LOCA… (i.e. gap inventory, design containment leakages, CSS operation and efficiency, and for iodine (differences in iodine form fraction, consideration of iodine chemistry or not), list of isotopes considered…). The conservatism and decoupled approaches have been reduced as much as possible in the updated calculations, nevertheless in very few of them due to low calculated PCT and no clad failure predicted some of them had to be retained.

LOCA DEC-A scenarios were analyzed by 5 partners, providing 13 calculations, and analyzing different kinds of reactor concepts (2 for VVER, 2 for PWR and 1 for BWR). In general, as recommended, realistic initial and boundary conditions were used for both sets of calculations. As for DBA, the main improvements made between 1st and 2nd set of calculations are related to fuel rod burst (**Figure 5**). For VVER this involves the use of thermomechanical models instead of the very conservative assumptions there all fuel rods would fail during the transient. For PWR-900, improvements of both clad burst models/criteria and core modelling were provided, while for BWR only the core ring nodalization was refined based upon information from detailed fuel performance code analyses. Finally, as for DBA, some partners, by using new tools in their calculation chains, improved their core modelling or the modelling of FP releases from fuel and transport up to containment. Thanks to this, as for DBA, the number of burst fuel rods in the core was generally reduced for all reactor concepts (especially for VVERs) except for the PWR cases where the number of failed fuel rods were either similar or increased. For PWR cases, the core modelling was refined in two different ways. The case using a 3D core modelling led to a slight increase of the failed rod number with a more accurate prediction of the clad burst as a function of the fuel assembly characteristics (burn-up, location, etc.). In comparison, the case using a more refined core modelling by just increasing the number of representative FA in each ring channels did not lead to any variations compared to initial results.

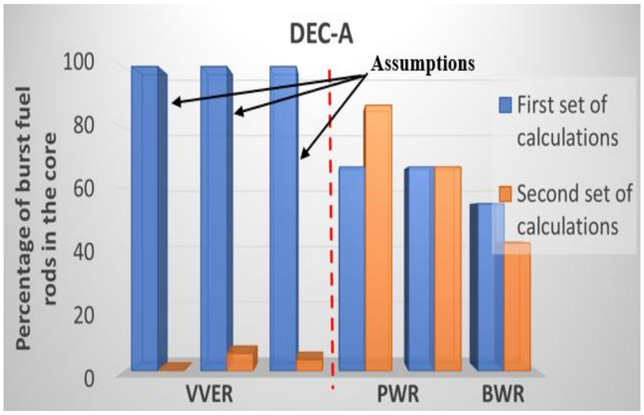

Figure 5. Rod burst ratio in LOCA DEC-A

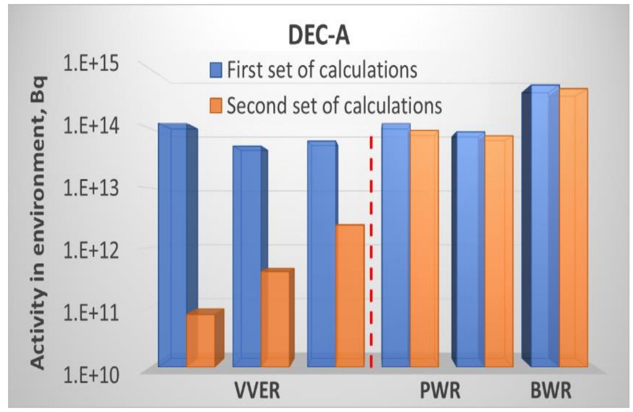

Figure 6. Activity releases in LOCA DEC-A

As for DBA, changes in burst fuel rod ratios directly influence the FP releases in the environment (**Figure 6**). That is why significant differences in environmental activities between the two sets of calculations were observed for VVER (2 to 3 order of magnitude difference). For PWR, the environment releases are quite similar, despite for one case a higher fuel rod burst ratio. This is due to more realistic assumptions regarding the containment leaking rates, where, for final calculations part of the containment leakages was considered recovered in the auxiliary buildings instead of fully direct containment leakages. In the end, the reduction in activity releases in the environment ranges from about 13 % (PWR, BWR) to 99% for VVER.
Due to the significant reduction in fuel burst ratio and FP releases in VVER, the differences between the different concepts (i.e. VVER/PWR) are finally more marked in final than in the initial calculations (4 orders of magnitude difference between minimum and maximum activity releases vs 1 respectively).
Radiological consequences (thyroid equivalent and total effective doses) evaluated for both sets of calculations are consistent with what is observed with the activity calculated in the environment: for most calculations, especially for VVER, radiological consequences are reduced in the second set of calculations compared to the first one.

