HORIZON EURATOM SASPAM-SA PROJECT: MAIN IDEAS AND FIRST OUTCOMES

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Abstract

Today, there is growing interest for light water integral PWRs (iPWR), that for their main specific features are considered one of the key technologies for the short-term nuclear technology deployment. In this framework, despite iPWRs show a reinforcement of the first three levels of Defence-in-Depth (DiD) due to the use of passive safety systems, independent features for preventing or mitigating hypothetical Severe Accident (SA) sequences have to be included in the design. Therefore, some scenarios that could lead to SAs need to be postulated and deterministically studied along the design and the safety review process. Considering that iPWRs technology comes from the Large LWR operational experience including evolutionary modifications, in order to speed-up the European licensing/siting process of iPWRs, the Horizon Euratom SASPAM-SA (Safety analysis of SMR with Passive Mitigation strategies – Severe Accident) project aims at investigating the applicability and transfer of the operating large LWR knowledge and know-how to iPWRs taking into account SA and Emergency Planning Zone (EPZ) European licensing needs. The project, coordinated by ENEA (Italy), started in October 2022 and involves 23 Organization. The paper aims at describing the pillars and goals of SASPAM-SA and provide the main results of the research activities performed in the first phase of the project.

1. INTRODUCTION

Light water Small Modular Reactors (SMRs) are emerging as a key option for the short-term deployment of new nuclear reactor technology. In Europe, there is growing interest in SMRs, with various research activities underway across several countries in the view of potential licensing needs. In particular, iPWRs are poised for licensing due to their foundation on the well proven technology coming from large LWRs, enhanced by its multi decades operational experience, and moderate evolutionary safety features to improve the inherent safety of the plant. However, even if a plant is designed with advanced inherent features, through the reinforce the first three Defence-in-Depth (DiD) levels, independent features for preventing and mitigating a SA sequence have to be included in its design (DiD level 4) together with the offsite emergency response (DiD level 5). Therefore, scenarios leading to SAs need to be postulated and deterministically studied throughout the reactor design and the safety review process. The Horizon Euratom SASPAM-SA project [1] aims to speed-up the European licensing/siting process for iPWRs by leveraging the operational experience and knowledge from large LWRs. This project, coordinated by ENEA (Italy) and involving 23 organizations, started in October 2022 and is focused on investigating the applicability and transfer of large LWR knowledge and know-how to iPWRs, considering European SA and Emergency Planning Zone (EPZ) licensing needs. To maximize knowledge transfer and project impact, two generic iPWR design concepts with different evolutionary innovations compared to larger LWR reactors have been selected for analysis: Design 1: iPWR characterized by a submerged containment and electric power of about 60 MWe, Fig. 1 (left); Design 2: iPWR characterized by the use of several passive systems, a dry containment and an electric power of about 300MWe, Fig. 1 (right). These two generic reactor concepts include the main iPWR features, considered in the most promising designs ready to go on the European market, allowing to assess in a wider way the capability of codes (SA and Computational Fluid Dynamics, CFD) to simulate the main phenomena typical of iPWR. The project's objective is not to evaluate the selected generic designs but to provide a broader understanding of the codes' applicability to current designs under postulated SA conditions. The specification of both designs is based on open literature and engineering assumption. More information on the two generic designs is in [2]. The achievement of the overall elements is assured by a consistent and coherent work programme, reflected in the technical Work Packages (WP) structure: WP1- Coordination; WP2- Input deck development and hypothetical SA scenarios assessment (SCENARIOS); WP3 - Applicability and Transfer of the Existing SA experimental database for iPWR Assessment (EXP); WP4 - Assessment of code capabilities to simulate and evaluate the In-Vessel Melt Retention (IVMR) in iPWRs (IVMR); WP5 - Assessment of the code capabilities to simulate IPWR containment and characterize mitigation measures efficiency (CONT); WP6 - Characterization of iPWR EPZ (EPZ); WP7 - Communication, dissemination and exploitation (DISSE).

