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"Energy and Nuclear Science and Technology"

Cycle XXX

Study on a submerged Small Modular Reactor: integral design, passive safety strategy and critical issues

Supervisor: prof. Marco E. Ricotti

Chair of the Doctoral Program: prof. Carlo E. Bottani

Doctoral dissertation of:
Marco Santinello

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Abstract

Following the Fukushima-Daiichi nuclear accident in March 2011, the nuclear community has highlighted that current nuclear power plants might not be prepared to face extreme events, which can lead to the failure of the core cooling process due to a prolonged loss of heat sink. Technological innovations are therefore needed to improve the reliability of new generation nuclear power plants. This work deals with the design and safety analysis of a submerged nuclear power plant, where a horizontal cylindrical hull, placed on the floor of a sea or an artificial lake, hosts an integral pressurized Small Modular Reactor. The reference concept is Flexblue, a 160 MW_{el} transportable and sub-sea based nuclear power unit developed by French company DCNS (now Naval Group). This type of reactor can rely on the presence of a large water reservoir, i.e. the water surrounding the containment, acting like an infinite heat sink. Such feature, combined with a fully passive safety strategy, can provide inherent protection towards Fukushima-like accident scenarios.

The research activity of the doctoral program has been targeted at bringing an advancement to the research about the submerged SMR concept. The thesis addresses several issues regarding the reactor design and the safety strategy. The focus is mainly on the thermal-hydraulic aspects. The first goal of this work is to define a reactor layout satisfying the constraints of a submerged environment, since no feasible proposal had been done in the past. The concept of an integral PWR (iPWR) SMR, suitable to operate in a submerged containment, is introduced, providing a preliminary sizing of the primary components and a thermal-hydraulic verification of the steam generator. Then, a fully passive safety strategy is proposed: two reference thermal-hydraulic accident scenarios are identified, i.e., "intact" and "broken" primary system, tackling the analysis of the core cooling in the long-term period. System codes Relap5-Mod3.3 and Apros 6 are used to simulate accident scenarios and perform a preliminary safety verification. The thesis also sketches two proposals for experimental activities to validate the numerical models. In conclusion, the most relevant critical issues for the deployment of submerged SMRs are identified and discussed.

The results of the numerical activities are encouraging, especially those regarding the safety analysis. The simulations predict a successful operation of the safety systems in the reference test cases, under conservative assumptions and boundary conditions. The designed systems allow exploiting the safety potentialities of the submerged concept. Also, the IRIS-like design of the reactor own interesting features, although it needs more detailed verifications and the design of a boron-free core, which optimizes the fuel cycle, represents a very challenging issue. Other engineering aspects, licensing procedures, international regulations and economic assessment are key challenges to be addressed to achieve the deployment of the reactor concept.

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List of acronyms

ADS	Automatic Depressurization System
ALFRED	Advanced Lead Fast Reactor European Demonstrator
AP1000	Advanced Passive 1000
BWR	Boiling Water Reactor
CEA	Commissariat à l'Energie Atomique
CFD	Computational Fluid Dynamics
CHF	Critical Heat Flux
CRDM	Control Rods Driving Mechanism
CVCS	Chemical Volume Control System
DBA	Design Basis Accident
DCNS	Direction des Constructions Navales Services
DEC	Design Extension Conditions
DVI	Direct Vessel Injection
DWO	Density Wave Oscillation
EHRS	Emergency Heat Removal System
ENEA	Ente Nazionale per Energia e Ambiente
EPR	European Pressurized Reactor
EU	European Union
GBS	Gravity-Based Structure
HEM	Homogeneous Equilibrium Model
HERO	Heavy liquid mEtal pRessurized water-cOoled tube
HTC	Heat Transfer Coefficient
HTGR	High Temperature Gas Reactor
I&C	Instrumentation & Control
IAEA	International Atomic Energy Agency
INSPIRE	INtegration of Smr's PotentIal Role in EU framework
iPW-SMR	integral Pressurized Water - Small Modular Reactor
IRIS	International Reactor Innovative and Secure
LFBR	Lead-Bismuth Fast Reactor
LOCA	Loss Of Coolant Accident

LOFW	Loss Of Feed Water
LOOP	Loss Of Off-site Power
LPA	Lumped Parameter Approach
LUHS	Loss of Ultimate Heat Sink
LWR	Light Water Reactor
NPP	Nuclear Power Plant
NRC	Nuclear Regulatory Commission
O&M	Operation & Maintenance
OCR	Organic Cooled Reactor
POLIMI	POLItecnico di MIlano
PRA	Probabilistic Risk Assessment
PWR	Pressurized Water Reactor
R&D	Research & Development
RC	Reactor Containment
RPV	Reactor Pressure Vessel
SBO	Station Black-Out
SCOR	Simple COmpact Reactor
SG	Steam Generator
SIET	Società Informazioni Esperienze Termo-idrauliche
SMART	System-integrated Modular Advanced ReacTor
SMR	Small Modular Reactor
SPES	Simulatore Pressurizzato per Esperienze di Sicurezza
ST	Safety Tank
TNPP	Transportable Nuclear Power Plant
VTT	Technical Research Centre of Finland
VVER	*Vodo-Vodyanoi Energetichesky Reactor,* Water-Water Energetic Reactor

Nomenclature

Latin letters

A	Cross area
c_p	Specific heat at constant pressure
d	Diameter
h	Enthalpy
k	Thermal conductivity
L	Length
m	Mass flowrate
N_r	Number of rows
Nu	Nusselt number
p	Tube pitch
Pr	Prandtl number
$\dot{Q}$	Thermal power
q''	Heat Flux
Ra	Rayleigh number
Re	Reynolds number
T	Temperature
v	Velocity
x	Quality
X_{tt}	Martinelli parameter

Greek letters

α	Heat Transfer Coefficient
μ	Dynamic viscosity
ρ	Density
σ	Surface tension

Introduction

The accident at the Fukushima-Daiichi nuclear power plant on March 11th, 2011 put on evidence the need for the nuclear community to be prepared for unexpected circumstances that may go beyond the design basis events. Even with large safety margins and good operation and maintenance practices, the, albeit remote, possibility of high consequence situations can never be excluded (IAEA, 2016-a). Fukushima-Daiichi accident was initiated by a series of three events: (i) off-site power distribution failed after the 9.0 magnitude earthquake, (ii) emergency diesel generators were flooded and thus unavailable and (iii) the transportation to the site and start-up of back-up equipment could be possible only several days after the reactor scram, because of the damages of the tsunami (Blandford & Ahn, 2012). Such sequence had not been predicted and the power plant was not prepared to handle it. Emergency cooling was not successful because several components, such as the Isolation Condenser (IC), the Reactor Core Isolation Cooling (RCIC) system and the High-Pressure Coolant-Injection system (HPCI), did not work properly in absence of electrical power. According to Tokyo Electric Power COmpany (TEPCO) estimates, Unit 1 was left without any water injection for 14 hours and 9 minutes, while Unit 2 and Unit 3 lost cooling capabilities for approximately 6 and a half hours (TEPCO, 2011). This led to the melting of all the fuel in Unit 1, 57% of the fuel in Unit 2 and a large part of the fuel in Unit 3 (Holt, et al., 2012).

Basically, Fukushima-Daiichi accident emphasized that operating nuclear reactors may show strong difficulties in facing the Loss of Onsite/Offsite Power (LOOP) scenario, which led to the Loss of Ultimate Heat Sink (LUHS). In the months following the accident, worldwide nuclear authorities prescribed stress tests and special measures to assess the safety procedures of existing power plants, especially considering those scenarios where AC power is unavailable. As far as new designs are concerned, the response to Fukushima included a very strong attention to the development of passive safety systems. After Fukushima, guaranteeing an adequate core cooling through natural circulation for a very long grace period, without electrical input or human intervention, has become an important feature for the safety strategy of many Gen III+ designs. Passive systems own a high potential to improve the safety of nuclear power plants: considering the case of the AP1000 (Advanced Passive 1000) (Schulz, 2006), a two-loop 1000 MW_{el} PWR developed by Westinghouse and currently under construction, the estimated core damage frequency is around 5.1×10^{-7} events/year (Matzie, 2008). In addition, passive systems allow a significant simplification of the layout with respect to conventional PWRs, requiring fewer components, tests, inspections, and maintenance cycles. In the AP1000 there are 50% fewer safety-grade valves, 35% fewer safety-grade pumps, 80% less safety-grade piping and 70% less cables. Similarly, there is 45% less seismic building volume as a result of the reductions in equipment (Matzie, 2008). Simplification is also a key factor in order to avoid difficulties, delays and extra-costs during the construction phase. The counter-example of the cases of the two European Pressurized Reactors (EPR) under construction in France (Flamanville) and in Finland (Olkiluoto) provides an additional confirmation of this fact. Roussely (Roussely, 2010) stated that the resulting complexity of the EPR, arising also from the redundancy of active safety systems, is certainly a handicap for its construction and therefore its cost.

However, the thermal-hydraulic reliability of passive safety systems still represents an important concern for its use in industrial systems. D'Auria (D'Auria, 2012) observed that one of the most critical matter regards the long-term stability: in comparison to forced flow, natural circulation driving force is small and even slight perturbations, which are not predictable or neither observable in the long-term period, can affect the stability and the successful operation of the entire system. Such situation becomes even more critical when boiling and condensation are involved. Jafari et al. (Jafari, et al., 2003) have analyzed several configurations of passive systems and have shown that, under certain methods, the estimated reliability of decay heat removal through natural circulation may turn out to be comparable to that of a pump installed in an "equivalent" forced flow system. The matter is still open and the research in nuclear thermal-hydraulics is nowadays strongly focused on the verification of the working principles of passive safety features. This type of research moves the attention from the reliability of active components, already well assessed in industry, to the analysis of physical phenomena related to passive safety, e.g. natural circulation, to demonstrate their effectiveness and stability. The former aspect was mainly demanded to constructors of pumps, valves etc., while the latter is a peculiarity of nuclear experts. Therefore, in the last three decades the need of verification for passive safety strategy stimulated large efforts among researchers in nuclear thermal-hydraulics, aimed at understanding the physics at the basis of physical phenomena and predicting the transient evolution of natural circulation and multiphase flow. Theoretical, numerical and experimental approaches are currently put on practice to support the design of safer, cheaper and sustainable nuclear reactors.

A passive safety strategy assumes paramount importance in Small Modular Reactor (SMR), where compactness and simplified layout are the key aspects of the design. Recently, the IAEA (IAEA, 2016-a) has discussed the most important safety features employed in existing reactors and advanced designs of water-cooled SMRs. Among the most innovative ideas, placing the nuclear reactor containment in a submerged environment has gained a lot of interest in recent years. The concept consists of having cold water that surrounds a large metal containment, which hosts the Reactor Pressure Vessel (RPV). Steam can be released in the internal atmosphere and condenses in contact with the inner surface, thus rejecting decay heat to the external water. This concept has been introduced in some innovative SMR designs, such as NuScale (Reyes Jr., 2012) and Westinghouse SMR (Smith & Wright, 2012): the reactor is immersed in a large pool, which offer a grace period determined by the total water inventory and heat transfer is effective until the water is sufficiently cold. The NuScale design (160 MW_{th}) relies on 30 days of cooling from the reactor pool, which is followed by air cooling to achieve an unlimited safe cool-down time.(Reyes Jr., 2012). Alternatively, if the metal containment is placed into the sea or in an artificial lake, the grace period given by the cooling process is potentially unlimited. Santinello et al. (Santinello, et al., 2017-a) performed a numerical investigation about this aspect: they observed that the decay heat cannot influence significantly the temperature of the sink and such heat transfer process is very effective. The water of the sea/lake is therefore a large reservoir acting as an infinite and permanent heat sink. Hence, the immersion offers unique safety features in terms of enhanced protection towards Fukushima-like accident scenarios.

The research activity of the doctoral program has demonstrated the viability and significantly enhanced safety benefits of the submerged SMR concept. The thesis addresses several issues regarding the reactor design and the safety strategy. The focus is mainly on the thermal-hydraulic aspects. The first goal of this work is to define a reactor layout satisfying the constraints of a submerged environment, since no feasible proposals had been done in the past. The concept of an integral PWR (iPWR) SMR, suitable to operate in a submerged containment, is introduced, providing a preliminary sizing of the primary components and a thermal-hydraulic verification of the steam generator. Then, a fully passive basic safety strategy is developed: two reference thermal-hydraulic accident scenarios are identified, i.e., “intact” and “broken” primary system, tackling the analysis of the core cooling in the long-term period. System codes Relap5-Mod3.3 and Apros 6 are used to simulate accident scenarios and perform a preliminary safety verification. The thesis also sketches two proposals for experimental activities to validate the numerical models. In conclusion, the most relevant critical issues for the deployment of submerged SMRs are identified and discussed.

The reference design adopted in the thesis is the Flexblue reactor (Haratyk, et al., 2014), a 160 MW_{el} nuclear power unit developed by French company DCNS - now Naval Group - conceived to operate on the seafloor. Flexblue adopts pressurized water technology and its safety relies on passive systems. *DCNS – Naval Group* company has sponsored part of the PhD activity, providing also technical inputs for the development of the long-term safety analysis with Relap5-Mod3.3. The benchmark with the system code Apros 6 has been done in cooperation with the code developer (*VTT Technical Research Centre of Finland*). In 2016, Flexblue has been proposed as the European reference SMR in the framework of an R&D project submitted to a H2020 Euratom call INtegration of Smr's PotentIal Role in EU framework – INSPIRE - (INSPIRE Consortium, 2016), led by the *Ente Nazionale per Energia e Ambiente (ENEA)* and supported by a consortium involving 13 organizations among universities, R&D centers and industries from 6 EU countries. In the PhD thesis the definition of the experimental part of the proposal has been developed, cooperating with the *Società Informazioni Esperienze Termo-idrauliche (SIET)*.

In brief, the work presented in this doctoral thesis is structured as the followings:

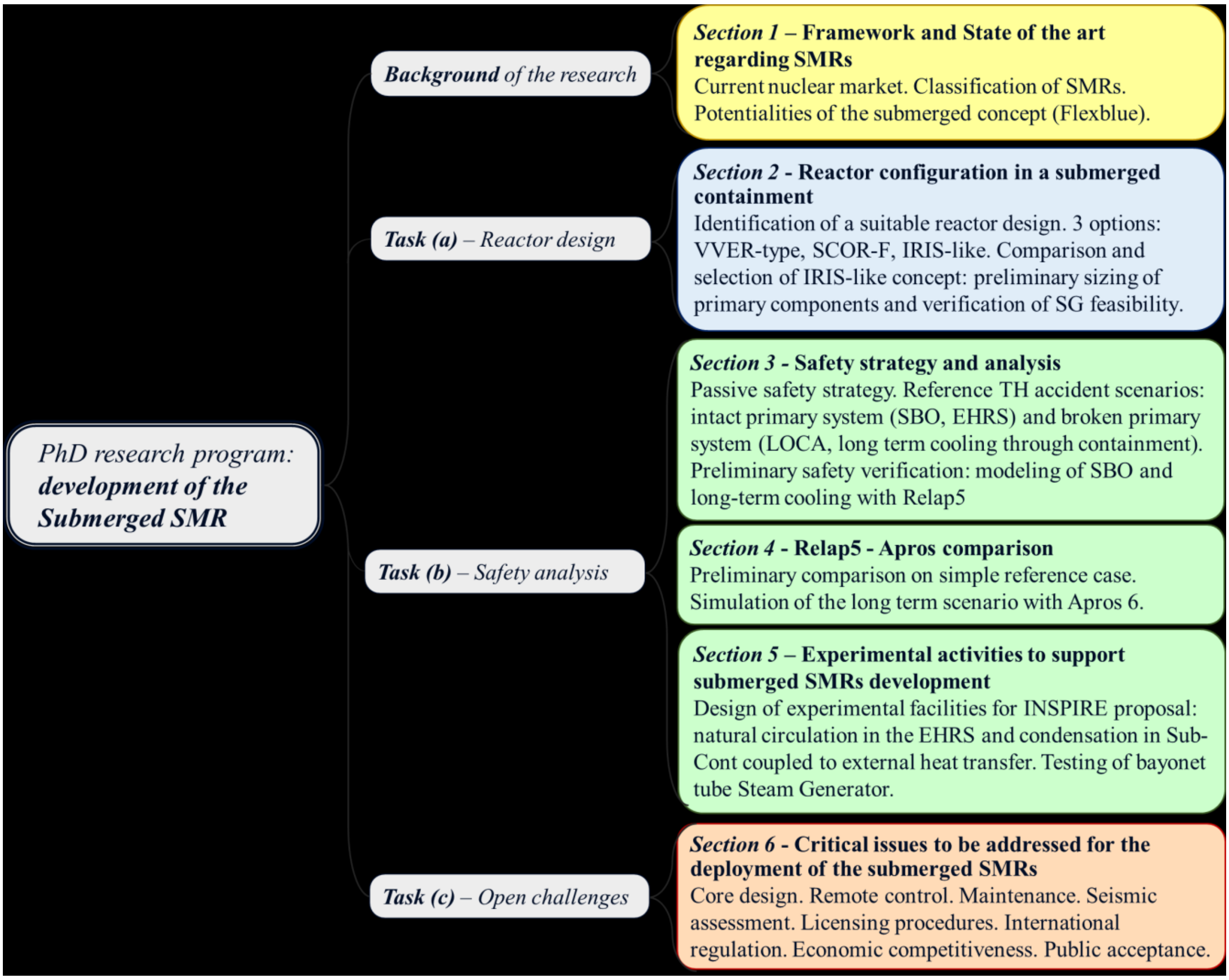
Section 1. *Background of the research.* The development of the submerged concept for nuclear reactors is included in a dynamic research framework about SMRs, where several designs of land-based and off-shore reactors are currently in design, licensing and construction phase. The potentialities offered by the immersion in the sea or in an artificial lake can address many challenges and provide consistent solutions to safety issues.

Section 2. *Reactor layout in a submerged containment.* The design of a submerged SMR needs to satisfy a manufacturing constraint given by the diameter of the containment: the reactor should be placed inside a horizontal cylindrical hull, whose diameter is currently limited by technological capabilities economic reasons and logistic issues. The definition of the reactor layout is quite challenging and requires ad-hoc studies. The thesis presents a preliminary design of an integral PWR layout, based on the IRIS reactor concept, in order to fit the 14m-diameter Flexblue hull. It also demonstrates that other concepts, already proposed for Flexblue (i.e. a VVER-type and the French SCOR-F), are not suitable to achieve design and safety targets.

Section 3. *Safety strategy and analysis. S*afety strategy and related thermal-hydraulic features of the submerged SMR concept are addressed. A promising passive safety approach has been implemented, with a suitable passive (two-phase flow natural circulation) safety systems combination among ex-hull heat exchangers, integrated steam generators, direct vessel injection lines and in-hull suppression pools. According to the Fukushima experience and the potential safety features of the submerged SMR concept, two refence accident scenarios have been selected and analyzed, to evaluate the potential safety performance of the SMR: i) the Station Black-Out (SBO), where the decay heat is removed through natural circulation in a condenser immersed in the seawater and ii) the long-term sump natural circulation, in which the decay heat is directly rejected through the submerged containment. The analysis has been carried out by means of the 1D system code Relap5-Mod3.3 (U.S., NRC). Simulations predict successful operation of the safety systems under the conditions tested.

Section 4. *Relap5 - Apros comparison.* The long-term cooling scenario has been used for a code comparison. The purpose is to compare results of Relap5-Mod3.3 simulations with the prediction of the system code Apros 6 (Finland, VTT), in order to obtain another verification of the feasibility of the safety concept. A preliminary analysis on a simple reference case, which tests the responses of both the homogeneous and the two-fluid models, already shows large discrepancies between the results of the two codes. The simulation of the long-term cooling scenario with Apros 6 resolves a different behavior of the natural circulation flow, with respect to Relap5, even though the cooling process of the core is still successful. This work also reveals an incorrect way to calculate the two-phase heat transfer coefficient under certain conditions and has turned out to be useful for the developer VTT to fix some issues in the Apros 6 code.

Section 5. *Experimental activities to support submerged SMRs development*. The thesis includes the design of two experimental facilities to support the development of the submerged SMR and to study its safety potentialities. The purpose is to cover some gaps of knowledge regarding both the operation of the Emergency Heat Removal System (EHRS) and the



Outline of the PhD research program

conditions of the flow in the Sub-Cont. This part represents an active collaboration within the INSPIRE proposal. Moreover, the results of an experimental campaign aimed at observing the performances of a bayonet tube steam generator are briefly illustrated and a simulation activity is proposed. Bayonet tubes can represent an interesting solution, alternative to the helical coil. for the design of the steam generator.

Section 6. *Critical issues to be addressed for the deployment of submerged SMRs.* Whilst the passive safety strategy owns a "ultimate" potential to solve the most challenging scenarios about nuclear safety, the deployment of a submerged SMR presents several other critical issues, which still need technical solutions. These issues include: design of a boron free core, remote operating and control, refueling and maintenance, seismic assessment, licensing procedures, international regulation, economic competitiveness and public acceptance.

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Section 1

Framework and State of the Art regarding SMRs

In the last four decades, the offer of nuclear power plants has been mainly characterized by large reactors, since the "economy of scale" principle has determined the higher convenience of such solution. Nowadays, the scenario is different: many developing countries are approaching the nuclear industry, but they cannot sustain large reactors on their grids. In Europe, China and United States utilities are facing serious difficulties in building new large reactors. Therefore, deliberately small and medium size reactors can find new spaces in the nuclear market, offering several advantages in terms of feasibility, economic sustainability, safety improvement and multi-purpose application. Currently, five SMRs are under construction in Argentina, Russian Federation and China. Four of them exploit pressurized water technology and three of them are designed to operate on a floating barge. Pressurized water is still considered the most reliable and proven technology: the integral SMRs designs, where all the primary components are arranged inside the reactor vessel, are very promising and close to near term deployment. The attention towards off-shore reactors (floating and underwater) has been increasing in the last years, especially thanks to the potential enhanced protection to Fukushima-like accident scenarios. In this sense, the development of the French design "Flexblue" allows exploring unique and innovative safety features, which can bring a great advancement in nuclear industry.

1.1. Framework of Small Modular Reactors development

In the history of nuclear industry, the "economy of scale" principle has often dominated the market of commercial reactors. Classic economic models show that the specific capital cost of a Nuclear Power Plant (NPP) tends to decrease with increasing power size, thanks to more efficient use of raw materials, higher performances of large components, e.g. steam generators, heat exchanger and pumps, and better amortization of set-up costs, e.g. for licensing, siting activities, civil works to access the transmission network (Locatelli, et al., 2014). Consequently, the current offer of NPPs is mainly composed of large-scale units rated at more than 1,000 MW_{el} and no commercial SMRs have been completed in the last two decades. However, in recent years several conditions have changed and nowadays the economy of scale principle shall be questioned in lights of new developments of the energy market. Large units can be suitable for the needs of large power grids such as in Europe, US or China, but they do not fit well in grids of smaller developing countries, whose energy demand is growing quickly. In such countries, a large reactor would represent more than 10% of the installed capacity and the electrical grid would not be able to support it. Conversely, in those countries where large reactors can be sustained, the utilities may have to face serious financial difficulties in affording large investments: cost and delay overruns due to problems with authorities and regulations are common in Western countries (Kessides, 2012) (Thomas, 2012). The cases of the European Pressurizer Reactors (EPRs) under construction in Olkiluoto (Finland) and Flamanville (France) are emblematic.[1] Moreover, today the construction of a large NPPs in western countries is a very peculiar "megaproject", if compared to other civil engineering works, such as roads or railways. The reason is the high political, social and mediatic exposure, which can undermine the reliability of the economic plans and discourage the investments of utilities. Finally, it should be also considered that the competition with fossil-fueled units and, in some areas, with renewable energies, is very harsh and may put concerns in the utilities about the actual convenience of investing many efforts and resources in a large NPPs. For all these reasons, the financing of a large nuclear reactor is nowadays complicated in western countries. Nuclear investments may represent a relevant industrial, economic, and financial risk for investors.

Small and medium size Modular Reactors (SMRs) represent a possible solution to these difficulties. According to IAEA definition, small reactors have electric output less than 300 MW_{el}, while medium reactors do not exceed 700 MW_{el}. Typically, first generation nuclear power plants did not have a large output and in principle they can be included in this definition. However, their power size was not deliberatively small, but determined by technological limitations. Nowadays, with the acronym SMR the nuclear community means those reactors intentionally designed to produce a smaller electric output, with respect to conventional NPPs (Ingersoll, 2009). The purpose of the reduced size is to exploit several advantages, like the modularity concept, that can make their deployment more feasible and convenient than large plants. In a recent work, Boarin et al. (Boarin, et al., 2012) compared a large nuclear reactor with an equivalent plant made of 4 modules. They showed that, in the complexity of current economic scenarios, the economy of scale may not be the only cost driver. Considering liberalized energy and capital markets, a more progressive investing strategy would allow reducing the financial exposure and a shorter construction time would facilitate the return on investment, also mitigating the impact of scenario uncertainty. In addition, a step-by-step deployment of the power units would require less capitals at risk. In Locatelli et al. (Locatelli, et al., 2014), it is concluded that SMRs are cost competitive with large reactors when the power required is 1–3 GW_{el}, since the economies of scale are compensated by the "economy of multiples". In those zones where the construction of a large nuclear reactors is not feasible for grid limitations or justified by energy needs, SMRs can also be competitive with fossil fuels. Carelli et al. (Carelli, et al., 2010) performed a preliminary characterization and modeling of integral SMRs economics, concluding that innovations brought by well-designed SMRs can compensate the disadvantage of the economy of scale and balance the competition with large plants. Abdulla et al. (Abdulla, et al., 2013) asked 16 experts of the nuclear industry to express a judgement about the competitiveness of a Light Water SMR with respect to large sized nuclear plants. Eleven experts estimated median costs for a 45 MW_{el}

[1] Constructions of Olkiluoto 3 and Flamanville 3 reactors started in 2005 and 2007, with expected connection to the grid within 4-5 years and cost of €3.0 and €3.3 billion, respectively. In October 2017, EDF (Electricité de France) announced that Flamanville reactor will be fully up and running in late 2019, while AREVA forecasts the connection of Olkiluoto reactor for May 2019. EDF has also estimated the amount of final budget for Flamanville in €10.5 billion (Reuters, 2017). According to Locatelli and Mancini (Locatelli & Mancini, 2012), the main reasons for budget overruns and delays are: overoptimistic estimations, first-of-a-kind issues and undervaluation of regulation requirements.

power unit between \$4,000 and \$7,700/kW$_{el}$ (average \$5,800/kW$_{el}$), while the overnight cost of a 1,000-MW$_{el}$ current generation reactor ranges from \$2,600 to \$6,600/kW$_{el}$. In addition, all experts agreed that the expected construction period would be considerably shorter.

Besides the economic aspect, SMRs present many other general benefits and challenges that must be considered for a proper evaluation of their potential role in the current and future nuclear market. Many authors in literature have reviewed and discussed advantages and disadvantages: a general outline is shown in Table I and Table II (Vujić, et al., 2012) (Ingersoll, 2009) (Liu & Fan, 2014). However, each SMR concept has its peculiarities, which determine the strong-points in terms of enhanced safety, easier construction and O&M etc., and its issues, which need to be accurately evaluated and addressed in the design phases.

Table I. Benefits of SMRs

Location	
(i)	Power generation in remote areas without infrastructures for electricity transmission
(ii)	Expanded potential siting options
(iii)	Easier transportation of components from factory to site
(iv)	Smaller nuclear island and footprint of the whole nuclear power plant
Construction	
(i)	Shorter construction time
(ii)	Fabrication in factory for several components, reducing the work on site
(iii)	Possibility to manage the whole construction in factory
Safety	
(i)	Design simplicity: prevention by design from many accident scenario
(ii)	Reduced inventory of radionuclides
(iii)	Passive safety strategy: decay heat removal through natural circulation
Operation	
(i)	Long-life cycle and reduced need for refueling
(ii)	Possible simplification and economization of operation and maintenance
(iii)	Flexibility to better match the required electrical output and keep the stability of the grid
Economic	
(i)	Lower impact of capital cost, lower risks and accelerated return of investment
(ii)	Multiple units (modularity concept): step-by-step deployment, shared infrastructures, economy of replication for components, process learning
(iii)	Easier financing strategy, thanks also to possible step-by-step deployment
Other	
(i)	Increased proliferation resistance
(ii)	Smaller water usage to reject heat to the environment
(iii)	Adaptability to non-electrical purposes (desalination, cogeneration, industrial process heat)
(iv)	Possible coupling to other energy sources, especially with renewables in smart grids.

Table II. Challenges of SMRs

(i)	Competitiveness in energy market to be demonstrated for the specific reactor concept
(ii)	Transport of spent fuel from isolated areas to reprocessing facilities
(iii)	Lack of regulation about small reactors and modularity concept
(iv)	Licensing and certification procedures of new designs
(v)	Public acceptance of new concepts

1.2. State of the art on Small Modular Reactors

In the last decade, significant efforts have been devoted to the development of new SMRs design, exploiting the know-how of past technologies and advancing new technical solutions. IAEA has highlighted more than 50 different innovative designs (IAEA, 2016). Another comprehensive review, periodically updated with the advancement status of each design, is available on www.world-nuclear.org (World Nuclear Association, 2017). Both these documents notice that designs at near term deployment mainly use pressurized water technology, but also some gas, liquid metal and molten salts SMRs are at well advanced development. Countries that are mostly involved in SMR development and deployment are Russian Federation, China and United States. Interesting and promising designs have also been proposed by Argentina, South Korea, South Africa and France.

