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Spanish research related to SMRs projects

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ABSTRACT

Small modular reactors (SMRs) are advanced nuclear reactors with a power capacity of up to 300 MW(e) per unit. SMRs encompass a variety of reactor technologies including light water reactors, high temperature gas reactors, molten salt reactors, liquid metal cooled fast reactors, and heat pipe technology-based reactors. The research and design of these diverse SMR types require a broad set of technological capabilities related to nuclear engineering and safety. Within this context, this article attempts to assess the current state of research and technological progress achieved by the Spanish research groups and companies. The results reveal a significant level of maturity among these groups and companies in various domains such as neutronic analysis, thermal hydraulic analysis, the improvement of models related to severe accident scenarios and an active involvement in the design of novel SMR.

1. Introduction

Small modular reactors (SMRs) are designed to produce mainly electricity (although they can also be designed to produce heat for industrial processes, such as water desalination, H_2 production, etc.) using nuclear units up to 300 MW(e), with some as multi-module power plants with 50 to 200 MW(e) per module. SMRs are appreciated for their simplicity, enhanced safety levels that come directly from their design, shorter construction times and faster return on investment due to their smaller initial costs. However, the economic benefits of SMRs compared to larger reactors should still be proven through their construction and operation. With their reduced footprint, SMRs could be a feasible choice for replacing conventional fossil fuel-powered stations as they become outdated or get closed down, thus catering to the electricity demand in distant regions while profiting from renewable energy synergies. The role of SMRs is not to replace large reactors for supplying baseload electricity to large-scale grids, even though some of them may be financially competitive compared to those large-scale reactors as claimed by their designers and manufacturers. Instead, SMRs are better understood as an alternative for replacing other generation systems. Additionally, SMRs could be a suitable technological option for market niches in countries with small grids or regions far from the main power grid. Along with renewable energy sources and large Nuclear Power Plants (NPPs), SMRs could play a significant role in reducing global warming and mitigating the effects of climate change.

There are dozens of SMR designs worldwide, promoted by reactor vendors like GEH, Westinghouse, NuScale, ThorCon, Newcleo, and EDF, as well as other industrial groups that recognize a market for this type of reactor. For now, there are two SMRs with different design under operation, KLT-40S and HTR-PM, and construction of the CAREM, ACP100 and BREST-OD-300 SMRs has already commenced. In addition, the following design are near-term deployment: NuScale, BWRX-300, RITM-200N, SMR-160, Xe-100 and MMR, (IAEA, 2021) (NEA, 2023).

The technologies used for SMRs differ significantly. The most common SMR technology corresponds to Light Water Reactor SMR (LWR-SMR). However, LWR-SMR include significant design modifications

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from the original LWR large reactor designs, as the implementation of the Helically Coiled Steam Generators (HCSG) or the incorporation of passive safety systems. Other SMR designs use advanced GEN IV technologies like Molten Salt Reactors (MSR), Sodium Fast Reactors (SFR), Lead Fast Reactors (LFR) and High Temperature Gas Reactors (HTGR), also known as Very High Temperature Reactors (VHTR). Every type of power reactor is suited for non-electrical applications as well, including cogeneration of electricity and other purposes such as desalination of seawater, hydrogen production, district heating, and process heat for industries. In fact, the widespread use of nuclear cogeneration, and non-electric applications of nuclear energy, could potentially be a transformative element in achieving a significant reduction in CO_2 emissions and mitigating climate change.

However, SMR designs must follow licensing procedures, as conventional reactors do, and the enhancements incorporated in the designs could present challenging issues when conducting the corresponding safety analysis in each country. In that sense, it is worth noting the nuclear harmonization and standardization initiative undertaken by the IAEA to achieve common rules for the licensing of SMR, regardless of their technology. Besides, conventional simulation tools should be adapted to incorporate the new SMR design features to accurately capture the physics inherent in each concept. This adaptation must be backed by appropriate experimental activities to facilitate validation. Therefore, it is highlighted that Spanish research groups, from universities, research centers, and even private companies, have been engaged in R&D efforts pertaining to GEN III- and GEN IV-SMRs over the last decade. Specifically, the Spanish Nuclear Fission Energy Technology Platform (CEIDEN) has recently launched an SMR Working Group that gathers all these institutions in a collaborative effort to explore specific work on SMRs in Spain.

Particularizing to this paper, the research activities developed in Spanish universities and research centers are described in Section 2. Later, the main industrial developments achieved in recent years by the Spanish companies regarding SMRs issues are commented in Section 3. Then, the expected new developments in Spain related to SMR are discussed in Section 4. Finally, the conclusions drawn from this work are presented in Section 5.

2. Research in universities and CIEMAT

The following sections present the SMR research activities conducted by Spanish universities, including the Universidad Politécnica de Madrid – School of Mining and Energy (ETSIME-UPM), the Universidad Politécnica de Madrid – School of Industrial Engineering (ETSII-UPM), the Universitat Politècnica de Catalunya (UPC), and the Universitat Politècnica de València (UPV), as well as the research center CIEMAT.

2.1. Status and prospects of the ETSIME-UPM research activities related to SMRs

Nowadays, the ETSIME-UPM research group is involved in three projects related to these SMR activities:

- 'High-Performance Advanced Methods and Experimental Investigations for the Safety Evaluation of Generic Small Modular Reactors' (McSAFER) European project, (Sanchez-Espinoza et al., 2021), which aims to advance SMR safety research by combining experimental investigations and different numerical simulations using conventional and multiscale/multiphysics modelling tools.
- 'Passive Isolation Condenser' (PIACE) European project, (Lorusso et al., 2023), which has the objective of developing a self-regulated isolation condenser with non-condensable gases.
- 'Integrated Safety Analysis of Modular and Evolutive Reactors' (ISASMORE) project, which focuses on analyzing the passive safety systems in CAREM-like and large Gen-III reactors.

Within this set of research projects, the following activities are being carried out:

- Review of Thermal-Hydraulics (TH) modeling needs for LWR-SMRs.
- Simulation of experimental tests carried out in several facilities related to SMRs.
- NuScale modeling with TRACE, PARCS and SUBCHANFLOW (SCF) codes.
- CAREM-like reactors modelling with 1D and 3D TRACE models.

Detailed descriptions of these activities are provided in the subsequent sections.

2.1.1. Review of thermal-hydraulic modeling needs for LWR-SMRs

The ETSIME-UPM group in collaboration with other research centers (Karlsruhe Institute of Technology (KIT), Instituto Balseiro and King Abdulaziz City for Science and Technology) carried out a review of the current requirements in the field of TH for LWR-SMRs. For this purpose, a series of activities were undertaken, as outlined in (Queral et al., 2020):

- Analysis of the characteristics of the different types of LWR-SMRs (e.g. natural or forced convection, use of boron for reactivity control, possible occurrence of sequences with spatial asymmetry, etc…).
- Review of the TH phenomena challenging the TH codes.
- Challenges found simulating the LWR-SMR behavior with TH codes.

The conclusions found during this review were:

- A physical sound simulation of the multidimensional flow (single, two-phase) requires 3D TH codes and/or multi-scale coupled codes (e.g. combination of CFD/system codes, system/subchannel codes).
- The physical phenomena on which the majority of passive safety systems rely to remove the decay heat require extensive re-evaluation and validation of the models of the system codes, e.g.: heat transfer and friction correlations for helical pipes in the inner surfaces and tube-banks; dryout conditions inside the helical tubes; condensation heat transfer correlations and flow-maps; CHF correlations for the different LWR-SMR operating conditions.
- The CFD codes also need to be validated/improved in LWR-SMR conditions: Flow mixing by means of turbulence; two-phase flow distribution, including bubbles drift by means of liquid recirculation; for example, bubbles drift from chimney to downcomer, during steady-state conditions.
- New experimental data for key LWR-SMR phenomena is also needed for the validation of the different kinds of TH codes in order to increase the confidence on their prediction capability e.g. related to helical heat exchanger, Passive Residual Heat Removal System (PRHRS) effectiveness, transition from forced to natural circulation within the core, CHF, etc.

2.1.2. Simulation of tests carried out in experimental facilities related to SMRs

The ETSIME-UPM group has participated in two European projects involving the TH simulation of experimental facilities by means of the TRACE code. Within the PIACE project, the modeling of the SIRIO facility has been performed (see ETSII-UPM section), and in the framework of the McSAFER project, the ETSIME-UPM is involved in the tasks related to the simulation of tests conducted in two test facilities: the MOdular TEst Loop (MOTEL) and the High-pressure WAter Test (HWAT):

- MOTEL facility, similar to NuScale reactor design, has been used to perform two experimental series, one is focused on investigating the behavior of the NuScale-like helical steam generator, while the other is devoted to studying the crossflows in the core region under asymmetrical power distribution conditions. The TRACE simulation results show a good agreement with the experimental data, see Fig. 1.

- The purpose of the HWAT experimental facility is to investigate the heat transfer in SMRs, including Critical Heat Flux (CHF) and condensation phenomena. Two test series will be conducted in HWAT based on analyzing the forced circulation steady state and on studying the transition from forced convection to natural circulation regimes. Preliminary calculations with TRACE code have been performed for the first test experimental series and similar results to those from GOTHIC code have been obtained.

2.1.3. Multi-scale and multiphysics analysis on NuScale

In the performed multiscale analyses, the objective pursued is based on the assessment of the potential applicability of a multiscale modelling tool to the NuScale design focusing on the different TH phenomena. On the multiphysics analyses regard, the improvements achieved by means of the application of the correspondent multiphysics modelling tools (3D-NK/TH) are also evaluated. At present, boron dilution, Steam Line Break (SLB) and Rod Ejection Accident (REA) have been analyzed and are described below.

Regarding NuScale multiscale analysis, ETSIME-UPM has built 1D and 3D TRACE models along with a TRACE/SCF model to simulate the boron dilution sequence. Best-estimate analysis of the boron dilution sequence has been performed using the point kinetics model to compute the power evolution considering the reactivity feedbacks and the actuation of the NuScale safeguards.

To do so, a 1D TRACE model of the Reactor Coolant System (RCS) including the HCSGs tubes was built as a first step, see (Sanchez-Torrijos and Queral, 2021) (Bencik et al., 2023), see Fig. 2. Later, a 3D TRACE model based on the previous one was built using two cylindrical VESSELs to consider the special geometry of the NuScale Reactor Pressure Vessel (RPV) and a Cartesian VESSEL to do the modeling of the core with a 1:1 ratio between the hydraulic channels and the number of FAs in the core, (Bencik et al., 2022). As the final stage of this study, a SCF core model has been coupled with the 3D TRACE model by means of the ICoCo (Interface for Code Coupling) methods MEDCoupling library (data is exchanged by means of the different functions and capabilities implemented in the ICoCo standard), and the SALOME platform (acting as the supervisor of the coupled calculation), (Queral and Sanchez-Torrijos, 2023), (Sanchez-Torrijos and Queral, 2023) and (Sanchez-Torrijos et al., 2023a). The SCF core model is built using a FA-wise modelling approach with a total of 37 subchannels for the whole core.

At this point, it is highly remarkable that the dedicated options in TRACE to do the modeling of the primary and secondary sides of the HCSGs were selected in the models ('Tube Bank Crossflow' and 'Curved Pipe' respectively) along with the implementation of the high order numerical technique Van Leer with flux limiters for solving the spatial differences to avoid numerical diffusion as much as possible.

The obtained results indicate that while similar results are obtained for the main plant parameters (boron concentration, reactivity, power, RCS pressure and mass flow rate), the coupled TRACE/SCF calculation

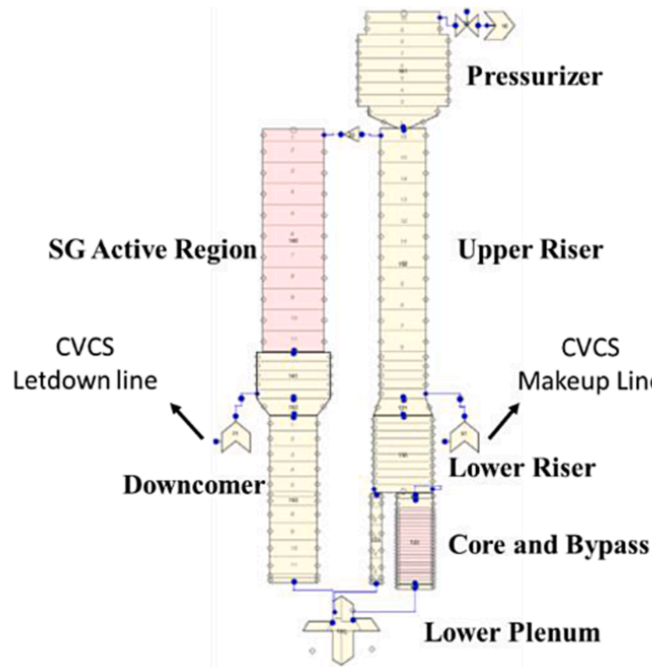

Fig. 2. NuScale RCS TRACE model.

could provide a more accurate understanding of the TH parameters in the core region. However, it is important to note that this improvement comes at the expense of higher computational costs and increased complexity in the modelling process, as stated in (Queral and Sanchez-Torrijos, 2023).

Related to the multiphysics analyses, the ETSIME-UPM is focused on building the inputs for the advance modeling tools TRACE/PARCS and TRACE/SCF/PARCS to perform the simulation of a SLB in the NuScale SMR design, see (Sanchez-Torrijos et al., 2023b) and (Redondo-Valero et al., 2022).

The previous 1D and 3D TRACE models used in the multiscale analyses were upgraded to a full-plant model of the NuScale Power Module up to the Turbine Stop Valve including the PRHRS and the reactor pool, see Fig. 3.