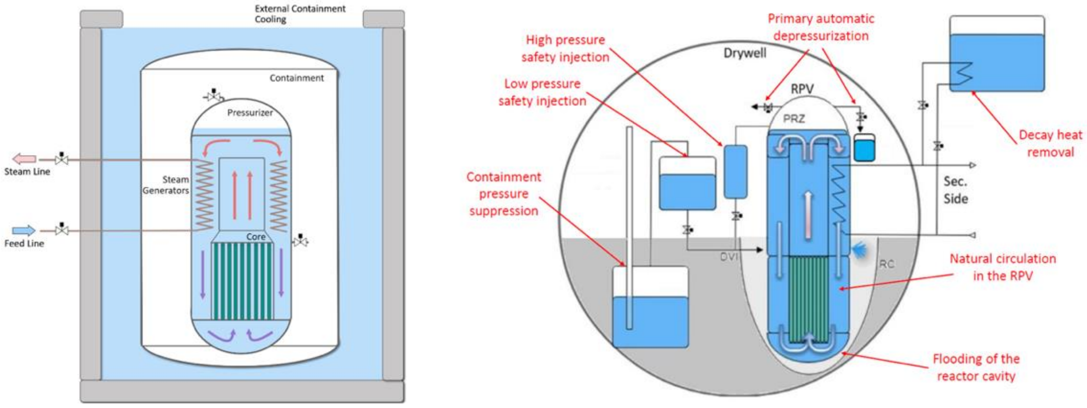

FIG. 1. Generic reactor concept layout: Design 1(left), Design 2(right).

2. WORKING PACKAGE 2: SCENARIOS

The main goal of the WP2 is evaluating the capabilities of state-of-art European and non-European integral codes and CFD tools, largely used in Europe, to simulate the dominant phenomena occurring during postulated DBA and SA scenarios in the two generic iPWR designs investigated in the project [1] [2]. In particular, the focus is on the analysis of the performance of such codes to predict the behaviour of the passive systems, their mutual interaction, and the effect on the iPWR scenarios progression of their postulated total or partial failure. Having this in mind, the WP2 is focussed on the following tasks: 1) assessment of the specifications of both iPWR designs based on open literature; 2) development of generic, but representative, SA and CFD codes' datasets of both designs; 3) postulation of hypothetical DBA and SA scenarios with the corresponding analyses; 4) assessment of a database of nuclear inventories at different burn-up steps from 20 MWd/kgU to 60 MWd/kgU. Note that the SA scenarios have been postulated (through the postulated failure of the correspondent valve to open) and selected based on their severity and not on the probability to occur, since no PSA assessments are planned in the SASPAM-SA project, since generic designs are considered. Finally, the outcomes of the calculation campaigns performed in the WP2 will be used as boundary conditions for the activities planned in the other WPs. Concerning the generic iPWR Design-1, four DBA scenarios have been postulated triggered by opening a break in the Chemical and Volume Control System (CVCS) make-up line of different sizes (100%, 35%, 20%, and 10% pipe section) by assuming the loss of AC power supply and the availability of the Reactor Vent Valves (RVV) and the Reactor Recirculation Valves (RRVs). Three SA scenarios have been postulated: SA-1) 100% break on the CVCS line, by assuming the loss of AC power, the availability of the RVVs, and the unavailability of the RRVs; SA-2) the inadvertent opening of one RVV (100%), by assuming the loss of AC power, the availability of the other RVVs, and the unavailability of the RRVs; SA-3) 100% break on the venting line, by assuming the loss of AC power, the availability of the RVVs, and the unavailability of the RRVs. The time-dependent behaviour of the pressure in the pressurizer (PRZ) and in the containment in the postulated DBA scenarios in the iPWR Design-1 computed by means of ASTECv3.1.1 and EDF-MAAP are shown in Fig. 2 (left) by using the 1973 ANS standard decay heat data increased by 20%.