1.2.1. SMRs under construction

In November 2017, five SMRs are under construction in three countries: an integral PWR in Argentina, three floating NPPs (two in Russian Federation and one in China) and a High Temperature Gas Reactor (HTGR) in China (World Nuclear Association, 2017).

- *CAREM – Argentina* – (Fukami & Santecchia, 2000) is an integral Pressurized Water Reactor (iPWR). A 27 MW_{el} prototype is under construction, while the final full-sized export version will be 100 MW_{el} or more. It has been designed by the Argentinian's National Atomic Energy Commission (CNEA), which exploited domestic technologies to achieve design simplicity and improved safety performances. The primary circuit is a natural circulation loop included in a self-pressurized reactor vessel. Recirculation pumps and pressurizer are not needed. Design simplicity inherently prevent from large Loss Of Coolant Accident (LOCA) and control rod ejection. In case of lack of energy sources (Station Black-Out scenario), two redundant decay heat removal systems ensure passive core cooling for 36 hours.
- *KLT-40S and RITM-200 – Russian Federation* – (Kuznetsov, 2012) (Zverev, et al., 2013) are, respectively, 35 MW_{el} and 50 MW_{el} pressurized water reactors designed by OKBM Afrikantov to operate on floating barges. The KLT-40S is conceived for a static operation to supply electricity in isolated zones, like the North-East of Siberia. Conversely, the RITM-200 is the new generation of Russian nuclear icebreaker and its main purpose is the naval propulsion. However, it is adaptable also for power generation and desalination. Both designs use pressurized water technology and two modules can be mounted on a single barge. Strong points of the floating concept regard (i) the long life-cycle of the fuel – 3 years for KLT-40S, up to 5-7 years for the RITM-200 because of lower utilization factor –, (ii) manufactory entirely done in shipyard and delivered to the sites ready for operation and (iii) high degree of freedom in site selection. Basically, the floating barge can be moored any costal region and no specific infrastructures are required for its installation.

- *ACPR50S – China* – (IAEA, 2016) is a 50 MW_{el} floating nuclear reactor designed by China General Nuclear Group (CGN). The concept is not dissimilar to the Russian barge-mounted NPPs, with some peculiar differences. The most important is the multi-purpose application: the reactor is designed to supply heat, electricity, fresh water and steam in cascade or combined mode. The concept is thus suitable to serve different users, e.g. oil drilling platforms, isolated coastlands and islands, offshore mines.
- *HTR-PM – China* – (Zhang, et al., 2006) consists of two pebble-bed reactor modules coupled with a 210 MW_{el} steam turbine. The design is by Tsinghua University's Institute of Nuclear and New Energy Technology (INET). It is a demonstrator of the HTGR concept for commercial electricity production, which offers unique inherent safety features thanks to the peculiar chemical composition of the fuel spherical elements. The purpose of this prototype is to prove economical sustainability and technology assessment. Reactor core is 11 m high and filled with more than 400,000 pebbles, which are continuously added and removed allowing online refueling. The safety relies on large negative temperature coefficients and a very high temperature that can be sustained without damages to fuel spheres. These features offer inherent protection against LOCA and reactivity accidents. Another peculiarity is decay heat removal through heat conduction and radiation within core internal structures, thanks to low power density.

1.2.2. PW-SMRs with integral layout

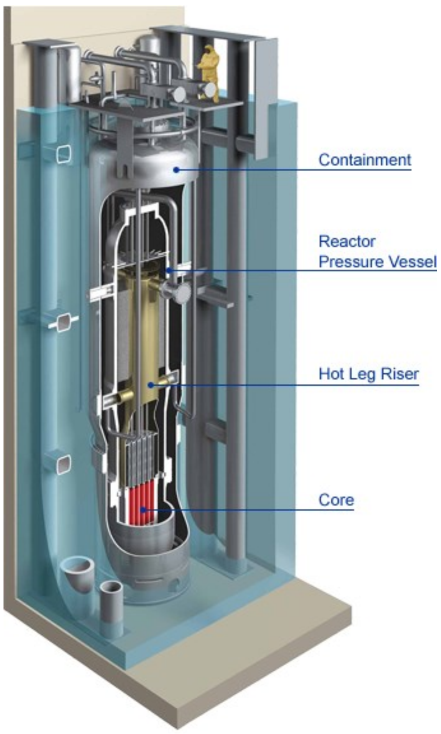
Historically, the pressurized water has been the most viable and reliable technology. It is adopted in more than 60% of operating NPPs and it is still used for Gen III+ large plants under construction (EPR, AP1000). Thanks to large industrial and operational experience, this solution has been widely considered also for innovative SMRs. When moving to smaller reactor sizes, several novel pressurized water designs implement the integral concept, with most of the primary components, i.e. core, control rods driving mechanism, steam generator, primary pumps and pressurizer, arranged inside the Reactor Pressure Vessel (RPV). The basic idea is to match the use of proven technology with innovative features, providing at least four unique advantages that can make these SMRs more suitable for today's market:

(i) Compact design and reduction of occupied volume.
(ii) Easier construction, with several components that can be manufactured and assembled in factory.
(iii) Layout simplification: less critical components, less RPV penetrations, inherent protection towards large break LOCA and other typical design basis accident.
(iv) Passive safety strategy: possibility to use natural circulation for emergency decay heat removal.

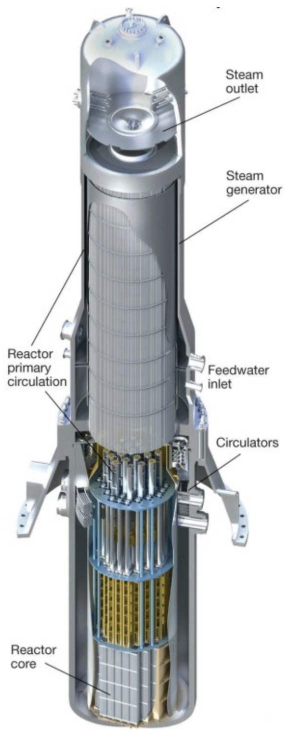
The integral PWRs (iPWRs) are the most promising SMR concepts that can be manufactured, constructed and operated in the short term (Ricotti, 2014). Besides the mentioned CAREM reactor under construction in Argentina, at date *SMART - South Korea* - (Kim, et al., 2014-a)*, nuScale* and *mPower - United States* - (Reyes Jr., 2012) (Halfinger & Haggerty, 2012) are integral SMRs very close to deployment. SMART (System-integrated Modular Advanced ReacTor) is a 100 MW_{el} multi-purpose application reactor designed by KAERI and received standard design approval from the Korean regulator in 2012. In recent years, the US Department of Energy has decided to fund the accelerated development of NuScale (50 MW_{el} modular unit by NuScale Power) and mPower (Babcock & Wilcox 180 MW_{el} factory-made reactor) with \$217 and \$226 million, respectively, although mPower design effort was completely stopped in 2016. Another key iPWR is IRIS (International Reactor Innovative and Secure) (Carelli, et al., 2004), whose development was prematurely stopped in 2011, a 335 MW_{el} medium reactor designed by an international consortium led by Westinghouse. IRIS development activities allowed a great technological advancement to integral SMRs. All these reactors present recurring characteristics in terms of integral reactor layout and safety strategy. To reduce the vessel dimensions, the design of in-vessel components is revisited with respect to classic models: for instance, the pressurizer is placed in the dome of the RPV and the steam generator modules have helical geometry to save space and enhance heat transfer. Free convection is adopted to remove decay heat in emergency situations. Small-size reactors (CAREM and NuScale) can use natural circulation also in normal operation, thus eliminating the need of primary pumps. The ultimate way to reject decay heat to the exterior is the passive containment cooling: reactor is depressurized, and the steam produced is condensed in contact with the surface of the containment, which is itself cooled from outside by a pool or with a gravity driven flow. Table III summarizes these features and Figure 1 shows integral SMRs layouts.

Table III. Distinctive features of integral PW-SMRs

	NuScale	mPower	CAREM	SMART	IRIS
Helical steam generator	✓		✓	✓	✓
Integrated pressurizer (or self-pressurized)	✓	✓	✓	✓	✓
Natural circulation in normal operation	✓		✓		
Natural circulation for emergency decay heat removal	✓	✓	✓	✓	✓
Passive cooling through containment	✓	✓			✓



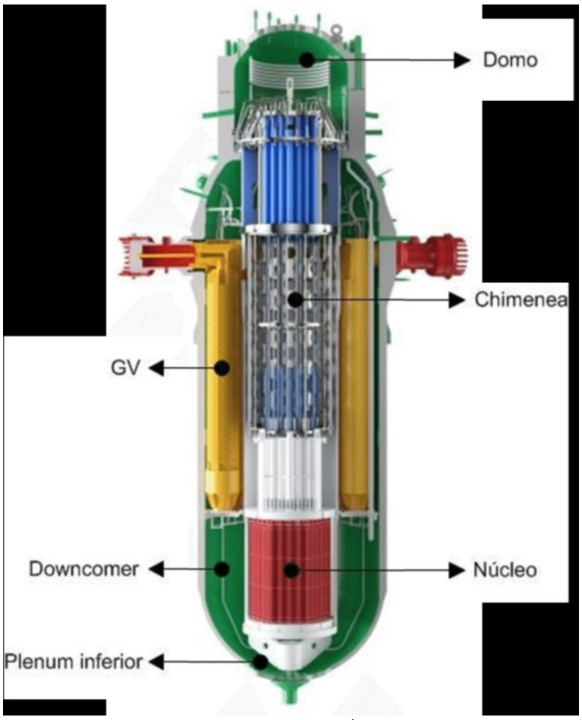
NuScale [2]



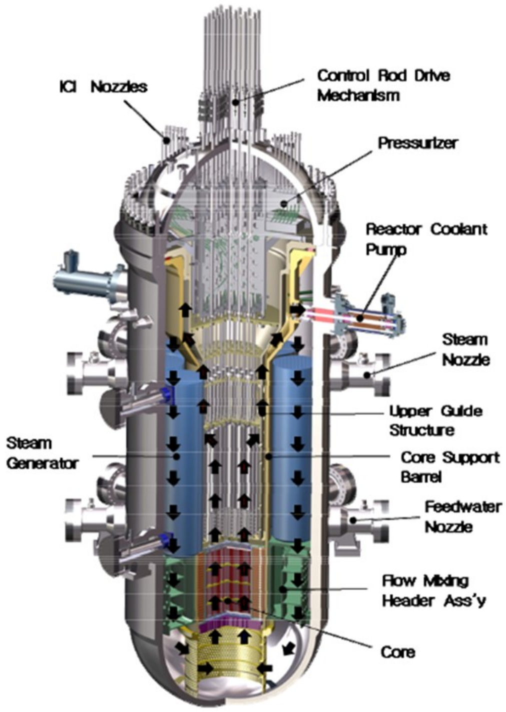
mPower [3]

[2] Source: Reyes Jr., J.N. & Lorenzini, P. (2010). NuScale Power: A modular, scalable approach to commercial nuclear power *Nuclear news* 53(7), pp. 97-107

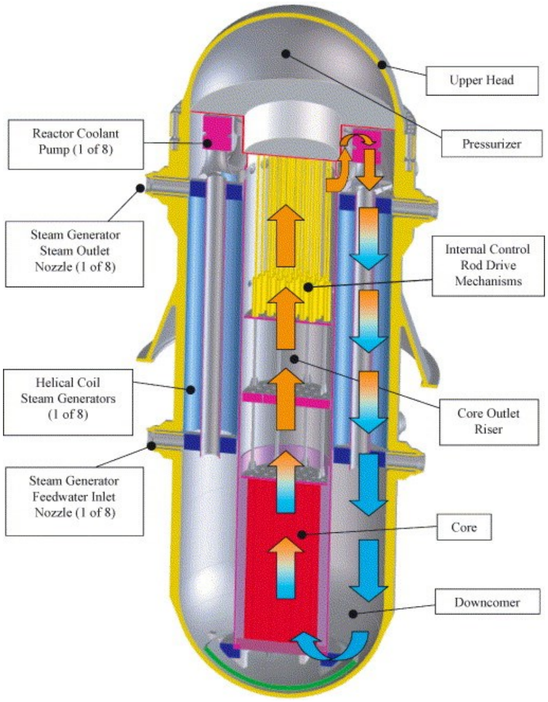
[3] Source: Halfinger, J. A., & Haggerty, M. D. (2012). The B&W mPower™ scalable, practical nuclear reactor design. *Nuclear technology*, *178*(2), 164-169



CAREM [4]



SMART [5]



IRIS [6]

Figure 1 Integral layout of NuScale, mPower, Carem, SMART and IRIS

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[4] Source: Catalá, L., Laiglecia, J., Maturana R. & Zanocco P. (2016). Analysis of process and regulations systems influence on the CAREM-25 reactor in a Loss-of-Coolant Accident *Proc. of 43° AATN "La mitigación del cambio climático tiene varias soluciones: la energía nuclear es la mas importante"*

[5] Source: Kim, K. K., Lee, W., Choi, S., Kim, H. R., & Ha, J. (2014). SMART: the first licensed advanced integral reactor. *Journal of Energy and Power Engineering*, *8*(1), 94.

6 Source: Carelli, M. D., Conway, L. E., Oriani, L., Petrović, B., Lombardi, C. V., Ricotti, M. E., ... & Grgić, D. (2004). The design and safety features of the IRIS reactor. *Nuclear Engineering and Design, 230*(1), 151-167.

1.2.3. Off-shore SMRs

Sea-based and transportable NPPs are SMRs designs conceived for off-shore operation. The purpose of these reactors is to satisfy the energy needs in regions of the world where land is scarce, isolated or just unsuitable for the construction of a nuclear or oil&gas power plants. This is, for instance, the case of remote areas with large natural resources, islands or highly populated areas under the threat of natural hazards. Off-shore SMRs can be classified into floating and steady operation (Figure 2). Floating barges hosting a small reactor for electricity production are the mentioned KLT-40S and the ACPR50S under construction in Russian Federation and China. An alternative solution to the floating barge consists in setting the reactor underwater, moored on the seafloor. This option is appearing quite attractive, as the Fukushima accident calls our nuclear industry to better consider extreme external events, like a tsunami, in the design of NPPs. Electric Boat (General Dynamics Electric Boat Division, 1971) and Herring (Herring, 1993) investigated subsea reactor designs in the 1970's and 1990's respectively. These projects stayed at the paper project stage. Nowadays, the progresses in subsea oil&gas technologies, submarine cables for offshore renewables and in shipbuilding techniques make offshore power reactors more feasible today than before. Based on its experience in the design, fabrication, maintenance and dismantling of nuclear-powered submarines and ships, in 2014 the French company DCNS (now Naval Group) presented the *Flexblue* concept (Haratyk, et al., 2014), a subsea and fully transportable nuclear power plant. Furthermore, other two concepts of offshore reactors can be found in literature. The *Offshore Floating Nuclear Plant* (OFNP) concept developed by Massachussets Institute of Technology (MIT - United States) represents a solution for stationary design (Buongiorno, et al., 2016): the reactor is built on a platform in a shipyard, transferred on the site within territorial waters and anchored in relatively deep water (100 m). Also, an ocean reactor based on the SMART design (Kim, et al., 2014-b) has been proposed by the Korea Advanced Institute of Science & Technology (KAIST – South Korea): the reactor operates on an offshore gravity-based structure, improving the safety from tsunamis and earthquakes.

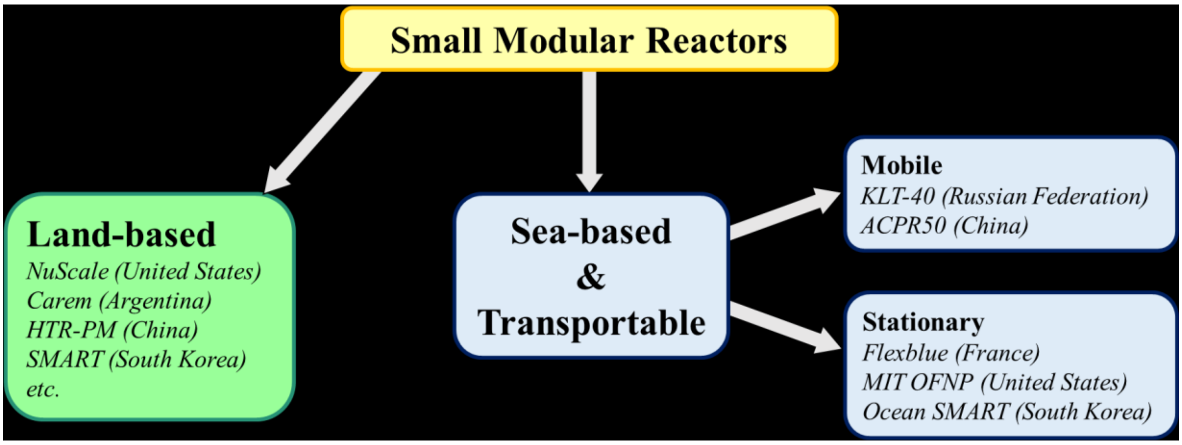

Figure 2 Site classification of SMRs

1.3. Safety potentialities of the submerged containment: NuScale and Flexblue

Placing a nuclear reactor in a submerged metallic containment is an innovative concept able to satisfy high levels of nuclear safety, owning some characteristics that are unique in the current nuclear scenario. With a fully passive safety strategy, this feature allows using natural circulation to cool the fuel rods for very long time. Decay heat rejection can follow two tracks: heat exchangers directly dipped into the water surrounding the containment and heat transfer through the metal containment.

This second case is a sort of ultimate solution for the cooling of the fuel rods, which can be actuated in case of failure of all the other safety systems. The core is depressurized, and the steam produced by the decay heat is vented in the closed containment atmosphere. The steam condenses in contact with the metal surface of the containment, which is cooled from the exterior with cold water. The condensate is then redirected into the core with a passive recirculation system. Some advanced SMR designs exploit this idea of removing the decay power from the core through the metal containment by placing the reactor in a pool. This is the case of NuScale. Other SMRs, such as SMART and mPower, and even a large-scale reactor, i.e., the AP1000 by Westinghouse (Schulz, 2006), implement a similar concept: the external surface of the containment is cooled with water drained by gravity from a large tank placed in a more elevated position. Flexblue and OFNP are designed to operate offshore and use the seawater to cool the containment.

1.3.1. NuScale

The key-point of NuScale passive safety is represented by the immersion of the RPV and reactor containment in a water-filled, stainless steel-lined concrete pool that resides underground. The pool is housed in a Seismic Category I building and acts as major heat sink. The pool is sized in order to be large enough to provide 30 days of core and containment cooling without adding water. The Containment Heat Removal System (CHRS) provides a mean to remove heat to the external pool in case of LOCA or if other primary cooling systems are unavailable. The primary coolant is vented directly into the containment, where it condenses on its internal surface. The condensate collects in the lower containment region. When the liquid level in the containment sump rises above the top of the recirculation valves, the valves are opened to establish a natural circulation path from the sump through the core and out of the reactor vent valves. The process is shown in Figure 3. After 30 days, the core decay heat generation is so small that the natural convection heat transfer to air and radiative heat transfer at the outside surface of the containment are sufficient to remove the core decay heat for an unlimited period. (Reyes Jr., 2012). However, this aspect of the safety strategy is feasible only with low reactor power output and large RPV external surface. Air cooling several days after the scram become possible because, with nominal thermal power 160 MW_{th}, the decay heat after 30 days is only 260 kW. In addition, NuScale RPV is 20.0 m tall, therefore the available heat transfer area for convective air cooling is very large.

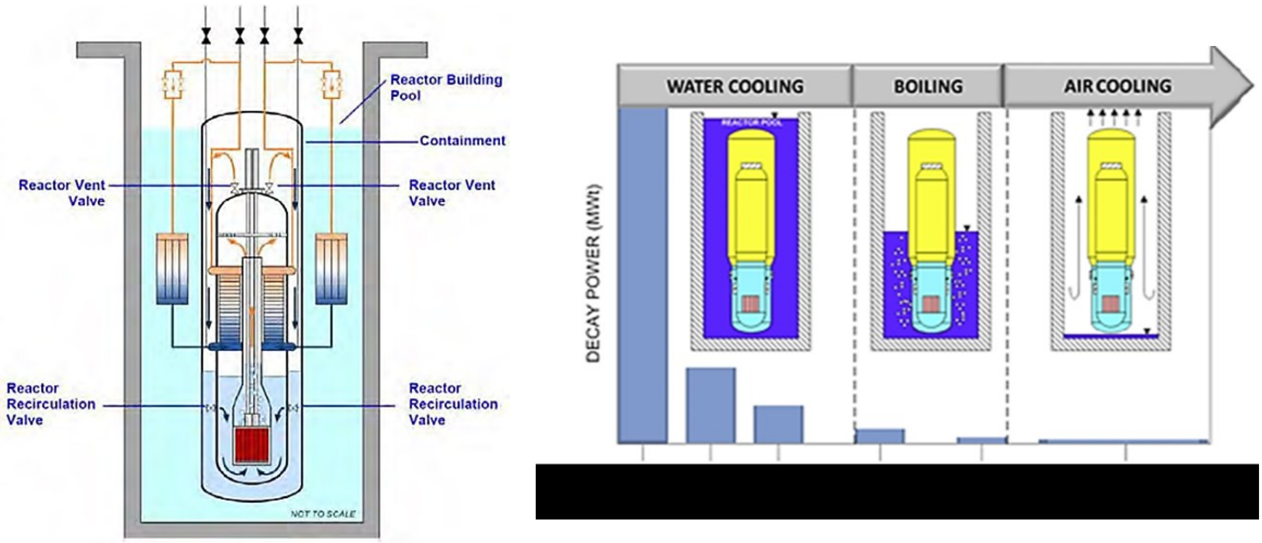

Figure 3 Schematic of NuScale Containment Heat Removal System and cooling strategy (Reyes Jr., 2012)

1.3.2. Flexblue

In NuScale design, there is a great advancement in term of protection toward the Loss of Ultimate Heat Sink (LUHS) accident, but that scenario cannot be totally excluded in case of events that can lead to the failure of the pool building, e.g., high magnitude earthquakes and fabrication errors. To inherently prevent from of the LUHS, the metallic reactor containment must be placed in a large submerged environment, such as the sea or an artificial lake. This ensures the presence of an always available heat sink and the grace period in case of accident becomes virtually unlimited. Santinello et al. (Santinello, et al., 2017-a) performed a CFD study on the external natural circulation driven by a large horizontal cylinder heated at 100°C, showing that the temperature of the heat sink is constant in almost all the simulated domain. The activity was supported by DCNS and aimed at defining the maximum thermal output of Flexblue.

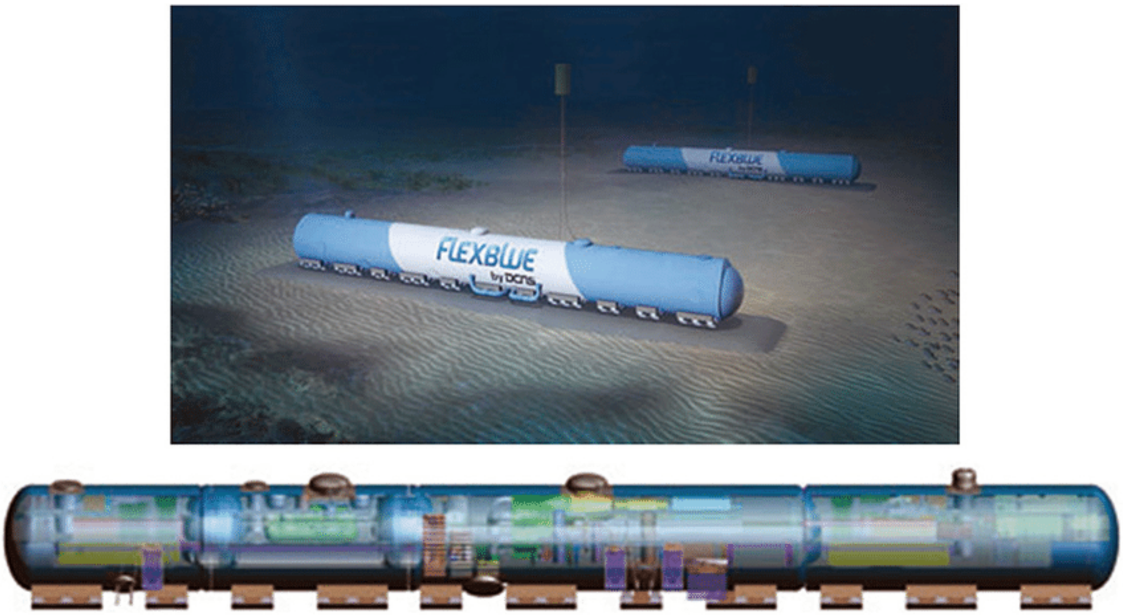

Figure 4 Flexblue plant layout (Santinello, et al., 2017-a)

Flexblue (Haratyk, et al., 2014) is a modular power unit that can supply 160 MW_{el} (500 MW_{th}) to the grid (Figure 4). It is immersed down to a hundred-meter depth, a few kilometers away from the shore, within territorial waters. Flexblue is entirely manufactured in factories and assembled in a shipyard per naval modular construction techniques. The module, a cylindrical hull of about 150 m long and 14 m diameter, is brought on site by transport ship and moored on the seafloor, where production takes place. Electricity is transferred to the mainland via submarine cables. Every 3 years approximately, electricity production stops for refueling: the module is removed and transported back to a coastal facility, which hosts the spent fuel pool. Flexblue uses typical pressurized water reactor technology, which exploits a considerable experience in commercial power plants and naval environments. Flexblue also benefits from factory fabrication process, enhancing the quality of manufacturing. The safety relies on the submerged environment. Immersion provides protection against external events, such as tsunami, earthquake (with opportune engineering solutions), flooding, extreme weather conditions and malevolent human actions. The availability of the seawater acting as a permanent and infinite heat sink allows exploiting fully passive safety features, where emergency core cooling is done without the need of AC power or human intervention. Although Flexblue development is at conceptual level only, some works have been performed to investigate the safety strategy in response to postulated accidents (Haratyk & Gourmel, 2015) (Gourmel, et al., 2016) (Santinello, et al., 2017-b). The cooling process in the long-term period is shown in Figure 5. Following a LOCA or a generic depressurization of the primary system, the reactor compartment is flooded with the water of an in-hull large safety tank. Steam is produced in the core and sump natural circulation flow is established thanks to recirculation lines, providing core cooling even when decay power has become very low. Ex-vessel flooding adds another channel to cool the reactor, since also the liquid water exchange to the exterior through the containment. In general, the safety relies on a large water inventory inside the hull, which absorbs the decay heat and then transfer it to the exterior through the containment.

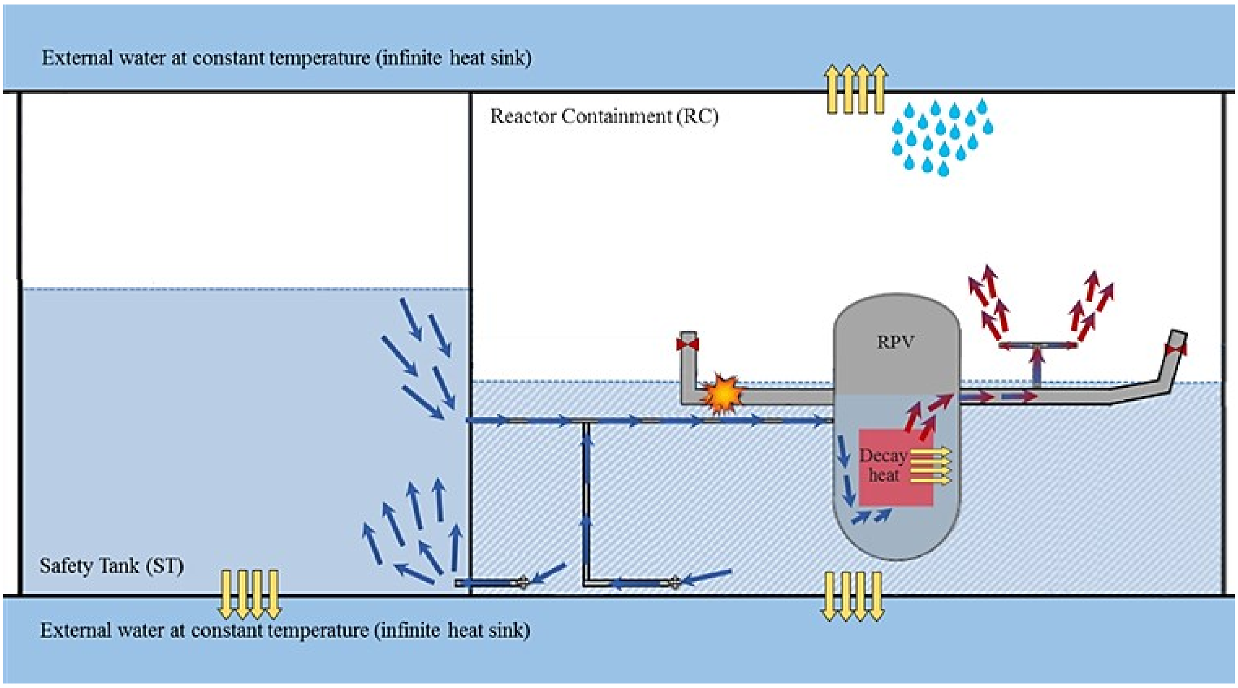

Figure 5 Working principle of long-term cooling (Santinello, et al., 2017-b)

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Section 2

Reactor configuration in a submerged containment

The basic reactor configuration of a submerged SMR operating in a horizontal hull, like Flexblue, is mainly influenced by two key parameters. The first one is given by the heat transfer capacity through the hull, which determines the maximum allowed power output of the reactor. The second one is the size of the containment: the reactor should be placed inside a horizontal cylindrical hull, whose diameter is limited by manufacturing capabilities and economic reasons. Therefore, definition of the reactor layout is quite challenging and requires ad-hoc studies. This section addresses the topic and presents a preliminary design of an integral PWR layout, based on the IRIS reactor concept. Components of the primary system have been revisited and scaled in order to reduce the power output to 500 MW_{th} and to fit the 14m-diameter of Flexblue hull. It is also shown that other concepts already proposed, i.e. a VVER-type and the French SCOR-F, are not suitable to achieve design and safety targets.