Furthermore, for the development of the PARCS model, the ETSIME-UPM group has received the two energy groups-cross sections calculated with the Serpent code by the VTT group as part of an internal collaboration agreement within the McSAFER project framework. The format of the cross sections has been transformed from the Serpent output to PMAX format by the ETSIME-UPM group using GENPMAX 6.3.1. Then, a PARCS model of the NuScale core using the NEMMG kernel has been developed for the following conditions: Hot-full power at beginning of cycle without Xe and Sm equilibrium. This calculation takes into account the accurate arrangement of the fuel elements as well as the axial and radial reflectors and the application of the assembly discontinuity

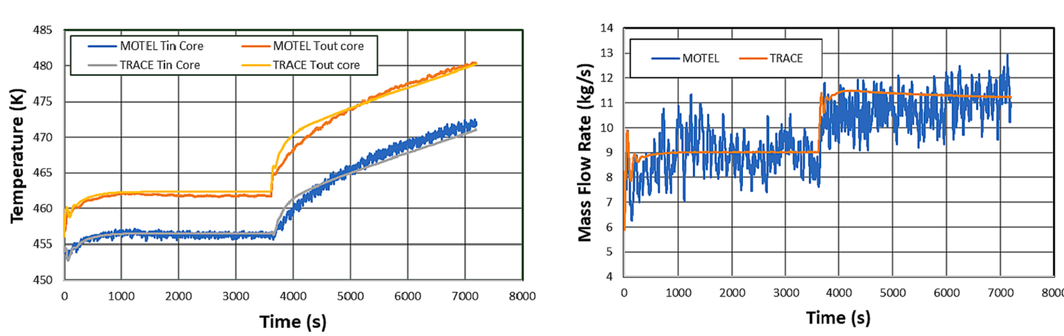

Fig. 1. TRACE MOTEL facility model vs experimental results (power step from 250 kW to 500 kW).

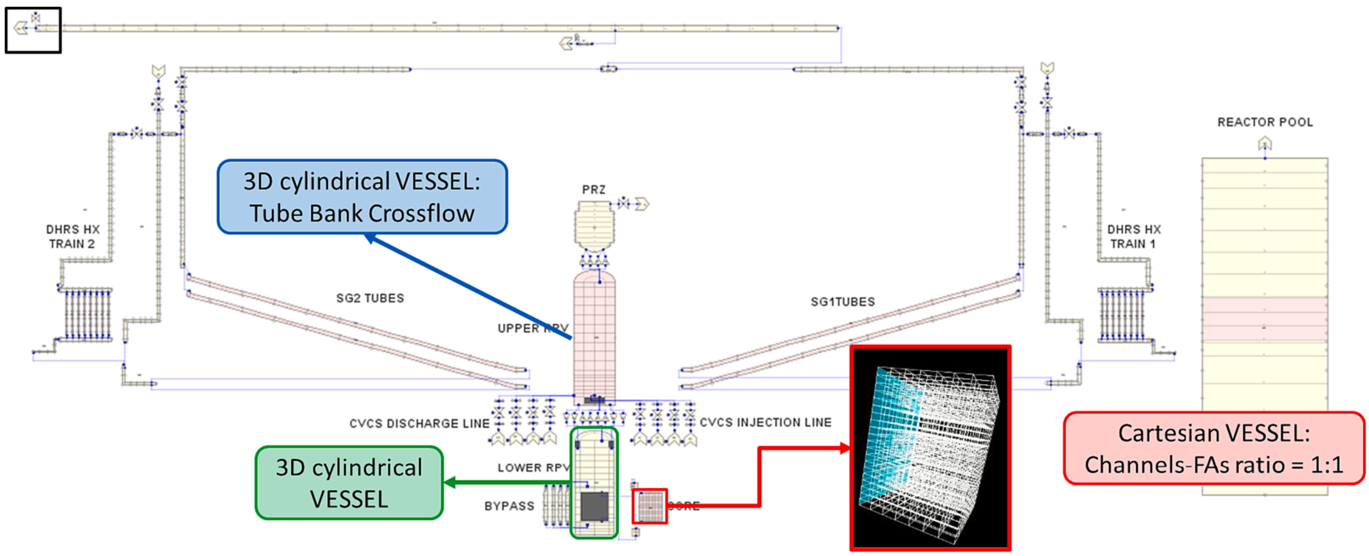

Fig. 3. NuScale nodalization scheme of the full-plant TRACE model.

factors.

The results obtained from the simulation using PARCS standalone, see Fig. 4, were verified against the computed with Serpent showing a very good agreement. By doing so, it was possible to perform the coupled calculations with the 1D-TRACE/PARCS, (Redondo-Valero et al., 2022), to simulate a SLB in the steam line section between the two main steam isolation valves, see Fig. 5.

The next stage of the SLB analysis was based on the usage of the TRACE/SCF/PARCS coupling tool. In this case, the 3D TRACE full-plant model along with the SCF core model, previously developed in the multiscale analyses, and the described PARCS model are externally coupled using the IcoCo methodology developed at KIT, (Sanchez-Espinoza et al., 2023).

Additionally, ETSIME-UPM has been involved in a complementary collaboration with the KIT group for multiphysics core analysis. As a part of this effort, a verification study using PARCS for neutronic and TWOPORFLOW for TH, both coupled using the IcoCo methodology, was performed for the simulation of a REA. The results show a good agreement between the theoretical expectations and the simulation conducted, see (Campos-Muñoz et al., 2023).

		1.09	1.02	1.09		
	1.03	1.16	0.85	1.16	1.02	
1.09	1.16	0.85	0.76	0.84	1.16	1.09
1.02	0.85	0.76	1.04	0.76	0.85	1.02
1.09	1.16	0.85	0.76	0.84	1.16	1.09
	1.03	1.16	0.85	1.16	1.02	
		1.09	1.02	1.09		

Fig. 4. NuScale radial power distribution obtained with PARCS.

2.1.4. *Modeling of CAREM-like reactors*

Regarding the CAREM-like activities, it is worth highlighting that ETSIME-UPM is leading the ISASMORE project, which is funded by the Spanish Ministry of Science and Innovation. This project aims to review and compare passive safety systems in large Gen-III and LWR-SMR reactors as well as the analysis of accidental sequences in CAREM-like reactors. In that sense, several 1D and 3D TRACE models of a CAREM-like reactor have been developed including the PRHRS, which correspond to an isolation condenser (IC), see Fig. 6. Simulations were performed to demonstrate the effectiveness of the IC design in dealing with Station Blackout (SBO) sequences. The SBO simulations indicated that upon reaching a specific pressure setpoint, the IC gets activated, thereby depressurizing the RCS, Fig. 7. This system is able to cool the reactor during 36 h with the actuation of a single train (1 out of 2 trains).

2.2. *Status and prospects of the ETSII-UPM research activities related to SMRs*

The works developed in the ETSII-UPM are divided in four different topics: GTHTR300C reactor simulation, participation in the PIACE European project for the design of passive safety systems, neutronic simulation of SMR-NuScale using the SEANAP system and application of the UPM multiphysics platform to NuScale.

2.2.1. *GTHTR300C TRACE model and non-electrical applications*

The GTHTR300C reactor (Gas Turbine High-Temperature Reactor for Cogeneration) proposed by Japan Atomic Energy Agency (IAEA, 2011) is one of the reactors with the largest amount of public available information. The GTHTR300C is a graphite-moderated and helium-cooled HTGR. Its nominal thermal power is 600 MWt, while its electrical power is between 300 MWe and 174 MWe, depending on if it is used for cogeneration or not (IAEA, 2011). TRACE system TH code was chosen to perform the simulations of the complete plant, studying the viability of the code to simulate HTGRs, as TRACE was originally developed for LWRs. The most complex part of the model to be represented in TRACE corresponds to the helical coil heat exchangers, which need to be adapted to introduce the proper parameters and have been modelled thanks to the "tube bank crossflow" option in TRACE. The results demonstrate the viability of the TRACE code to simulate HTGRs. The absolute error is within the expected limits for a TH simulation of this type (less than 2 %), so the model could be considered a good starting point for performing further transient analysis. In Fig. 8 a SNAP view of the model is presented.

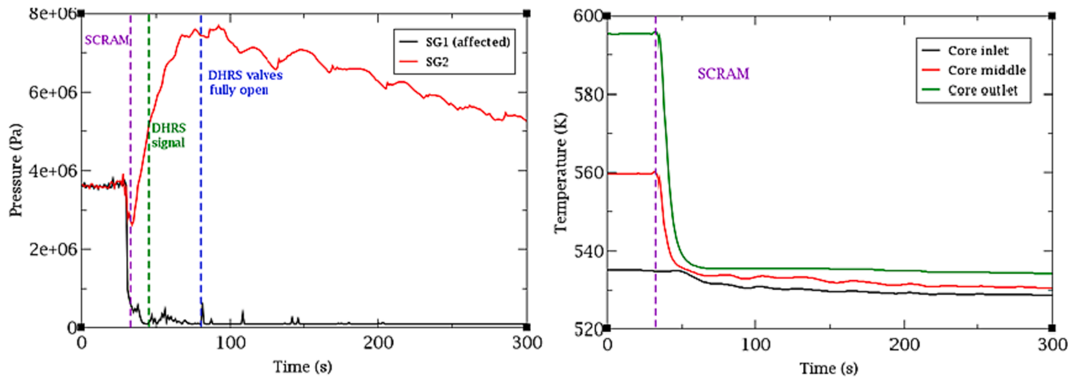

Fig. 5. SLB calculation with TRACE/PARCS model: Secondary side pressure (left) and core temperatures (right).

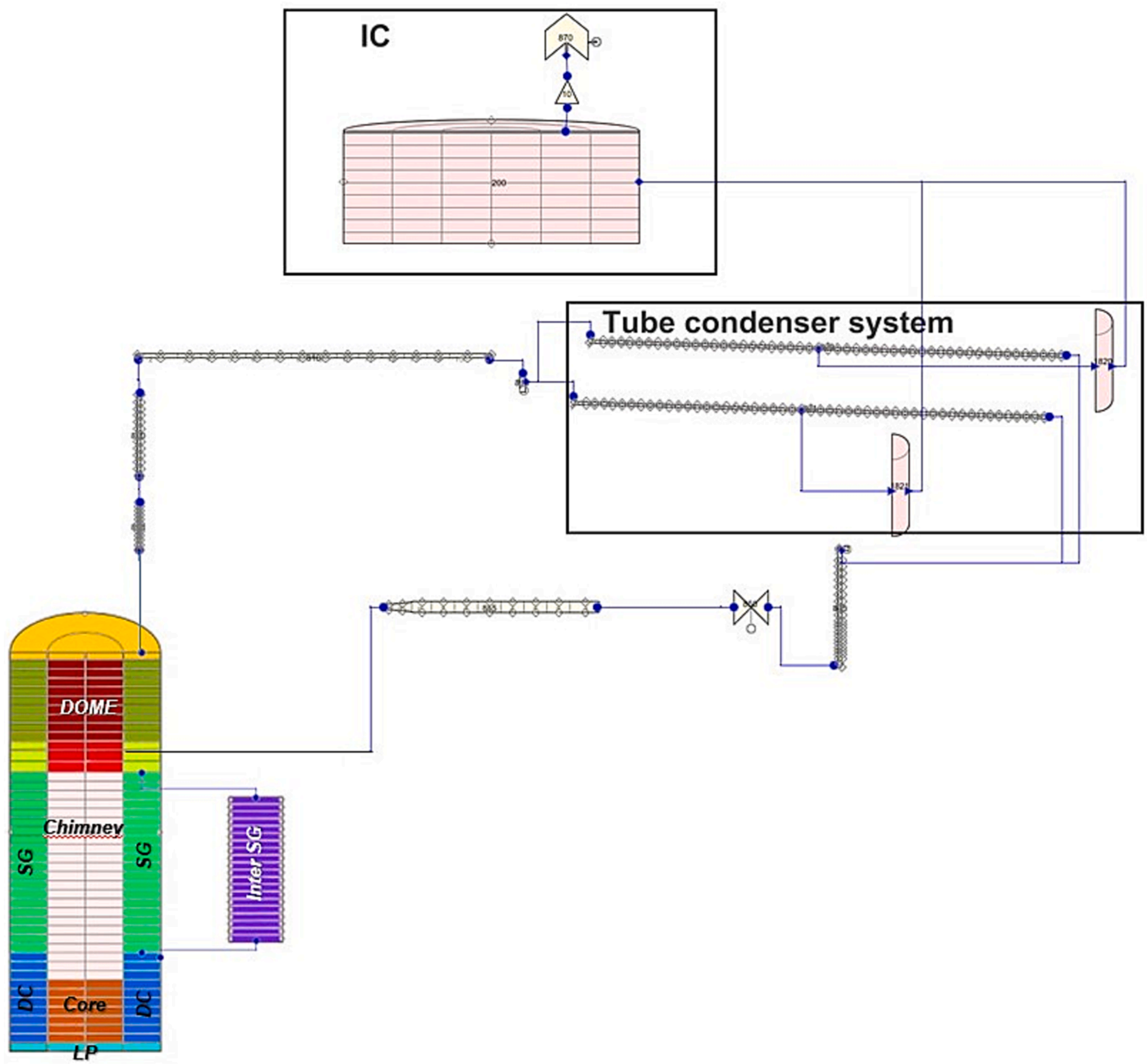

Fig. 6. CAREM-like model for TRACE code including RCS and isolation condenser system.