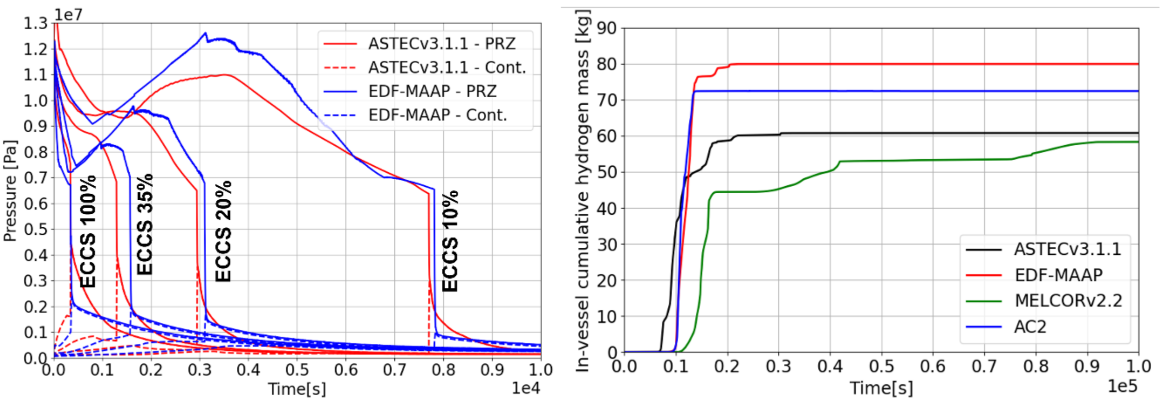

FIG. 2 DBA scenarios in the iPWR Design-1: pressure in the RPV and in the containment computed by ASTECv3.1.1 and EDF-MAAP (left) and in-vessel cumulative hydrogen mass during the SA-1 (right).

Following the opening of the break, the pressure in the Reactor Pressure Vessel (RPV) drops down (continuous lines), while the pressure in the containment increases (dotted lines). Scenarios with smaller breaks have slower initial pressure decreases in the RPV, as would be expected. Furthermore, the results show an increase of the pressure in the RPV, since not all the decay heat can be evacuated through the break. As a result, the pressure in the RPV increases up to reaching a maximum after which the Emergency Core Cooling Systems (ECCS) is activated leading to the water refill of the core by gravity. The results show a rather good agreement among the codes on the behaviour of the pressure and on the timing of activation of the ECCS, which show a deviation among AC2, ASTEC, EDF-MAAP, and MELCORv2.2 ranging between 500 s and 1000 s. As for the DBAs, a quite large calculation campaign of postulated BDBA has been performed for the identification of SA scenarios. The results show that no failure of the RPV is predicted by all the codes due to the ex-vessel reactor cooling for any of the SA scenarios postulated. The findings also demonstrate that the in-vessel degradation process proceeds more quickly the greater the decay heat used. Note that the decay power computed in the WP2 is ~30% lower than the 1973 ANS standard data. The results concerning the in-vessel hydrogen production from the different codes during the SA-1 is shown in Fig. 2 (right). The deviation among the codes on the hydrogen onset is ~1,000-2,000 s and on the in-vessel hydrogen production is ~20 kg. Concerning the core degradation, the preliminary analysis of the results shows a quite general agreement among the codes. In particular, by using the same decay heat data, a deviation of the timing of material relocation to the Lower Plenum (LP) is about 1,000 s, while the maximum deviation on the mass of relocated material is ~1 ton (range between 8 ton and 9 ton). The results therefore reveal that the codes employed are able to simulate the main expected thermal-hydraulics phenomena in the iPWR Design-1 during the postulated DBA and SA scenarios and the main degradation phenomena in SA.

Concerning the generic iPWR Design-2, one DBA scenario has been postulated initiated by a guillotine break of one Direct Vessel Injection (DVI) line, by assuming all the safety systems in operation. Three SA scenarios have been postulated by assuming a guillotine break of one DVI line in conjunction with: SA-1) the unavailability of both the Emergency Heat Removal System (EHRS) and the Automatic Depressurization System (ADS)-1; SA-2) the unavailability of all the safety systems; SA-3) the unavailability of the EHRS only. Furthermore, a sensitivity on the decay heat was carried out, considering the data produced in the framework of the WP2, the 1973 ANS standard, and the 1973 ANS standard increased by 20%. The results of the integral codes employed (ASTECv3.1.1, MAAPv5.06, MELCORv2.2, and AC^2) for the DBA scenario in the iPWR Design-2 show a quite good agreement with respect to the timing of the key events characterizing the transient. As an example, the behavior of the pressurizer (PRZ), containment, and secondary pressure computed by ASTECv3.1.1 and AC^2 is shown in Fig. 3 (left). The results show a quite good agreement between the codes.