2.1. Former proposed configurations for Flexblue

2.1.1 Overview

At the current stage of its development, Flexblue still does not present a final reactor design. While it surely adopts pressurized water technology, no decisions have been taken about the reactor layout to be placed inside the horizontal hull. The two main constraints come from (i) safety and (ii) manufacturing reasons. (i) The power output is determined by the heat transfer capacity through the hull, to avoid overheating of the fuel rods during the emergency decay heat removal just after the scram. A study by Santinello et al. (Santinello, et al., 2017-a) faced this problem, observing that the presence of a painting layer on the external surface, necessary to protect the hull from chemical degradation, drastically reduces the heat flux. However, the study also allowed determining a suitable power output at 500 MW_{th}. (ii) To ensure transportability and in-factory fabrication, the reactor must lie in a cylindrical hull, whose diameter is limited by manufacturing capacity and economic reasons. For Flexblue, the reference hull diameter is 14 m (Haratyk, et al., 2014). Other requirements regard the adaptability to a fully passive safety strategy. In the past years, a couple of options have been proposed for the Flexblue case, i.e. a loop-type reactor and SCOR-F reactor from CEA. For different reasons, these two solutions may be inappropriate for a submerged reactor.

A work by Shirvan et al. (Shirvan, et al., 2016) examined five nuclear technologies in relation to their adaptability for an offshore underwater SMR: Lead-Bismuth Fast Reactor (LBFR), Organic Cooled Reactor (OCR), Superheated Water Reactor (SWR), Boiling Water Reactor (BWR) and integral PWR. They concluded that all these technologies are good for a fully passive safety strategy. In addition, LBFR and OCR are the most volume saving technologies, while OCR and BWR seem to allow cost reduction. However, this study does not properly consider that LBFR and OCR can rely on a very little experience in civil nuclear industry. Lead-bismuth coolant was used only for military submarine propulsion and organic fluids were investigated only up to the ’60s and then the concept had been abandoned. The development of such technologies would require a huge amount of work in terms of research and experimental testing. Moreover, these concepts need to be adapted for an underwater SMR, where reactor control is in remote and maintenance cycles are necessarily quite long. For these reasons meeting the requirement of safety authorities would be very difficult and not feasible in the short-medium term.

2.1.2 Loop-type design (VVER-like)

A loop-type reference design was adopted by DCNS for illustration purposes and in some preliminary safety analysis. It exhibits two primary loops, with two primary coolant pumps and two recirculation Steam Generators (SGs) (Figure 6). The distinguishing feature is the horizontal layout of the SGs, which make it similar to the VVER (*Vodo-Vodyanoi Energetichesky Reactor,* Water-Water Energetic Reactor) developed and built in Soviet Union since the ‘70s. DCNS proposed such layout as a preliminary choice. The reason is the necessity to reduce the total height of the reactor in order to fit in the containment. Manufacturing constraints are therefore satisfied, but a VVER-like design presents some downside to adopt a fully passive safety strategy. The reactor configuration is not compact, resulting unsuitable for natural convection. The horizontal SGs do not offer gravity head to natural circulation and introduce large hydraulic resistances in terms of friction and form losses. Such SGs would not operate efficiently in emergency cooling circuits. The consequence is that the reliability of emergency loops with ex-hull heat exchangers connected to the secondary circuit is questionable, since in these conditions natural circulation equilibria are very delicate and exposed to instabilities. Therefore, the safety strategy would be mainly based on Direct Vessel Injection (DVI) systems and core cooling must rely almost only on internal safety tanks, which should be large enough to avoid boiling or overheating. Basically, a VVER-like reactor would introduce difficulties to achieve well-designed passive safety systems. In addition, the number of required RPV penetrations would be considerably high, also because of redundancy policies, thus increasing the risk for large and small break LOCA.

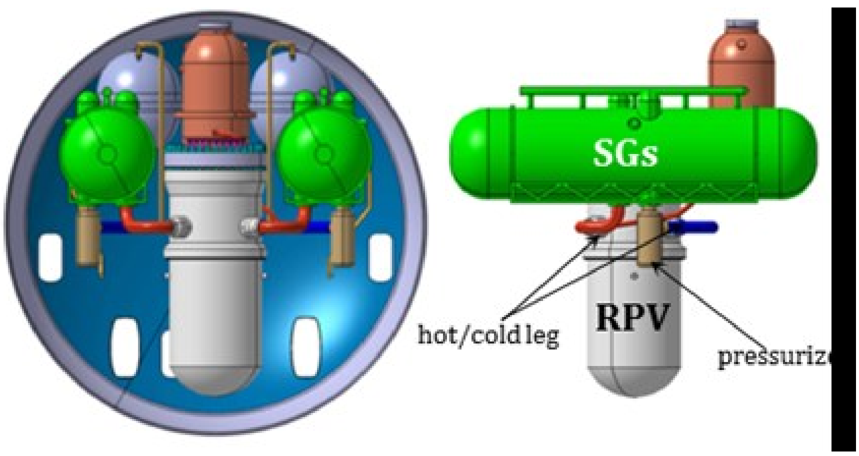
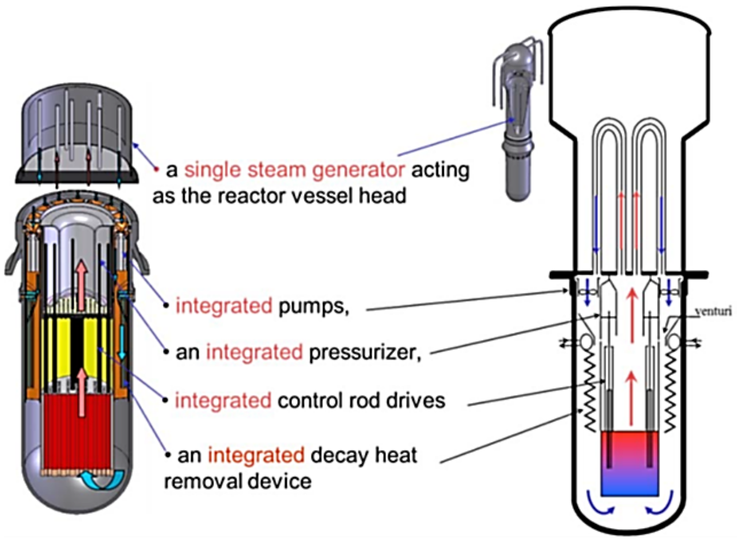


Figure 6 VVER-like configuration for Flexblue containment (Gourmel, et al., 2016)

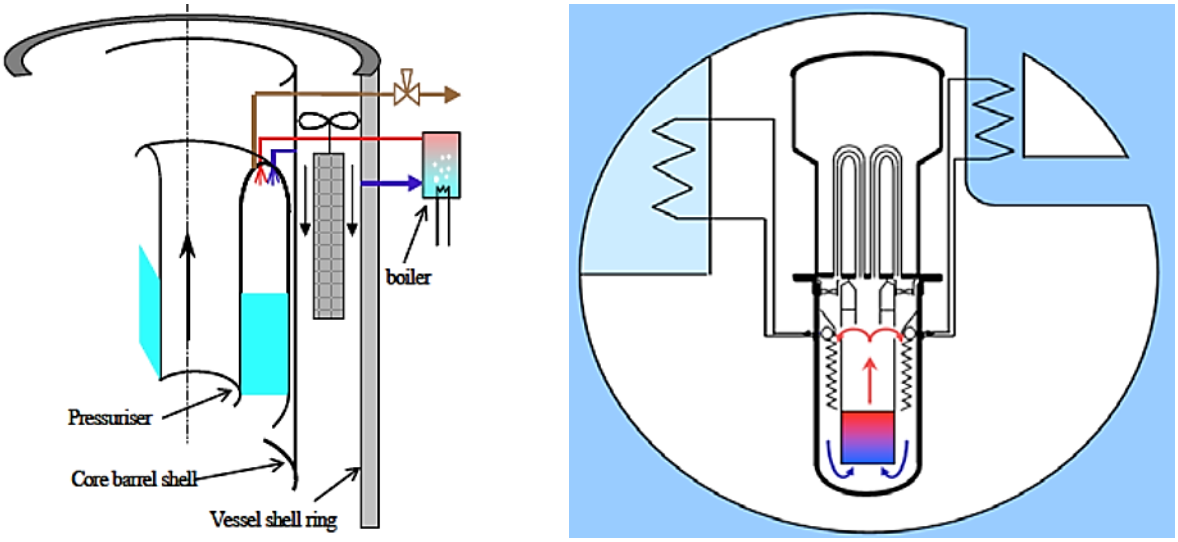
2.1.3 SCOR-F

SCOR-F (Figure 7a) was the integral reactor design proposed by NuSMoR project to be arranged in the horizontal containment of Flexblue (NUSMoR consortium, 2014). The activity was included in a call within the framework of the Horizon 2020 Euratom work program. SCOR-F is a reduced power version of SCOR (Simple COmpact Reactor), a medium size - 630 MW_{el} - integral PWR developed by CEA (Commissariat à l'Energie Atomique) in France (Gautier & De Masi, 2005). The initial idea of the proposers was to adapt the SCOR design to a submerged containment, but a complete design about SCOR-F concept has not been developed yet and no data are available. The SCOR design is based on a compact reactor pressure vessel that houses the main primary system components including core, pressurizer, steam generator, reactor coolant pumps, control rod drive mechanism and heat exchangers of the decay heat removal system. Its peculiarity is the vertical U-tube SG placed right above the core, acting also as the reactor vessel head. The pressurizer also has an uncommon configuration: it is integrated into the upper part of the riser, just below the SG, and designed in an annular shape as of an inverted-U with hot coolant flowing in its center (Figure 7b). Electric resistances and sprayers are outside the RPV. Concerning safety, the residual heat removal in primary circuit is the main system devoted to decay heat removal. It is made of a certain number of bayonet tube heat exchangers integrated into the RPV (Figure 7c) and connected to safety pools or directly to the seawater. During normal operation, this system works in series with the SG, while in emergency situations a venturi component operates a SG by-pass and allows establishing natural convection in safety circuits.

SCOR-F is suitable for a fully passive safety strategy and its integral layout needs only small vessel penetrations, thus eliminating by design the Large Break LOCA (LB-LOCA) accident scenario. However, pressurizer and, especially, SG layouts put severe concerns about the adaptability of SCOR-F for a submerged containment. Pressurizer inverted-U design needs accurate verifications and sizing, since the volume available inside the RPV is quite limited. In addition, remote control of ex-vessel sprayers and resistances poses another issue on this pressurizer concept. On the other side, the problem with the SG is that the position over the riser and the vertical layout lead to doubt about the possibility to place the reactor inside Flexblue hull. With preliminary calculations and reasonable assumptions, the total height of SCOR-F has been estimated, showing its unsuitability for the 14-m diameter containment.



(a)



(b) (c)

Figure 7 SCOR-F configuration: (a) Reactor Pressure Vessel; (b) pressurizer; (c) Residual heat Removal in Primary circuit (NUSMoR consortium, 2014)

In principle, it is possible to assume that the fuel assembly would have active height around 2 m, which becomes 3 m considering the gas plenum and core support plates. The height of the control rods and drive lines is typically twice the length of the fuel assembly plus the handling mechanism. The resulting height is roughly 5.5 m. Other contributions to the total height come from the lower plenum, i.e. ≈ 1.5 m, and from the plate separating primary and secondary sides, i.e. 0.3 m. Concerning the SG height, rough calculations with a lumped parameter approach allow a preliminary estimation. The length of the SG tubes has been subdivided in three zones as in Figure 8: heat transfer in Zone 1 and Zone 3 (tube inlet and tube outlet) is liquid-to-liquid, while Zone 2 sees the two-phase mixture on the secondary side. The saturation point determines the limits between the zones. Since the design of SCOR-F is not defined, operating conditions and constructive data needed for the are assumed from SCOR, i.e. core inlet-outlet temperature, flowrate per SG tube, tube diameter and thickness, secondary fluid sub-cooling.

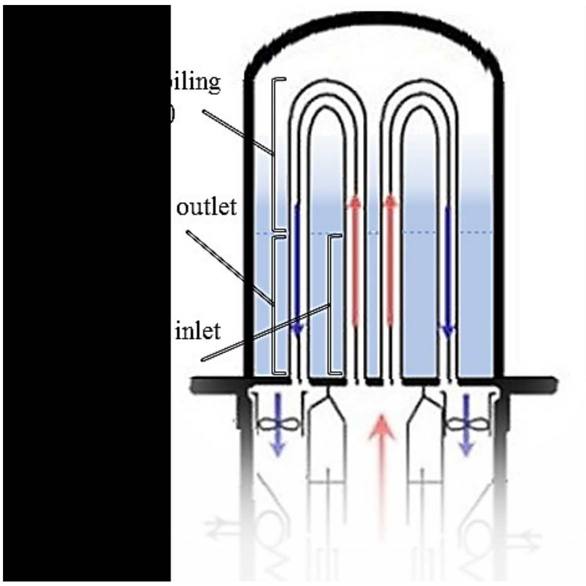


Figure 8 Sketch of SCOR-F steam generator

The system in eq. (1) provides the two primary temperatures at the limits between zones, i.e. T_{1-2} and T_{2-3}. However, this system is underdefined: given the inverted-U shape of the steam generator, both Zone 1 and Zone 3 contribute to the enthalpy jump in the liquid part of the shell side and it is not possible to determine $\dot{Q}_1$ and $\dot{Q}_3$ from the system only. $\dot{Q}_2$ is known from eq. (2), considering the enthalpy balances in the shell side.

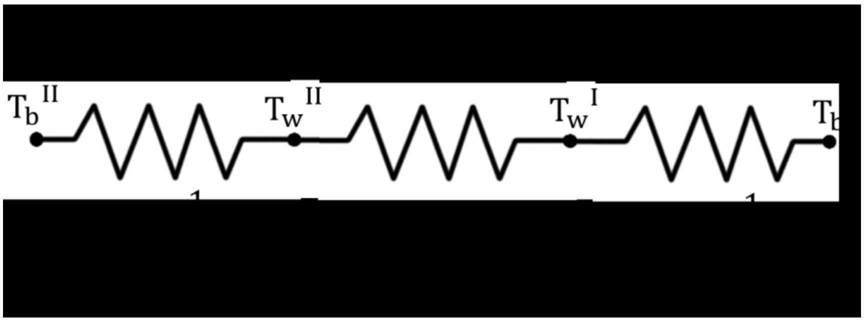
$$\begin{cases} \dot{Q}_1 = \left[mc_p(T_{in} - T_{1-2})\right]^{I} \\ \dot{Q}_2 = \left[mc_p(T_{1-2} - T_{2-3})\right]^{I} \\ \dot{Q}_3 = \left[mc_p(T_{2-3} - T_{out})\right]^{I} \end{cases} \tag{1}$$

$$\dot{Q}_2 = \left[m\left(h_{out} - h_{sat}^{liq}\right)\right]^{II} \tag{2}$$

Superscripts I and II indicate tube and shell SG sides, respectively.
Subscripts 1, 2 and 3 indicate the number of the Zone

$\dot{Q}_1 = \dot{Q}_3$ has been assumed as guess initial hypothesis and then the system has been solved with an iterative routine, which returns $\dot{Q}_1, \dot{Q}_3, T_{1-2}, T_{2-3}$ ensuring that $\dot{Q}_1 + \dot{Q}_2 + \dot{Q}_3$ gives the total thermal power of SCOR-F. T_{1-2} and T_{2-3} allow defining preliminary mean zone tube bulk temperatures T_b for the lumped parameter approach. For each zone, the system in eq. (3) has been set by imposing the continuity of the heat flux at the tube interfaces: the unknowns are the average superficial temperatures. Calculations are based on the electric analogy (Figure 9). α^I and α^{II} are the convective Heat Transfer Coefficients (HTCs), calculated with empirical correlations. For two-phase flow heat transfer, i.e., in Zone 2 at the shell side ($\alpha_2{}^{II}$), the correlation by Kandlikar has been adopted (Kandlikar, 1990), while for all the other zones with single phase heat transfer the Dittus-Boelter correlation was suitable (Dittus & Boelter, 2000). Correlations are reported in Table IV.

$$\begin{bmatrix} \dfrac{1}{R_w} & -\left(\dfrac{1}{R_w} + \dfrac{1}{R^{II}}\right) \\ -\left(\dfrac{1}{R_w} + \dfrac{1}{R^{I}}\right) & \dfrac{1}{R_w} \end{bmatrix} \begin{bmatrix} T_w{}^{I} \\ T_w{}^{II} \end{bmatrix} = \begin{bmatrix} -\dfrac{T_b{}^{II}}{R^{II}} \\ -\dfrac{T_b{}^{I}}{R^{I}} \end{bmatrix} \tag{3}$$



[redacted]

Figure 9 Electric analogy of the SG system, with formulas for thermal resistances

Table IV. Empirical correlations used to determine HTCs

Author(s)	Correlation		Applicability
Dittus & Boelter	$\mathrm{Nu_D} = 0.023\mathrm{Re}^{0.8}\mathrm{Pr}^{n}$ $n = 0.3$ *for fluid being cooled* $n = 0.4$ *for fluid being heated*	$\mathrm{Re} > 10^4$ $0.6 < \mathrm{Pr} < 160$ $L/D = 10$	All zones of tube side Zone 1 and 3 of shell side
Kandlikar	$\dfrac{\alpha_{tp}}{\alpha_{lo}} = C_1 Co^{C_2} + C_3 Bo^{0.7}$ *α_{lo} is liquid single-phase HTC to be calculated with Dittus-Boelter equation.* *Water in two-phase flow* *Vertical geometry*	$Co = \left(\dfrac{1-x}{x}\right)^{0.8}\left(\dfrac{\rho_g}{\rho_l}\right)^{0.5}$ $Bo = \dfrac{q''A}{mh_{fg}}$ *HTC best estimation is the highest between convective and nucleate coefficients.* **Convective** / **Nucleate** **C_1**: 1.1360 / 0.6683 **C_2**: -0.9 / -0.2 **C_3**: 667.2 / 1058.0	Zone 2 of shell side

The results provide the temperatures on the metal surfaces. These allow calculating the axial heat flux in each zone and then deducing the required lengths to exchange the thermal powers $\dot{Q}_1, \dot{Q}_2, \dot{Q}_3$. The Kandlikar correlation introduces the boiling number Bo, which depends on the heat flux. In this use, the correlation is implicit and consequently the solution of the system in eq. (3) requires an iterative procedure for Zone 2. In addition, it is necessary to impose that the lengths of zones 1 and 3 are equal: also this operation needs an iterative procedure, since the variation of zone length modifies $\dot{Q}_1, \dot{Q}_3$ and mean primary temperatures. All calculations have been solved with Matlab-R2017b. Figure 10 shows the temperature profiles in the SG as a function of the length of the tubes. The estimated length of the SG tubes necessary to transfer the whole thermal power is around 15 m. Therefore, even considering a mild curvature, the overall height of the inverted-U SG would not be lower than 7.0 m, which becomes even higher if steam separators are arranged inside the RPV.

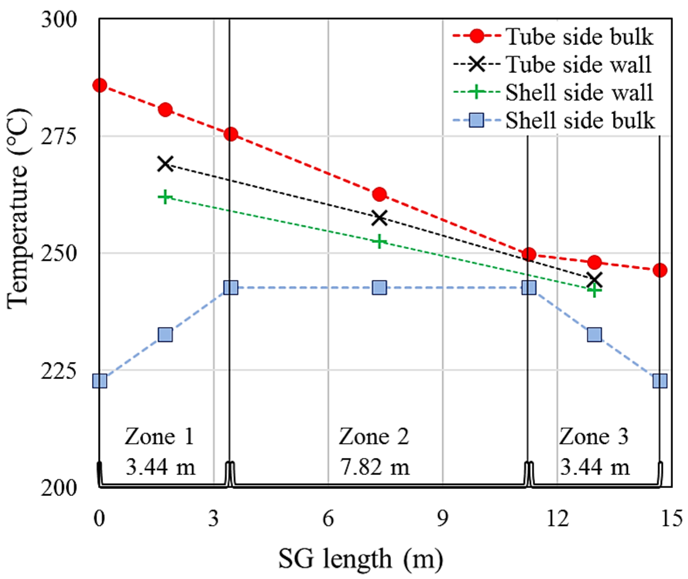


Figure 10 Estimated temperature profiles in SCOR-F steam generator given as function of SG tube length

The resulting height of SCOR-F is summarized in Table V. The position and the shape of the SG makes this concept unsuitable for the 14-m diameter of Flexblue hull and, in general, for submerged and transportable SMR.

Table V. Estimated height of SCOR-F components

RPV thickness (x2)	0.3 m
Lower plenum	1.5 m
Reactor core	3.0 m
Control rods driving mechanism	5.5 m
Primary/secondary separating plate	0.3 m
Steam generator (no in-vessel steam separators)	7.0 m
Total estimated minimum height	**17.6 m**

2.2. IRIS primary system as reference configuration

The International Reactor Innovative and Secure (IRIS) (Carelli, et al., 2004) is a reactor design developed from 2001 to 2011 by an international consortium led by Westinghouse and involving 26 contributors from ten counties among industry, university, laboratories and power companies. IRIS is an integral, modular, medium size (335 MW_{el}) PWR. The vessel houses not only the reactor core, but also all the primary components: eight spool-type primary pumps, eight once-through helical steam generator modules, the control rod drive mechanisms and the pressurizer integrated in the upper vessel. Also, a steel reflector, which surrounds the core, improves neutron economy and provide a first radiation shielding, is arranged in the RPV. This configuration requires a larger RPV diameter with respect to conventional pressurized reactors, but, thanks to the integral layout, the size of the containment results considerably smaller.

Reactor core uses 89 17x17 fuel assemblies with design similar to conventional Westinghouse PWR elements. Core active height is 4.267 m, producing 1,000 MW_{th} as nominal thermal power. The designed fuel is UO_2 enriched to 4.95% in ^{235}U. Reactivity control is accomplished through solid burnable absorbers, control rods and soluble boron. Fuel cycle is 3-3.5 year long, with half core-reload at each refueling to optimize the burnup.

The *Control Rods Driving Mechanism (CRDM)* is placed inside the RPV, in the region above the core and surrounded by the steam generators. This carries two advantages: (i) the rod ejection accident is eliminated by design, because there is no differential pressure to drive out the CRDM extension shafts; (ii) there are no nozzle penetrations on the upper head of the RPV. In IRIS design, the CRDM was placed above the core and actuated with electromagnetic or hydraulic mechanism.

Coolant pumps are of an axial “spool-type” located entirely within the reactor vessel, with only small penetrations for the electrical power cables and for water cooling supply and return. They are placed in hot leg position, to achieve better integration in the primary circuit and easier access, being far from the core. On the other side, in such position pumps must work with hotter fluid and are more exposed to cavitation risks, thus requiring higher attention in maintenance operations.

The *steam generators* (Figure 11a) are once-through, helical-coil tube bundle design with the primary fluid outside the tubes. Eight SGs are arranged in the annular space between the core barrel and the reactor vessel. For each module, a central inner column supports 656 tubes and the entire secondary circuit is designed to bear the primary pressure in case of SG tube rupture. The helical-coil design provides compactness and enhanced heat transfer capabilities with respect to conventional SGs. In addition, it allows accommodating thermal expansion without excessive mechanical stress and has high resistance to flow-induced vibrations.

The *pressurizer* (Figure 11b) is integrated into the upper head of the reactor vessel. The pressurizer region is defined by an insulated, inverted top-hat structure that divides the circulating reactor coolant flow path from the saturated water. This configuration provides a very large water and steam volume, as compared to plants with a traditional, separate, pressurizer vessel, i.e., total volume ≈71 m^3, steam volume ≈49 m^3. Thanks to the large available volume, IRIS pressurizer does not require sprayers.

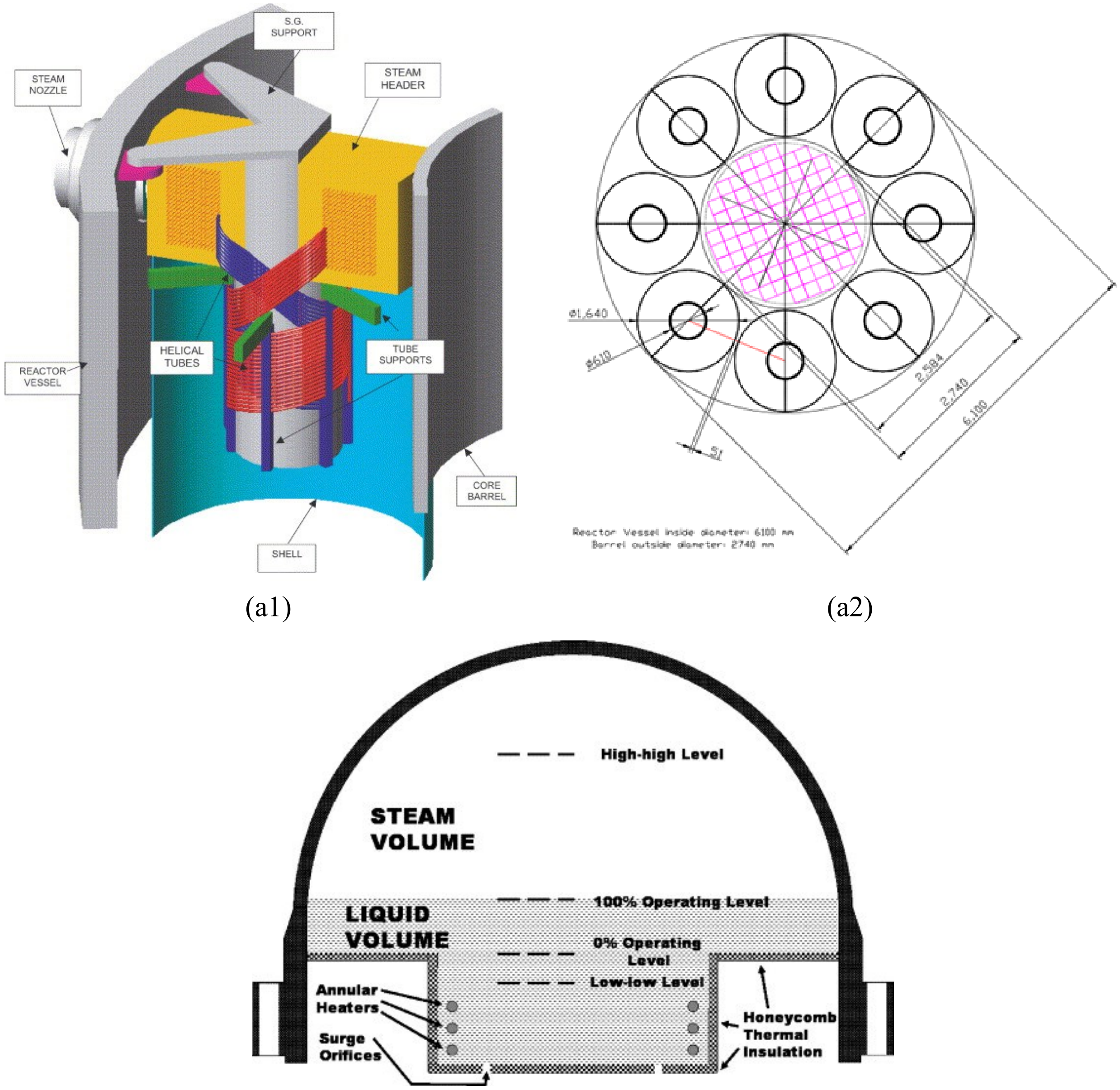


Figure 11 IRIS steam generator – pictorial view (a1), layout (a2) – and pressurizer (b) (Carelli, et al., 2004) (Cinotti, et al., 2002)

2.3. Revisited reactor layout: IRIS-160

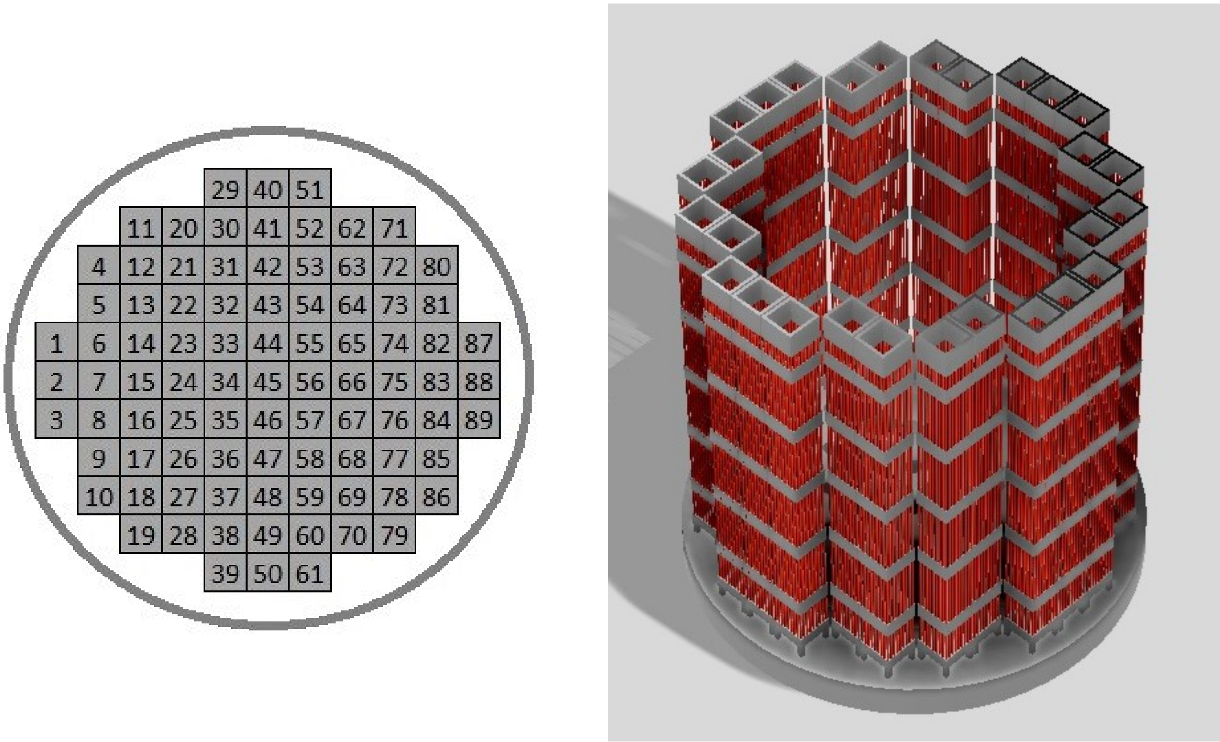
2.3.1 Introduction

The purpose of this section is to introduce the concept of an integral PWR (iPWR) SMR, suitable to operate in a submerged containment and based on a scaled version of the IRIS design. Scaling IRIS is not a new operation: in 2009 Petrovic et al. (Petrovic, et al., 2009) proposed the concept of IRIS-50, a reduced power (50 MW_{el}) version of the standard 335 MW_{el} IRIS design, conceived to better address cogeneration purposes and to supply electricity to remote or isolated areas. This concept was approached utilizing as much as possible IRIS design efforts. The number of steam generator modules was reduced from eight to two. The authors also discussed the use of natural circulation in normal operation, but the idea seemed unsuitable because it required to increase the height of the RPV. Reactor core was designed with 37 fuel assembly and, unlike IRIS, the use of soluble boron was avoided. Fuel composition was decided to extend the refueling interval to at least 4 years, to make the reactor suitable to operate in isolated areas, although this may go to the detriment of economy.