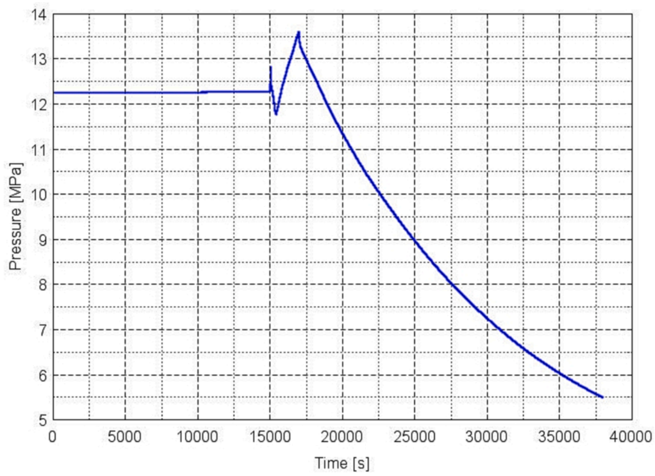

Fig. 7. Primary system pressure evolution during a SBO (CAREM-like reactor).

Another study done at the ETSII-UPM group focusses on one of the advantages of SMRs: the non-electrical industrial applications (Larriba and Jimenez, 2021). The cycle is analyzed with the EES software (Engineering Equation Solver) (F.-Chart-Software, 2018). EES is a program that can solve coupled non-linear algebraic and differential equations. But perhaps the most important characteristic of this software is its fluids thermodynamic database and its capacity to change units and easily define thermodynamic cycles. Four applications are considered: district heating or hot sanitary water, desalination, oil refining, and hydrogen production. For each one, an EES diagram is created and adapted to the problem. The simplest cycle, without modification, was taken to establish the nominal parameters.

Once the main parameters of the reactor of interest are set, the potential for accomplish the requirement for each application should be tested, as it depends on the reactor pressure and temperature at the outlet, as well as the mass flow rate in the circuit (Kasahara et al., 2016). Then, it is possible to choose the most efficient point in the cycle to establish the heat exchanger associated with cogeneration studied in each case. It could be stated that for higher pressures and temperatures, the corresponding heat exchanger should be located after the nuclear vessel, and for lower pressures and temperatures, before the vessel. Since the technology associated with each application has different technical requirements, two industrial applications (high- and low-pressure) plus electricity production were considered in the study, which will be an interesting case of study.

For low-pressure applications, only an improvement in the efficiency is found, but when both applications (high- and low-pressure ones) are included at the same time, the global efficiency is improved and it is possible to adapt the different products to the necessities in a flexible way. For oil refining industries, a high-pressure application, a CO_2 reduction is possible, adding to the electricity demand (and cost) reduction.

In addition, while it is possible to produce pink hydrogen using nuclear reactors, steam methane reforming remains the most cost-effective method. However, Sulfur-Iodine (S-I) process is competitive with steam methane reforming technology if it is considered that S-I is CO_2 free. Indeed, the benefit is higher even than steam methane reforming with carbon capture, which is a more sustainable process.

2.2.2. PIACE HORIZON-2020 project

The PIACE project (Lorusso et al., 2023) has the main objective to support the technology transfer from the research to industry in the area of safety of nuclear installations. An innovative PRHRS for nuclear reactors, presently under technology validation in relevant environment (SIRIO facility) (Herranz et al., 2022a), Fig. 9, will be scaled-up to achieve a system prototype demonstration in operational environment, relevant for LFRs, Accelerator Driven System (ADS) and LWRs.

For the UPM, the assigned technology is the Boiling Water Reactor (BWR). To accomplish WP2 objectives, the following steps were followed. Firstly, in order to understand the behavior of the experimental facility, a TRACE5p5 model of SIRIO was obtained from an available RELAP5-3D model prepared by ANSALDO (Lorusso et al., 2023); afterwards, the ALFRED pre-test simulations results of both models were compared (Redondo Valero et al., 2020). After a deep understanding of the BWR model with the innovative IC system defined in WP1 (de la Fuente Garcia et al., 2020), the next goal was to perform its scaling to

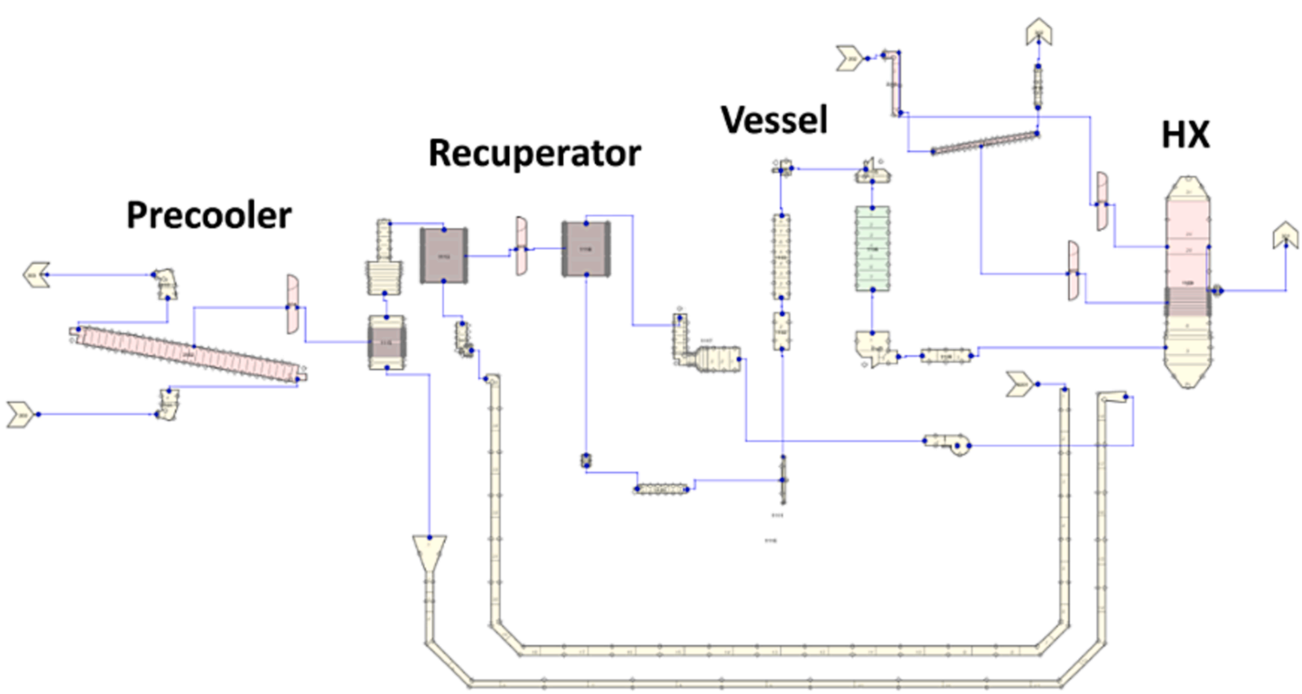

Fig. 8. GTHTR300 TRACE model.

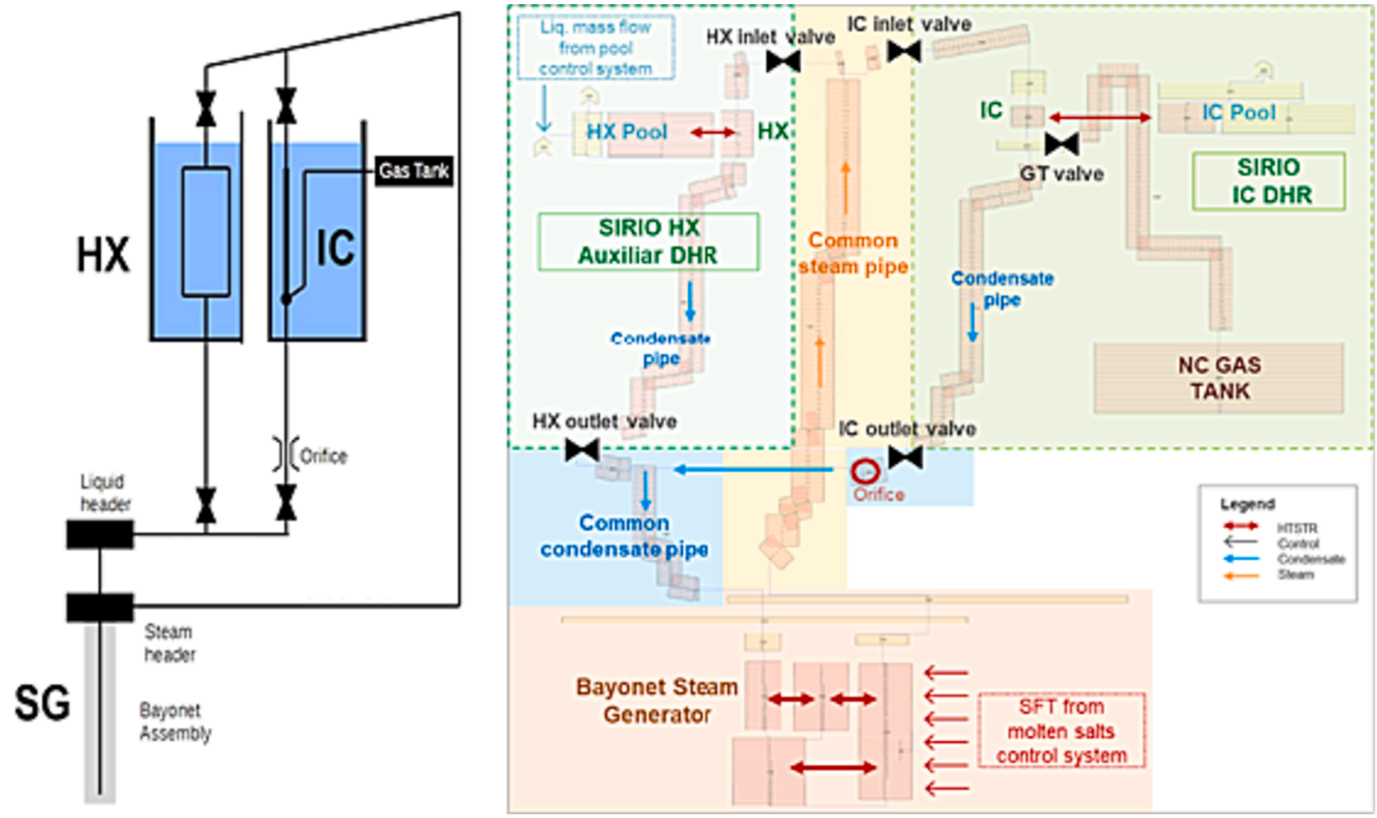

Fig. 9. SIRIO facility scheme with main components.

SIRIO. Some simulations have also been performed comparing TRACE with RELAP5 with the Pressurized Water Reactor (PWR) proposal scheme in collaboration with the Josef Stefan Institute (Larriba et al., 2022), with the aim to learn the behavior of the codes when the system has non-condensable gases and natural convection. In order to adjust the scaling correctly, some SIRIO modifications were gathered with the objective of capturing the TH behavior shown in the full-scale model results. Then, several pre-test simulations with the BWR modifications were performed to obtain the two final experiment configurations. These experiments provide a first insight on how the proposed BWR IC system with non-condensable gases will behave during the enveloping transient, confirming its applicability to SIRIO. It is expected that this technology will be adapted to new design of SMR in a follow-up project, thanks to lesson learned in this project.

2.2.3. Neutronic simulation of NuScale using the SEANAP system

The SEANAP system (Anhert et al., 1999) is an integrated system of computer codes developed at the UPM since 90s. SEANAP provides a full and independent PWR core analysis capabilities. The nuclear design analysis and fuel loading pattern evaluations have been carried out in more than 90 cycles of six Spanish PWR units (Almaraz I and II, Ascó I and II, Vandellós II and Zorita) in the last 30 years.

In recent years, the UPM has developed new capabilities in SEANAP, the most important of which are those aimed to the capability to manage different Advanced Technology Fuel (ATF) concepts, using different nuclear data libraries for the WIMSD5b lattice code and adapting the codes to work with PWR-SMRs.

Then, the neutronic design analysis of a SMR PWR-type using the updated SEANAP system is performed for the NuScale design. For this analysis, SEANAP can predict the boron let-down curve, power peaking factors, axial and radial power distributions, reactivity coefficients, differential and integral rod worth and xenon oscillations for a control bank movement.

Examples of nuclear design parameters for the first cycle at different burnup conditions are given in Fig. 10 which shows the power peaking factor as a function of power level at BOC and EOC. These values are calculated for All Rods Out (ARO) and Power Dependent Insertion Limits (PDIL) conditions.

The effect of different nuclear data libraries in the NuScale design analysis is shown in Fig. 11. The ENDF/B-VII.1 is used as a reference library, and differences in critical boron-let down are shown as a function of the burnup for the JEF-2.2, JEFF-3.1, JEFF-3.3 and ENDF/B-VIII.0 nuclear data libraries. Significant differences are shown for JEFF-3.3 with a slope greater than for other evaluations.

Finally, UPM has been participating within the OECD/NEA/WPEC-SG46 on "Efficient and Effective Use of Integral Experiments for Nuclear Data Validation" for the prediction of the target accuracy requirements (TAR) (OECD/NEA, 2023) in NuScale. The uncertainty quantification in keff due to nuclear data uncertainties gives an uncertainty of 748 pcm in ENDF/B-VII.1, 522 pcm in ENDF/B-VIII.0, 669 pcm in JEFF.3.3 and only 397 pcm in JENDL-5.0.