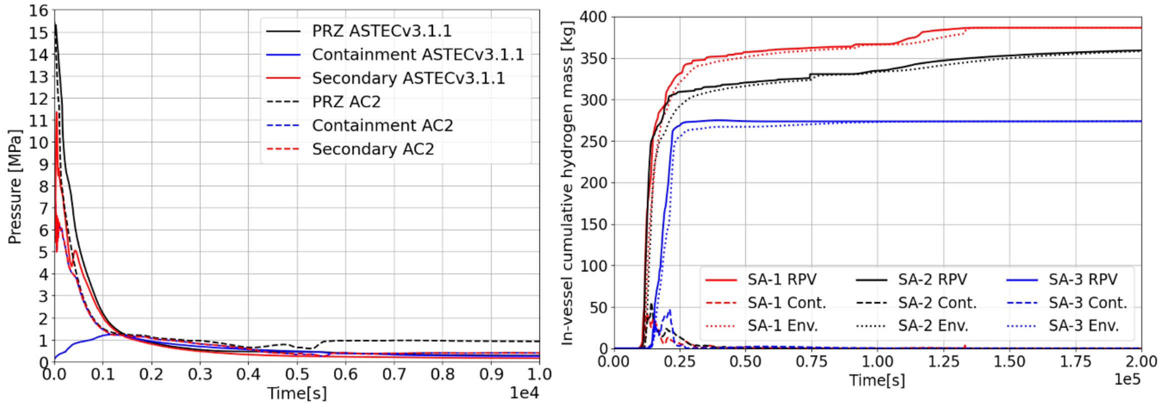

FIG. 3. DBA scenario in the iPWR Design-2: pressure in the RPV, in the containment, and in the secondary computed by ASTECv3.1.1 and AC^2 (left) and in-vessel cumulative hydrogen mass computed by ASTECv3.1.1 (right).

Following the DBA analysis, a wide calculation campaign of postulated BDBA has been performed and SA scenarios has been identified by using the fuel inventory evaluated in the WP2. The results from the integral codes predict no RPV failure due to the effect of the ex-vessel reactor cooling and the reaching of the design pressure. Note that no containment leakages have been modelled in the datasets, such analyses being planned in the WP5. The maximum deviation on the prediction of the beginning of the in-vessel degradation progression, namely the instant of the first core failure, in the three SA scenarios is ~3,000 s. The severity of the postulated scenarios increases with the increase of the unavailability of the safety systems from the SA-3 to the SA-1. The codes show a good agreement in predicting such increase and, more in general, a qualitative agreement is observed on the behaviour of the accident progression, see also [2]. For example, the maximum mass of the molten material in the core ranges from ~60 ton (SA-3) to ~100 t (SA-1), with a deviation among the codes of ~10 ton. As an example of the consequences of the postulated SAs, the ASTECv3.1.1 results for the in-vessel hydrogen production in all the analysed scenarios are shown in Fig. 3 (right). Depending on the scenario, the hydrogen onset occurs between 7,000 s and 11,000 s and the cumulative hydrogen production ranges from 270 kg and 380 kg, a preliminary comparison of the results among the codes showing a deviation of ~20 %.

A CFD model has been developed for the design1 with ANSYS CFX (preliminary mesh of about 370000 nodes) and with containmentFOAM for design 2 (preliminary mesh using between 200000 and 750000 mesh for representing one quarter of the containment). The first results show rather stable simulation and good qualitative assessment of the expected phenomenologies.