The IRIS-50 case is interesting, since it shows how the scaling operations, which are subjected to the proposed use of the reactor, provide not only limitations, but also some advantages. However, the case of a submerged reactor presents constraints and peculiarities that are considerably different. Requirements for the design of a submerged SMR regard the power output, the total height of the RPV and the adaptability to a fully passive safety strategy. The new proposal is named IRIS-160. The primary components have been revisited in order to reduce the thermal power output from 1,000 to 500 MW_{th} and the reactor height from 22 to less than 14 m. The analysis regards the reactor core, the control rods driving mechanism, the steam generator, the primary pumps and the pressurizer.

2.3.2 Reactor Core

The revisited reactor core design considers a standard PWR fuel assembly as adopted in IRIS. The fuel assembly is made with 264 fuel rods in a 17×17 square array. The central position is reserved for in-core instrumentation, and 24 positions have guide thimbles for control rods. As in IRIS, 89 assemblies are positioned in the core, as in Figure 12a. The resulting diameter is around 2.75 m. To reduce the power output, the active height of IRIS-160 fuel elements has been scaled down: it must be roughly halved with respect to the 4.20 m fuel assembly height adopted in IRIS. Therefore, an element with active height around 2.0-2.5 m should be designed. Among the current or proposed offer of fuel elements, some products seem to be suitable for the purpose, e.g. Framatome, Lo-Lopar, M-Power and Westinghouse SMR, all with active height around 2.5 m (NEI, September 2014). The active height of NuScale and Smart reactors is 2 m and CAREM-25 adopts 1.4 m height elements (IAEA, 2016-b), although these reactors are designed with power output smaller than IRIS-160. In principle, a 2-m value for the fuel assembly active height can be reasonably assumed. Considering gas plenum and core support plates, the overall height of IRIS-160 core would be in the range of 3.00-3.20 m. The solution seems acceptable and can rely on technologies already available in industry, albeit neutronic verification must be performed to assess the viability of such a core. Unlike IRIS, an underwater reactor cannot use soluble-boron to control reactivity, since recycling systems of borated water are voluminous and requires frequent maintenance. This aspect still need to be addressed for the IRIS-160, but in literature the work by Ingremeau and Cordiez (Ingremeau & Cordiez, 2015) faces the problem for Flexblue. The authors designed a boron-free core and also performed an economic optimization.



(a) (b)

Figure 12 Core configuration and isometric view of IRIS-160 core

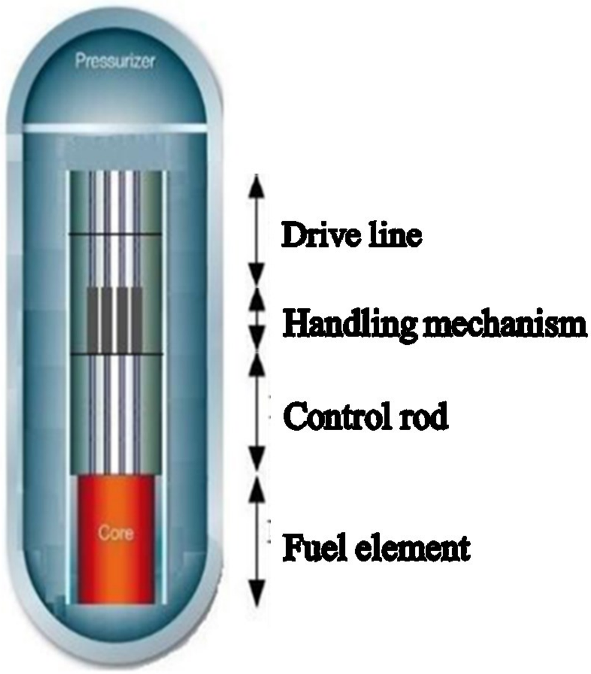


Figure 13 General layout of IRIS-160 control rods driving mechanism

2.3.3 Control rods driving mechanism

In the IRIS design, the Control Rods Driving Mechanism (CRDM) was placed above the core and actuated with electromagnetic or hydraulic mechanism. For IRIS-160, a similar approach is maintained. In a preliminary estimation, the height of the CRDM is twice the total length of the fuel assembly, to host the withdrawn control rods and the drive line, plus the height of the handling mechanism. The overall height can be estimated between 5.5 and 6.0 m. Figure 13 presents a global layout of the CRDM.

2.3.4 Primary pumps

Like in IRIS, for the IRIS-160 the use of axial "spool-type" pumps has been assumed. This technology has been used in marine applications and in chemical plant requiring high flow rates and low developed head. A discussion about the reliability of the spool pump and the advantages brought by its integration in IRIS RPV has been made by Kujawski et al. (Kujawski, et al., 2002). The authors observed that the spool concept allows eliminating large penetrations in the pressure boundary, requiring only a small penetration for power input. Many other advantages on the operational side are achieved by integrating the electric motor with the impeller, over conventional arrangements. Figure 14 shows the pump basic component, as designed by the Electro-Mechanical Division of the Westinghouse Government Services Company. Given the low head required, four units would be enough to provide circulation. Pumps would be placed above the SG modules, in the annulus between the barrel and the RPV. In the 4 SG modules - 8 headers configuration, each pump is positioned between two upper headers (Figure 15). As for IRIS, pumps will stay in hot leg position for better integration in the primary layout. Overall dimensions of pump and diffuser is, approximately, 1.5 m height, 1 m width and 1 m depth. At this preliminary stage, no pump model is chosen.

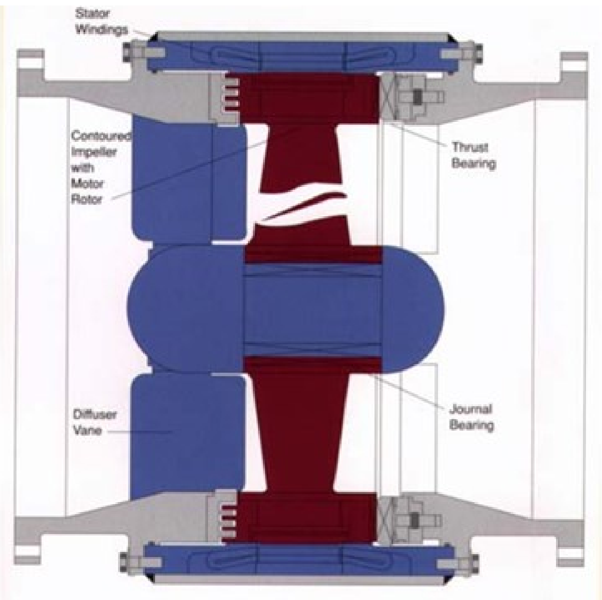


Figure 14 Westinghouse EMD IM/P Basic Components (Kujawski, et al., 2002)

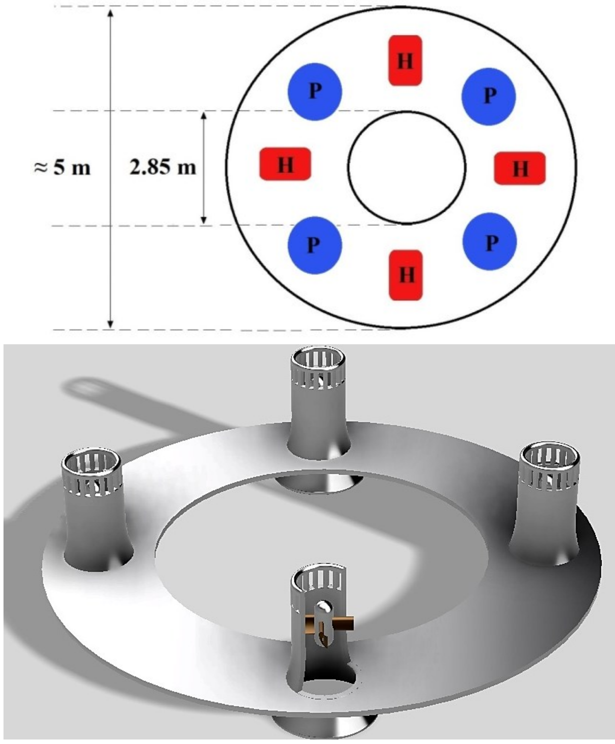
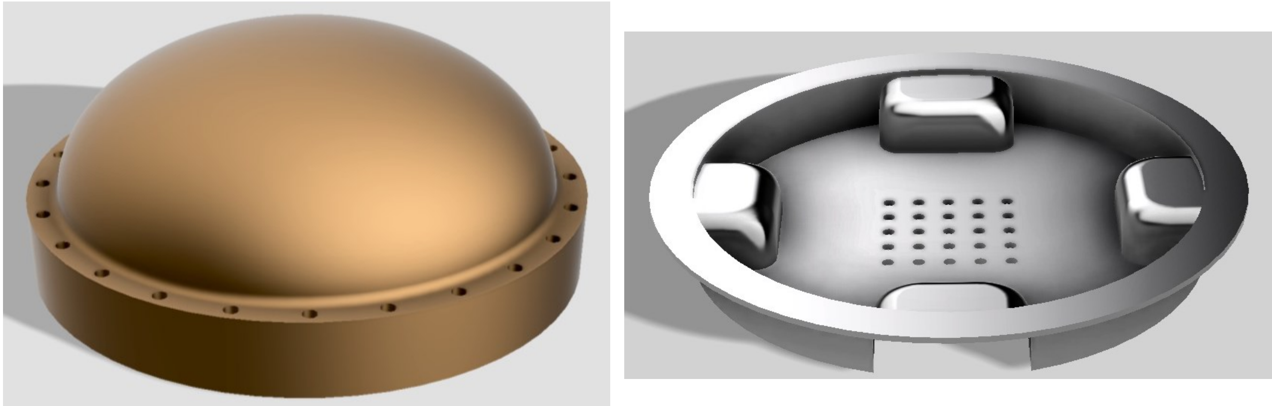


Figure 15 Integration of spool-type pumps in IRIS-160 layout

2.3.5 Pressurizer

The IRIS-160 preliminary pressurizer design has been made keeping the same volume/power ratio of IRIS: basically, IRIS-160 needs half the pressurizer volume of IRIS. Anyway, to reduce the total height of the reactor, the shape of the dome is not spherical, but ellipsoidal (Figure 16a). The pressurizer brim of the inverted top hat can be shaped on pumps to increase the volume of the pressurizer (Figure 16b). With this configuration, the necessary volume for the pressurizer (roughly 40 m^3) can be obtained with an elliptic dome with 4.7 m base diameter and less than 2 m height. Pressurizer volume is still much larger in comparison to PWRs with similar power, thus avoiding the necessity of sprayers. In addition, the integration on the RPV dome may allow air convective cooling on the external surface to reduce internal pressure.



(a) (b)

Figure 16 Pressurizer ellipsoidal dome (a) and lower plate (b)

2.3.6 Steam generator sizing

The Steam Generator (SG) design for IRIS-160 has undergone large modifications with respect to the IRIS original design. The solution with eight helical coil SG modules placed around the barrel and with a 1.5-meter diameter is not feasible for IRIS-160. Due to the reduction of reactor power output, for economic reasons it is desirable to reduce the vessel diameter as much as possible. Therefore, a layout with two or four helical SG modules co-axial to the barrel is proposed. This solution is similar to NuScale SG (IAEA, 2016): a NuScale power unit adopts two once-through helical-coil modules located in the annular space between the hot leg riser and the RPV inside diameter wall. Preheated water enters in bottom SG headers and superheated steam is collected at the outlet from to headers.

To determine a SG configuration to be used for preliminary calculations, two constraints have been imposed: a restriction on the length of each helical tube of 32 m (due to manufacturing reasons, same for IRIS) and the SG module height, limited to 4 m since there must be room for headers and pumps within the limit of the CRDMs clearance. For a preliminary design, the same tube diameter and pitches adopted in IRIS have been maintained. Given that primary and secondary pressures would be similar to those of IRIS, also the same tube thickness has been considered. Main geometrical parameters are given in Table VI. The resulting SG module outer diameter as a function of the number of tube rows is given in Figure 17. For each number of rows and depending on the number of modules, the optimized average length of tubes and the available number of tubes has been calculated. These results are shown in Figure 18.

The preliminary sizing has been performed with a 0D model based on a Lumped Parameter Approach (LPA), equivalent to the method used for SCOR-F in paragraph 2.1.3. Two configurations, i.e. 2 SG modules (4 headers) and 4 SG modules (8 headers), have been considered. The sizing calculations aim at determining the heat exchange surface, and consequently the number of tubes and rows, needed to transfer the thermal power from the primary to the secondary side. The total power is 500 MW_{th}: it is less than 530 MW_{th} used for SCOR-F sizing because the helical once-through SG produces superheated steam and therefore the efficiency of the secondary cycle is expected to be higher. LPA employs energy balances, Newton's law of cooling and empirical correlations for heat transfer coefficients. Primary fluid flows down across the tube bundle, while steam is produced and superheated inside the helically coiled tubes. Since the thermal power of the scaled version is roughly halved with respect to standard IRIS value, on the primary side two options are possible: the SG inlet specific enthalpy and (a) the SG outlet specific enthalpy, or alternatively (b) the total mass flow rate can be maintained equal to those adopted in IRIS. Consequently, suitable outlet values satisfying the energy balance must be taken for (a) the total mass flow rate or (b) for the outlet specific enthalpy, respectively. The advantage of choice (a) is a very strong reduction of pressure losses on the primary side due to the decrease of the total mass flow rate, while choice (b) represents the best solution to maximize the temperature difference between primary and secondary fluid and therefore to enhance the heat transfer process. Both alternatives have been analyzed. As far as secondary side flow rate is concerned, it has been reduced in order to obtain superheated steam at outlet, with the same specific enthalpy of the standard IRIS. The same primary and secondary operating pressures of IRIS have been considered. Operating conditions adopted for the LPA are shown in Table VII.

Table VI. SG operating conditions

Primary pressure	15.5 MPa
Secondary pressure	6.2 MPa
Primary inlet temperature	328 °C
Secondary inlet temperature	224 °C
Primary mass flowrate	2250 - 4500 kg/s
Secondary mass flowrate	251.5 kg/s

Table VII. SG geometrical parameters

Tube outer diameter	17.46 mm
Tube thickness	2.11 mm
Vertical pitch	23.00 mm
Horizontal pitch	23.85 mm
Barrel outer diameter*	2.85 m

* *corresponding to SG module inner diameter*

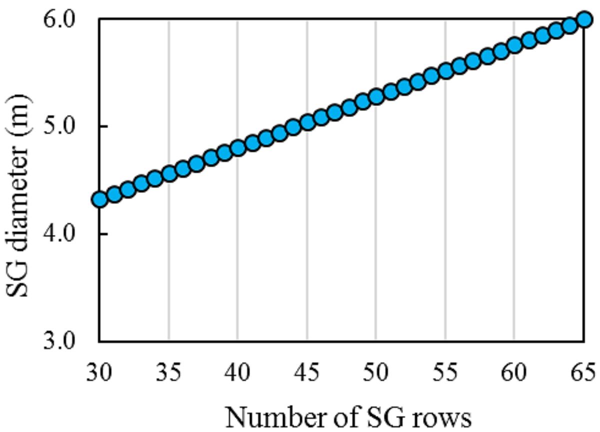


Figure 17 SG diameter given the number of rows

To consider the different heat transfer regimes of secondary side two-phase flow, the tube has been divided into four zones: (1) subcooled liquid, (2) bulk boiling, (3) CHF and post dryout, (4) superheated steam (Figure 19). The heat transfer problem in each zone is solved by means of the electrical analogy, considering primary and secondary convective resistances, plus the conductive resistance for cylindrical geometry accounting for the tube wall. Based on Newton's law of cooling, the system in eq. (3) can be written for each zone:

$$\begin{bmatrix} \dfrac{1}{R_w} & -\left(\dfrac{1}{R_w}+\dfrac{1}{R^{II}}\right) \\ -\left(\dfrac{1}{R_w}+\dfrac{1}{R^{I}}\right) & \dfrac{1}{R_w} \end{bmatrix} \begin{bmatrix} \mathrm{T_w}^{\mathrm{I}} \\ \mathrm{T_w}^{\mathrm{II}} \end{bmatrix} = \begin{bmatrix} -\dfrac{\mathrm{T_b}^{II}}{\mathrm{R}^{II}} \\ -\dfrac{\mathrm{T_b}^{I}}{\mathrm{R}^{I}} \end{bmatrix} \tag{3}$$

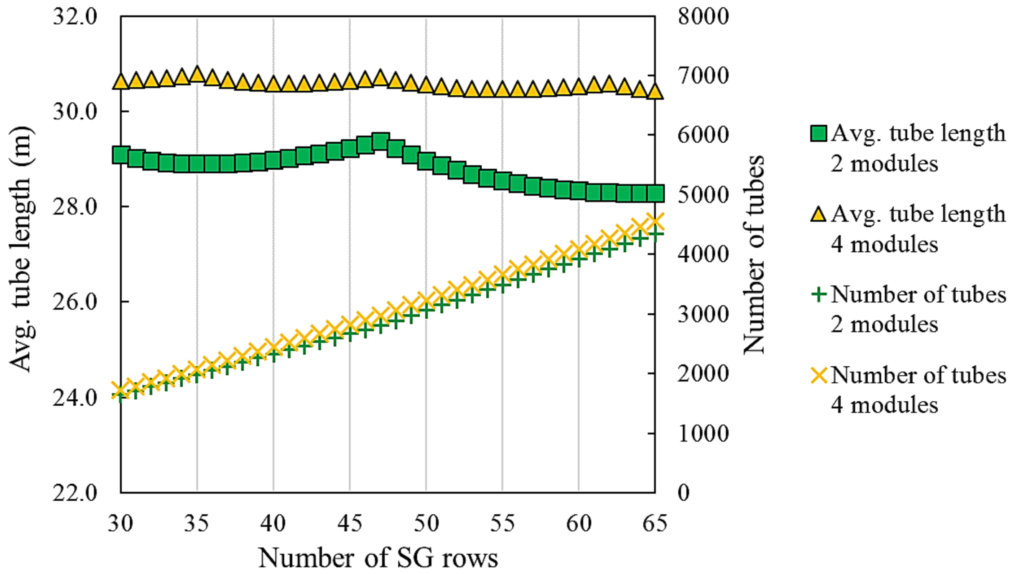


Figure 18 Characterization of 2 modules and 4 modules SG

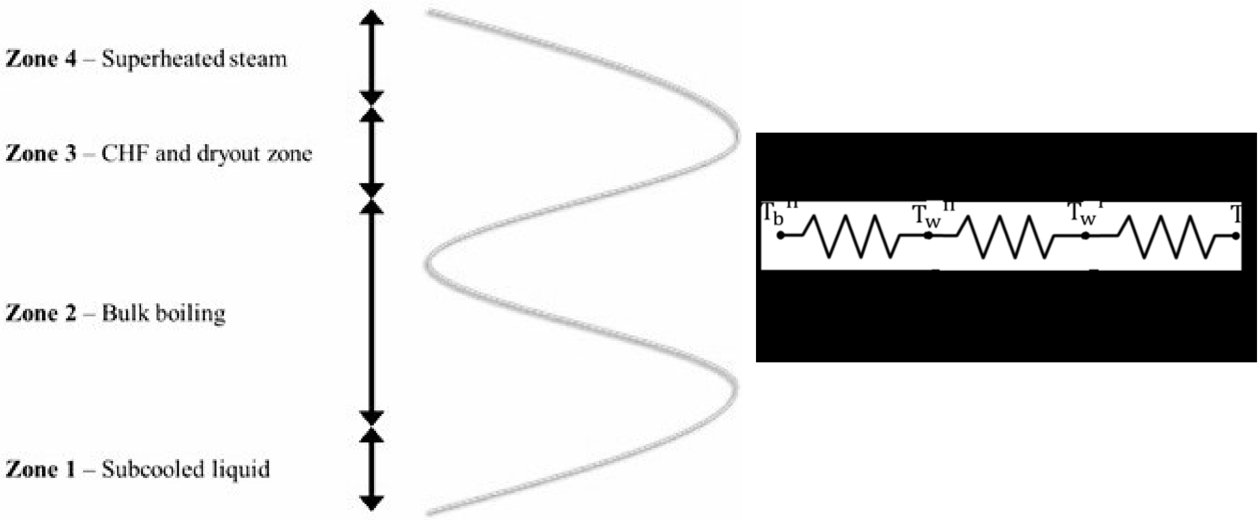


Figure 19 Zone division and electric analogy

Bulk temperatures have been derived from the enthalpy jumps in each zone, which can be obtained from the information on the secondary side provided the quality at dry-out with a specific correlation (saturation minus inlet enthalpy for Zone 1, dryout minus saturation enthalpy for Zone 2 etc. To determine the Heat Transfer Coefficient (HTC) α in the primary side and in the four secondary side zones, suitable empirical correlations have been used, as reported in Table VIII.

Table VIII. Empirical correlations used to determine HTCs

Author(s)	Correlation	Applicability
Žukauskas (Žukauskas, 1987)	$Nu = 0.021 Re_{D,max}^{0.84} Pr^{0.36} \left(\frac{Pr}{Pr_w}\right)^{0.25}$ Number of tubes in flow direction > 20 $Pr = 0.1\ to\ 500$ $Re = 2 \times 10^5\ to\ 2 \times 10^6$	**Primary side** **All zones** Heat transfer coefficient of single-phase fluid in cross flow over a bank of tubes.
ESDU (ESDU, 2001)	$\mathrm{Nu} = \frac{f/2\, \mathrm{Re_D Pr}}{1.07 + 12.7\sqrt{f/2}\left(\mathrm{Pr}^{\frac{2}{3}} - 1\right)} \left[1 + 0.059 \mathrm{Re_D^{0.34}} \left(\frac{D_c}{D_t}\right)^{-0.68}\right] \left(\frac{\mu}{\mu_w}\right)^{0.11}$ $f = (1.58 \ln \mathrm{Re_D} - 3.28)^{-2}$ Reynolds: $1.5 \times 10^3 - 3 \times 10^5$ Prandtl: 0.7 – 11.6 Coil to tube diameter: 4.9 - 271	**Secondary side** **Zone 1** Single phase heat transfer in curved tubes.
Chen (Chen, 1966)	$\alpha_{conv} = \frac{k_f}{d_t} 0.023 \left(\frac{4\dot{m}(1-x)}{\mu_f \pi d_t}\right)^{0.8} \mathrm{Pr_f^{0.4}} F$ $\alpha_{nb} = 0.00122 \left[\frac{\left(k^{0.79} c_p^{0.45} \rho^{0.49}\right)_f}{\sigma^{0.5} \mu_f^{0.29} h_{fg}^{0.24} \rho_g^{0.24}}\right] \Delta T_{sat}^{0.24} \Delta p^{0.75} S$ $\alpha_{tot} = \alpha_{conv} + \alpha_{nb}$ $\Delta T_{sat} = T_w - T_{sat} \quad \Delta p = p(T_w) - p(T_{sat})$ $S = \frac{1}{1 + 2.53 \times 10^{-6} \mathrm{Re}^{1.17}}$ $F = 2.35 \left(0.213 + \frac{1}{X_{tt}}\right)^{0.736}$ Pressure: 1.7 – 35.0 MPa Inlet velocity: 0.06 – 4.5 m/s Heat flux up to 2.4 MW/m^2 Quality up to 0.7	**Secondary side** **Zone 2** Heat transfer of two-phase turbulent flow.
Groeneveld (Groeneveld, 1973)	$\mathrm{Nu} = 0.00109 \left\{\frac{4\dot{m}}{\pi D_t \mu_g} \frac{x}{\varepsilon_{hom}} \left[x + \frac{\rho_g}{\rho_f}(1-x)\right]\right\}^{0.989} \mathrm{Pr_g^{1.41}} Y$ $Y = \left[1 - 0.1 \left(\frac{\rho_f - \rho_g}{\rho_g}\right)^{0.4} (1-x)^{0.4}\right]^{-1.15}$ Diameter: 0.0025 – 0.025 m Pressure: 6.8 – 21.5 MPa Mass flux: 700 – 5300 kg/m^{2}s Heat flux: 0.12 – 2.1 MW/m^2 Quality: 0.1 – 0.9	**Secondary side** **Zone 3** Heat transfer in post dry-out.

Hadaller and Banerjee (Hadhaller & Banerjee, 1969)	$Nu = 0.0101 Re^{0.8744} Pr^{0.6112} \left(\frac{L}{D_t}\right)^{-0.0328}$ Reynolds: $10^4 - 6 \times 10^5$ Temperature: 295 – 590 °C Pressure: 2.0 – 21.4 MPa	**Secondary side Zone 4** Heat transfer to superheated steam.