2.2. SGTR: Modelling improvements and source term re-assessments

The presence of leaking fuel rods in normal operation, due to defects, is one of the main bases of the considered source term in SGTR. Occurrence of these defects, depending on NPPs, is detected by primary circuit contamination measurements, where sometimes several orders of magnitude of specific FP activities were observed between situations without or with leaking fuel elements. The occurrence of defective fuel rods, becoming today rarer, primary circuit contamination is reduced.
To calculate properly the radiological consequences of SGTR DBA or DEC-A transients, the main phenomena to be considered are related to:

- Thermal-hydraulic (in coolant circuits, safety injection systems, SGs…),
- FP releases from defective fuel rods,
- FP transfer from primary-to-secondary coolant systems,
- FP transfer between gas and liquid phases in the failed SG (incl. flashing/atomisation/partitioning…).

As for LOCA, initial calculations originating from different backgrounds greatly varied, then improvements were needed. The results of initial source term evaluations were consequently rather scattered [2-. Varying degree of conservatism, arbitrariness or empiricism were used, including for DEC-A scenarios, to compensate the lack of models and/or measurements (i.e. primary coolant activity in normal operations, spiking activities during SGTR and iodine gas-liquid transfer in failed SG…). Also, iodine speciation in RCS (then gaseous iodine fraction), directly impacting the source term evaluation, was always imposed.

Then, although significant improvements have been made regarding transient FP release from fuel during a SGTR [7-], the main modelling improvements that have benefited radiological source term re-assessments have concerned the iodine spiking in RCS and flashing/partitioning in the failed SG.

2.2.1. Modelling improvements: iodine spiking in RCS & transport in failed SG

The release of FP, particularly iodine, from defective fuel rods and its transport into the secondary side of the failed SG were considered as areas where further improvements might have a significant impact on environmental activity releases as generally conservative and/or empirical assumptions were currently used to take them into account. Amongst these improvements, the refinement of spiking models (i.e. enhanced transient releases of FP gap inventory during a transient) appears of upmost importance whatever SGTR scenarios (overflowing or non-overflowing). The improvements made during the project, for the most part, were based on legacy or new NPP measurements (**Figure 7**) and concerned especially iodine. They were made in different types of code (i.e T/H, Severe Accident codes…) considering not only the reactor power & primary pressure variations but also the time point in the fuel cycle as influent parameters and greatly differed in complexity (ranging from simple correlations to more mechanistic models where the latter appears more appropriate to be easily extending to any operating NPP concept).