3. WP3: ASSESSMENT OF THE RELEVANCE AND APPLICABILITY OF EXISTING EXPERIMENTAL DATABASES TO IPWR

The objective of WP3 is to evaluate the relevance and applicability of the existing experimental database to iPWR. Based on the plant SA scenarios identified and investigated in the WP2, the main boundary conditions and the specific features of iPWRs are determined and compared to those of large LWR. Based on this comparison, the applicability of the existing experimental data to iPWR is assessed for in-vessel, containment, source term and ex-vessel coolability (if applicable) phenomena [3]. The specific data to be evaluated include, e.g., natural circulation, debris bed formation, liquid melt spreading, steam explosion, re-flooding of an overheated core, in-vessel melt pool formation, corium cooling under water, hydrogen distribution, combustion and mitigation, wall condensation, aerosol transport and hygroscopic growth, iodine speciation and mitigation, pool scrubbing, and fission product composition. The work in WP3 was started by developing a methodology to assess the applicability of existing data to iPWRs, as reported in FIG. 4. Based on DBA and postulated SA scenarios for designs 1 and 2 in WP2, the main boundary conditions and the specific phenomenological features of the two selected iPWR designs were determined. Comparison with experimental data is done using the main parameters and dimensionless numbers determined based on the governing equations for each phenomenon. Comparison with DBA and SA conditions for the iPWR designs 1 and 2 are done using a table (or plot) in which all the main parameters and, when deemed useful, dimensionless numbers are presented. Based on the comparison of the experimental data with the iPWR designs, assessment of the applicability of the data to iPWRs is made. The data are divided into three categories: 1) data which are directly applicable to iPWR; 2) data which can be used by developing extrapolation and other methods to extend their applicability to iPWR conditions, and 3) data which are not applicable to iPWRs. After classification, methods are developed to extend the applicability of existing data for the data in category 2) to be able to apply them for different iPWR designs. In the end of the activity, the potential need for new experiments is determined. The first results have been discussed in [3].

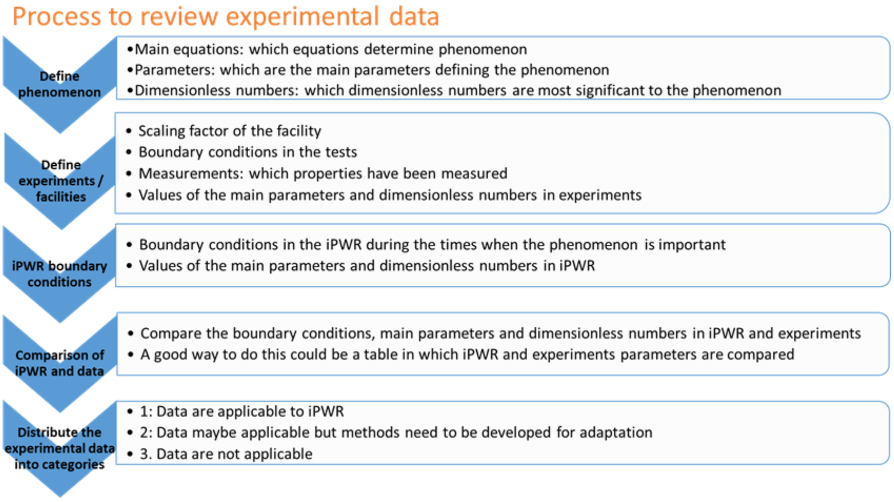

FIG. 4. Proposed methodology for evaluation of the applicability of existing experimental data to iPWRs