Zukauskas correlation regards a tube bundle and required some considerations about the calculation of cross areas and flow velocities. $Re_{D,max}$ is the Reynolds number in the zone where the flow velocity is maximum, i.e., the velocity where the cross area is minimum. This quantity can be derived from a mass balance:

$$\dot{m}^I = \rho v A_{tot} = \rho v_{max} A_{min} \tag{4}$$

v is the undisturbed velocity which means the flow velocity if there were no tubes in the space between the barrel and the vessel (A_{tot}). A_{min} is the total area minus the cross area occupied by tubes.

$$A_{tot} = \frac{\pi}{4}\left(d_{vessel}^2 - d_{barrel}^2\right) \approx N_r d_{c\_av} p_h \tag{5}$$

$$\begin{aligned} A_{min} &= A_{tot} - \sum_{j=1}^{n°\text{of rows}} \frac{\pi}{4}\left(\left(d_{c_j} + 2d_{t_{ext}}\right)^2 - {d_{c_j}}^2\right) \\ &= \sum_{j=1}^{n°\text{of rows}+1} \frac{\pi}{4}\left({d_{c_j}}^2 - \left(d_{c_j} - 2\left(p_h - d_{t_{ext}}\right)\right)^2\right) \approx N_r d_{c\_av}\left(p_h - d_{t_{ext}}\right) \end{aligned} \tag{6}$$

Assuming that the annular areas can be approximated with rectangular areas, given by the product of d_c with the total width of the pitches, as done in eq. (5) and eq (6), the balance in eq. (4) becomes:

$$v_{max} = \frac{p_h}{p_h - d_{t_{ext}}} v \tag{7}$$

Correlations used for zones 2, 3 and 4 are not validated for helical geometry, but these have been used anyway because of the lack of specific correlations in open literature. This is reasonable, considering that the coil curvature is very mild: the average coil diameter to inner tube diameter ratio is very high, i.e., around 300, thus softening the effects of centrifugal force. To determine the position of the ending on Zone 2 / beginning of Zone 3, that is to say the beginning of the thermal crisis, Ruffel correlation (Ruffel, 1974) for first quality dryout has been used.

$$x_{dryout-int} = 1 - 0.0053\left(\frac{\rho_g}{\rho_l}\right)\left(\frac{d_c}{d_t}\right)\left(0.001\frac{\dot{m}}{A}\right)^{-1.5}(0.01q'')^n \qquad n = 0.75\left(0.001\frac{\dot{m}}{A}\right)^{0.5} \tag{8}$$

The system equal to that in eq. (3) has been solved for each zone with a Matlab routine. An iterative procedure was applied, because of the presence of non-linear terms. Then, the energy balance in eq. (4) provides the value of the total length of the jth zone:

$$\frac{1}{R_w} L_j \left({T_{w_j}}^I - {T_{w_j}}^{II}\right) = m^{II} \Delta h_j^{II} \tag{9}$$

The term $\dot{m}^{II}$ is the secondary flowrate divided by the total number of tubes, which depends also on the number of rows, and $\Delta h_{j_{II}}$ is the increase of fluid specific enthalpy in the jth zone.

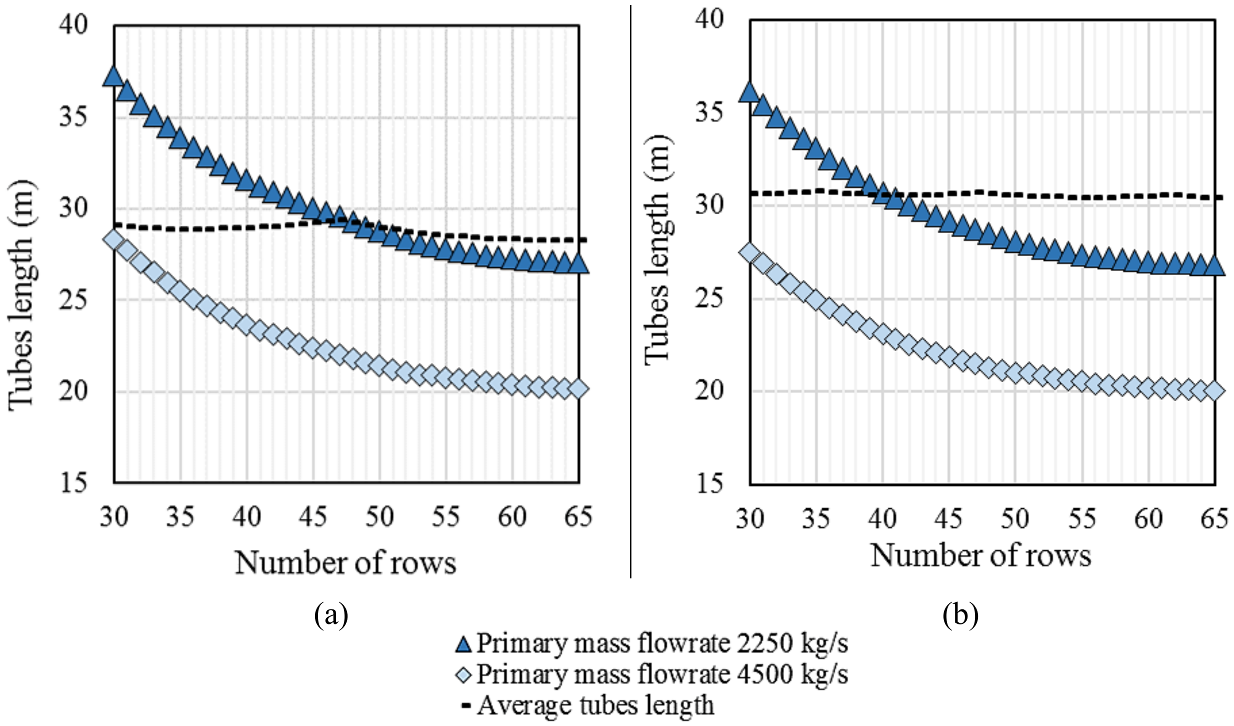


Figure 20 Required tube length necessary to transfer 500 MW_{th} from primary to secondary side for (a) two SG modules / four headers and (b) four SG modules / eight headers

The results for the two cases simulated (two and four SG modules) are reported in Figure 20. The graphics show the length of SG tubes needed to transfer the whole thermal power for number of rows between 30 and 65. The SG is considered feasible if this value is lower than the available average length of the tubes, which slightly varies with the number of rows. LPA calculations reveal that, in principle, the outer diameter of the helical SG module can be lower than 5 m. Increasing the number of rows, the curves seem to tend to a horizontal value: despite the enlargement of the heat transfer surface, with many tubes the secondary flow velocity becomes very low, thus reducing the heat transfer coefficients.

Calculations with total or half primary flowrate, with respect to IRIS operating value, are also shown. Case with half flow rate penalizes too much heat transfer and the necessary average tubes length is very often beyond the constructive limit defined by each configuration. On the other side, case with total flow rate can provide an average tubes length much lower than the limit, but the reduction of the flow areas would determine a large increase of pressure drop. In principle, it will be possible to define a suitable value for the primary flowrate, which optimize the RPV diameter and the head of primary pumps. However, the final value should consider also safety needs. Graphics also put on evidence that configuration with four modules and eight headers permits to employ a lower number of rows, given the same level of flow rate, thus allowing a further reduction of RPV diameter.

2.3.7 Steam generator verification with Relap5

The results of the LPA have been matched with a 1D system code analysis, using Relap5 system code and benchmarking versions Mod3.2 and Mod3.3. The description of Relap5 features is given in Appendix A. A SG with 45 rows and 5 m diameter has been modeled, considering the average tube length of the 4-module configuration. The purpose is to calculate the thermal-hydraulic conditions at the outlet of tube and shell sides and verify if the tested configuration can transfer 500 MW_{th}. The

Relap5 model is similar to the proposal by Cioncolini et al., made for the study of IRIS SG (Cioncolini, et al., 2003). Simulations adopt the same operating and boundary conditions of the LPA calculations. The SG model is made of two rectilinear pipes, characterized by the total flow area and the hydraulic diameter of shell and tube sides (Figure 21). The tube side has the same inclination of the coil. The model does not consider the curvature of the tubes, since Relap5-Mod3.2 and Mod3.3 do not include correlations for helical geometry. Headers at the ends of the tubes are connected with infinite tanks acting as flowrate source/sink. SG pipes are split into 160 elementary volumes. A heat structure couples shell and tube sides, simulating the metallic part of the tube.

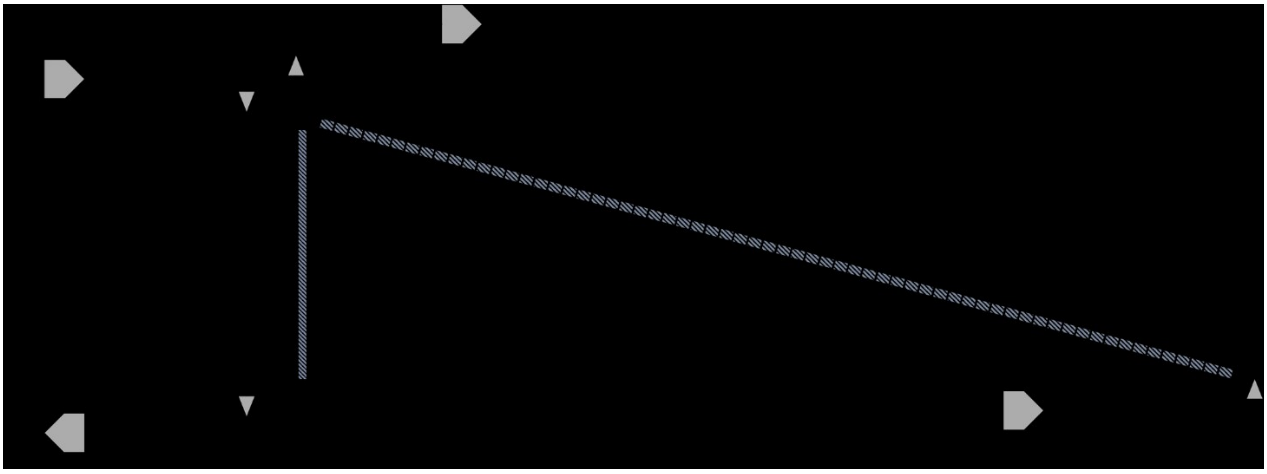


Figure 21 Sketch of Relap5 model simulating IRIS-160 steam generator (not on scale)

The LPA-0D and 1D models use a different perspective: with Relap5 it is required to fix the geometry of the helical pipes and calculate the exchanged power, while in the LPA the total power of 500MW_{th} is used as boundary condition, then obtaining the necessary tube length for the heat transfer. Relap5 results of the total power exchanged through the SG are shown in Table IX. All the estimations are around the desired power output of IRIS-160 steam generator, thus confirming the feasibility of the concept from the operating viewpoint. Comparisons of temperature profiles with the LPA model are given in Figure 22 and Figure 23. In general, a qualitatively good agreement can be appreciated, despite the different lengths of the tubes in the cases with high primary flowrate. The thermal crisis effects are clearly visible on the temperature profiles: there is a sudden increase of wall temperatures corresponding to the drying of the inner tube surface and consequent decrease of the HTC. Remarkable discrepancies between predictions of Mod 3.2 and Mod 3.3 have been found, especially concerning the behavior of temperature profiles and the position of the thermal crisis: these are discussed in Appendix B. However, these effects do not affect much the quantitative results, therefore they do not have consequences on the general conclusions of the analysis.

Table IX. Relap5 results – exchanged thermal power

	Mod3.2	**Mod3.3**
$\dot{\mathbf{m}}^{\mathbf{I}}$ **= 2250 kg/s**	504.23 MW	488.24 MW
$\dot{\mathbf{m}}^{\mathbf{I}}$ **= 4500 kg/s**	508.67 MW	506.42 MW

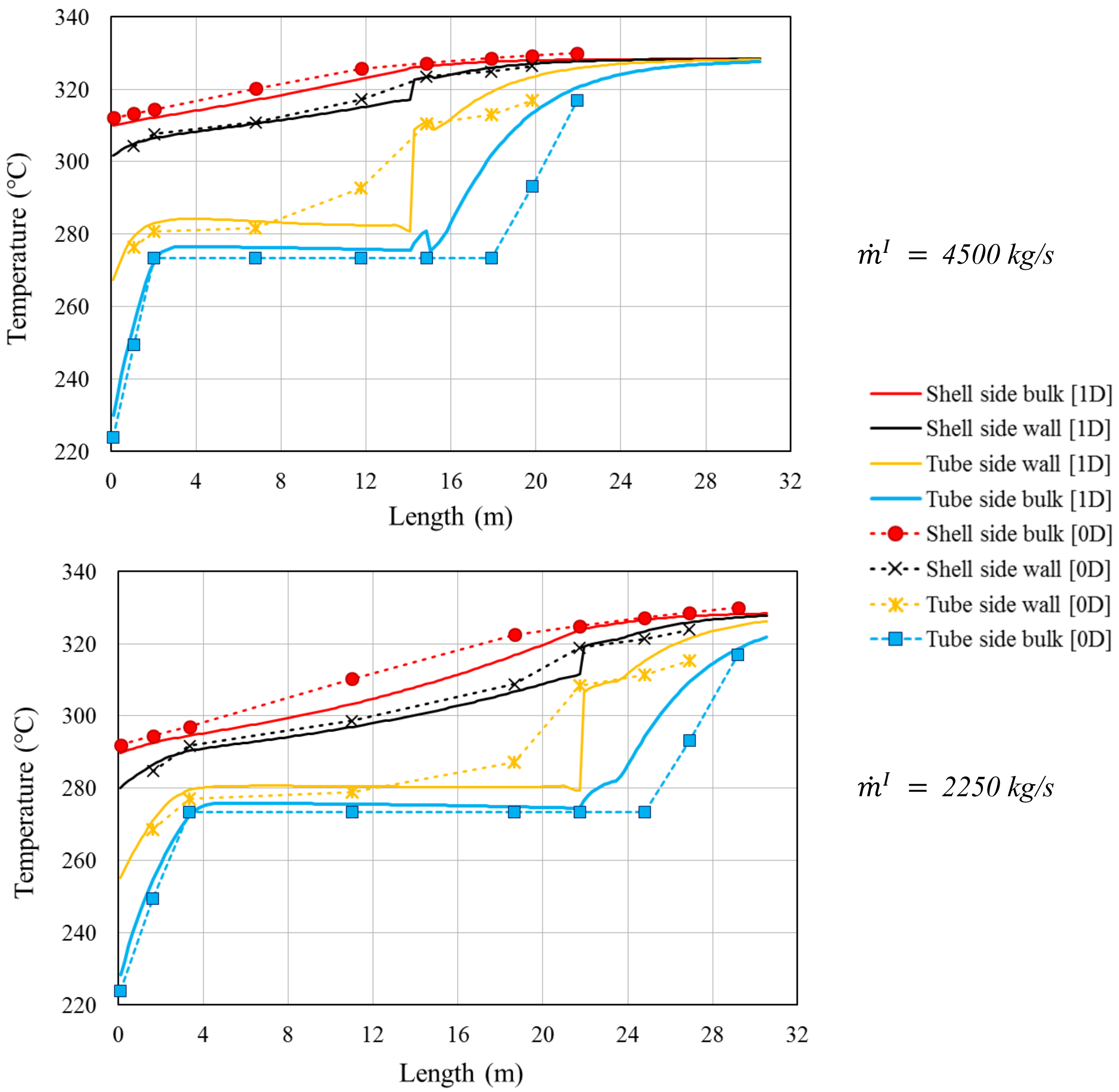


Figure 22 Results of Relap5-Mod3.2 simulations

A parametric analysis about the variation of the total exchanged power as function of the number of the coils has been performed also. Starting from the reference model, coded for a 45-rows configuration, tube and shell flow areas have been varied to represent configurations with different RPV diameter. Primary and secondary flowrates have not been modified. The total exchanged power has been calculated for RPV diameter varying between 4.3 and 6.5 m. Results in Figure 24 show that the enlargement of vessel diameter beyond 5.0 m does not carry benefits to the overall heat transfer. The reason is shown in Figure 25: increasing the shell side flow area causes a substantial reduction of primary Reynolds number and, consequently, a lower primary HTC. Therefore, worse heat transfer performances compensate the larger available surface. Basically, 5 m represents a sort of optimum value, over which the heat transfer has no gain from a further increase of RPV diameter. In conclusion, this preliminary analysis provides reasonable evidence that the layout of the SG can be suitable for the concept of the IRIS-160, with a RPV internal diameter lower than 5 m.

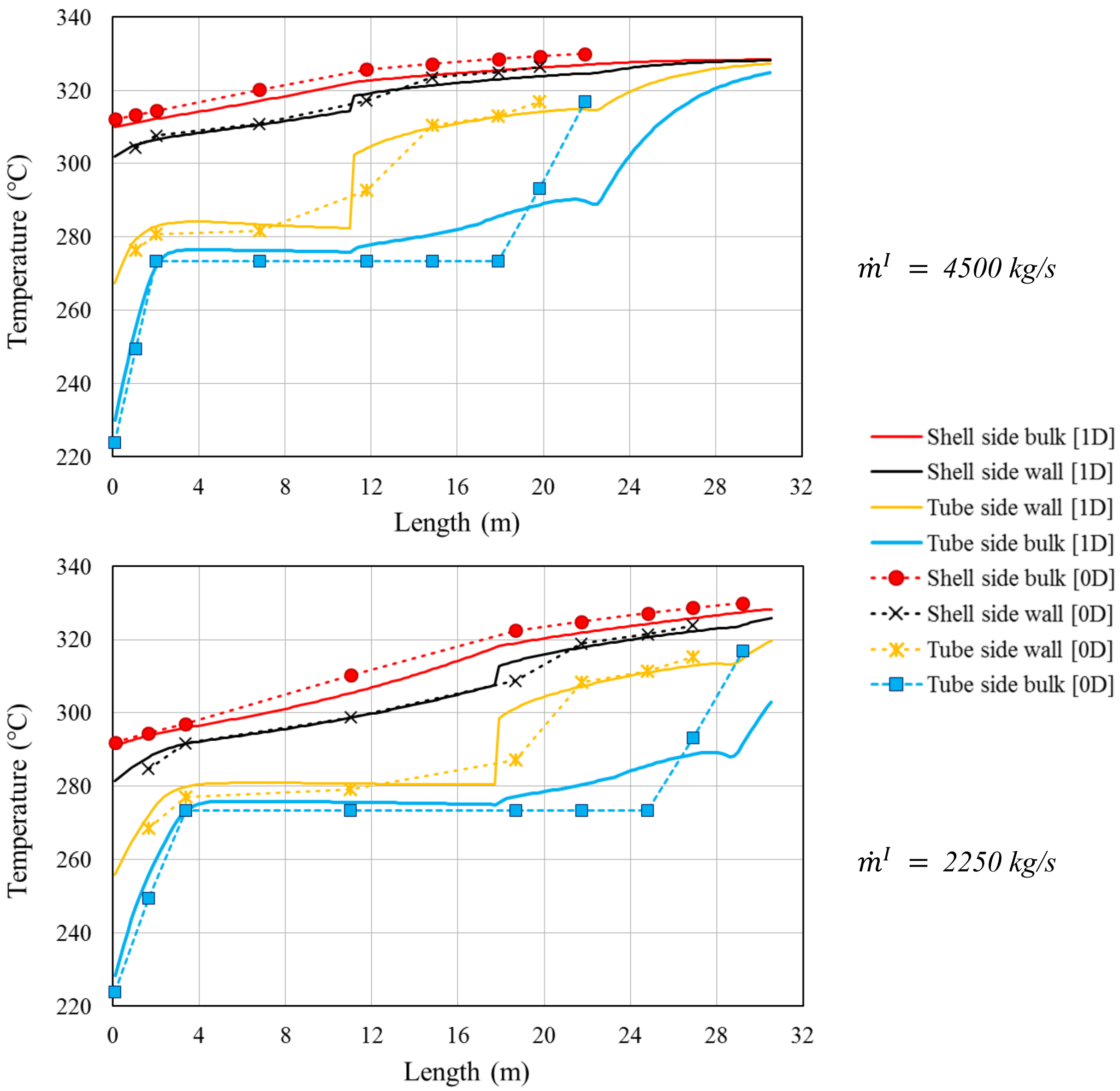


Figure 23 Results of Relap5-Mod3.3 simulations

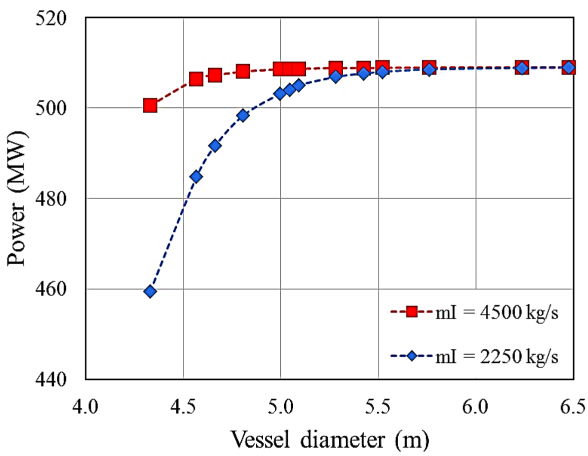


Figure 24 Power transferred through the SG vs. RPV diameter (Relap5-Mod3.2)

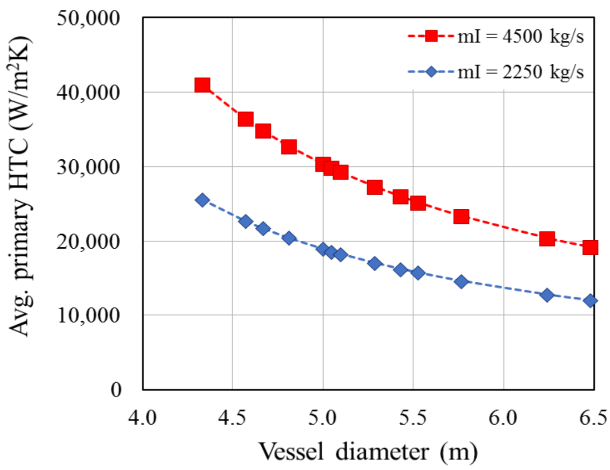
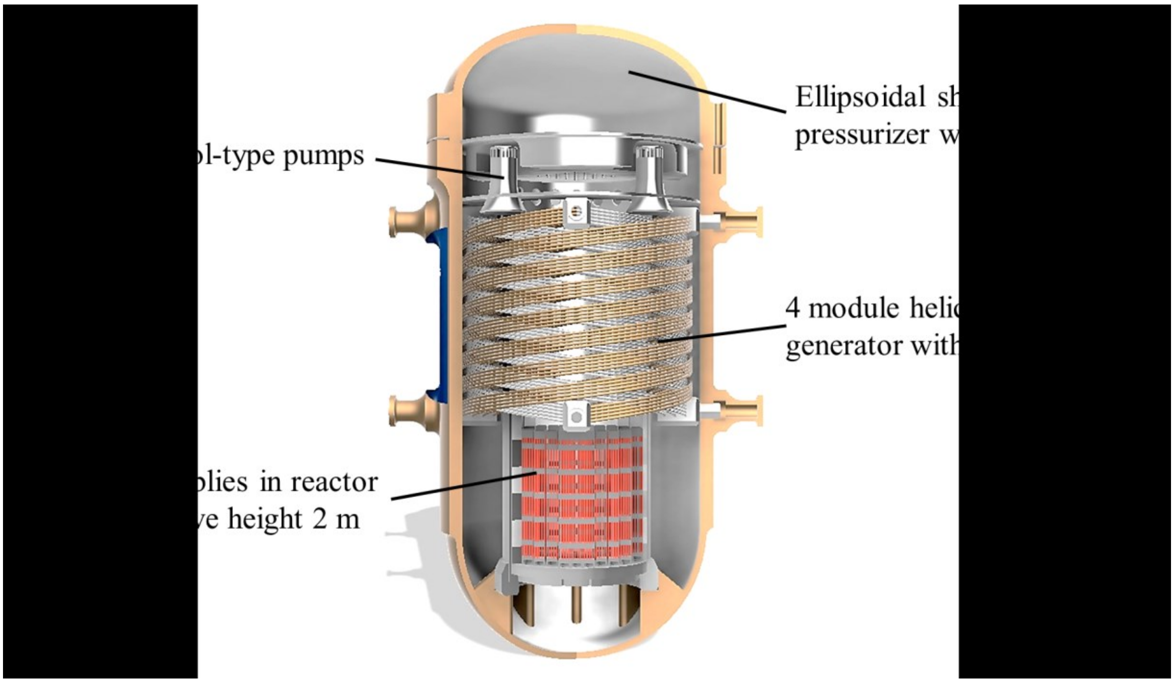


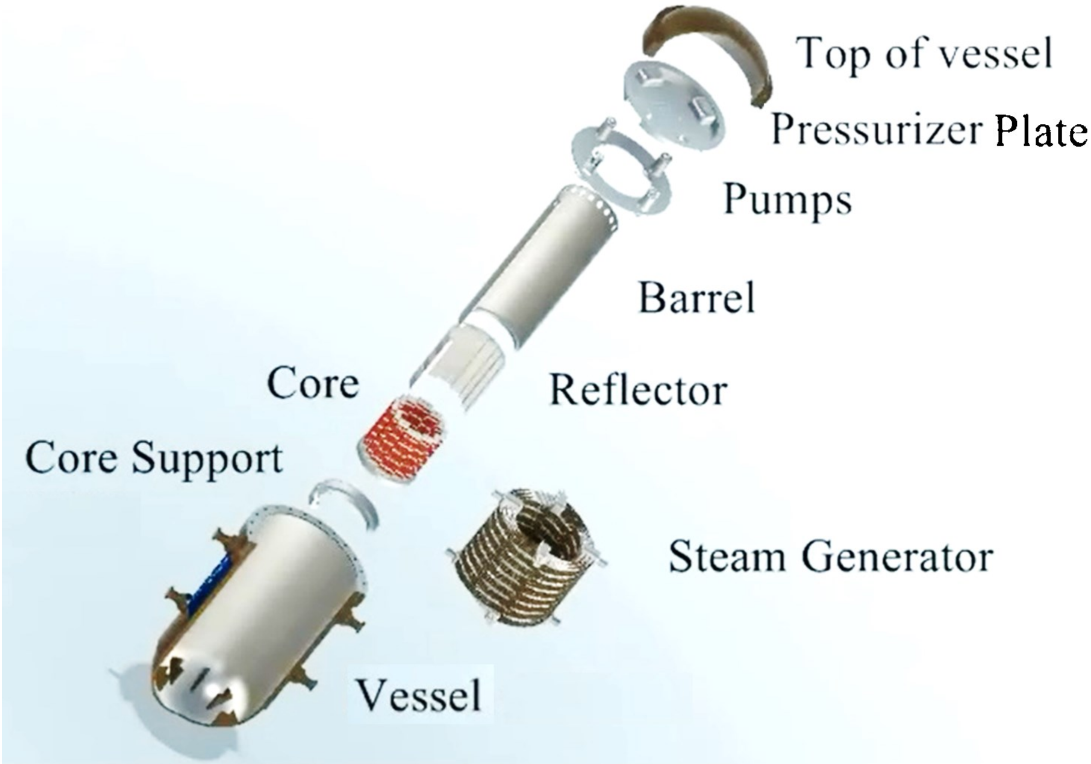
Figure 25 Avg. primary HTC vs. RPV diameter (Relap5-Mod3.2)

2.3.8 Complete layout

The assembly of the primary components is shown in Figure 26. The total height of the integral RPV has been estimated and in principle it seems possible to keep it below 13 m. Similarly, the integral layout has also the potential to keep the RPV diameter below 5 m. Table X shows the details of height and diameter calculations. Final design sizes depend on the definition of operating flowrate and on safety considerations.



(a)



(b)

Figure 26 Assembly (a) and components (b) of IRIS-160 reactor layout

Table X. Summary of estimated lengths and diameters of primary components

<table>
<tr><th colspan="2">RPV Height</th><th>RPV Diameter</th></tr>
<tr><td colspan="2">RPV thickness ≈ 0.15 m</td><td>Core ≈ 2.75 m</td></tr>
<tr><td colspan="2">Lower plenum ≈ 1.20 m</td><td>Barrel ≈ 2.85 m</td></tr>
<tr><td colspan="2">Reactor core ≈ 3.20 m</td><td>Steam Generator ≈ 4.75 m</td></tr>
<tr><td rowspan="2">CRDM ≈ 5.50 – 6.00 m</td><td>Steam Generator ≈ 4.20 m</td><td>Outer diameter ≈ 5.25 m</td></tr>
<tr><td>Pumps ≈ 1.50 m</td><td></td></tr>
<tr><td colspan="2">Pressurizer (including plate) ≈ 2.00 m</td><td></td></tr>
<tr><td colspan="2">RPV thickness ≈ 0.15 m</td><td></td></tr>
<tr><td colspan="2">Total height ≈ 12.40 – 12.70 m</td><td></td></tr>
</table>

2.4. Summary and next steps

A preliminary feasibility study concerning a suitable reactor that could be arranged inside the hull of a Flexblue-like submerged SMR has been performed. The investigation led to two main results.

i) Doubts about the fittingness of SCOR-F configuration seem justified: the calculated overall height exceeds of more than 20% the limit given by the reference diameter of Flexblue hull. In addition, the VVER-type layout is useful to minimize the total height, but the concept is not appropriate for a fully passive safety strategy.
ii) Encouraging results have been obtained from the preliminary design of an IRIS-like integral concept, which could be placed inside the hull with more than one meter of margin. Results of preliminary calculations have also shown that, with a helical SG placed around the barrel, the diameter of the RPV can be lower than 5 m. The IRIS-160 can provide compactness and adaptability to passive safety.

The work here performed represents a preliminary investigation of the integral concept in a submerged SMR, which requires further verifications to prove the feasibility of the IRIS-160 reactor. In particular, the neutronic verification and control strategy of the core are the most critical aspects, which need to be carefully addressed.