In addition, this work has allowed to identify the uncertainty reduction needed in nuclear data to achieve a target uncertainty of 300 pcm in the prediction of keff. The TAR exercise gives a target value in the thermal energy range of 0.3 % for the 235U(nubar) and 0.6 % for the 235U(n,gamma) and 238U(n,gamma). This information will be useful to create a new entry in the NEA/High Priority Request List (HPRL) which certainly may motivate new experiments and evaluations in nuclear data.

2.2.4. Application of the UPM multiphysics platform to NuScale

The UPM simulation platform utilizes the standard two-step methodology for reactor core analysis. In the initial stage, a lattice calculation using the deterministic code NEWT from the SCALE system is employed to solve the 2D neutron transport equation for each fuel assembly (FA) within the core. This process takes into account the neutron flux energy,

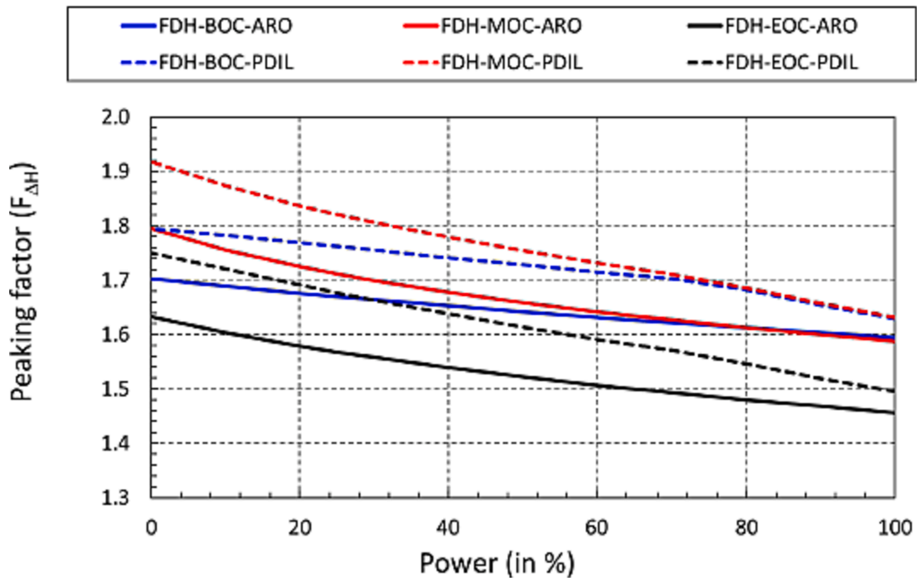

Fig. 10. Enthalpy peaking factor – FDH in NuScale, 1st cycle. Calculations with SEANAP code – nuclear data ENDF/B-VII.1.

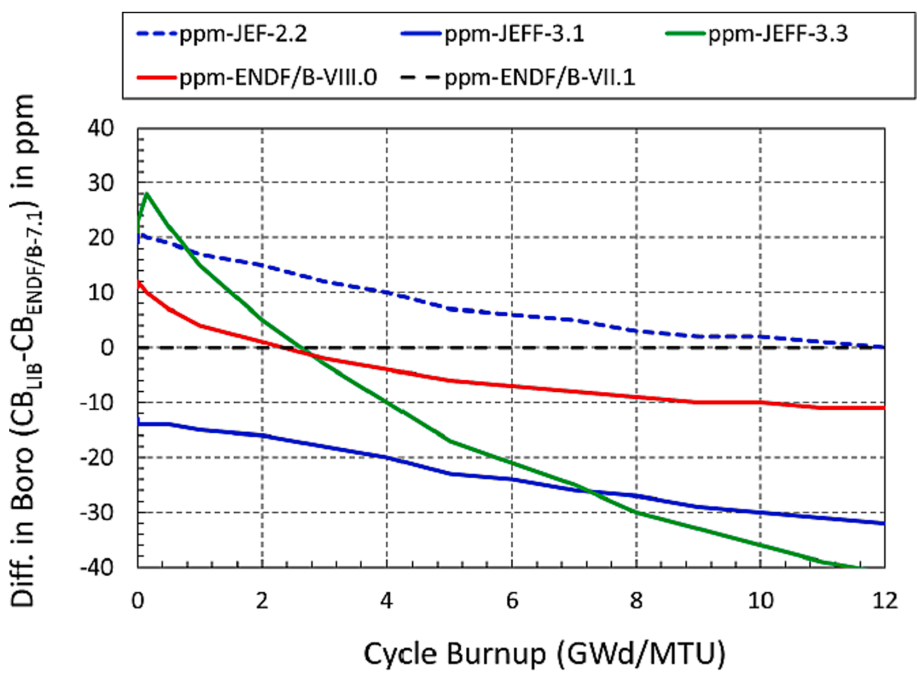

Fig. 11. Differences in critical boron let-down as a function of burnup in the 1st cycle NuScale. Calculations with SEANAP code – nuclear data ENDF/B-VII.1 is reference.

spatial dependencies, and the resonance self-shielding effect, allowing for precise homogenization and two-group collapsed cross-sections at a nodal or a pin-by-pin resolution.

In the second step, a full core 3D coupled simulation is performed using the diffusion code COBAYA4, incorporating the previously obtained cross-sections. Additionally, the simulation accounts for the TH feedback to ensure a comprehensive and accurate analysis of the reactor core behavior. For this feedback, two options are available. In case the TH feedback has not important impact in the results, the SIMULATH module using an enthalpy increase closed channel approach for fluid temperatures calculation may be used. When feedback has a strong impact, the two-fluid three-fields core TH code CTF can be use. This type of 3D calculations may be performed at both, nodal or pin-by-pin resolution.

This system has been developed through the participation of UPM in several European projects, NURESIM, NURISP and NURESAFE, which had as main goal the development of a common European platform for Nuclear Reactor Simulations and Safety Analysis. During these projects, some capabilities of the UPM in-house code COBAYA4 were implemented (García-Herranz et al., 2017).

Application of the UPM platform to water based SMR analysis is part of the next steps which will be developed. For this, the NuScale reactor has been selected preparing the bases for a Best-Estimate Plus Uncertainty (BEPU) approach. The first step has been to perform an estimate for the numerical bias and of the uncertainty when lattice calculations are applied for cross sections generation for this reactor (Duran-Vinuesa and Cuervo, 2021). The analysis of the discretization bias reveals that employing a 16x16 pin-cell resolution effectively limits the FA *k*eff bias to below 20 pcm, leading to significant computational savings of approximately 400 % in CPU time.

The UPM platform has been also applied to the REA at the Beginning of Life, considered in the NuScale Design Certification Application (DCA), see (Durán-Vinuesa et al., 2022). A Best Estimate approach is used to calculate the fuel enthalpy rise and it is compared to the standard adiabatic heat-up model used in the REA NuScale Methodology. The core TH model extends through the space existing between the lower and upper FA nozzles. The boundary conditions are the inlet coolant mass flow rate and temperature and system pressure. These conditions correspond to the nominal operating conditions range given in the NuScale REA Methodology. The realistic model's performance has been verified, illustrating also that the adiabatic heat-up model produces considerably pessimistic outcomes concerning the maximum fuel enthalpy rise (Fig. 12). The differences are maximized for the highest reactivity insertions in sub-prompt critical scenarios.

2.3. CIEMAT contribution to the research on SMRs

CIEMAT contribution to the SMRs investigation may be synthesized according to the different technologies addressed along the last decade: HTGRs, SFRs and LWR-SMR.

In addition to these studies, which describe roughly a decade of research, CIEMAT developed capabilities closely related to the water-cooled SMR technologies related to safety passive engineering features. In particular, CIEMAT developed phenomenological models of the passive containment cooling systems of mid- and large-size nuclear reactors that have inspired some of those included in the SMR designs (Herranz et al., 2000; Herranz et al., 1998; Herranz et al., 1997; Muñoz-Cobo et al., 2005). The main contributions made for the three technologies listed above are summarized in the next sections.

It is worth mentioning that such works were developed under a diversity of frameworks, from bilateral agreements with nuclear utilities, like General Electric and PBMR Ltd., to collaborations with Academy (Universidad Pontificia COmillas, UPCO-ICAI; University of Wisconsin, UW; UPV).

2.3.1. High temperature gas-cooled reactors

Given the distinctive nature of He as a coolant compared to water, the confinement/containment system of an HTGR becomes a key element. CIEMAT modeled the performance of a postulated HTGR vented confinement under prototypical accident conditions resulting from a small and a large breach in the He pressure boundary (Fontanet et al., 2009). By using the ASTEC and CONTAIN codes, the TH response was thoroughly characterized. As for the aerosol behavior, the analytical tools agreed that most of the fission products inventory coming into the confinement would be eventually depleted onto the walls and only about 1 % of the aerosol dust would be released into the environment. The effect of having a scrubbing pool in the pathway of fission products towards the confinement vent was also modelled with ASTEC 1.3 for a very large break accident (Herranz and Fontanet, 2013). Water pools would strongly change the TH evolution within the building and would become efficient aerosol traps, which scrubbing efficiency would depend on their configuration (i.e., vent cross section and pool submergence). A correlation was developed for the Decontamination Factor (DF) in terms of the pool geometrical features (i.e., pool submergence, H; and venting cross section, S):

$$DF_{global} = 3.54 \cdot H \cdot S^{-0.25}$$

In comparison with the dry confinement configuration, the source term reduction in the wet version was roughly a factor of 10. Fig. 13 compares the normalized mass released to the environment of a dry vented confinement compared to two wet-confinement configurations.

Outside the safety domain, in collaboration with UPCO-ICAI, CIEMAT explored the potential of VHTRs to enhance their thermal efficiency by using Brayton cycles instead the traditional Rankine cycles used in large LWRs. Some specific aspects, like regenerative reheating and the use of inter-cooling and reheating (Herranz et al., 2009), were proved to have the potential to significantly impact thermal efficiency. Some interesting concepts explored in these works were also nuclear cogeneration and load follow-up adapted configurations.

2.3.2. Sodium-cooled fast reactors

In the domain of SFRs, CIEMAT focus has been on modeling radioactive transfer from Na pools to the gas atmosphere by two mechanisms: vaporization from hot pools and Na-pool fires. Both scenarios might set up if a disruption of the core happened and caused a gross rupture of fuel pins and some FAs. The fuel-coolant interaction could lead to a foaming and expanding bubble of vaporized sodium with core debris, fuel components and gases. The mechanical uplift of the overlying Na would then impact the reactor vessel head causing a breach in the primary system and the ejection of contaminated Na into the containment. There, the Na

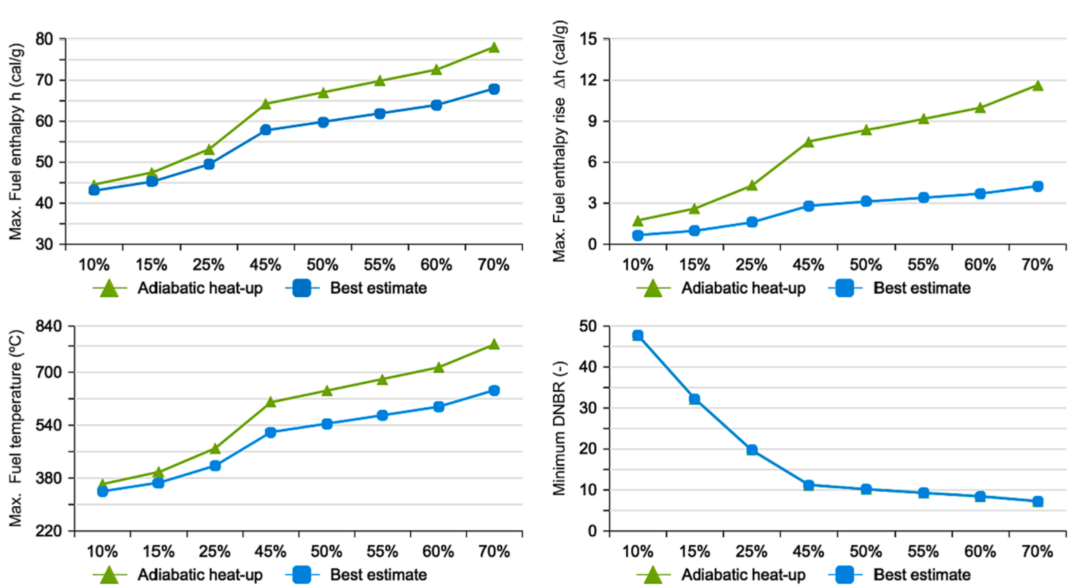

Fig. 12. REA NuScale analysis results (realistic and adiabatic models).

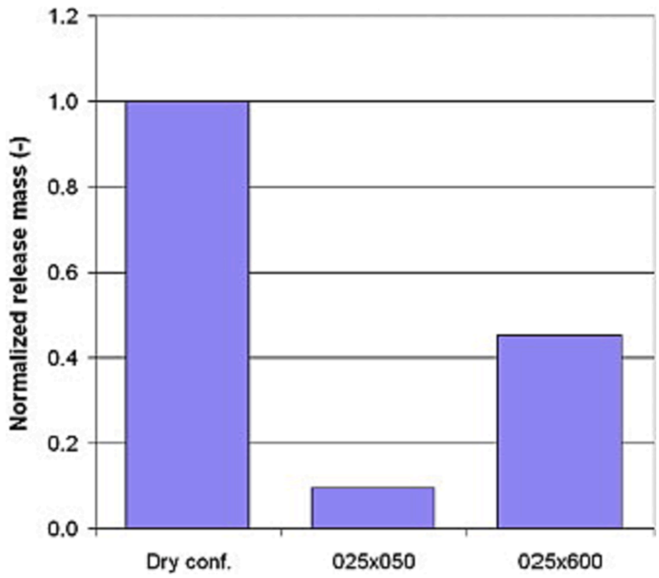

Fig. 13. Normalized released mass to the environment: dry (1st column) vs. wet confinement (2nd and 3rd).

solution of fission products (FP) might get hotter due to FP decay and, even, get oxidized to burning in the presence of oxygen.