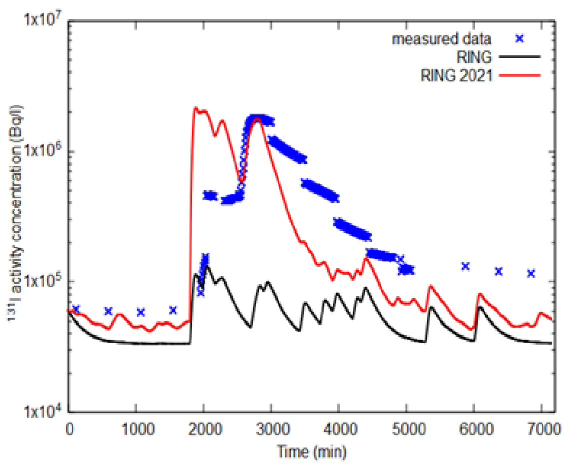

Figure 7. I-131 spike in RCS during SGTR: RING code results vs measurements

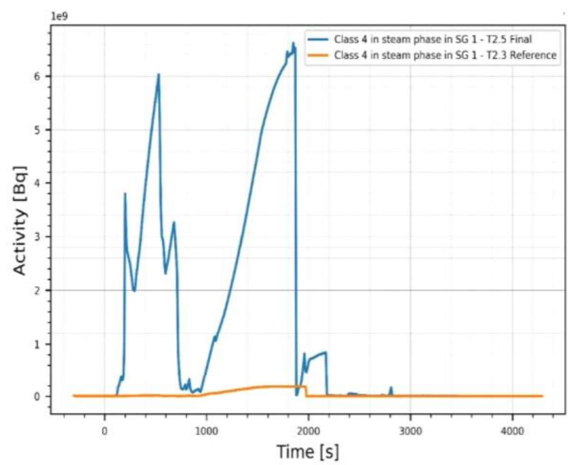

Figure 8. I_2 partitioning in failed SG: MELCOR code initial vs final calculations [8-]

The modelling of the FP (particularly iodine) transport within conditions representative of failed SG was also especially enhanced, as, often, no specific models were available. The scarcity of relevant data addressing those phenomena (such as flashing, atomization, partitioning) under the prevailing conditions made this more complex and uncertain. The main modelling improvements used in re-assessments concerned iodine flashing and partitioning (**Figure 8**). They often led to higher iodine source term in the failed SG gas phase, thus potentially a further increase in environmental activity releases.

2.2.2. SGTR source term re-assessments

SGTR source term re-assessments benefited from two types of improvements related to:

1. The code modelling: i.e. use of models for FP dilution, transport in RCS, use of updated FP (mostly iodine) spiking models, use of updated or new FP transport models in failed SG (such as for partitioning, flashing and atomization also mainly for iodine), use of refined T/H models for SG relief/safety valve releases …,

2. The calculation chains: i.e. use of new modules/simulation tools, used of mechanistic codes in support of less detailed codes… For SGTR calculations, both detailed T/H and integral codes were used.

In addition, implementation and application of new optimized EOPs were performed in some calculations.

DBA transients were analyzed by 8 partners, providing 23 calculations. Main improvements made by the partners between 1st and 2nd set of calculations are all related to RCS radionuclide inventory and transport aspects both in RCS and failed SG, except for one partner, who also improved the SG relief/safety valves thermohydraulic correlations. Thanks to this, partners achieved a 23% to 80% reduction of the activity releases in the environment (**Figure 9**), except for one partner, whose environmental radiological releases are two times greater in the final calculations than in initial ones. This partner used a new calculation chain including a new module for FP transport in RCS and failed SG directly impacting the pre-spiking RCS contamination and spiking, as well as the FP distribution between the liquid and gaseous phase in the failed SG. Finally, this partner also used a detailed code in support for predicting the iodine speciation in RCS and then better calculate the iodine flashing rates in the failed SG.