Publications

The contents of sub-section 2.3 are published in:

Santinello, M., Ricotti, M.E., 2018. Preliminary analysis of an integral Small Modular Reactor operating in a submerged containment. *Progress in Nuclear Energy,* Volume 107, pp. 90-99

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Section 3

After Fukushima: safety strategy and analysis

Following the Fukushima Daiichi event, nuclear safety directives for new reactor designs are strongly focused on scenarios considering the simultaneous loss of AC power and ultimate heat sink. A safety strategy suitable for emergency decay heat removal in a submerged SMR has been developed. The target is to implement a fully passive safety approach, which does not require electricity and human interventions, exploiting the seawater as a permanent heat sink. The integrity of the primary system has been used as reference criterion to determine two basic scenarios: the decay heat is removed through ex-hull heat exchangers with intact primary circuit and through the metal containment if the system is broken or depressurized. These situations have been simulated with Relap5 code, showing the consistency of the main safety principles. The simulation of a Station Black-Out scenario presents a good response of the emergency heat removal system. The simulation of the sump natural circulation heat transfer through the submerged containment addresses the decay heat removal in the long-term period and provides a verification of the feasibility of the safety concept.

3.1. Reference IRIS safety principles

3.1.1 IRIS safety components

The focus of IRIS concept is to propose important enhancements to the safety of new designs: the reactor layout has been conceived to eliminate by design certain accidents and to introduce innovative safety features that reduce the probability and mitigate the consequences of many other critical scenarios (Carelli, et al., 2004). For instance, the primary fluid never gets out of the RPV during operation, inherently preventing from the large-break LOCA. Compared to the reactor power output, the primary fluid inventory is also larger than in conventional power plants, since the RPV is taller and larger: this feature mitigates the consequences of a small-break LOCA. The emergency cooling relies on passive strategy (Figure 27). Five key systems implement a simplified strategy based on natural circulation. The working principle of these systems has been assessed towards transients and postulated design basis accidents, e.g., small-break LOCA, SG tube rupture, decrease/increase in heat removal from the secondary side, loss of coolant flowrate.

(i) A passive *Emergency Heat Removal System (EHRS)*, made of four independent subsystems, connects SG steam/feed lines to the refueling water storage tank. The system operate in case of loss of secondary system heat removal capability and each single subsystem is sized on the entire decay power.

(ii) Two *full-system pressure emergency boration tanks* provide makeup water by gravity through the Direct Vessel Injection (DVI) lines and represent an alternative shutdown system.
(iii) A small *Automatic Depressurization System (ADS)* from the pressurizer steam space operates in case core coolant level becomes too low, allowing feedwater injection.
(iv) A *containment pressure suppression system*, made of a spherical steel containment, six water tanks and a common tank for non-condensable gas storage, operates when steam is released through the ADS. Steam is vented in the containment and condensed in the pools, separating non-condensable gases. The suppression system limits the peak containment pressure to less than 1.0 MPa even in case of the most limiting blowdown event. In case of failure of the EHRS, direct cooling of the containment outer surface is provided, making this way the ultimate solution for decay heat removal.
(v) A *lower containment volume cavity* housing the RPV is filled with water above the core level, creating a gravity head of water sufficient to provide coolant makeup to the reactor vessel through the DVI lines.

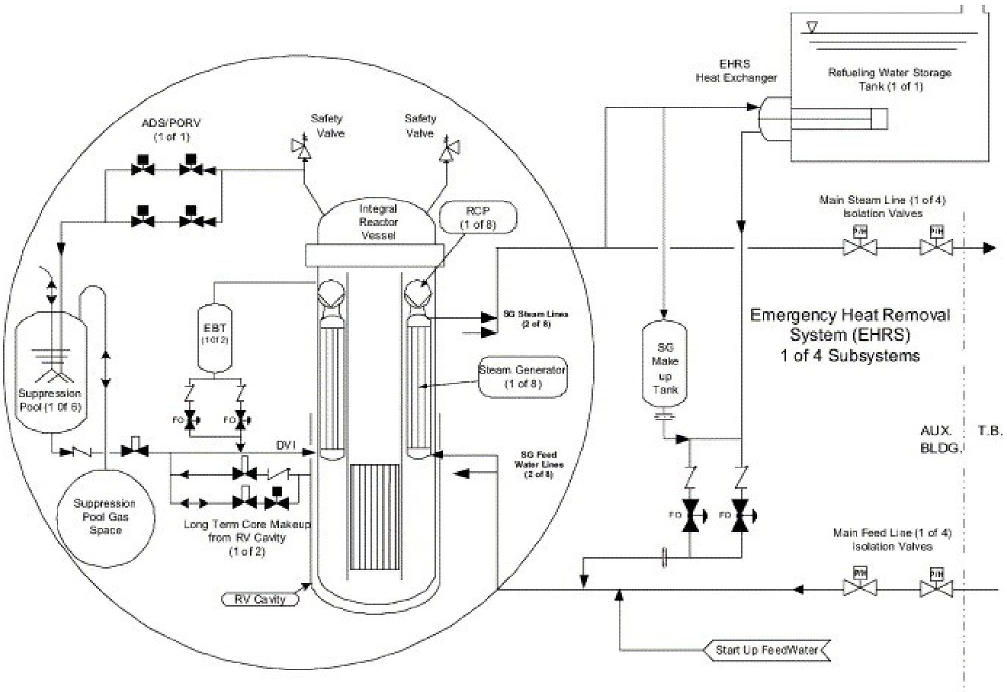


Figure 27 IRIS passive safety system schematic (Carelli, et al., 2004)

3.1.2 IRIS thermal-hydraulic accidental scenarios

Ricotti et al. (Ricotti, et al., 2002) performed a preliminary safety analysis of IRIS, considering three postulated accident scenarios: a Loss Of primary Flow Accident (LOFA), which is basically a station black-out, a small-medium break Loss Of Coolant Accident (LOCA) and a Loss Of Feed Water (LOFW) on the secondary side. LOFA and LOFW regard all the situations in which primary and/or secondary flows stop. Reactor is automatically scrammed and the scenario is faced with the activation of the four-module passive EHRS, which provides natural circulation cooling. The authors simulated 6,000 s long transients with Relap5, showing the good response of the systems. To cool the fuel rods in a LOCA scenario, the RPV is connected with the containment, releasing steam that is condensed into the suppression pool. Reactor cavity is then filled with water by redundant paths, so that the cooling can always rely on gravity driven injection. The features of IRIS containment and the response in case of LOCA have been descripted in detail by Conway et al. (Conway, et al., 2001).

3.2. Basic safety strategy for decay heat removal in IRIS-160

3.2.1 Safety systems

The safety target for emergency decay heat removal operations in a submerged SMR concept is to implement a fully passive safety approach, which does not require AC power or human interventions and can rely on the water surrounding the containment as a permanent and infinite heat sink. The achievement of this goal is fundamental for an underwater reactor, because its peculiar position would make challenging to manage in remote the safety operations in emergency situations. Passive safety would prevent by design from control errors. Beside this necessity, a fully passive strategy would represent a significant breakthrough for the nuclear safety and would practically allow eliminating the Fukushima-like accident scenarios. The research purpose of this section is to provide strong scientific-based arguments aimed at assessing that passive safety systems of a submerged SMR can be well designed and ensure the safe cooling of the fuel rods for an indefinitely long grace period.

A promising set of safety systems refers to: (i) two (or four, to be defined by PSA considerations) trains of Emergency Heat Removal Systems (EHRS); (ii) two trains of in-pool heat exchangers, directly connected to the integral RPV; (iii) the reactor containment (dry-well); (iv) a pressure suppression pool (safety tank), with direct injection lines to the RPV and to the reactor containment.

(i) The *Emergency Heat Removal System* is a two-phase flow natural circulation loops, connecting one in-vessel helical-coil SG module to two ex-hull condensers on the top of the hull, in direct contact with external water. The great advantage of this system is that ex-hull heat exchangers reject the decay heat to a sort of "infinite reservoir": safety operations cannot influence the temperature of the cold sink, which remains constant (Santinello, et al., 2017-a). Furthermore, placing the condenser on the top of the hull maximizes the elevation difference between the inlet of the heat exchanger and the outlet of the steam generator.

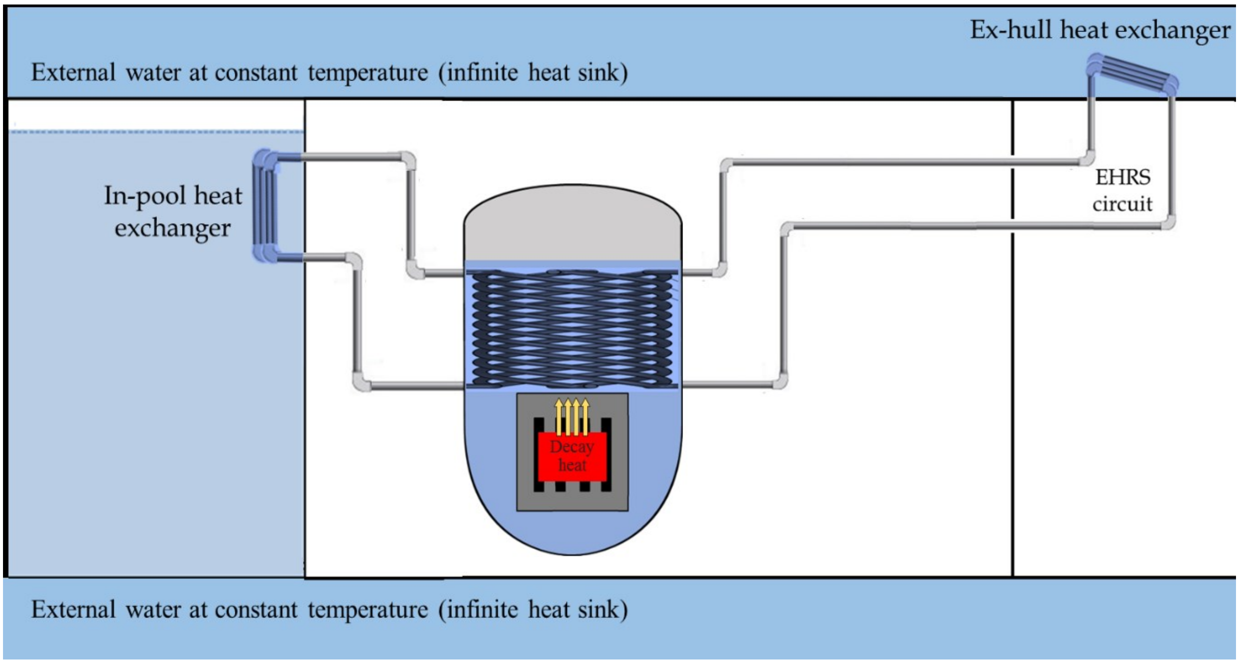
(ii) The *in-pool heat exchangers* are directly connected to the RPV and provide natural circulation cooling through the SG. These operate in parallel with the EHRS. In-hull heat exchangers own the advantage to operate in a protected environment, while ex-hull components are exposed to corrosion and external hazards.

(iii) The *reactor containment* is a dry-well filled with nitrogen or inert gas. Following a tube rupture or Automatic Depressurization System (ADS) valves opening, steam is flashed/released in the dry-well atmosphere. The containment offers steam condensation capability on the metal surface in contact with the external water. The condensate is collected on the bottom of the containment.

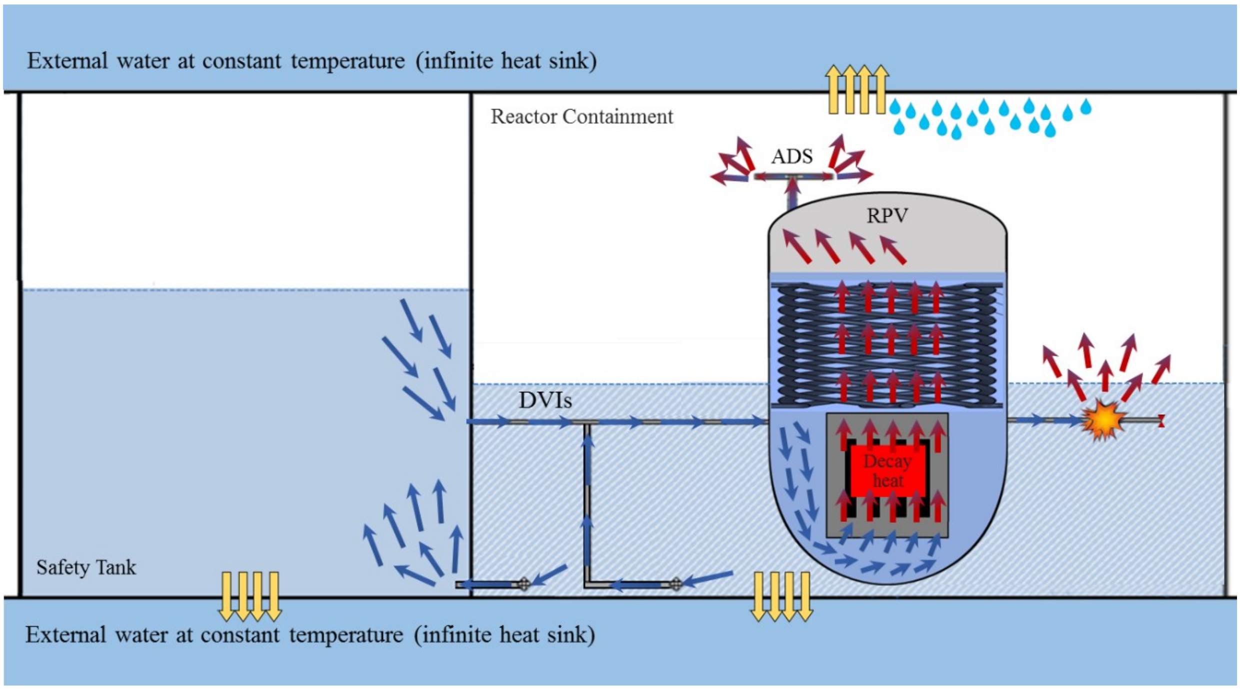
(iv) The *pressure suppression pool* is a large Safety Tank (ST) placed at the side of the reactor compartment. Its primary function is to condense the steam produced by a primary break or induced depressurization and reduce the pressure peak in the reactor compartment. The steam/non-condensable gas mixture is directed into the suppression pool, generating an over-pressure that can be exploited for RPV injection. In addition, it has several other basic purposes: it is an in-hull source of cold water for direct vessel injection, it acts as a heat sink for pool heat exchangers, its water is used to flood the reactor compartment and activate sump natural circulation flow.

3.2.2 Response to design basis scenarios

According to a Fukushima-like scenario, the reference accident related to thermal-hydraulic is only the Station Black-Out (SBO), since the concurrent Loss of Ultimate Heat Sink (LUHS) is assumed as practically impossible. Hence, the basic accident scenario begins with the loss of ordinary active cooling capabilities, automatic reactor scram and actuation of passive safety systems. Then, two reference situations have been selected according to one single criterion: the integrity (or not) of the primary circuit. In case of an "intact primary" (non-LOCA-SBO) accident, the EHRS is the key passive safety system, while for a "non-intact primary" (LOCA-SBO) scenario, the submerged reactor containment plays the main role. The latter represents also a backup strategy in case of failure of other safety systems.



(a)



(b)

Figure 28 Principles of safety strategy for intact (a) and non-intact (b) primary system scenarios (dimensions are not representative).

The safety procedure adopts, in an *"intact primary" (non-LOCA)* scenario: the passive EHRS, to reject the decay heat to the infinite heat sink (sea or lake) and/or the in-pool heat exchangers, to reject the decay heat to the suppression pool. In a *"non-intact primary" (LOCA-like)* scenario, after the immediate emergency injection from high-pressure systems the strategy is: moving of steam and non-condensable gases into the suppression pool and then direct injection lines to the integral RPV (exploiting pool over-pressure), plus flooding of the reactor compartment and condensation on the inner wall of the containment. The two safety strategies are sketched in Figure 28. Thanks to the integral layout, the large break LOCA accident is prevented by design, therefore the "non-intact primary" scenario refers only to small break LOCAs, e.g., in case of rupture of a Direct Vessel Injection line.

3.3. Simulation of a Station Black-Out scenario

3.3.1 Overview

A Station Black-Out event (SBO) occurs in case of complete failure of both off-site and on-site AC power sources. Primary and secondary pumps stop and the removal of residual heat from the fuel rods can be performed by passive systems only. In this scenario, the primary system does not undergo ruptures, loss of coolant and depressurization. A numerical investigation of this scenario has been performed with Relap5-Mod3.3. Salient features of Relap5 code are described in Appendix A. In the simulation case, after the reactor scram decay heat removal is demanded only to the EHRS. The purpose is to analyze the effectiveness of the EHRS and underline the potentiality of the safety concept: the expected outcome is the success/failure of the EHRS to bring the system to a safe state without the intervention of other systems, e.g., depressurization valves, direct vessel injection. Simulations allow observing if in this way it is possible to depressurize the system only by the effect of the cold heat sink at constant temperature, to ensure sufficient core cooling by natural convection and to evacuate the decay heat to the exterior.

3.3.2 Model

To study the SBO problem with Relap5, the system under investigation must be represented with a one-dimensional logic. The 1D model, set up also by exploiting the experience of a previous work (Ricotti, et al., 2002), consists of: (i) the primary circuit, which includes the *core*, the *pressurizer*, the *primary side of the SG* and *other minor components*; (ii) the secondary circuit, which includes the *secondary side of the SG*, the *EHRS* exchanging directly with seawater, a *water tank* and *connecting piping*. For the geometric characterization of the primary circuit, the IRIS-160 design has been adopted. No detailed design exists about the ex-hull heat exchanger. Therefore, a preliminary sizing has been made, assuming that this component should be capable to transfer the decay power five seconds after the scram, i.e., roughly 25 MW_{th}. A straight tube bundle placed on the external upper part of the hull has been considered, to take advantage of the highest possible placement. Pipes are slightly inclined to better integrate with the cylindrical shape of the hull. To maintain symmetry and increase redundancy, in this rough sizing the tube bundle is split into two identical modules, placed symmetrically on the two descending sides of the top of the hull, but a four-module design would be better from a safety perspective. For a conservative estimation, the sizing has been done considering liquid to liquid heat transfer and Churchill & Chu correlation (Churchill & Chu, 1975) has been used to calculate the external HTC.

An overview of the nodalization is given Figure 29 and Table XI reports geometric parameters.

The *core* component has been modeled as a single pipe subdivided into 12 elementary volumes, with hydraulic equivalent diameter of a single channel and flow area equivalent to the total core flow area. A heat structure, which act as a power source, is linked to the pipe and is able to simulate the thermal resistance of the UO_2 pellet and the Zircaloy cladding. The *pressurizer* has been modeled as a single volume, initially filled with water at quality 0.255 (initial value). A source/sink volume with fixed pressure has been connected to the top to simulate the control on pressure when the system is working in operating conditions. Two volumes, a separator component and a single volume, simulate the *upper plenum* under the plate of the pressurizer. The upper plenum is the linking element among the riser, the SG and the pressurizer. The separator component implemented in Relap5 is useful to help directing liquid and steam into the SG and pressurizer, respectively, and avoiding unphysical where the liquid rises and the steam descends. The separator is connected to the pressurizer and to a bypass volume, which models the annular space around the upper plenum under the pumps and flows into the SG. The *steam generator* has been modeled as described in paragraph 2.3.7. Other primary components are: the *lower plenum*, a volume situated under the core; the *riser*, which occupies the internal space of the barrel and it connect the upper part of the core to the upper plenum; the *downcomer*, placed under the steam generator and with ideal annular shape; *primary pumps*, modeled with a flowrate source and isolated at the beginning of the SBO.

The *EHRS* model include the SG tube side, the ex-hull heat exchanger and connecting piping. The *ex-hull heat exchanger* works as a condenser in the first phases of a SBO scenario, since there is steam production in the secondary. It is modeled with a single pipe owing the total flow area. A heat structure simulates the heat transfer to the exterior. A *water tank* has been added to the EHRS circuit to increase the total inventory cold water. The water tank is a volume of 50 m^3 and help controlling the pressure decrease. *Connecting piping* is quite long, since the ex-hull heat exchanger position must be shifted in order to avoid hull penetrations on the reactor compartment. *Trip valves* isolate the turbine section and enable the operation of the passive EHRS after the loss of secondary forced flow.

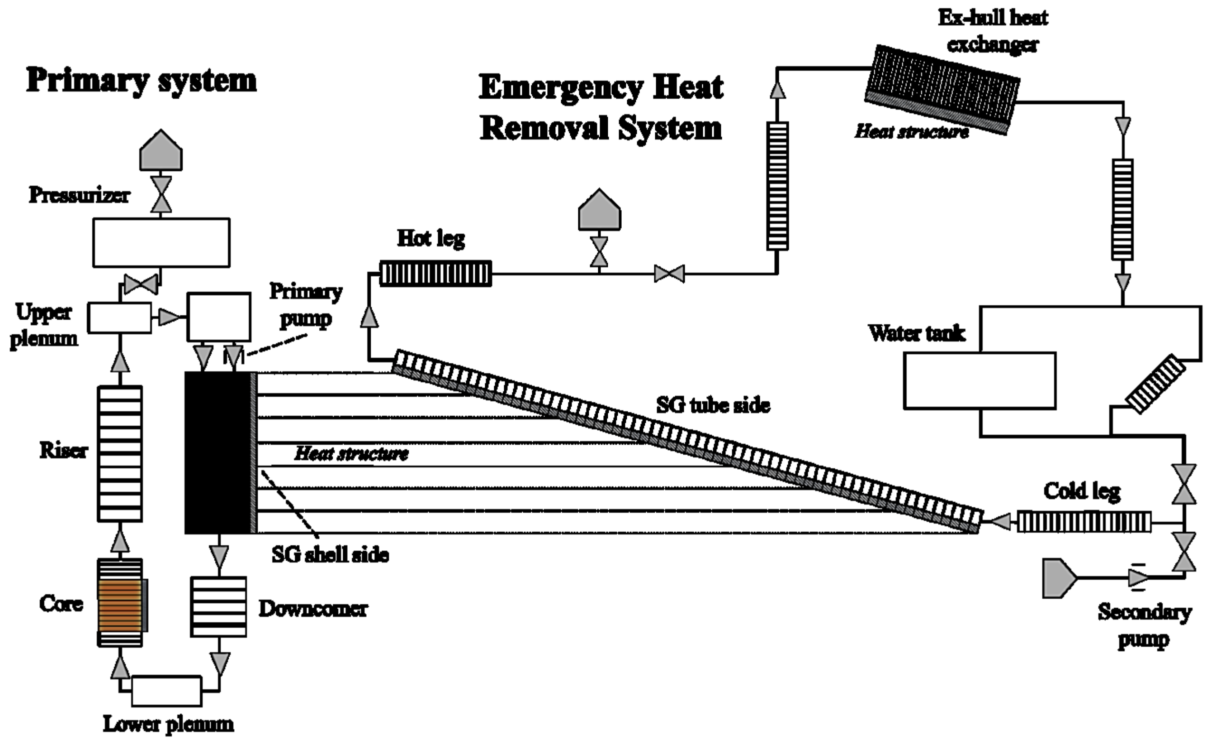


Figure 29 Complete Relap5 nodalization of the primary system + EHRS

Table XI. Geometry and Nodalization of SBO model main components

	Component	Geometry		Nodalization
Primary circuit	*Reactor core*	Active length Total length Hydraulic diameter Flow area	2.0 m 3.1 m 16.1 mm 2.462 m^2	10 elementary volumes for active zone, 4 and 3 for inferior and superior zones, respectively
	Pressurizer	Total height Volume	3.0 m 42 m^3	Single volume + time dependent volume
	Upper plenum	Total height Volume	0.4 m 6.0 m^3	Separator + single volume
	Steam generator shell side	Flow area Height Hydraulic diameter	8.43 m^2 4 m 22.03 mm	160 elementary volumes + heat structure
	Riser / Downcomer	Height riser Diameter riser Height downcomer Hyd. diameter downc.	4.2 m 2.7 m 3.3 m 2.194 m	10 elementary volumes for riser, 5 for downcomer
EHRS	*Steam generator tube side*	Flow area Tube length Tube diameter Tube thickness Tube inclination	0.35 m^2 30.6 m 12.53 mm 2.465 mm 6.5°	160 elementary volumes + heat structure
	Ex-hull condenser	Flow area Tube length Tube diameter Tube thickness Tube inclination Heated surface	0.147 m^2 3.0 m 20.4 mm 2.5 mm 23° 106.84 m^2	60 elementary volumes + heat structure
	Water tank	Volume	50 m^3	Single volume
	Connecting piping	Hot leg length Cold leg length Pipe diameter	16.17 m 20.04 m 0.3 m	33 elementary volumes for hot and cold legs

3.3.3 Boundary and initial conditions

In order to avoid simulating the external seawater, the strategy adopted has been to calculate the external HTC as a function of the pipe surface temperature, imposing a convective condition on the external side of the heat exchanger. Undisturbed water temperature has been assumed at 20°C and 0.6 MPa outer pressure has been considered, with salinity at 35 ppm (corresponding saturation temperature at 159°C). External HTC depends on fluid thermodynamic conditions. The HTC profile has been subdivided into three zones, according to the Nukiyama curve:

1) the single-phase zone, i.e. $T_{tube} - T_{sat} < 5$;

2) the nucleate-boiling zone, i.e. from $T_{tube} - T_{sat} > 5$ to thermal crisis point determined with Zuber correlation for Critical Heat Flux (CHF) (Zuber, 1958);
3) the transition-boiling zone.

For zone 1, HTC is determined with equation proposed by Churchill and Chu (Churchill & Chu, 1975) for external natural circulation from a horizontal pipe, which is very simple and widely employed and validated. Even though this correlation is specific for a single-tube and not for a tube bundle, its use is justified by conservativeness. Churchill and Chu correlation can be used also in zone 3, assuming a gaseous single-phase. For zone 2, an equation recommended by Palen (Palen, et al., 1972) for nucleate boiling convection around a horizontal tube bundle has been used. The resulting HTC profile of the three zones, as a function of the temperature difference between the outer surface of the tube and undisturbed water, is shown in Figure 30.

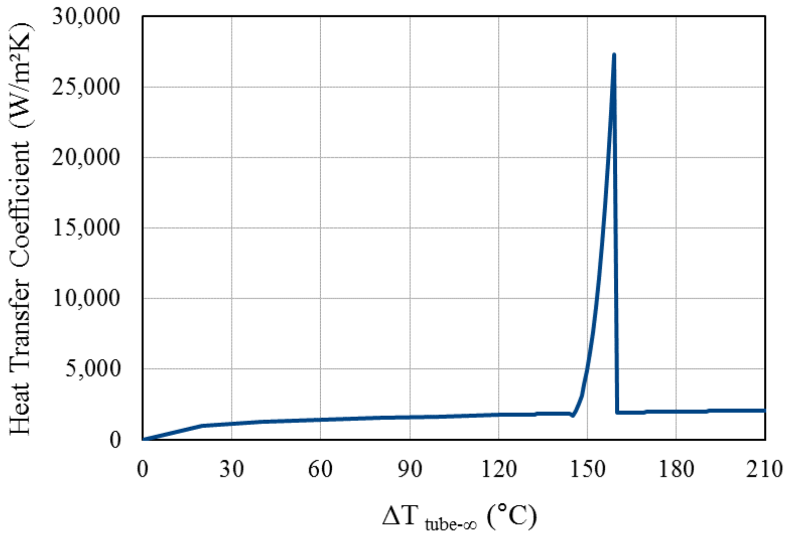


Figure 30 HTC profile calculated as a function of $\Delta T_{t-\infty}$ for the ex-hull heat exchanger

The power source in fuel rods has been modeled imposing a volumetric heat source in the heat structure coupled to the flow channel. The heat structure represents the solid part of the core. The power source has a total initial value of 500 MW_{th} and it has a cosine-shape axial distribution. The first part of the transient is aimed at reaching the nominal operating state, so that the SBO event can start from a realistic situation. After 1500 s of nominal operation, SBO occurs: pumps stop, pressurizer control is disabled, and some trip valves isolate the turbine sector and connect the SG secondary side to the EHRS. Reactor is scrammed, and power generation follows the decay curve. Initial conditions in primary circuit and in SG tubes are taken from nominal values. The EHRS does not operate until the scram, therefore at the beginning of the simulation the setting of a cold and pressurized state has been used. Table XII reports the list of all the initial conditions.

Table XII. Initial conditions for SBO simulation

	Primary	SG tubes	EHRS
Temperature	326°C	278°C outlet 224°C intlet	20°C
Pressure	15.5 MPa	6.2 MPa	6.2 MPa
Mass flow rate	2250 kg/s	251.47 kg/s	--

3.3.4 Transient and numerical approach

The transient study investigates how the system reacts and evolves to the SBO event from an operational full-power state. The simulation never reaches a steady state, because the decay power has an exponential decrease. The investigation is focused on the first 5 hours (18,000 s) from the start of the SBO. The scramming event has been simulated imposing a sudden power drop from a situation of constant full power. From this moment on, the core power decreases following the typical exponential law. At the SBO event, primary and secondary forced flowrates start decreasing and become zero within 60 seconds. Then, only natural convection driven flows can cool the core. A set of auxiliary valves isolate the pumps (flow generators in the model), the pressurizer and the steam /feed line from their respective loop. Other valves connect the secondary loop to the EHRS. Valves are actuated with 6 second delay from the scram.