FPs dissolved in the hot sodium pool diffuse away from the liquid surface at the same time as sodium vaporization takes place enhancing the FPs release. In other words, there are two driving mechanisms for FPs to get transferred from the pool to the gas phase: diffusion of FPs and convective dragging by the Na vaporization from the pool. A pseudo-mechanistic approach based on the diffusion film theory, the analogy of heat-mass transfer and the Raoult's law has been proposed (Gas Mass Transport, GMT model; (Garcia et al., 2014)). The results of this model were compared to data from the NALA II experimental program in terms of the NaI pool Retention Factor (RF), as shown in Fig. 14. The results were discussed to mean a substantial enhancement of qualitative and quantitative predictability, and no less important, in all the cases they were conservative.

In the presence of oxygen, combustion of Na would result in the formation of huge amounts of Na-oxide aerosols that might become the governing source of airborne radioactivity inside the containment. This, together with the potential harm associated with the chemical species resulting from the Na-oxides reaction with water vapor present in the atmosphere, would be responsible to a great extent for the radiological and chemical impact of any release to the environment. A phenomenological Particle Generation (PG) model covering sodium-vapor burning and formation of sodium-oxide aerosols above an evaporating sodium pool has been developed with the objective to calculating the characteristics (number and size) of the particulate source term to the containment (Garcia et al., 2016). Based on a flame sheet approach, the model articulates a suite of individual models: Na vaporization (diffusion layer approach), O2 transport by air natural circulation (3D flow pattern modelling), Na-O2 chemical reactions (instantaneous reactions and energy input) and vapor-to-particle conversion of Na-oxides (i.e., classical nucleation theory and heterogeneous condensation). Fig. 15 shows a schematics of the main pillars of the PG model.

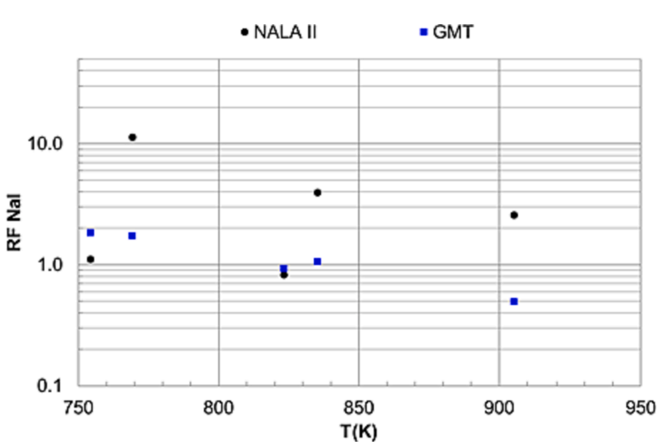

Fig. 14. NaI RF vs. pool temperature.

Once validated through sodium pool experiments, a derivation of suitable analytical correlations for use in a severe-accident code allowed its implementation in the ASTEC-Na code (Herranz et al., 2017). The extended ASTEC-Na code predictions showed a promising response in terms of order of magnitude of airborne concentration, dominant depletion mechanism and particle size variation when compared with experimental databases (Herranz et al., 2018).

2.3.3. Water-cooled SMRs

Recently, CIEMAT joined the first European joint venture addressing severe accidents in LWR-SMRs (the EC SASPAM-SA project). The highest-level objective of the work is to achieve a sound knowledge and know-how transfer from the experience gained in the LWR severe accident domain and adapt whatever necessary. The major challenges CIEMAT is currently addressing are: exploration of potential accident scenarios that could lead to severe consequences in an integrated PWR (iPWR), assessment of the applicability of the existing pool scrubbing database (Herranz et al., 2022b), and reaching the feasibility and consequences of Accident Management (AM) measures, like In-Vessel Melt Retention (IVMR) and Filtered Containment Venting (FCV), among others. This work is foreseen to be developed in the next three years.

2.4. Status and prospects of the UPC and ENSO research activities related to SMRs

Energy Software Ltd. (ENSO), in collaboration with the UPC, has developed an iPWR input nodalization for both US Nuclear Regulatory Commission (NRC) RELAP5 mod3.3 (R5m33) and ISS RELAP/SCDAPSIM (RS35) codes. This work was carried out within the framework of the CAMP-Spain project and the participation in the IAEA CRP "Advancing the State-of-Practice in Uncertainty and Sensitivity Methodologies for Severe Accident Analysis in Water Cooled Reactors of PWR and SMR Types". The accident scenario simulated was a SBO with all the Emergency Systems fully available at low pressure, and the calculations were stopped by a trip of creep rupture signal, or a maximum end time set to five days. A BEPU analysis was carried out in order to advance in the application of Uncertainty Quantification methodologies in Severe Accident simulations.

The reference plant for this project is an iPWR with natural circulation operation. This model was generated from publicly available data of CAREM reactor and from engineering judgment and sensitivity analyses. The nodalization (see Fig. 16) includes the major components of a typical PWR primary system within the reactor pressure vessel as well as passive safety systems. The number of control blocks and trips is very reduced because only one active system is defined for the feedwater. This system adjusts the RPV pressure to the set point value. Otherwise, the different passive safety systems included in the nodalization are:

- High pressure gravity driven injection system with highly borated water to shutdown the reactor if there is a malfunction in the rod insertion.
- Emergency accumulator system that injects subcooled water when RPV pressure drops below 1.5 MPa.
- PRHRS: which condensates the steam of the RPV through nearly horizontal tubes placed in a 16 m^3 tank heat exchanger.

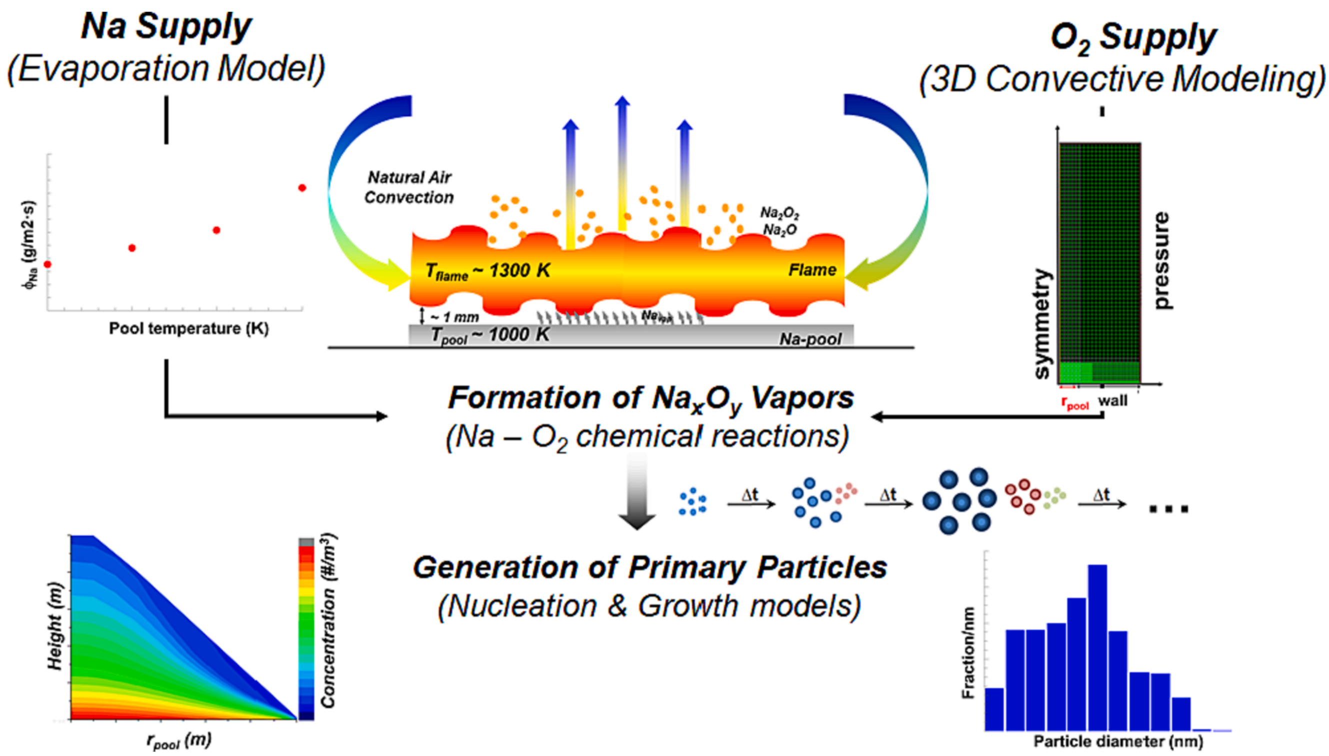

Fig. 15. Key elements of the PG model.

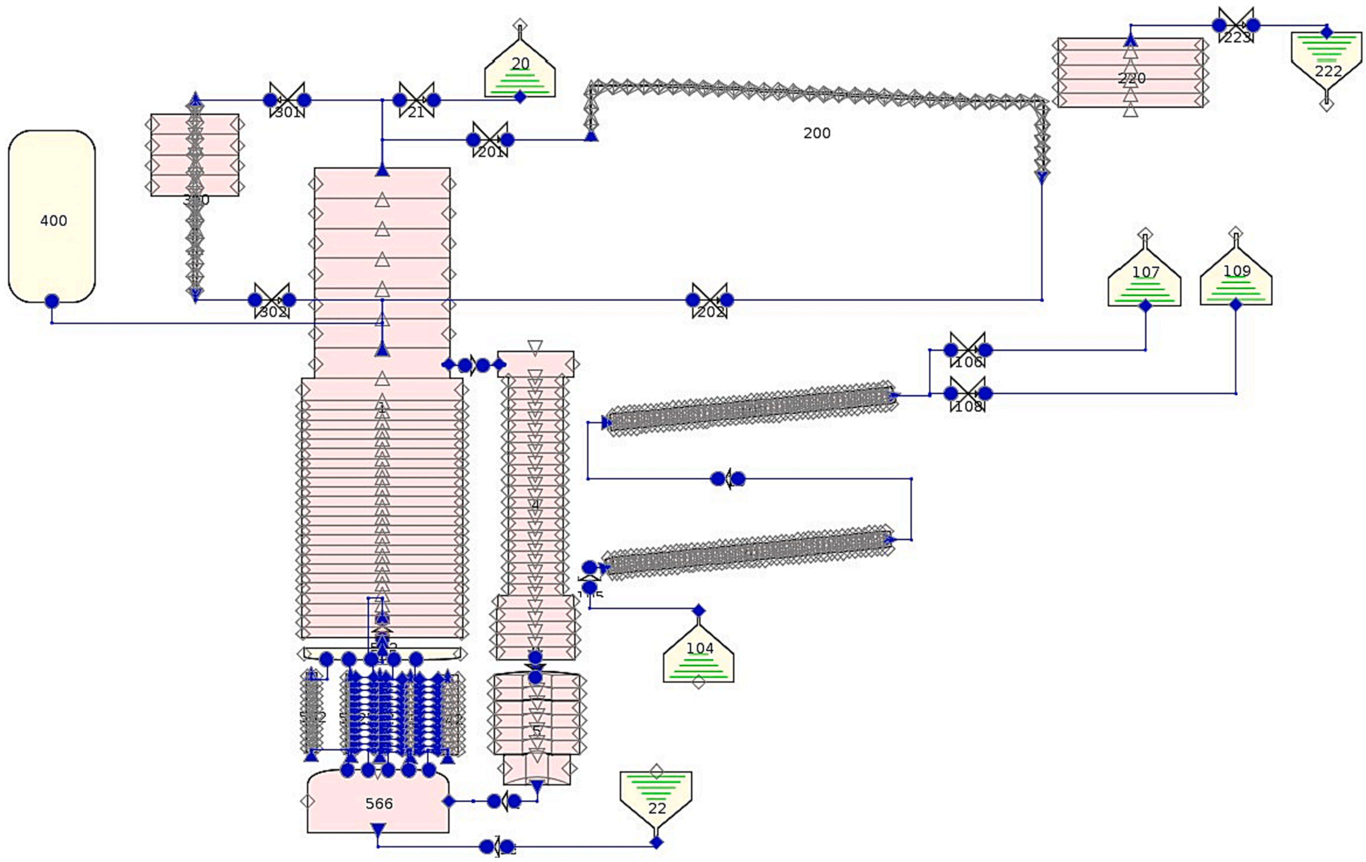

Fig. 16. iPWR nodalization in R5m33.

- Pressure Relief Valves: passive control system to avoid pressures higher than 15 MPa in the RPV. It also depressurizes the RPV if core dryout and steam superheating is detected at the core exit temperature.

The results of the steady state simulations, Table 1, showed a close agreement in all the simulations with an acceptable maximum deviation of the 1 % for the RPV mass flow rate of the RS35 simulations. In this sense, it is important to remark that hydrodynamics equations and convection heat transfer modes of RELAP/SCDAPSIM are based on models and correlations of RELAP5mod3.2, hence slight differences could be expected between R5m33 and RS35 simulations. Furthermore, it was observed that the form losses had to be modified in the core junctions of the RS35 SCDAP nodalization to obtain steady state conditions (from K = 1.8 to K = 1.2). These modifications seemed to be related to the SCDAP components that increase the frictional effects in the hydrodynamic components that are coupled. With the adjusted K losses, the results were quite similar to those obtained for the nodalization with standard RELAP5 heat structures.