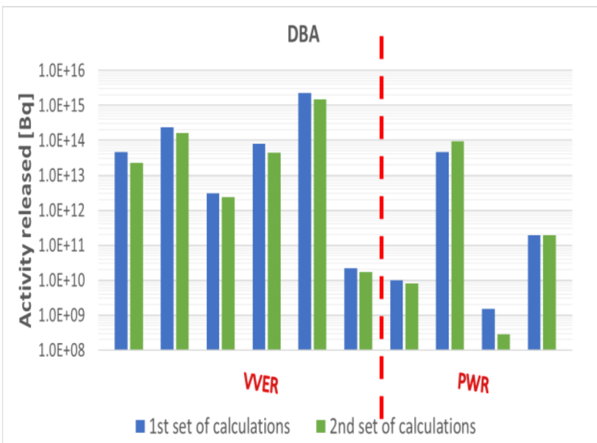

Figure 9. Activity releases in SGTR DBA

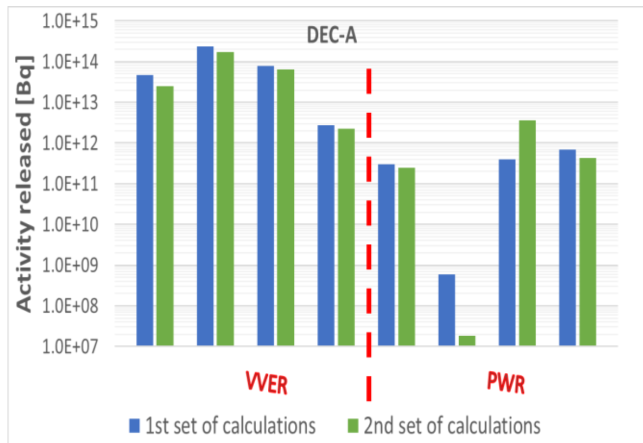

Figure 10. Activity releases in SGTR DEC-A

Large discrepancies between the activity releases (up to 7 orders of magnitude difference between minimum and maximum) can be observed even for the updated calculations. There are several reasons for this, such as: differences on the thermohydraulic evolution during the transient (due to break size, operators' actions, initial conditions, thermohydraulic code and correlations considered etc.), differences in the assumptions regarding the initial RCS FP inventory (including spiking), differences on the list of isotopes considered, differences in the modeling (i.e. on FP dilution in RCS, on FP transport phenomena considered in the failed SG (such as atomization, flashing, partitioning, scrubbing), or sometimes also differences in the consideration of activity releases (from steam only or from both liquid/steam phases). In the end, some partners tend to be more conservative while others may be more realistic (i.e. the highest releases calculated by one partner in both sets of calculations due to maximum allowable RCS activity considered, a slow RCS and SG depressurization leading to a break flow rate that doesn't cancel out in time and continuous activity release to environment through the relief valve). Conservatisms have been however as much as possible reduced in the updated calculations through model/calculation chain refinements. Nevertheless, despite such improvements, discrepancies between partners are in the second set of calculations as high as in the first ones.

DEC-A scenarios were analyzed by 6 partners, providing 21 calculations. As for DBA, the main improvements made by the partners between 1st and 2nd set of calculations were all related to RCS radionuclides inventory and transport aspects in RCS and failed SG, except for one partner, who considered optimized EOPs for

operators' actions to depressurize RCS as fast as possible. Thanks to this, partners achieved a 17% to 97% reduction of the activity releases in the environment (**Figure 10**), except for one partner, whose environmental radiological releases are greater in the final calculations than in initial ones due to the consideration of activity release through the liquid pathway in addition.
As for DBA, there is also a large discrepancy between the partner results (up to 7 orders of magnitude difference between minimum and maximum activity releases) for the same reasons. Even though more realistic conditions were generally used for DEC-A, the discrepancies were even greater. This is mainly due to the results of one partner who, unlike the others, have considered a SGTR + isolable Steam Line Break which led to stop the releases to the environment very early during the transient.
Eventually, radiological consequences (thyroid equivalent and total effective doses) were evaluated for both sets of calculations. The results are consistent with what is observed with the activity releases to the environment: for most partners, radiological consequences are reduced in the second set of calculations compared to the first one, except for two SGTR transients.

3. MAIN PROGRESSES IN ACCIDENT MANAGEMENT AND PREVENTION

Several studies were performed during the project to cope with SGTR radiological consequences, resulting through containment by-pass in direct FP releases to the environment.