In Relap5.Mod3.3 (Information Systems Laboratories, Inc., 2003), the 1D fluid equations solution is achieved with a finite difference semi-implicit algorithm. Pressure gradient is evaluated implicitly. Averaged hydraulic properties are evaluated in the center of the cell, which is the control volume for mass and energy quantities, except for velocities, which are evaluated in junctions, namely momentum quantities control volumes. To check the acceptability of the solution at each time step, Relap5 include several criteria, e.g., Courant limit checks, mass error checks, and material properties out of defined ranges, water property errors, excessive extrapolation of state properties in the metastable regimes. Heat structures permit the calculation of heat transfer across solid boundaries of hydraulic volumes. Heat flow path are modeled in a one-dimensional sense, using a finite difference mesh to calculate temperatures and heat flux vectors. In the present case, coupling of heat transfer and hydro-dynamics is made implicitly. Quantities correlated to transverse gradients, such as friction and convective heat transfer, are not directly solved from equations, but calculated empirically. Relap5 owns several flow maps and heat transfer modes for this purpose. Further details are given in Appendix A.

3.3.5 Results

Relap5-Mod3.3 simulation predicts that the nominal configuration can remove the decay power from the core. Figure 31 compares the decay power with the power exchanged in the SG and the power dissipated through the ex-hull heat exchanger. At the beginning of the transient, the EHRS must reject not only the energy coming from the natural decay process in the fuel rods, but also the heat stored in the coolants and in the metallic structures. This is more evident after 6000s, where the ex-hull heat exchangers dissipate more power than the heat transfer rate in the SG, while before this moment the situation is the opposite. This means that until 6000 s after the scram, the SG is cooling the primary circuit, but the EHRS is not capable to transfer all the power to the ocean: the result is an accumulation of energy in the secondary circuit. This effect generates some high vapor production in the SG, but it seems not to pose any critical issue. After a sufficiently long time, a substantial overlapping of the three curves is expected. While they indefinitely continue to decrease, the system reaches an equilibrium between the power generated and dissipated in the external heat sink.

The pressure curves shown in Figure 32 have a strongly decreasing trend. At the end of the simulation time, the pressure decreases to very low values. This is probably due to the effect of the very cold heat sink and, if verified, it could allow avoiding the need to actuate an Automatic Depressurization System (ADS) for this type of scenario. ADS would anyway operate in case of failure of the EHRS. A similar pressure decrease was observed by De Rosa et al. (De Rosa, et al., 2014) in a simulation of a SBO accident in the SPES3 facility, using Relap5 and TRACE system codes, although in that case the heat sink is not kept at constant temperature. Secondary pressure profile shows a spike just after the beginning of the SBO and pressure suddenly rises to 10.0 MPa. This behavior is likely to be caused by the loss of secondary forced flow and by the delay between the stop of mass flow and the bypass of the turbine section and activation of the EHRS: the secondary coolant velocity in the SG drops, generating pressure and temperature increase (Figure 33). The pressure spike does not pose problem in terms of mechanical stress of the secondary circuit, since in an IRIS-like integral design the secondary side is dimensioned to bear the primary pressure in case of SG tube rupture. Mass flowrate profiles (Figure 34) show that natural circulations flows are always effective and flowrates never stop during the transient simulated.

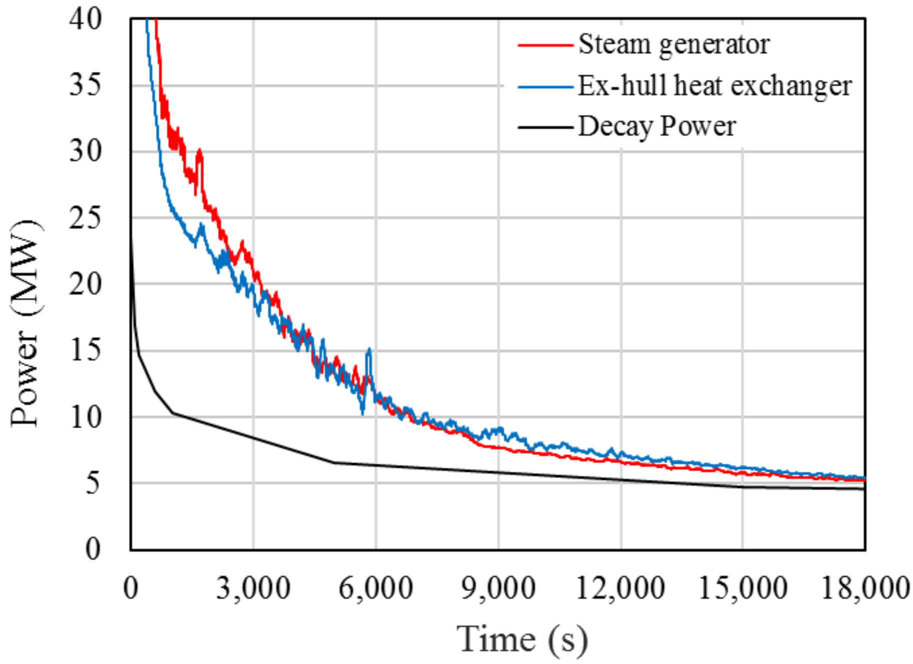


Figure 31 Comparison among decay power (black curve), power exchanged through the SG (red curve) and power exchanged through the ex-hull heat exchanger (blue curve)

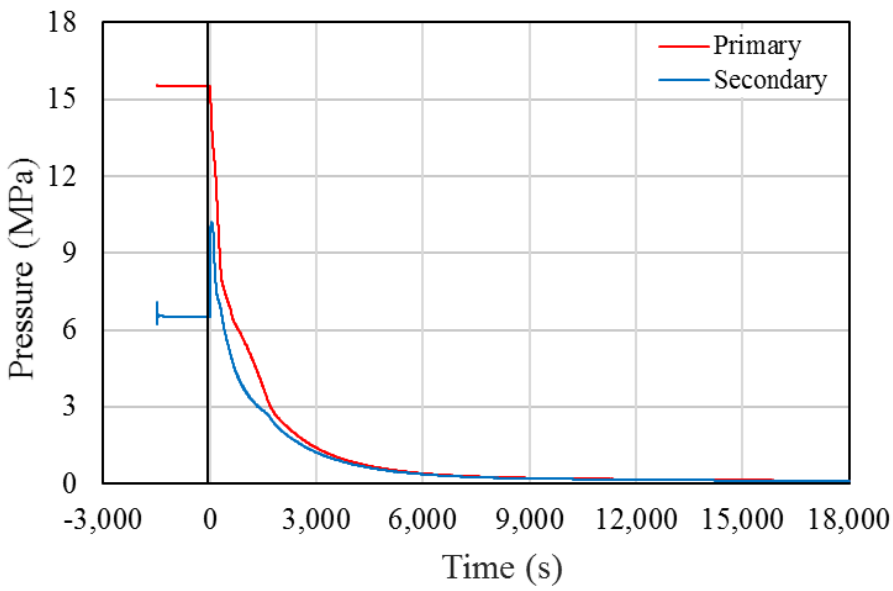
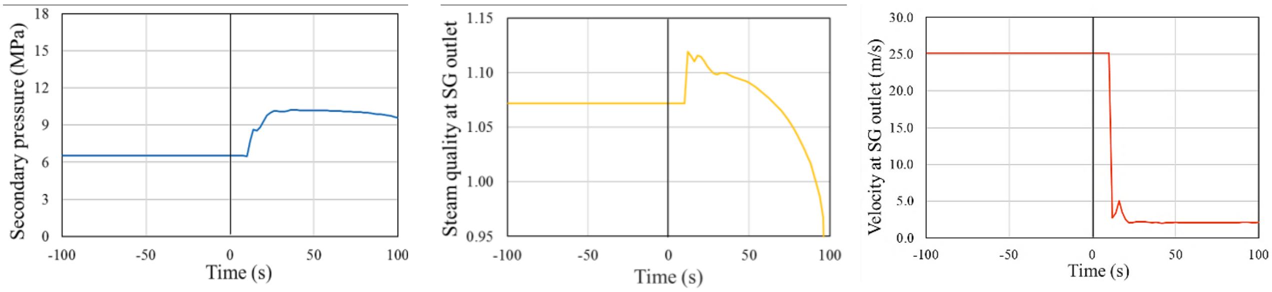


Figure 32 Primary and secondary pressure



(a) (b) (c)

Figure 33 Detail of pressure (a), steam quality (b) and steam velocity (c) at the SG outlet

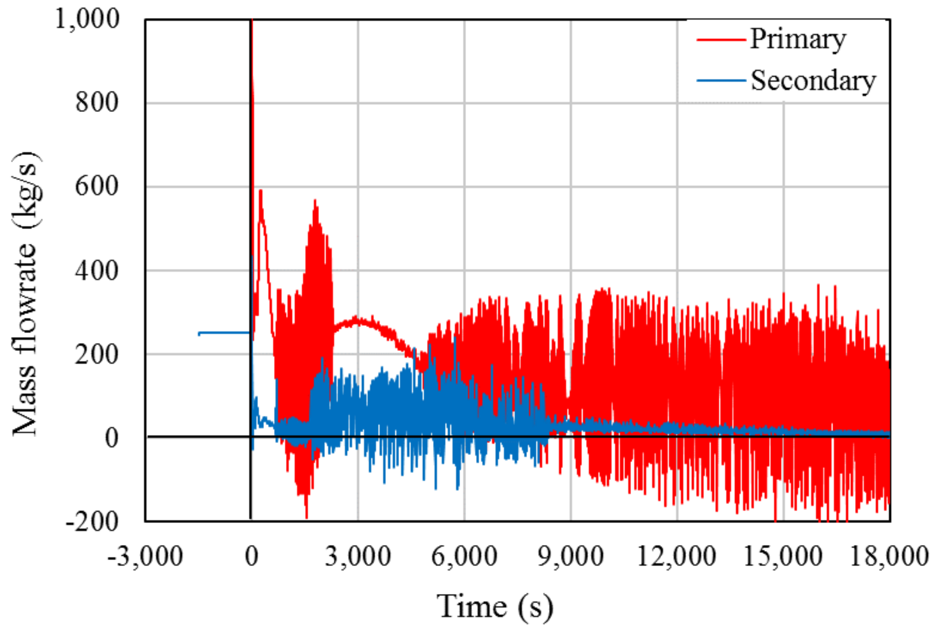


Figure 34 Mass flowrate profiles

The core never reaches a critical situation, since subcooled water always enters the core and the quality at the outlet is limited to acceptable values. (Figure 35). This situation not only assures the continuous wetting of the fuel rods, but also enhances the heat transfer, reaching a slightly bubbly flow. Thanks to the low amount of steam produced in the core, the collapsed liquid level (Figure 36) is always far above safety margins. Considering zero as the base of the active core, the collapsed liquid level never goes below 5 m: the level is always located between the upper plenum and the upper half of the riser, far from the top of the active core, which is at 2 m. Therefore, the Relap5 simulation evidences that the natural circulation flows transfer the decay heat to the exterior without risks of primary coolant overheating and core uncovering.

The behavior of the flow in the EHRS is driven by the ex-hull heat exchanger. It is possible to notice that on the first instants, until 1800 s, the outer wall temperature (Figure 37) is high and it is likely to cause a slight boiling of the external water (Figure 38). The tube side of the SG always sees production of steam (Figure 39), which is then condensed. The two-phase regime enhances the efficiency of the EHRS.

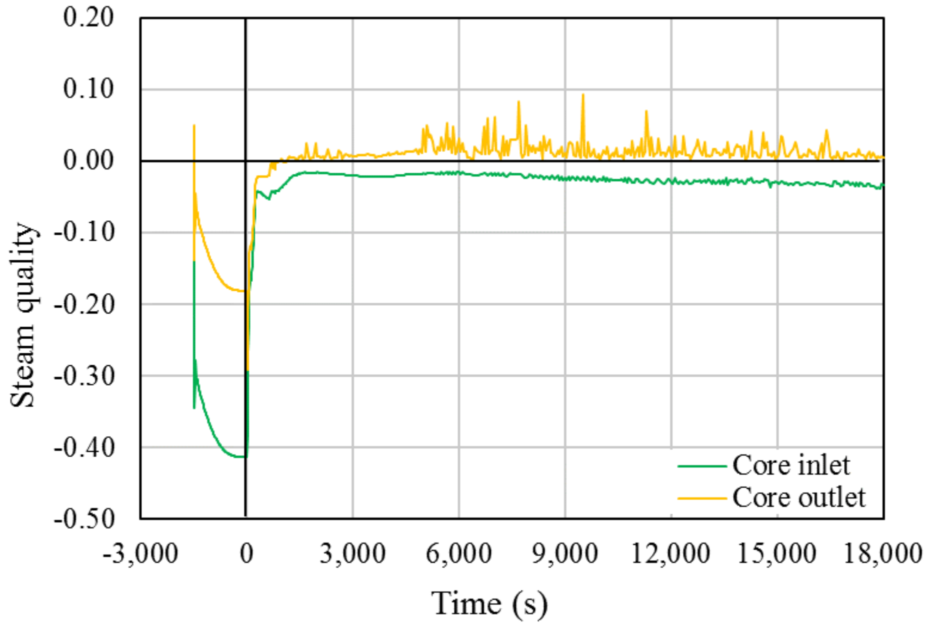


Figure 35 Steam quality at core inlet and outlet

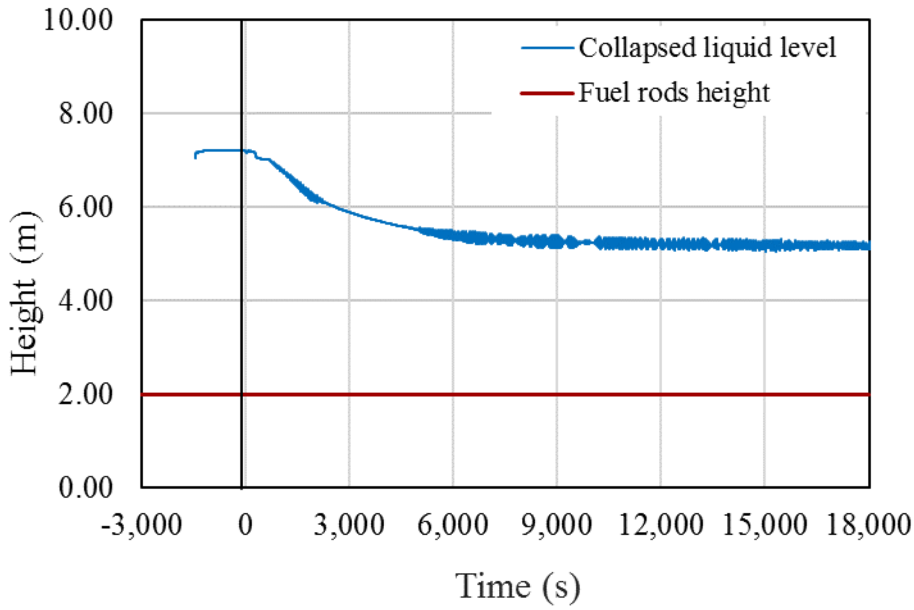


Figure 36 Collapsed liquid level in core barrel (zero is the base of active core)

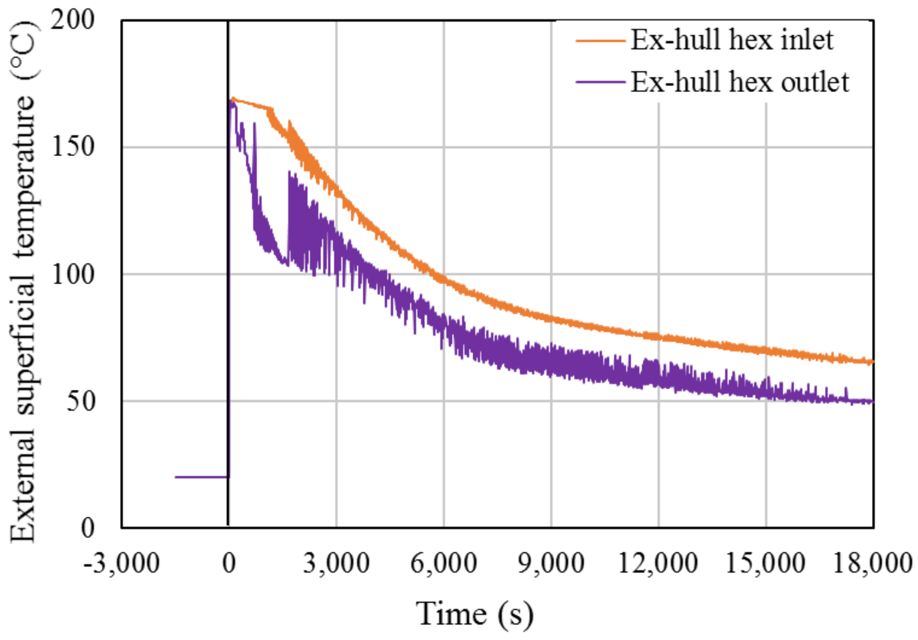


Figure 37 External superficial temperature profile of the ex-hull heat exchanger

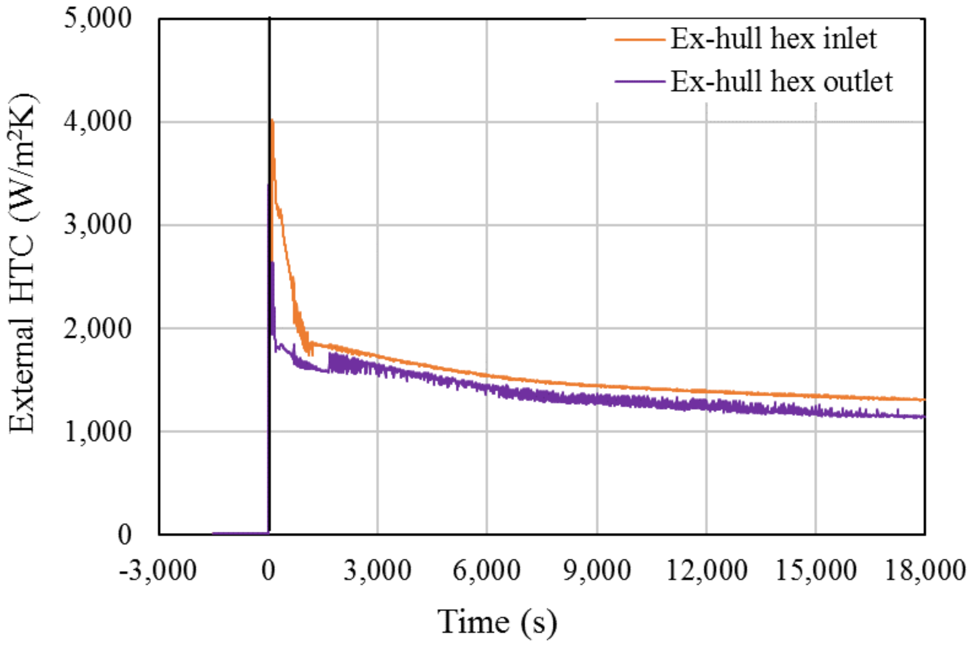


Figure 38 External heat transfer coefficient profile of the ex-hull heat exchanger

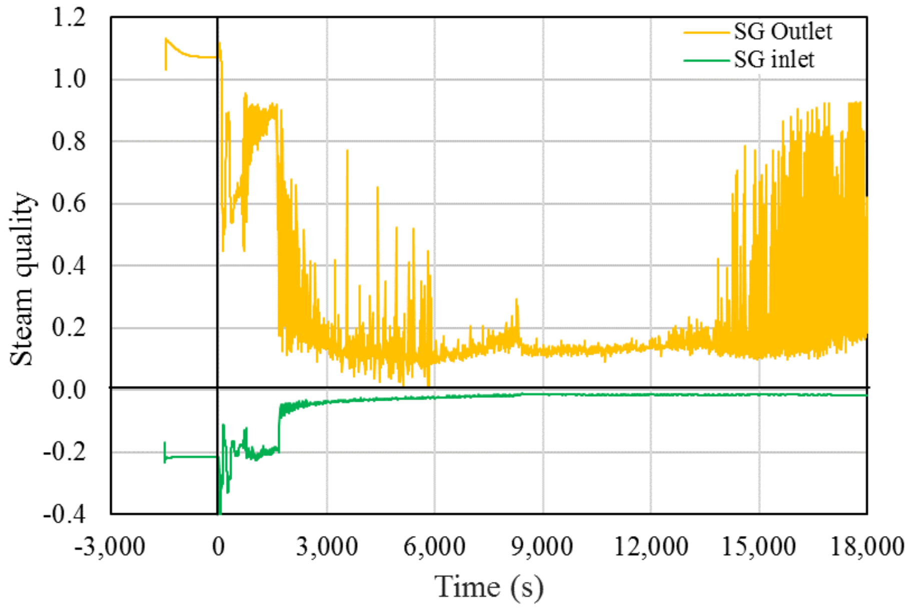


Figure 39 Steam quality at steam generator inlet and outlet

3.3.6 Parametric analysis: failure of one SG module

Considering a SG configuration made by two modules, the failure of one module, with consequent unavailability of two condenser trains, halves the available heat transfer surface of the heat exchangers and the cross area of the EHRS. This fact has consequences on the thermal-hydraulic conditions of the primary circuit and on the cooling of the fuel rods. Basically, a nuclear reactor is a system that operates in a heat flux imposes condition. This mean that, when almost-stationary conditions are reached and heat transfer processes are in equilibrium, the total decay power is always transferred to the secondary side and to the external sink and does not depend on the thermal resistances of the heat exchangers. What is changing is the global temperature difference needed to compensate an increase of the thermal resistance due to the reduction of the heat transfer areas. If one SG module fails during a SBO, the mean temperature of the primary circuit needs to be higher. Consequently, there is a larger production of steam in the core with respect to the case with the total SG area available.

The simulations are aimed at verifying if, during the transient, the thermal-hydraulic conditions at which heat transfer occurs are safe for the fuel rods. The model for this parametric case is equal to that used in the reference case and shown in Figure 29. The only modifications regard the geometric characterization of the steam generator tube side, the ex-hull heat exchanger and the piping of the EHRS: heat transfer areas and cross section of the tube sides have been halved.

Few hours after the scram, the collapsed liquid level in the core (Figure 40) drops very dangerously: at t = 9,000 s it is very close to the top of active core. Then, it rises a little bit thanks to the decreasing of decay power. The situation is much more critical than in the reference case, where the collapsed liquid level remained always by far higher than fuel rods height. The total power exchanged through the SG and condenser (Figure 41) is anyway greater than the decay power throughout all the transient, since at the beginning of the transient the system must exchange also the heat stored in the primary fluid. For roughly two hours, the heat exchange through the SG is higher than through the condenser. In this period the heat is accumulated in the secondary circuit. The quality at the outlet of SG secondary side (Figure 42) is quite low and in the final part of the transient the fluid is almost single-phase liquid. This is an effect of the imposed heat flux regime: since there is the need of a higher temperature difference, the equilibrium of the natural circulation flows is that on average the primary side is hotter and the secondary colder, with respect to the reference case. The reduced enthalpy jump of the half surface case is balanced by the mass flow rate on secondary side, which is higher in half surface case than in reference case (Figure 43a) because of lower pressure drop of the liquid single-phase flow regime with respect to the two-phase one. On the contrary on the primary side the flowrate is much lower than in the reference case (Figure 43b).

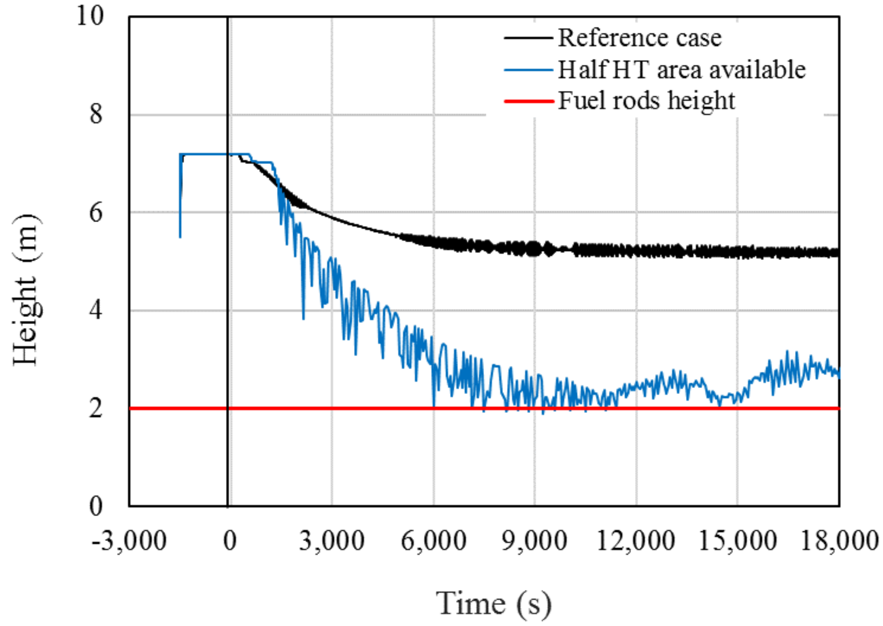


Figure 40 Collapsed liquid level in core barrel (zero corresponds to the base of active core) [parametric case]

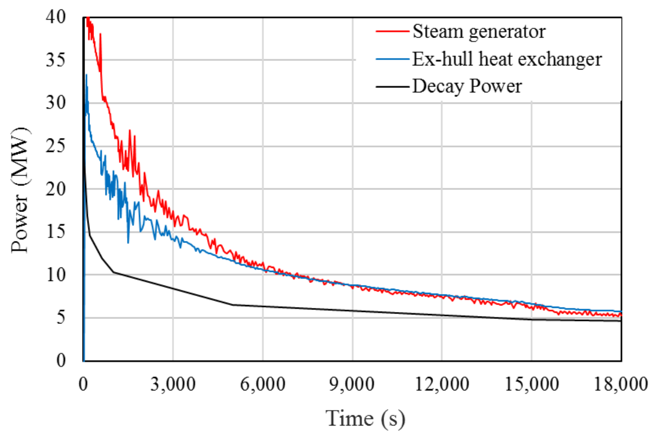


Figure 41 Comparison among decay power (black curve), SG power (red curve) and ex-hull heat exchanger power (blue curve) [parametric case]

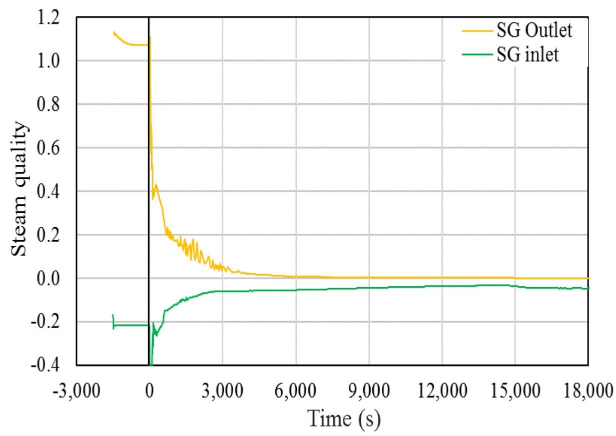
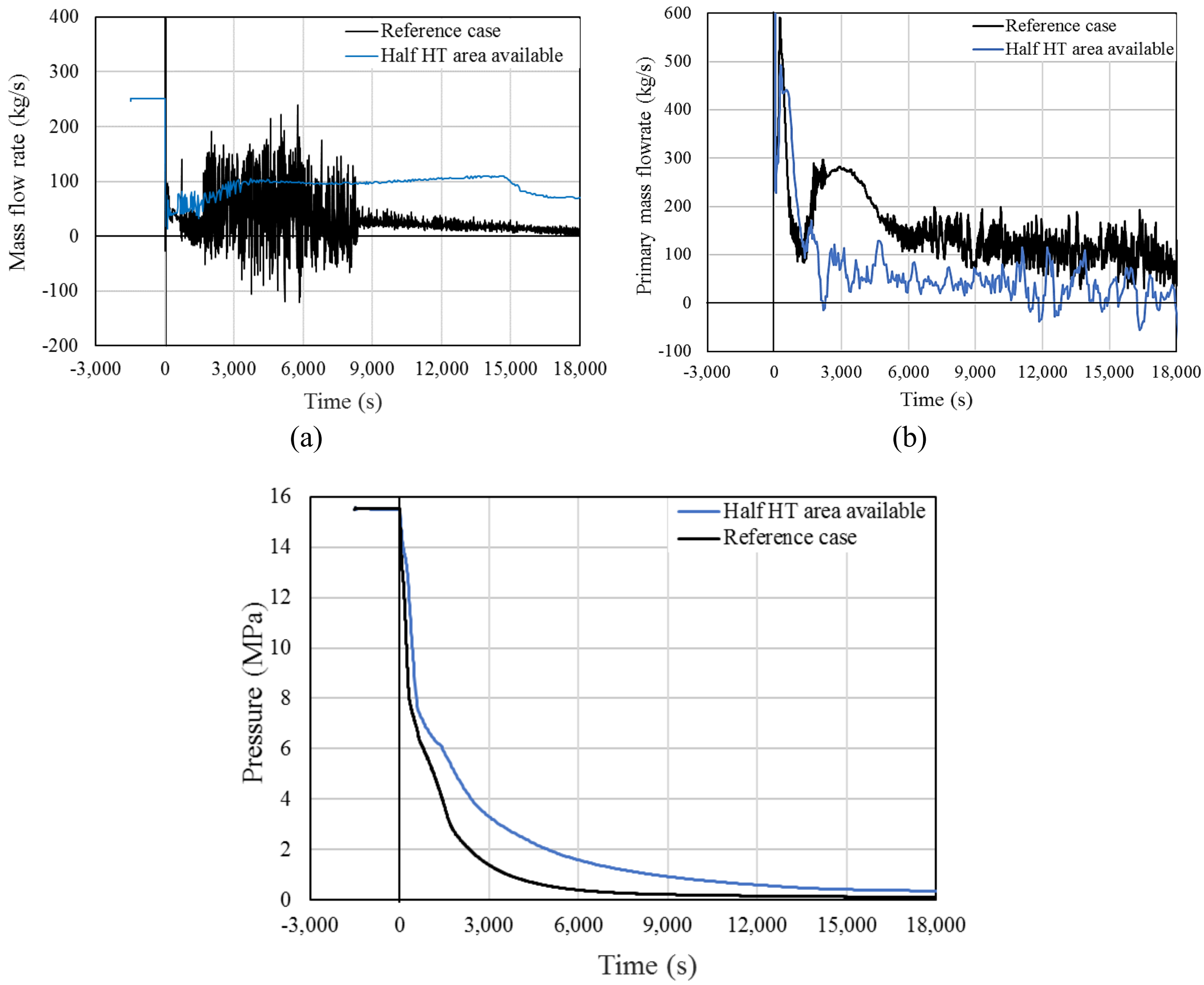


Figure 42 Profiles of quality at secondary SG inlet and outlet [parametric case]



(a) (b)

(c)

Figure 43 Comparison of profiles between reference and parametric cases: EHRS mass flowrate (a) primary mass flowrate (b), primary pressure (c)

The lower flowrate is the cause of the low collapsed liquid level. Given that the thermal output of the core is a fixed condition of the system, if the heat transfer area at the SG is halved the system start compensating it by producing more steam. This would increase the average temperature difference in the SG and also the HTC because of higher condensation rate. As balancing effect, the greater production of steam keeps the primary pressure, and consequently the saturation point, higher (Figure 43c), so in principle the higher pressure is a negative feedback and it would lead to a reduction of steam generation and to an equilibrium. However, this is not happening because the greater production of steam tends to reduce primary flowrate. The primary system is working in two-phase natural circulation at fixed power and in such conditions, there might be an inverse relation between mass flowrate and pressure drop. In a normal situation, a reduction of flowrate typically goes with a reduction of pressure drop. In a natural circulation boiling heated channel, an increase of the production of steam may change the flow regime to a condition with much higher friction: without a strong external input to the flowrate, i.e. a pump with strongly negative characteristic, the driving force of natural circulation is too weak to sustain the pressure loss and the result is a working point with reduced mass flowrate and higher steam quality. This situation is not desirable because it lead to the low collapsed liquid level evidenced by the Relap5 simulation and it is critical also because it is much exposed to Leddinegg and pressure drop instabilities in case of system perturbation. Therefore, the Relap5 simulation results bring to the conclusion that the EHRS alone may not be able to safely manage an SBO scenario if one SG module is not available.