The results for the SBO simulation showed consistency and quite good agreement between codes, for both the main events and the relevant phenomena (see Fig. 17). In addition, results of the simulations seemed to confirm the 36 h grace period for SBO scenario of the CAREM design plus the extended 36 h grace period associated to the availability of EIS in Loss of Coolant conditions reported by designer. Further details regarding the iPWR nodalization and the BEPU analysis for an iPWR SBO scenario can be found in references (Ahn et al., 2022) and (Martinez-Quiroga et al., 2022).

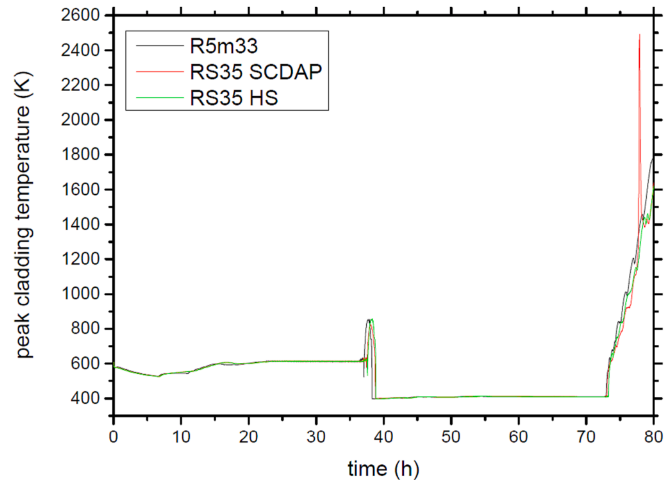

Fig. 17. Peak cladding temperature in SBO simulations with R5m33 and RS35.

2.5. *Thermal-hydraulic-Neutronic analysis of the NuScale reactor at UPV*

UPV, together with Universidade Federal de Minas Gerais (UFMG), developed, on the one hand, a TH model of the NuScale SMR and, on the other hand, neutronic models of both the core and the fuel elements to generate a cross-section table for future use in a 3D TH-neutron kinetics coupled calculation with TRACEv5p7 and PARCSv3.4 (SP3 solver) codes. These works are being performed in the framework of the Code Application and Maintenance Program (CAMP), sponsored by the US NRC and the Spanish Nuclear Regulatory Body (CSN).

The TH model in TRACE consists of a cylindrical VESSEL in which the core is modeled with a Cartesian VESSEL. Both VESSELs are generated automatically from a program developed in Matlab©. This model is improved with the inclusion of the HCSGs as well as the secondary system with boundary conditions. TH steady state results have been verified against NuScale DCA data (NuScale Power LCC, 2020) obtaining good agreement results.

Regarding the neutronic part, the NuScale reactor is modeled with Serpent 2.1.32 using the data available from the DCA (NuScale Power LCC, 2020), see Fig. 18. The reactor core is, for now, modeled without boron. Only a little information has been provided regarding fuel rods, guide tubes, and the arrangement of Gd fuel rods. Since the assembly design is based on an existing 17x17 PWR, the FAs were modeled according to a PWR standard design. At the same time, the FA rods with burnable poisons were divided into 10 concentric rings. Based on the simulation of NuScale core model with CASL VERA code (Suk et al., 2021), it was decided to vary the concentration of Gd_2O_3 between 0 % and 8 %, with an interval of 2 %.

The neutron model defined in PARCS requires the data of kinetic parameters and cross-sections defined at the operating point to be simulated. This process of obtaining cross-sections is currently being carried out. Specifically, UPV is working with the Serpent and SCALE (Polaris module) codes to obtain the cross-sections library necessary to complete the neutron model. In the first study, the JEFF-3.3, ENDF/B-VII.1, and ENDF/B-VIII.0 have been used, and the results obtained with Serpent are shown in Table 2. The neutron population parameters were set at 2200 cycles and 10,000 neutrons per cycle. A NuScale reactor model is also being developed using KENO-VI, NEWT, and POLARIS codes from the SCALE package to compare the results with Serpent and generate a library of collapsed and homogenized cross-sections for nodal codes.

3. Industrial activities related to SMR projects

The SMR projects in which Spanish industry is participating are presented in the following sections. The industries involved are IDOM Consulting, Engineering, Architecture SAU, Tecnatom and Empresarios Agrupados Internacional (EAI).

3.1. *IDOM consulting, engineering, architecture SAU*

Since 2019 IDOM is one of the providers for engineering and consulting services related to the development of SMR technologies and its applicability for a variety of industrial and transport applications.

Table 1
iPWR steady state results.

Parameter	Units	Expected Value	R5m33 K = 1.8	RS35 HS K = 1.8	RS35 SCDAP K = 1.2	Deviation (%)
Thermal power	MW	100.0	100.0	100.0	100.0	0.0
Primary pressure	MPa	12.25	12.25	12.25	12.25	0.0
Secondary pressure	MPa	4.7	4.7	4.7	4.7	0.0
Core inlet temperature	K	557.0	561.0	561.6	561.6	0.8
Core outlet temperature	K	599.0	599.2	599.3	599.3	0.0
RPV mass flow rate	kg/s	410.0	407.9	406.1	406.1	1.0
RPV collapsed liquid level	M	–	6.7	6.7	6.7	–
Secondary inlet temperature	K	473.15	473.15	473.15	473.15	0.0
Secondary outlet temperature	K	563.2	565.0	565.2	565.2	0.3
Secondary mass flow rate	kg/s	–	48.4	48.4	48.2	–

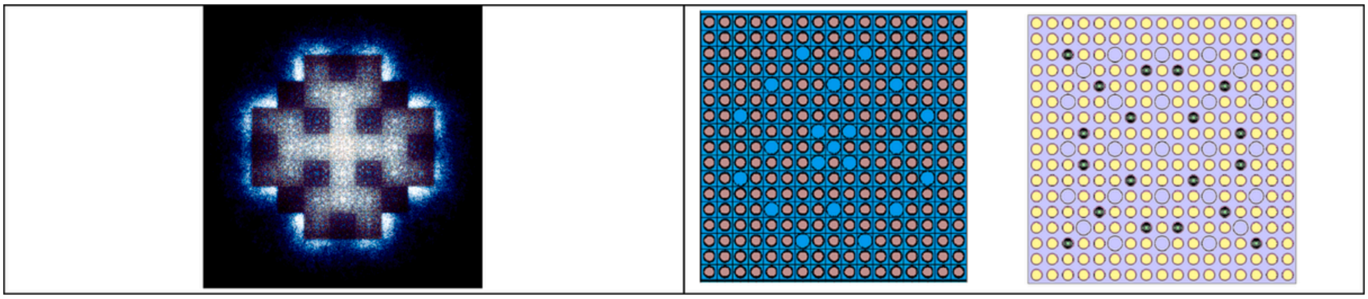

Fig. 18. Neutron models in Serpent of the NuScale core and fuels.

Table 2
Keff values for different libraries with their mean values and standard deviation.

Gd2O3 (%)	0 %	2 %	4 %	6 %	8 %
Keff (ENDF/B-VII.1)	1.37994±0.00014	1.17997±0.00017	1.17180±0.00017	1.16721±0.00017	1.16359±0.00018
Keff (ENDF/B-VIII.0)	1.37669±0.00014	1.17771±0.00017	1.16949±0.00017	1.16522±0.00018	1.16184±0.00017
Keff (JEFF-3.3)	1.38233±0.00014	1.17354±0.00017	1.16472±0.00017	1.16009±0.00018	1.15670±0.00017
Mean Keff value	1.37965	1.17707	1.16867	1.16417	1.16071
Standard deviation	0.0023	0.0027	0.0030	0.0030	0.0029

3.1.1. SSR-W reactor and WATTS reprocessing facility

Moltex Energy Canada Inc. (MEC) is developing its patented unique designs for the SMR Stable Salts Reactor Waste-burner (SSR-W) and Waste to Stable Salts reprocessing facility (WATSS), Fig. 19 see also (Moltex Energy Canada Inc, n.d.) and (IAEA, 2020). IDOM, as a strategic partner of MEC, has had a key contribution to accomplish the pre-licensing Vendor Design Review (VDR) phase 1 for the SSR-W conducted by the Canadian Nuclear Safety Commission (CNSC) covering several key areas like FA5 – I&C –, FA7 – Emergency Heat Removal System –, FA8 – Containment –, FA11 – Pressure Boundary –, FA14 – Out of Core Criticality –, and contributing in other 6 areas out of 14. The final outcome from this pre-licensing stage is that there is no significant

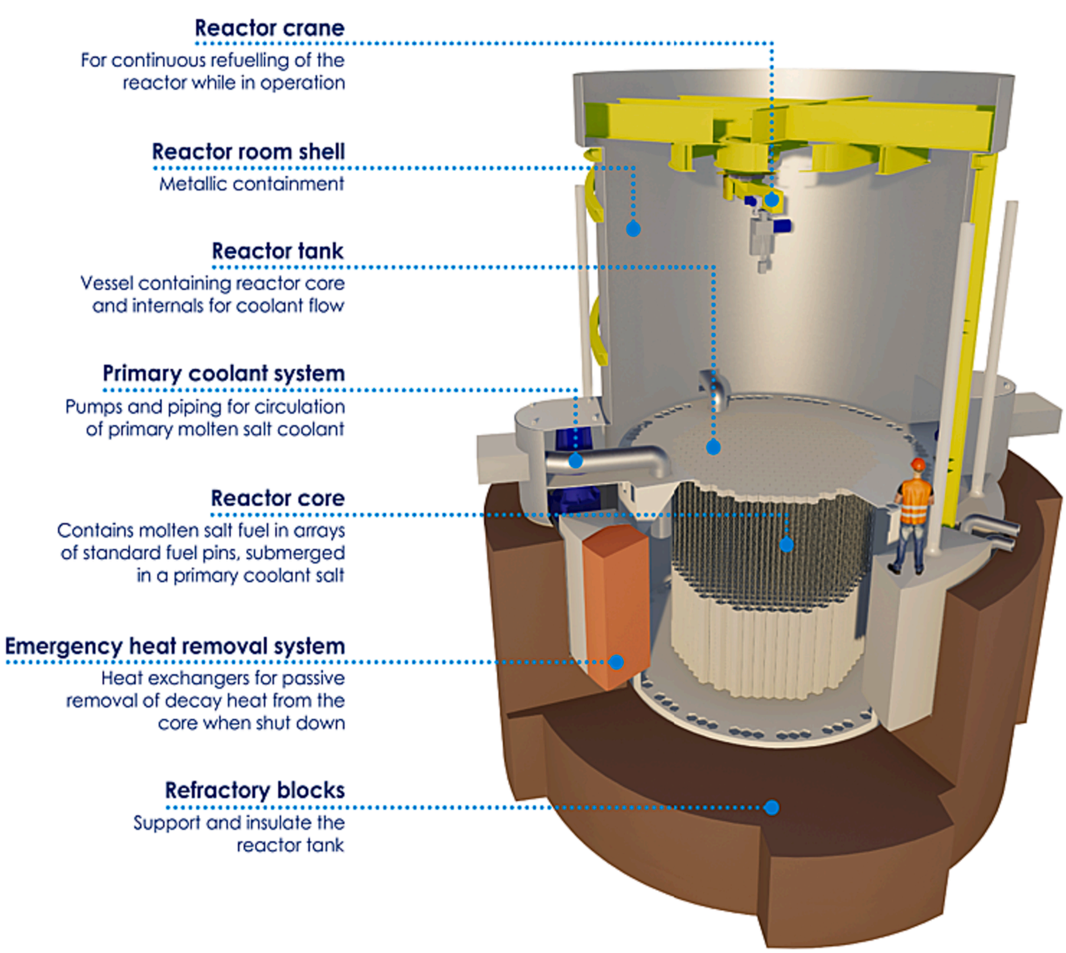

Fig. 19. SSR-W Reactor.

barriers for a future reactor licensing.

IDOM engineering has leaded or collaborated also, at feasibility or conceptual or option-engineering level, in the SSR-W design of:

- Fuel Assemblies, mechanical finite elements analysis have been performed in order to check the structural integrity and material selection to withstand not only with the forces transmitted by the flow patterns to these structures, but also with the working conditions inside the reactor core. Also CFD analyses have been conducted to check heat removal from the assemblies when placed in reactor environment out of the molten salt cooling pool.
- In vessel flow patterns have been modelled in Star CCM+ software to check heat flow removal from the core in operation – direct flow – and shutdown conditions – reversal flow -. Several in vessel components like bottom diagrid, downcomers, or main pipes suction localizations were designed and also at FA level, different components like wrapper, bottom nozzle or pin pitch. Other CFD analyses have been carried out pre-size the Emergency Heat Removal System based on air natural convection surrounding the reactor vessel.
- Reactor vessel and means for supporting were modelled in finite elements with ANSYS in order to provide a preliminary design of these components complying with the mechanical stresses allowable.
- Primary Heat Exchangers different configurations were assessed in an option-engineering approach in order to find out the most suitable configuration complying with the technical and economical requirements.
- Several neutronics models have been assessed to check among others reactor core features like Clorine-36 generation, practicability of the core based on chloride fuel salts and cooling salts – parametric modelling carried out with Serpent, Helium generation in fuel pins cladding, etc.

IDOM has also participated in the preliminary and conceptual design of the WATSS facility (Moltex Energy Canada Inc, n.d.) – from different approaches: some key tests, to be performed in Canadian National Labs (CNL) in order to verify the actinides extraction performance of the process, have been outlined and also high-level layouts and main process and auxiliary systems definitions have been carried out.