3.1. AMP optimization

The general approach of optimizing an Accident Management Procedure (AMP) [14- using the Downhill Simplex method [15-, earlier applied to a station blackout in VVER1000 [13-, was refined in the frame of the R2CA project and applied to a DEC-A SGTR scenario for a generic PWR 3750 MW_{th}. In this scenario, the Safety Relief Valve (SRV) of the affected SG was assumed to remain stuck open after the first opening (direct release path) and that none of the trains of Low-Pressure Injection System (LPIS) was working due to technical failure.
The AM strategy for such events aims at reducing the primary system pressure as fast as possible to prevent further loss of coolant to the secondary system (i.e through cooling down of the primary system with the three remaining intact steam generators, actuation of the pressurizer spray system, opening of the pressurizer relief valve (PORV) and finally safety injection switch off). In this analysis, it is further assumed that the operator would then initiate cool down of the primary via depressurization of the secondary and that two trains of the High-Pressure Injection System (HPIS) would be disconnected. The Simplex method was the used to optimize the timing of three further operator actions: opening the PORV, disconnecting train 3 of HPIS, and finally train 4 while reconnecting the make-up system.
The simplex method provides a very robust numerical algorithm to find a local minimum of a function $F:\mathbb{R}^n \rightarrow \mathbb{R}$. First step is to create an initial, arbitrary simplex in $\mathbb{R}^n$ (i.e. the simplest geometrical shape, e.g. a triangle in $\mathbb{R}^2$ or a tetraeder in $\mathbb{R}^3$). The function F is then evaluated at each corner of the simplex. The highest point is moved according to the algorithm and a new simplex is constructed, the function is again evaluated, and so on. The series of simplexes slowly converges to a local minimum. To use the simplex method for AM optimization, the independent parameters of the function were chosen to be the timing of the three above mentioned operator actions. Then, a thermal hydraulic system code (the function) evaluated the plant response. As dependent variable, a figure of merit was constructed, that evaluated the AM-Measures based on a.) the radioactive releases to the environment, b.) the primary pressure at the end of the transient and c.) the water level in the reactor core. The simplex method was then applied to find the optimum timing for the operator actions. After thirteen iterations, the algorithm converged to a minimum. The results showed that the opening of the PORV is most efficient later in the transient, while shutting down one HPIS early, and the second one late in the transient, provides good results. Overall, the iodine release to environment could be significantly reduced through optimization of the timing of investigated operator actions (**Figure 11**).

3.2. Development of Neural networks for earlier diagnosis

During NPP normal operations, the fuel rod cladding may be damaged due to various phenomena. Such deterioration and subsequent activity releases having safety and economic consequences, the early detection of defective fuel rod is of primary importance. Different approaches to this end have been developed over the years, based mostly on physical considerations [16-.

On another hand, the improvement of data-driven methods led to their successful use as anomaly detectors in various fields of industry. These were consequently tested in the framework of R2CA project: since the occurrence of a defect leads to a change of primary activity compared to an activity level labeled as normal, the efficiency of autoencoders shaped as Recurrent Neural Networks [17- (RNN), Gated Recurrent Units [18- (GRU) networks, or Long Short-Term Memory networks (LSTM) [19-], fitted for time related variables, were tested. As no experimental data were available, a workflow was established to build artificial data.

First, a simple physical model was built [20-, physically relevant enough to render various situations such as the opening of a defect, a pre-existing defect, no defect at all among others. The physical model considers the release from a fuel pellet by diffusion or recoil, a generalized diffusion and first-order kinetic model for the fission products in the gap region, and a first-order kinetic model in the coolant region. Considering that the use of physical model can be CPU intensive, a fast surrogate model was built, based on a moderate sized physical database, taking as input parameters both literature values and TRANSURANUS [21- calculations results. The surrogate model turned out to be precise as the maximum relative error was about 2.6%, and less CPU intensive by a factor 1000 than the physical reference model. This allowed the generation of several tens of thousands of activity sequences in a small amount of time. Finally, all tested anomaly detectors gave 100% satisfactory results (**Figure 12**), and the difference between them relied in learning speed.