4. WP4: ASSESSMENT OF CODE CAPABILITIES TO SIMULATE AND EVALUATE CORIUM RETENTION IN IPWRS

Based on the Designs 1 and 2 input decks and SA scenarios developed in WP2, the objective of this WP is to investigate IVMR strategy in a postulated SA scenario in iPWR, assess the capability to simulate the main phenomena characterizing the IVMR in iPWR, and characterize its feasibility by guided calculations. The WP4 objectives will be addressed by: 1) Analysing the processes leading to the formation of a molten pool in the LP and the possible oxide/metal stratification (impact of cladding material will be analysed at this stage); 2) Identifying/Implementing modifications to the models used for operating PWR in SA codes; 3) Evaluating the safety margin for a few simplified iPWR designs and assessing the degree of confidence. As a preliminary study, a steady state 0D model was developed [4] following the approach initially developed for the AP600 design. Some adaptations of models were necessary (see next sub-section). As expected, the first results show that in-vessel retention strategy appears feasible, with a good safety margin. The main results are summarized in Table 1. First results show that the maximum heat flux is small enough to be extracted by pool boiling, in principle. The residual vessel thickness is large enough to ensure mechanical resistance with a good safety margin (comparable or better than for VVER-440 reactors where IVR strategy is currently implemented). One remarkable feature of the two designs studied is the quite large amount of power lost by radiative heat transfer to the top structures of the vessel, as illustrated in Fig. 5. However, detailed data and complete calculations would be necessary to go further. It will be the objective of the next steps of the project. Some specific features of IVMR in iPWRs appear: a larger fraction of power transferred to the top of the oxide pool because of the low aspect ratio (shallow pool), the existence of a rather thick oxide crust and the occurrence of a thin metal layer but a limited focusing effect because of a large fraction of power lost by radiative heat transfer. Regarding the oxide crust thickness, results yield average values of one order of magnitude higher than for High-Power Reactors (HPR). All these specific features must be considered in models. The impact of aspect ratio (height to radius, H/R) can be treated by simply selecting correlations which already include the H/R parameter. The presence of thick oxide crusts requires a modification of existing models which usually neglect the crust. A model was proposed in [4]. Finally, a more accurate analysis of top radiative heat transfer must be done as it appears to be significant for the evaluation of focusing effect and the maximum heat flux to the vessel. Some of those issues will be addressed in the next steps of SASPAM-SA WP4, by developing new models and/or performing transient calculations of the stratified pool.

Table 1: Main results for IVMR analysis of Designs 1 and 2

	Unit	Design 1	Design 2
Average ablated thickness	cm	6.5	4.4
Min. residual vessel thickness	cm	22.1	8.7
Max. heat flux to vessel	MW/m^2	0.133	0.335
Power fraction towards top metal	-	0.657	0.655
Radiative power fraction in metal	-	0.872	0.662

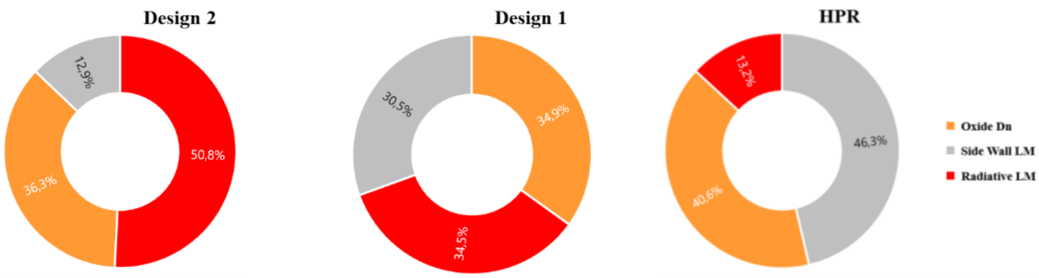

FIG. 5. Heat removal distribution in Design 1, Design 2 and HPR.

5. WP6 EMERGENCY PLANNING ZONE SUMMARY

WP6 provides the final step in the assessment of hypothetical postulated SA of iPWRs. From the foundation laid by other WPs (core inventories, SA scenarios and phenomena, including those in the containment) the purpose is to proceed from possible accidental atmospheric releases to the actual potential harm to the health of the off-site population. The radiological risk is based on predicted individual effective or organ-specific doses, which are then used, after comparing with certain dose limits, to set a distance (size of a circular zone) within which emergency planning is necessary. The limits can be e.g. the generic limits expressed as doses in IAEA GSR Part 7 (Appendix II), the dose thresholds to initiate certain protective actions in a country, or directly EPZ-setting dose values, like in STUK Decree Y/2/2024 in Finland. Generally, for iPWRs and many other SMRs, it is claimed that they are inherently safer than operating large NPPs, partly because of the smaller radioactive inventory and partly because of advanced passive safety systems. Consequently, lower radiological impact and smaller EPZ have been expected. The main objective is to provide recommendations for a rigorous and justified, scientifically sound basis for the assessment of iPWR EPZ size (single unit is currently considered in the WP6), regarding both the methodology of its determination and actual results. However, it is recognized that the EPZ size, even for the same plant, is not necessarily universally the same in all countries, because there are many affecting factors, some also non-technical. So far in WP6, the work was started with a review phase investigating the present methods and regulations for offsite dose projection and EPZ determination in various countries or proposed by the IAEA. The review covered e.g. implications of DiD, phases of a nuclear emergency, definitions of dose limits and protective measures, and suitability of atmospheric dispersion and offsite dose projection models, including code tools. Some potential challenges may be encountered e.g. because of urban environment and the requested fidelity of modelling in the very near range. The review of EPZ-related regulations covered e.g. Bulgaria, Finland, Lithuania, Ukraine, and the USA. The implications of EU standards and directives for several EU countries were considered. EPZ methods and regulatory schemes can be classified in several ways, e.g. as generic/universal or generalizable (more case-based); or deterministic, risk-informed or risk-based. The practical, quantitative work in