3.1.2. MSR stable salts reactor

MoltexFlex is developing the design of the MSR Stable Salts Reactor Uranium based see (Moltex Energy Ltd, n.d.) and (IAEA, 2020), Fig. 20. IDOM has actively participated in the conceptual design of this reactor performing preliminary civil calculations for the reactor pit, option-engineering analysis for tank vessel design and manufacturability, neutronic benchmarking with Serpent to assess core performance, desalination application assessments and ROM5 cost assessments for the complete nuclear island.

3.1.3. Nuproship-I initiative

The ultimate objective of Nuproship-I initiative is to identify which GEN-IV nuclear technology, if any, can be developed further for commercial shipping in direct unsubsidized competition with Heavy Fuel Oil (HFO) while satisfying all requirements from all stakeholders.

IDOM is part of a consortium, led by the Norwegian University of Science and Technology, that is developing the NuProShip-I project. IDOM is responsible for conducting the Nuclear Island analyses. This task involves addressing a range of issues related to nuclear physics, reactor physics, and nuclear engineering, as well as ensuring nuclear safety and security, radiation protection, land infrastructure, decommissioning, and management of spent fuel and radioactive waste. IDOM is also in charge of the selection of the most suitable SMR concept to be applied to this particular application; the Analytical Hierarchy Process, a method developed by the professor Thomas L. Saaty – (Saaty, 2010), is being applied as a technique to organize and analyze the complex selection decision. Preliminary results after the application of this methodology have provided interesting conclusions regarding the best designs to be set up on a ship.

3.2. Tecnatom SMR simulators and human factors engineering

There is an extensive experience in the use of full scope simulators for training activities of operating personnel and familiarization of other technicians. The scope in systems, fidelity and benefits is well defined and based on standards (ANSI/ANS 3.5) with a broad international consensus.

However, a training simulator developed under these standards can also be a very powerful tool, and even unique, which can add value in other areas. These are full-scope simulators which integrates all the

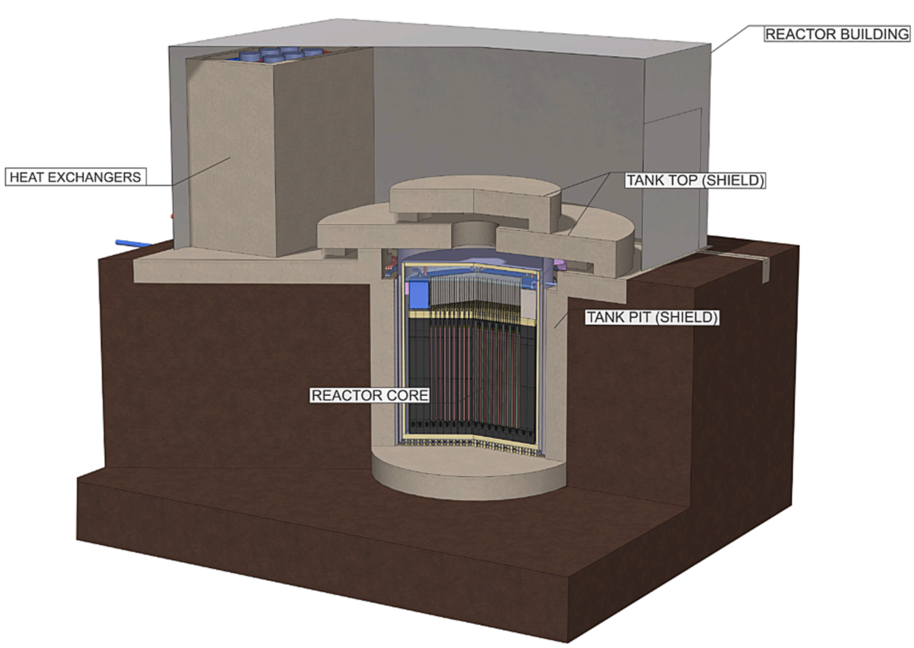

Fig. 20. Moltex Flex Reactor Nuclear Facility.

dynamic systems, control and protection loops, and all the operating modes of the plant with interface in the control room under a replica appearance. The dynamic models use the state of the art in the TH simulation codes of general purpose. The control and protection systems reproduce exactly the real control loops using automatic translation of the algorithms and the operation displays, or even their stimulation (software or hardware in the loop concepts).

It is, therefore, a unique environment to evaluate the design of the new SMR plant in all operating situations and considering all the interrelationships between physical and control systems. This is where also, the discipline of Human Factors Engineering (HFE) applied to new SMR plants can be benefit from Full Scope Simulators. These ideas are being established in the community through the SAE (Simulation Assisted Engineering) concept for which there is currently no consensus standard, but on which there are already some initiatives in that direction (ANSI/ANS 3.5.1).

Tecnatom has developed two SMR Simulators, both of them with Tecnatom simulation Technology, Fig. 21. The first one, a generic iPWR design provided to IAEA and another 12-units iPWR recently provided to IFE (Institute for Energy Technology in Norway) for HFE purposes mainly.

3.2.1. Tecnatom SMR simulators

One of the SMR simulators developed by Tecnatom is the IAEA SMR simulator. This is a full operational scope simulator between hot standby and 100 % power, of a generic iPWR type plant. The Nuclear Steam Supply System (NSSS) and Balance of Plant (BOP) are solved with the Tecnatom best estimate code TRAC_RT, the containment is solved with a second instance of a TRAC_RT 3D model, the neutronics uses the NEMO_RT 3D simulation code, and the auxiliary and safety systems are solved using our TH graphical model builder TEAM_FLOW. The simulator admits several configurations on the basic design (natural or forced circulation, cold source through cooling towers or lake).

The control and protection logics has been developed ad hoc for the simulator by using a graphical simulation tool, TEAM_LOGIC. The simulator supports various control configurations on top of the basic design (turbine follows reactor or vice versa). This has been perhaps the most innovative element, since it required the design from scratch of stable control logics, especially with regard to the level control on the secondary side of feed water/steam to the turbine. The simulator belongs to the IAEA's public battery of simulators, which allows its dissemination for the knowledge of the fundamentals of technology related to SMRs (IAEA, n.d).

Another SMR simulator developed by Tecnatom is the IFE simulator. It includes a full operational scope between cold shutdown and 100 % of a generic plant with up to twelve iPWR-type reactors in natural circulation and the heat sink in the lake. The twelve reactors are autonomous (each of them can be in a different operational state or at a different time in the cycle) up to its electrical generator, but there are only two connection points to the electrical network from two-unit transformers, two parks of batteries and two diesel generators common to all the reactors (grouped in packs of six, but interconnectable).

The NSSS, BOP and containment are solved using the best estimate TRAC_RT code, neutronics are solved using the 3D NEMO_RT code, while auxiliary and safety systems are solved using the graphical model builders TEAM_FLOW and TEAM_ELECTRIC. The control and protection logic has been designed ad hoc for the simulator by using graphical simulation solutions. The control strategy is based on the reactor-follow-turbine concept, but the simulator also supports the reverse option.

The development of this simulator, Fig. 22, has required innovative developments in several concepts such as the existence of twenty-four instances of TRAC_RT and twelve of NEMO_RT cycling coupled, simultaneously in real time; the development of tools for generating particular initial conditions with any combination for operating situations of its different reactors; specific control strategies for common systems; operating strategies for several reactors with a reduced number of operators; hierarchical systems for the presentation of information, etc.…

Additionally, the simulator incorporates specific communication protocols to enable operation from panels/displays developed by third parties (hardware/software in the loop concept). The initial objective of the development of this simulator is the deepening and consolidation of said the aforementioned operation strategies and other aspects related to HFE like how the information is displayed (NEI, n.d.).

3.2.2. Human factor engineering program in SMR

SMRs are coming to compete in the energy market through a reduced

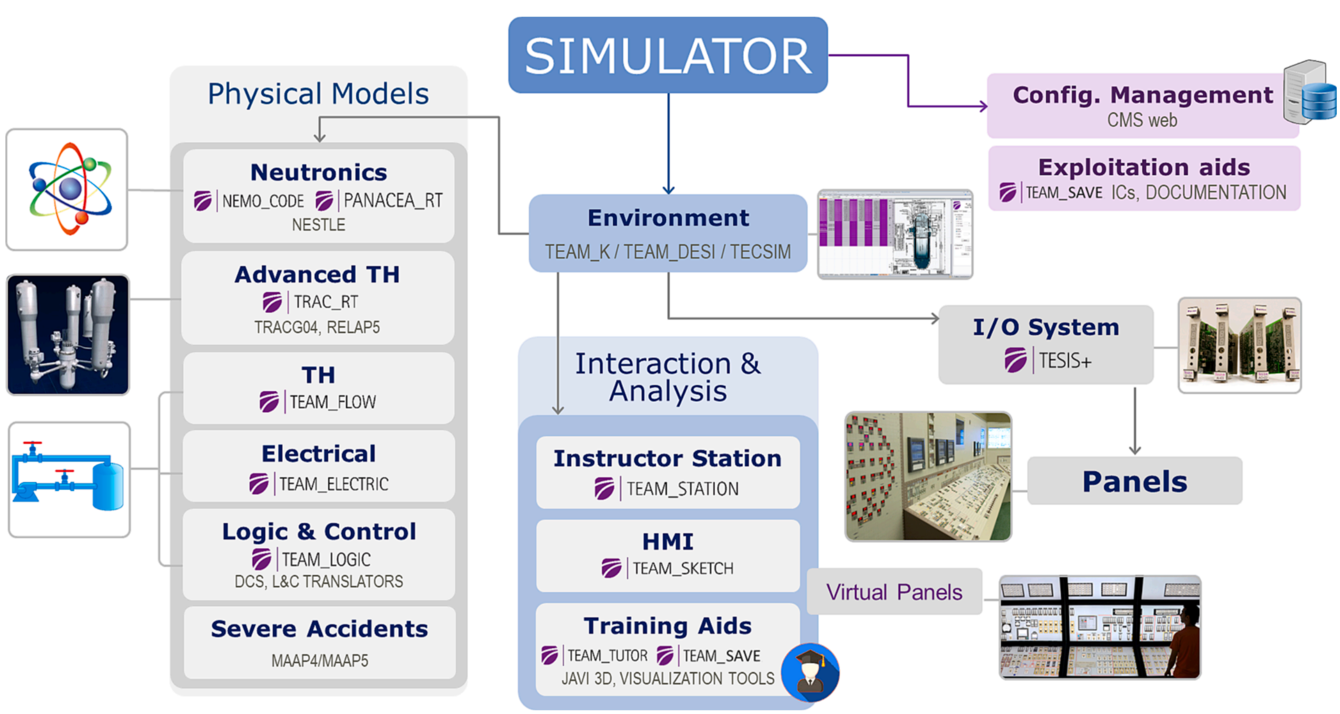

Fig. 21. Tecnatom Simulation Technology.

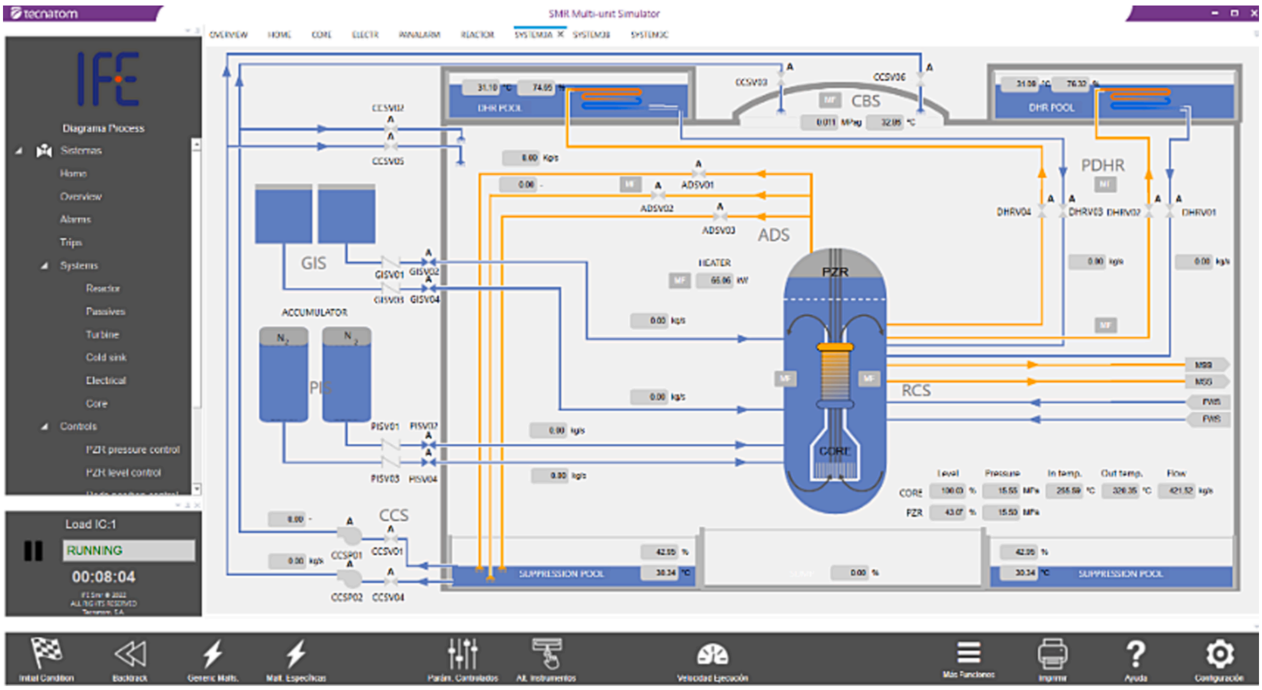

Fig. 22. Reactor Display of IFE SMR Simulator.

operation cost. To achieve this goal, reducing permanent staff at site is crucial. In large conventional nuclear power plants, where more than one reactor is installed, each unit is supervised by one crew in a dedicated main control room (MCR). New SMR designs, which are typically conceived to be installed in multi-unit sites, are so automated that one MCR crew can safely supervise several reactors. Proving that such a multi-unit supervision does not reduce the plant safety level represents one of the most important challenges for SMR deployment.