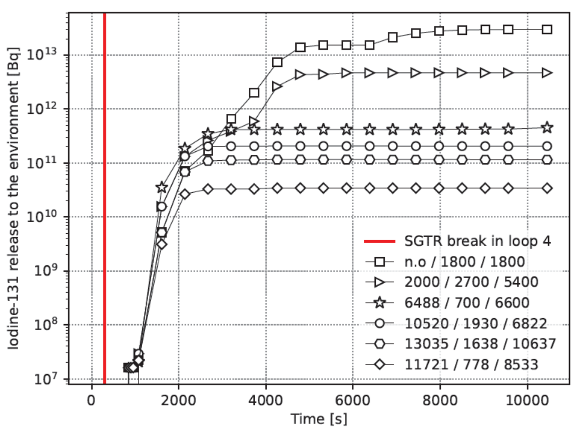

Figure 11. I-131 release in environment as a function of operator action timing

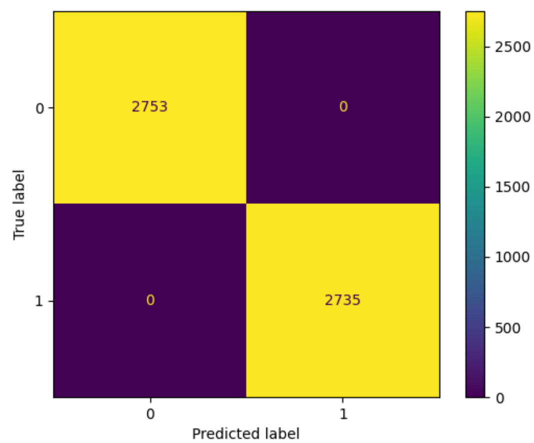

Figure 12. Anomaly detector performance

4. FINAL REMARKS

Thanks to the developments/improvements of models, simulations tools and calculations more realistic RC evaluations of both LOCA and SGTR scenarios within DBA and DEC-A conditions have been provided that led to reduce some conservatisms. This represents a first step towards best estimate evaluations. These latter needed to be associated to systematic Sensitivity Analyses an Uncertainty Quantifications were beyond the scope of the project. However, from the updated evaluations, the parameters having the most impactful effect on results were identified and some recommendations formulated as:

For LOCA transients:

- Better prediction of failed fuel rods is of first importance. To do so, expressing specific fuel rod initial conditions (i.e., burn-up, internal rod pressure, specific power…) for thermomechanical calculations (incl. axial gas communication) and use refined criteria of cladding burst temperature and strain, specific to DBA and DEC-A conditions is important.
- To have a coupling between T/H and thermomechanical calculations. To this end, the use of a detailed 3D core modelling allowing for better differentiation of FA characteristics (power distribution) could be beneficial. It also allows to consider the asymmetric character of LOCA transient. However, the level of details in the model must be balanced with computational capabilities.
- Refine the FP gap inventory calculations by considering differences between fuel rods/assemblies (i.e., provide multi-inventory, 3D burn-up distribution…) is also recommended.
- Regarding FP behaviour in containment, use of a dedicated calculation tool/module for modelling the complex and time dependent iodine behaviour/chemistry is necessary.

For SGTR transients:

- Initial RCS contamination and spiking is of first importance. Real NPPs data can be preferred to be more realistic but it will be difficult to extend them to other operating or future NPPs (i.e. for different types of fuel, operating conditions…). Then, the development of more complex mechanistic models was recommended, at least in support of less detailed codes.
- All transport phenomena through SGTR break and in failed SG (atomization, flashing, partitioning, scrubbing) should be simulated based on feedbacks from the TH part. Iodine speciation in primary circuit conditions also (determining its volatility) as it could impact its flashing fraction at SGTR break and its partitioning from secondary liquid phase.
- For transfer to environment, as consequences of activity releases in liquid water or in steam phases are not the same, each phase must be considered separately in failed SG. In addition, for non-overflowing scenarios, aerosol retention in upper SG structure (size-dependent) must be calculated or deduced from the use of existing test facility results.

In parallel, many other modelling improvements were made (generally more mechanistic), not reflected in the re-evaluation of the scenarios, which will be of benefit in future reactor calculations (i.e. internal clad secondary hydriding, FP releases from fuel during transients, high burn up structure and releases...). Regarding code improvements, enhanced couplings of fuel performance code (such as TRANSURANUS) with FP codes were also performed and their capabilities extended to study new fuel types in the future, such as near-term accident tolerant fuels (ATF).

Finally, exploratory work on AMP showed that the increased capabilities of numerical tool and the use of expert methods could be advantageously used to increase the NPP safety.

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