WP6 EPZ is proceeding in two successive phases: First the participants are doing dispersion, dose and EPZ calculations using simple, but also then by necessity more conservative methods (to be completed in 2024). Conservativeness can exist at many levels, like choice of source term nuclides and their released activities (generally the main factor affecting doses), other release parameters (effective release height and release rate temporal trends), weather parameters or choice of statistical weather fractile, and a large number of dispersion and dose modelling internal parameters. Conservativeness is a subject of debate, and it is well recognized that differing choices may lead to a spectrum of results. The most cautious possible view would be to choose conservatively (i.e. in the direction of higher predicted doses) at every step, as emergency preparedness is usually defined to exist for the unthinkable, independently from plant design. Differing results may also be caused by site characteristics and dose limits in different countries. The codes used for release through water volumes (like reactor pool) were SPARC-90 for pool scrubbing and IMPAIR for iodine chemistry and release. For the atmospheric dispersion and dose calculations, MACCS, JRODOS, or the VTT in-house code ARANO were used. A subsequent best-estimate (BE) phase will be carried out, in a mechanistic fashion by dedicated SA codes, like MELCOR or ASTEC. Not having to resort to conservative expert judgement, it remains to be seen whether source terms and consequently offsite doses and EPZ sizes will turn out smaller than in the first phase. In the final report, evaluations of code suitability, recommendations on appropriate EPZ determination methodology, and also some numerical values for Design 1 and Design 2 accidental offsite doses and EPZ sizes will appear.

6. CONCLUSIONS

The SASPAM-SA project has made significant progress in advancing the safety analysis practice for iPWRs. WP2 identified DBA and BDBA scenarios to evaluate the capability of state-of-the-art codes to simulate the main features of iPWRs. The SA scenarios have been identified, and the code capability to predict degradation phenomena has been tested. The conditions in the containment and in the vessel that characterize iPWR scenarios have been identified, and a dedicated database has been fully developed to be used in WP3 to assess the applicability of existing experimental data to iPWRs. The results of the codes have been compared to evaluate potential differences. Across all simulations of the postulated SA scenarios and using all SA codes, no lower head failure was observed in WP2 analyses; WP4 analyses support these findings, indicating that the maximum heat flux can be managed by pool boiling, and the residual vessel thickness ensures mechanical resistance; therefore, as expected, the first results show that in-vessel retention strategy appears feasible, with a good safety margin. This has allowed the project to build the know-how for the analysis of IVMR phenomena and to identify specific IVR features and near-term code development needs. WP6 is assessing the EPZ for iPWRs by analyzing SA scenarios and their radiological impacts. Initial findings suggest iPWRs may have lower radiological impacts and require smaller EPZs compared to large NPPs, facilitating their siting near populated areas. WP5, related to containment analysis, is set to begin. These code applications enable, on one hand, the building of expertise among code users for SA in iPWRs and the training of new code users (e.g., younger generations), and on the other hand, the assessment of code guidelines and best practices for the simulation of iPWRs. These efforts highlight the project's contribution to advancing the understanding on safety of iPWRs, supporting the possible licensing review process in Europe.

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