Nuclear reactor designers establish a HFE Program as part of the design of the nuclear plant, especially for the design of the MCR. HFE is the application of knowledge about human capabilities and limitations to the design of the plant, its systems and equipment. The HFE Program ensures a human-centered approach is used in the design through a series of activities including engineering analyses, design options studies, verification that HFE principles are met, and performance-based testing. Data gathered and analyzed within the HFE Program is evaluated in the assessment of the safety level of the operations performed in the MCR.

A nuclear plant simulator is a powerful tool in a well-developed HFE Program. The simulator evolves together with the plant design and serves at various stages of the HFE Program. A simulator can be used to evaluate different design alternatives, so that the decision-making process is based on testing. A simulator of a multi-unit plant, where scenarios affecting several units can be assessed, is the best tool to identify human factors issues related to the staffing configuration, which can be addressed during the design and re-tested. Close to the end of the design process, a final human factors validation is performed at a full-scope simulation, Fig. 23, evaluating the most challenging scenarios for the MCR staff. When successful, the multi-unit plant is ready to be supervised by one MCR crew, see references (Hervás, 2021a; Hervás, 2021b).

3.3. Experience of Empresarios Agrupados Internacional (EAI) in SMRs

EAI with its more than 50 years' experience in the design of Generation III, III+, and IV nuclear reactors, is nowadays involved in multiple research and industrial projects related to SMRs or SMR technologies. EAI Experience with SMRs can be broken down into four major blocks,

- European Research Projects (described in Section 3.3.1).

Fig. 23. Use of the Simulator for HFE validation. Courtesy of IFE.

- Participation in the Thorcon MSR-SMR design (described in Section 3.3.2).
- Participation in the South African Pebble Bed Modular Reactor (PBMR) design from 2006 to 2009.
- Participation in SNETP and Nucleareurope SMR Task force supporting the promotion of SMRs in Europe.

3.3.1. European research projects: EURATOM and "Horizon Europe"

Since 2004, EAI has participated in EURATOM R&D programs to develop reactor technologies with application in SMRs. For example, in GEMINI+, EAI made a coupled model of the reactor and the secondary and tertiary loops (EAI, 2019). These European projects have analyzed the LFRs (LEADER, ELSY, SILER), HTGRs (HYCYCLES, HYTECH, ADEL, RAPHAEL, ARCHER, GEMINI+) and SFRs (EISOFAR, ESFR), as well as the ADS reactors (ACCELERATOR).

Additionally, EAI also participates in other "Horizon Europe" programs:

- In the PIACE project, as described before, EAI designs a passive isolation condenser that can be used in SMR design.
- For the TANDEM project EAI develops SMR operating conditions in hybrid scenarios of generation in combination with renewable energies.
- Regarding the ANSELMUS project EAI contributes significantly to the safety assessment of heavy-liquid-metal (HLM) systems, using ALFRED and MYRRHA as study cases due to the maturity of their design. The conclusions of these studies will be applicable to SMRs based on HLM systems.

Within TANDEM project, and among other tasks, EAI performs simulations to evaluate multiple study cases related to different optimized architectures of SMRs integrated in hybrid energy systems. For ANSELMUS project, EAI also performs simulations of two technologies (hydrogen production and thermal storage) to obtain their key parameters that will be used as input for further studies within the project. For all these simulations EAI uses its own proprietary software: EcosimPro. This software is able to model any kind of dynamic system represented by differential-algebraic equations (DAE) or ordinary-differential equations (ODE) and discrete events. The tool provides an object-oriented approach for creating reusable components libraries.

3.3.2. Thorcon SMR: A molten salt reactor

Empresarios Agrupados has been chosen as Architect Engineer for ThorCon International's TMSR-500 MSR-SMR. On this project, maritime codes will be used for the conventional part along with nuclear design codes, as the reactor will be built on a floating platform (see layout in Fig. 24).

This MSR-SMR uses a low-pressure reactor core that reduces the complexity of the design and the increased the range of materials that can be used. It also has three low pressure molten salt loops and a supercritical steam system with standard coal plant conditions (550 °C, 250 bar). The primary loop consists of a low pressure and high temperature fuel salts loop containing Uranium fluoride. The secondary loop uses a mixture of sodium and beryllium fluorides and the tertiary loop a mixture of sodium and potassium nitrate, commonly called solar salt due to its use as an energy storage medium in solar plants.

As for this TMSR-500 reactor, EAI is responsible for the design, procurement and commissioning as well as providing support for its licensing by BAPETEN (Indonesia's regulatory agency)(EAI, 2023). At this stage, the conceptual design is completed and the basic design is in progress. The main challenges EAI are facing currently are the site preparation and the qualification of the supply chain.

4. Expected new developments in Spain

The main research topics to be addressed by each of the groups on issues related to SMRs are:

- UPM will be devoted to the analysis of accidental sequences with the NuScale and the CAREM-like TH models developed for the TRACE code. The objective is to further evaluate the performance of the safety systems, with particular emphasis on the passive ones. The main new sequences to be studied in both models are LOCA, and SBO sequences. Other topics of interest to be addressed in the following years include the incorporation of ATF to analyze the increase in available time and the verification of the most important event trees from the point of view of risk for each technology. Such verification will be carried out by performing simulations with TRACE that include the corresponding human actions contemplated in the PSAs.
- CIEMAT will focus on applied research that helps paving down the path of water cooled SMRs implementation in the coming decade. In particular, the activities will focus on two main topics. On the one hand, a sound demonstration that, based on design bases and operational procedures, DEC-B scenarios cannot be feasible. And, in the event they are, that their consequences will not need to set emergency planning zones, such as implemented in the level 5 of the Defence-in-Depth. This will require to develop MELCOR models for selected scenarios in pressurized water cooled SMRs and specific studies of the effect of new reactor features (i.e., flooded containments) on the Source Term to the environment. The second topic will be related to the assessment of passive systems performance under anticipated DEC-B conditions. To do so, ASTEC heat transfer models and their enhancement will be used under conditions to be set at ad hoc experimental setups.

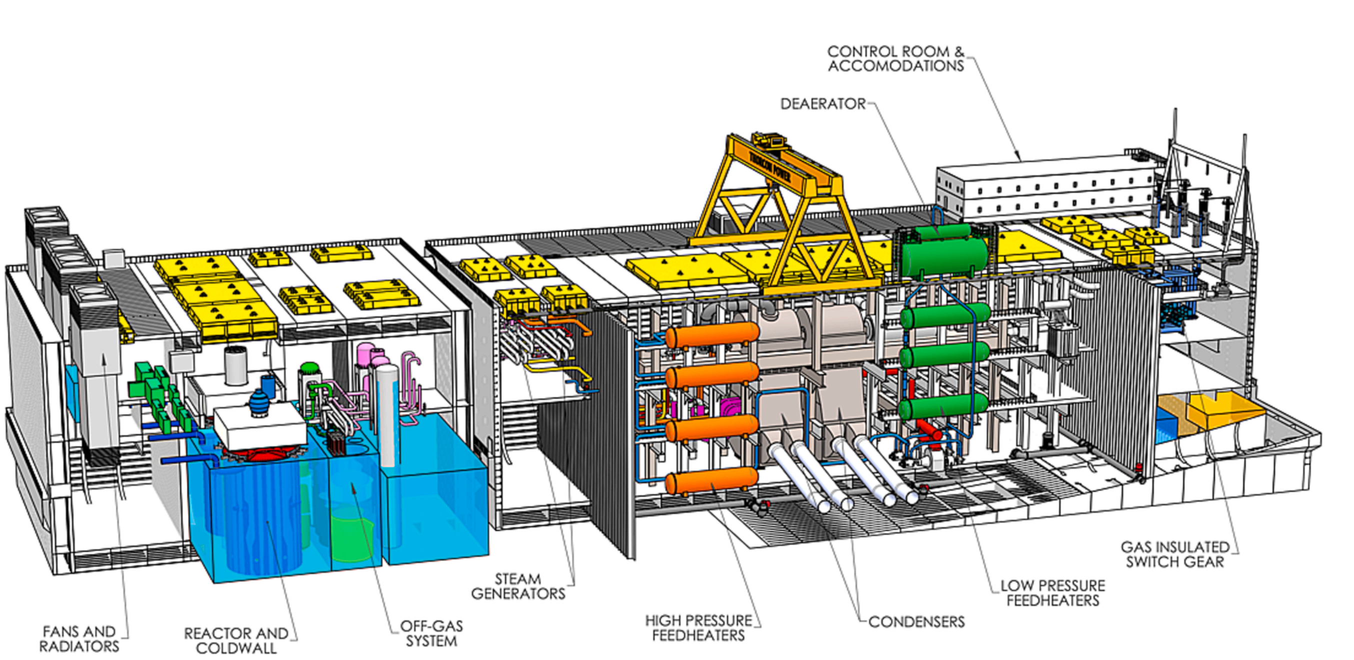

Fig. 24. ThorCon SMR Layout.

- UPC and ENSO will be devoted to assess the capabilities of system codes to reproduce the relevant phenomena associated to the actuation of new passive systems, with particular attention to those that occur at conditions for which the codes were not designed (natural circulation with limited or negligible sources and/or sinks of energy and momentum). Another key issue that needs to be addressed in the near future is how to make the most of the large database of separate and integral tests carried out to date for the licensing of LWR and the validation of computer codes. In the short time UPC and ENSO aim to develop scaling strategies for applying this data and knowledge to validate the system codes for iPWR designs, in particular for those relevant phenomena also reported in their Safety Analysis Report.
- UPV is working in Monte Carlo and in-house deterministic transport codes and high-fidelity models (SP3 and Discrete Ordinates) of different SMR technologies, as well as the needed nuclear parameters generation methodologies for these deterministic codes. Both for stand-alone and for coupled TH-neutronic steady-state and transient analyses.
- IDOM is still involved in the Moltex projects and currently contributing to the next engineering phases for the development of the different technologies Moltex is promoting. Apart from that, IDOM will be part of the consortium for Nuproship-II initiative, once phase-I concludes, where basic design and supply chain qualification assessments will be made for nuclear propulsion civil ships. Currently, IDOM is part of the Spanish SMR working group promoted by CEIDEN and is member of the NUCEAL partnership which has EDF as sole client and NUWARD SMR engineering development within its targeted scopes.
- EAI plans to increase its contribution to the SMR development and deployment by using its extensive nuclear experience and its knowledge in LWRs, MSR, SFR and LFR technologies. EAI is participating in R&D proposals for the Horizon Euratom 2023 and an internal initiative is ongoing for the study of the replacement of conventional fossil fuel power plants by SMRs.

5. Final remarks

The main findings can be summarized in four key SMR categories: LWR-SMR, HTGR, MSR and SFR:

A. LWR-SMR.
- TRACE and RELAP5 codes have shown their capabilities to be able to simulate accidental sequences in LWR-SMRs involving passive safety systems. Besides, ASTEC, MELCOR and RELAP/SCDAPSIM models are under development to find out potential severe accident scenarios and identify the major phenomena governing the accident progression.
- Spanish universities have significant knowledge in the simulation of experimental SMR-related facilities, acquired with SIRIO, MOTEL and HWAT. For the PIACE project proposals have been made for scaling the facility to a BWR reactor and simulations of the scaling to a PWR reactor have been carried out, comparing the results given by TRACE and RELAP5.
- The results obtained from applying multiscale (TRACE/SCF) and multiphysics (TRACE/PARCS) showed that these tools represent a better approach to properly understand the phenomena occurring in SMR accidental sequences such as boron dilution and SLB.
- UPM and UPV have acquired the capability to employ the PARCS code to simulate the neutronics of the NuScale reactor core, using cross sections from Serpent and SCALE. This capability can also be expanded to cover simulation efforts for other SMR designs.

B. HTGRs.
- A complete model of the primary circuit of the GTHTR300C was created by UPM with data from the open bibliography. The results demonstrate the feasibility of using system codes to simulate HTGRs.
- HTGR is a promising technology with interesting safety features, which third barrier deserves specific investigation to make the most out of its performance, since what expected in case of in-containment source term has specific features far from LWR source term ones.

C. MSRs,
- The participation of Spanish companies in this field shows a high degree of involvement in several projects like TMSR-5000, SSR-W and SSR-U.

D. SFR.
- The results of the analysis show potential safety issues in the event of cooling disruption. The large differences in the coolant properties (water and sodium) make their interaction with environments and, particularly, fission products a domain that should be further explored and more soundly modelled.

A common conclusion to all types of SMRs discussed in this paper is that passive systems are highly valuable enhancements to nuclear safety. Nonetheless, their reliability under all accidental conditions foreseen should be proven for the diverse SMR technologies (LWR, HTGR, MSR, SFR…).

On the other hand, it is also important to highlight that the Spanish industries, EAI and IDOM, are involved in the design of several SMRs. Furthermore, Tecnatom is actively contributing to the development of SMR simulators and is developing projects to study the human factor engineering.

CRediT authorship contribution statement

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Declaration of competing interest

The authors declare that they have no known competing financial interests or personal relationships that could have appeared to influence the work reported in this paper.

Data availability

The data that has been used is confidential.

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