3.3.7 Final remarks

In this sub-section, a 1D model of the IRIS-160 primary circuit coupled with the EHRS has been built and simulated with Relap5-Mod3.3 during a SBO scenario. Nodalization and modeling rely on previous works and empirical correlation. Results evidence that in reference configuration the system can adequately remove the decay heat from the core during the first 5 hours after the loss of forced flow. The quality of the water in the core remains always below safety margins and fuel rods are covered by at least 3 meters of liquid. The situation shows some criticalities if half of the heat transfer area and tubes cross section in the EHRS is unavailable. In this case, the mean temperature difference between primary and secondary sides needs to be higher, thus leading to a larger production of steam in the core and a reduction of primary natural circulation flowrate. The collapsed liquid level drops very close to the top of active core. Such a situation should be investigated in detail, since it represents a potential weak point of the EHRS. However, if the production of steam in the core becomes too high, the safety strategy allows opening the ADS, condensing the steam in the reactor containment, enabling the direct vessel injection from the safety tank and the flooding of the reactor compartment. This is the ultimate way to remove decay heat, which can operate in case of failure of all the other safety systems.

3.4. Long-term cooling analysis

3.4.1 Background

The keystone of submerged SMR safety strategy is the demonstration that the designed systems can provide passive decay adequate heat removal under any circumstances and for an indefinitely long time. This condition depends on the feasibility of the “depressurized and flooded” safe state, i.e. a targeted situation where the reactor containment is flooded by the injection of water from a large safety tank to the depressurized primary system. Depressurization may be the consequence of a LOCA or induced by ADS opening in case of failure of other safety systems. Decay heat is removed by sump natural circulation and rejected through the metal containment to the surrounding seawater, which acts as an infinite heat sink. This process is expected to provide a continuous and efficient cooling of the fuel rods, ensuring a potentially unlimited grace period. Such a concept owns several distinctive features of the emergency cooling system adopted by NuScale and Westinghouse SMR. The reactor metal containment is immersed in a pool and steam produced by the decay power condenses on the internal surface of the containment, thus rejecting heat to the exterior. The concept used in AP1000 and mPower is quite similar: water sprays and air circulation remove heat from the external surface of the metal containment. Those designs rely on the presence of large pools or tanks, therefore the amount of water that can be stored may determine a limit to the grace period or a strict threshold to the design power output of the reactor. In addition, protection toward the Loss of Ultimate Heat Sink (LUHS) event is not inherent, since a damage to the pool/tank can cause a loss of the cold water. The submerged SMR, relying on an infinite and permanent heat sink, brings an innovation in this field.

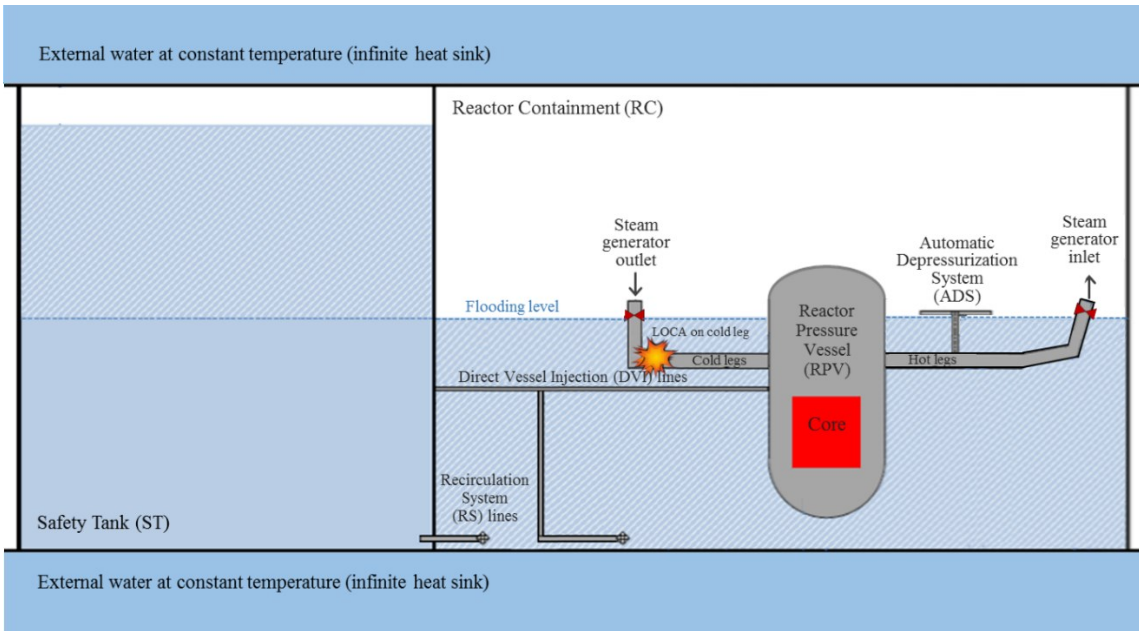


Figure 44 Sketch of the system considered in this work (dimensions are not representative)

3.4.2 Description of the problem

This subsection presents the modeling and results of a 1D system-code numerical investigation about the long-term core cooling process of a pressurized SMR placed on the seafloor, after a rupture of the primary circuit. Simulations have been performed using the 1D system code Relap5/Mod.3.3 and following the guidelines provided by U.S. NRC Nuclear Safety Analysis Division (U.S. NRC Nuclear Safety Analysis Division, 2003). The reference situation is a 500 MW_{th} PWR-like reactor placed in a submerged large horizontal cylinder, which undergoes a LOCA. Following the initiating event, reactor scram automatically occurs and high-pressure emergency injection and steam suppression systems operate in the period immediately after the beginning of the accident. They are aimed at managing the pressure peak and removing heat from the core when the decay power is high. Haratyk and Gourmel (Haratyk & Gourmel, 2015) described and simulated such sequence for a LOCA in Flexblue reactor with a VVER-type configuration, while Papini et al. (Papini, et al., 2011) did it in IRIS. After this first phase, which is estimated to last approximately 7-8 hours, the targeted situation is the "depressurized and flooded" safe state. The transient considered in the activity begins 7h30' after the reactor scram, when the water stored in a large safety tank has been already released to flood the reactor containment. At that time, the decay heat produced by the fuel rods is around 4 MW_{th}. In this analysis, the LOCA event and the operation of components to provide immediate coolant injection have not been simulated, since the focus is the long-term decay heat removal. The two reference simulations explore the heat transfer process in the first day after the scram and until the core power is 1 MW_{th}, i.e. approximately 21 day later.

A sketch of the system considered for this activity is shown in Figure 44. The system is composed of three macro-components, i.e. the Reactor Pressure vessel (RPV), the Reactor Containment (RC) and the Safety Tank (ST), jointed by three groups of piping, i.e. the Direct Vessel Injection (DVI) lines, the Automatic Depressurization System (ADS) and the Recirculation System (RS). The reactor type here considered is not the integral IRIS-160, but a VVER-like pressurized reactor: this study has been supported by DCNS, who advised to simulate the reactor configuration used for Flexblue preliminary safety analyses. However, the main characters under investigation here are the reactor containment and the recirculation lines. Primary system is depressurized and direct vessel injection is activated, while the SG and the secondary circuit are isolated and not simulated. Therefore, the layout of the reactor is believed to have a limited impact on the behavior of the natural circulation. The results of this analysis can be significant also for the IRIS-like case.

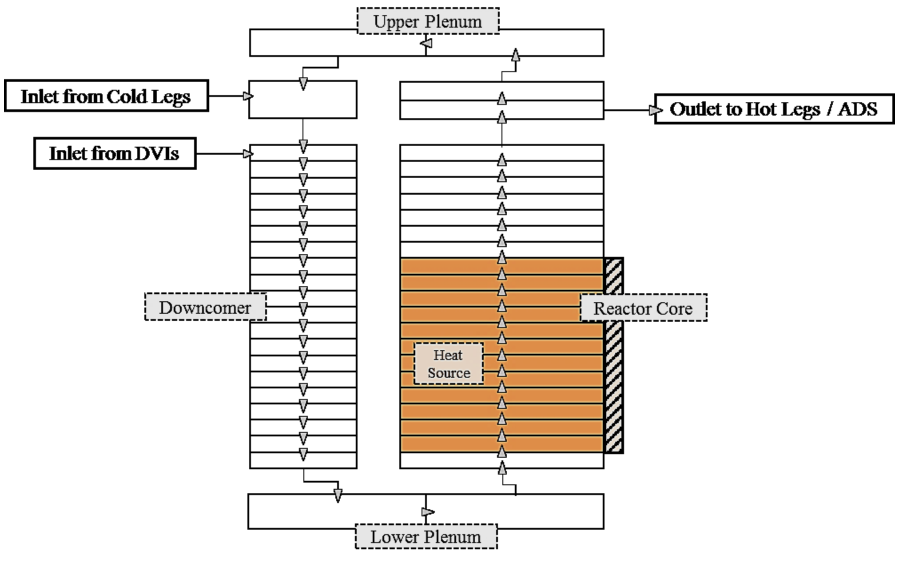
Table XIII. Geometry and nodalization of long-term cooling model

	Component	Geometry		Nodalization
Reactor Pressure Vessel	*Core*	Active length Total length Hydraulic diameter Flow area	2.3 m 3.0 m 11.3 mm 3.763 m^2	12 volumes for active zone, 2 volumes for core inlet
	Lower plenum	Total height Volume	1.10 m 3.25 m^3	2 single volumes with cross-flow connection
	Upper plenum	Total height Volume	1.3 m 5.75 m^3	2 single volumes
	Upper head	Total height Volume	0.1 m 2.15 m^3	2 single volumes with cross-flow connection
	Downcomer	Flow area Height Hydraulic diameter	2.68 m^2 3.8 m 290.0 mm	20 elementary volumes (1 specific for DVIs inlet)
	Riser	Flow area Height Hydraulic diameter	6.4 m 1.0 m 2.85 m	6 elementary volumes
	Hot legs	Number of tubes	2	17 elementary volumes for each tube
Connection lines	*Direct Vessel Injection lines*	Number of tubes Connection	2 ST-RPV	70 elementary volumes for each tube
	Recirculation lines	Number of tubes Connection	4 RC-DVIs (2 tubes) RC-ST (2 tubes)	27 elementary volumes for each RC-DVI tube, single junctions for RC-ST
	Automatic Depressurization System	Number of tubes Shape Connection	2 T-junction RPV-RC	12 elementary volumes for each ADS line
Containment	*Reactor Containment*	Diameter Length Water inventory Occupancy rate	14.0 m 25 m 1,344,300 kg 30%	Two parallel pipes made by 56 elements and cross-flow connected. Volumes and flow areas are shaped to represent the horizontal cylindrical geometry
	Safety Tank	Diameter Length Water inventory Occupancy rate	14.0 m 18.1 m 1,247,500 kg 10%	

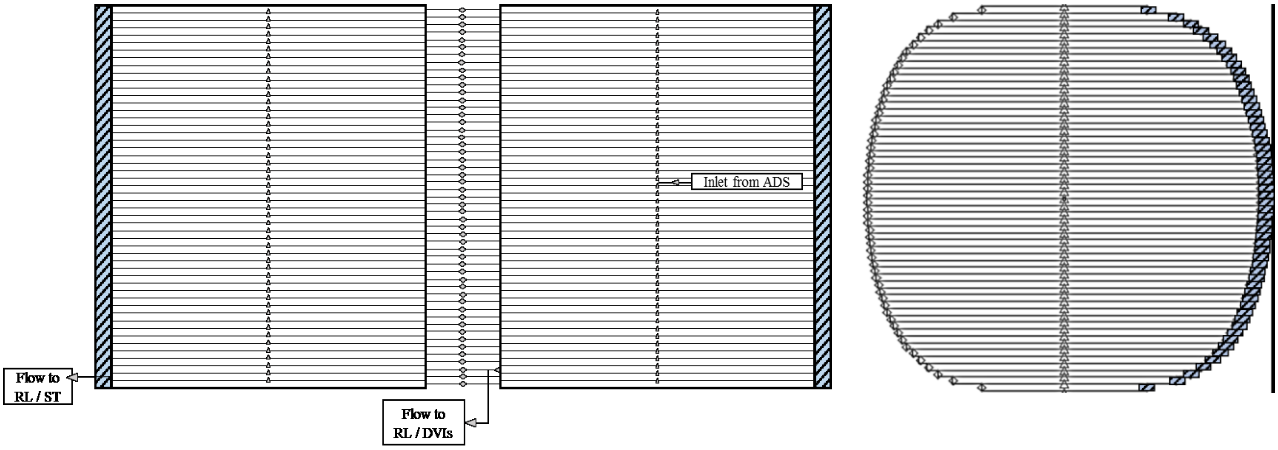
3.4.3 Modeling and nodalization

Geometry and nodalization features are given in Table XIII and shown in Figure 45 and Figure 46. The RPV model (Figure 45a) reflects the simplified design of a typical pressurized water reactor: it is made of the downcomer, the lower plenum, the core and the upper plenum. Each component is modeled using elementary volumes connected by junctions. To avoid the onset of unphysical recirculation flows, the model of the core is made with a single vertical channel. The heat source is placed into the active zone of the core, which is made of 12 elementary volumes. Since only one pipe is used to simulate the core region, the radial power distribution in the core is neglected. On the contrary, the axial cosine-shaped distribution is considered. Form losses coefficients are set into the core region to simulate the concentrated pressure drop given by the spacing grids. The model also considers friction and typical tube entrance/exit losses. However, without a specific RPV model, it is not possible to have a precise form pressure losses characterization of the model.

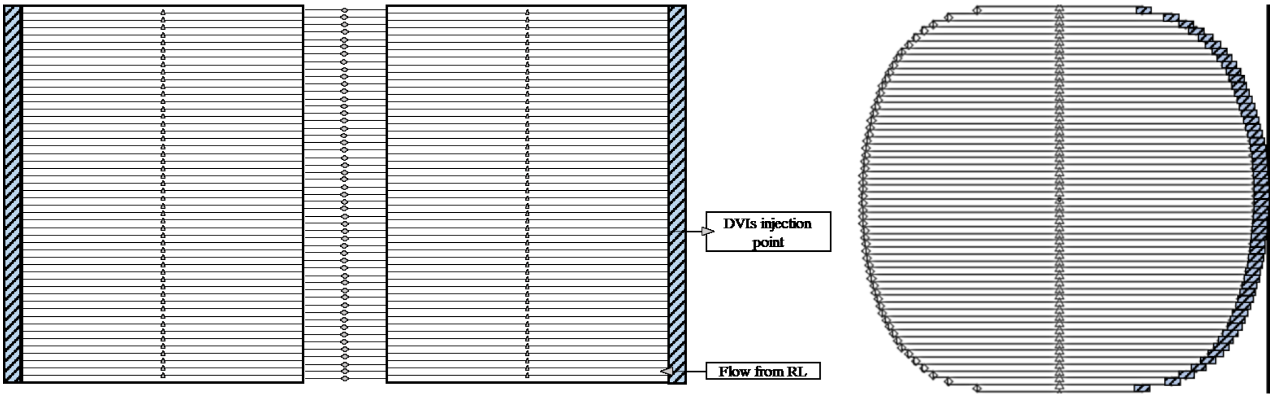
The nodalization of the two containments, i.e. RC and ST, in Relap5 is a hard task, since it is a “pipe oriented” code and it is not optimized for analysis of large volumes. A sliced model is here used: the total volume of each containment is subdivided into two parallel pipes, upward vertically oriented, made of 56 elementary volumes connected by transversal crossflow junctions, as shown in Figure 45b and Figure 45c. Each element is characterized by volume, hydraulic diameter and heat transfer area. Each junction is characterized by the cross section. A similar approach was proposed by Papini et al. (Papini, et al., 2011): they tested the sliced model on a case-study and compared the results with the predictions of the code GOTHIC, a specific tool for the simulation of large containments, observing an acceptable agreement. Pipes are provided with heat structures that simulate the conductive thermal resistance of the containment and the convective heat transfer given by the natural circulation of the external water. This approach to the spatial discretization may present some criticalities, such as the modeling of the heat structure geometry: Relap5-Mod3.3 allows only rectangular, cylindrical and spherical geometry and the rectangular option has been selected for this work. However, it offers at least three advantages: (i) it permits a good vertical resolution of the liquid and gas phases in the containments; (ii) thanks to the two pipes, it permits to observe flow recirculation; (iii) it allows using a non-uniform discrete profile for the external heat transfer coefficient.



(a) Reactor Pressure Vessel



(b) Reactor Containment (front view and side view)



(c) Safety Tank (front view and side view)

Figure 45 Nodalization of RPV (a), RC (b) and ST (c)

The zones indicate the presence of a heat structure coupled to the component.

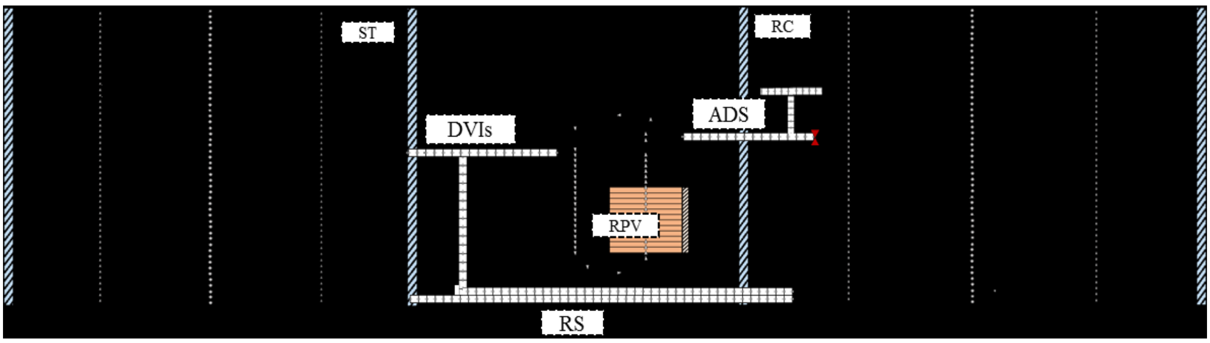


Figure 46 Complete model (dimensions are not representative)

3.4.4 Boundary and Initial Conditions

To study the system under investigation with a 1D code, two thermal boundary conditions are required where heat structures simulate boundary heat transfer processes. Second type boundary condition (fixed-power conditions) has been adopted to simulate decay heat generation in the fuel rods and third type boundary condition (convective condition) for the external seawater natural circulation. In-vessel retention has been neglected in this study. Initial conditions at 7h30’ after the scram are determined thanks to conservative and reasonable assumptions.

The heat structure that simulates fuel rods is composed of an inner UO2 cylinder, which contains a volumetric heat source, surrounded by a Zircaloy-4 annulus, which represents the cladding. The decay curve proposed by ANS 2005 Standard is used (Shwageraus & Fridman, 2012) to simulate a 25-hour-long reference transient. In addition, an accelerated decay curve is used besides the standard one. The accelerated transient allows investigating the behavior of the sump natural circulation flows up to 21 days after the scram. Both curves assume the decay power 7h30' after the scram as initial value. A comparison between the two profiles is given in Figure 47. The use of such accelerated decay curve is an obligated choice in order to simulate, within a manageable computational load, the behavior of the system up to several days/weeks after the scram, thus addressing the main purpose of the study. However, such approach represents a strong hypothesis on the system evolution and poses some warning about the conservativeness of the assumption: although it reduces the buoyancy force in the core more rapidly than in the real case, it also underestimates the amount of heat transferred to the RC within the simulation period. Notwithstanding this consideration, one can observe that the thermal inertia of the water inventory in the ST and in the RC is very large. Therefore, the impact of that underestimation on the results is believed to be quite limited and allow focusing the attention on the hydro-dynamic aspects of the sump natural circulation flow when core decay power is low. To complete the investigation, an additional simulation is performed considering a constant and very low core power, equal to 0.4 MW. This value represents the decay heat of the reactor approx. 4 months after the scram. For this case, the output of the simulation with the accelerated curve is used to determine initial conditions. The conclusions of the study are then deduced from all the three cases.

The heat structures associated to RC and the ST take into consideration the conductive thermal resistance of the reactor containment and the external natural convection. A convective boundary condition is used to avoid modeling the external water. The results of a previous study (Santinello, et al., 2017) (Figure 48) about the external natural circulation from a submerged horizontal cylindrical containment are employed to define the heat transfer coefficient (HTC). Seven discrete constant values are set, as shown in Figure 49, since the profile of the HTC is not uniform along the perimeter of the containment. Undisturbed seawater temperature has been fixed to 20°C.

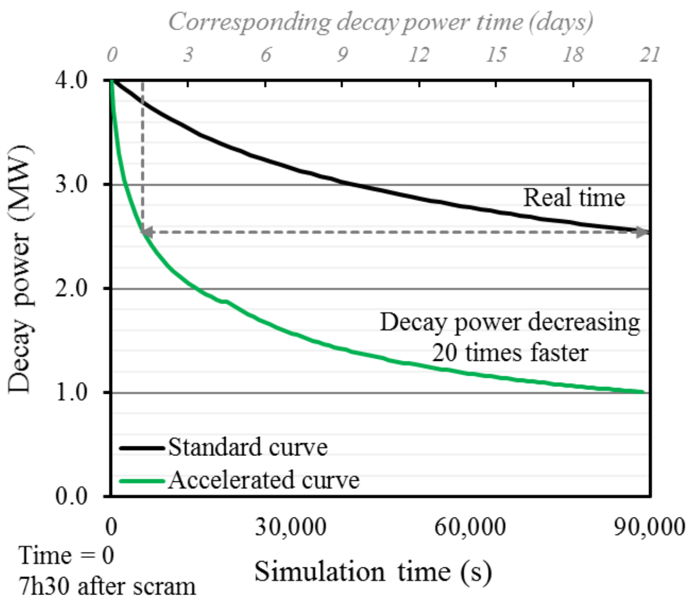


Figure 47 Decay curves used for the simulation. The zero point on the x-axis represent the starting point of the simulation, i.e. 7h30' after the scram

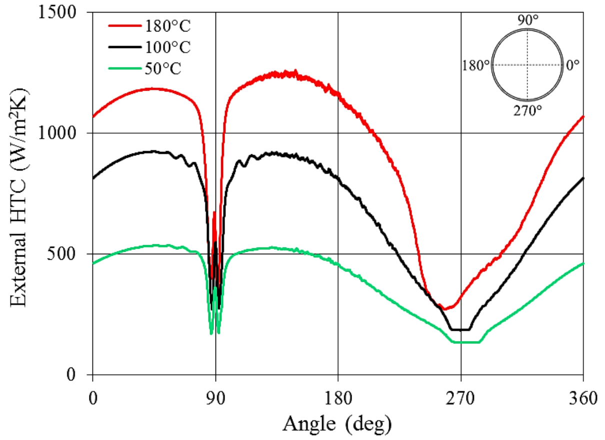


Figure 48 Results of the CFD study about the external HTC in (Santinello, et al., 2017), fixed internal temperatures and T_∞=35°C

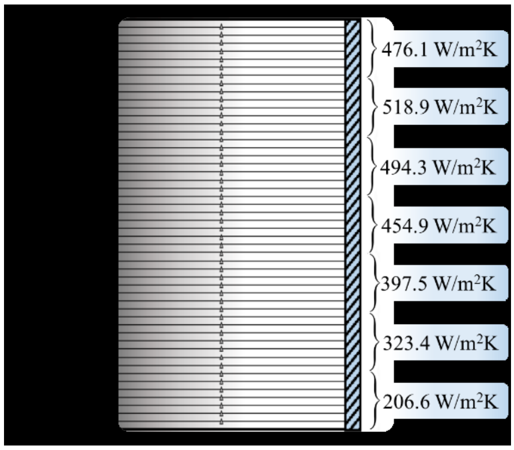


Figure 49 HTC values used in this study, based on 50°C case (green curve in Figure 48)

Definition of initial conditions is a difficult aspect of the study, since the conditions of the system 7h30' after the initiating event of the accident are not known. During the initial transient, the system evolves without any human intervention or AC power: several heat transfer, phase change and mixing processes occur and the thermal-hydraulic conditions of the system are not predictable. Approximated and/or conservative estimations have been made a priori for pressures, temperatures and phases conditions. The chosen initial conditions are reported in Table XIV. In particular, the initial temperature of the water stored in RC and ST, equal to 50°C, is strongly conservative: if all the decay heat produced in 7h30' after the scram was transferred to the water stored only in the RC with no heat transfer to the exterior (as in eq. 10), the temperature would increase from 20°C to 46°C.

$$T_i - T_{ext} = \frac{\int_{scram}^{7.5h} \dot{Q}(t)dt}{m_{RC}c_p} \approx 26°C \tag{10}$$

T_i = initial RC temperature (°C)
T_{ext} = external temperature (°C)
$\dot{Q}(t)$ = decay power function
c_p = specific heat (J/kgK)
m_{RC} = water inventory in RC (kg)

Table XIV. List of initial conditions for cases with standard and accelerated curves

	Reactor Pressure Vessel	Reactor Containment	Safety Tank
Pressure	0.2 MPa	0.2 MPa	0.2 MPa
Temperature	Saturation in upper plenum 100°C elsewhere	100°C for gas zone 50°C for liquid zone	50°C for both liquid and gas volumes
Level of liquid	Steam in upper plenum, liquid elsewhere	7 m	7 m
Non-condensable	-	100% nitrogen	100% nitrogen

3.4.5 Results

The combination of three simulations, i.e. standard curve, accelerated curve and low-power, allows investigating the sump natural circulation flow from the early phases few hours after the scram to the long-term period. The analysis with the standard curve firstly shows that the containment is capable to reject a large amount of heat even when the decay power is high. Then, the accelerated transient and the low-power case reveals that the sump natural circulation flow should be still effective several days/weeks after the scram. Steam is produced into the core, condenses in contact with the RC wall and the condensate falls by gravity into the flooded zone. Pressures and levels of liquid in RPV, RC and ST are summarized in Table XV. Such values confirm the feasibility of the natural circulation flow. Because of the presence of nitrogen and steam, pressure in RC is higher than in ST, hence the flow in the recirculation line is such that liquid water flows from RC to ST, increasing the level of liquid of the latter with respect to initial conditions. The head in the ST is sufficient to push cold water in the DVIs and to cool the fuel rods. The feedwater in the DVIs flows from the ST/RC to the RPV, ensuring a continuous and stable injection. This process is sketched in Figure 50.

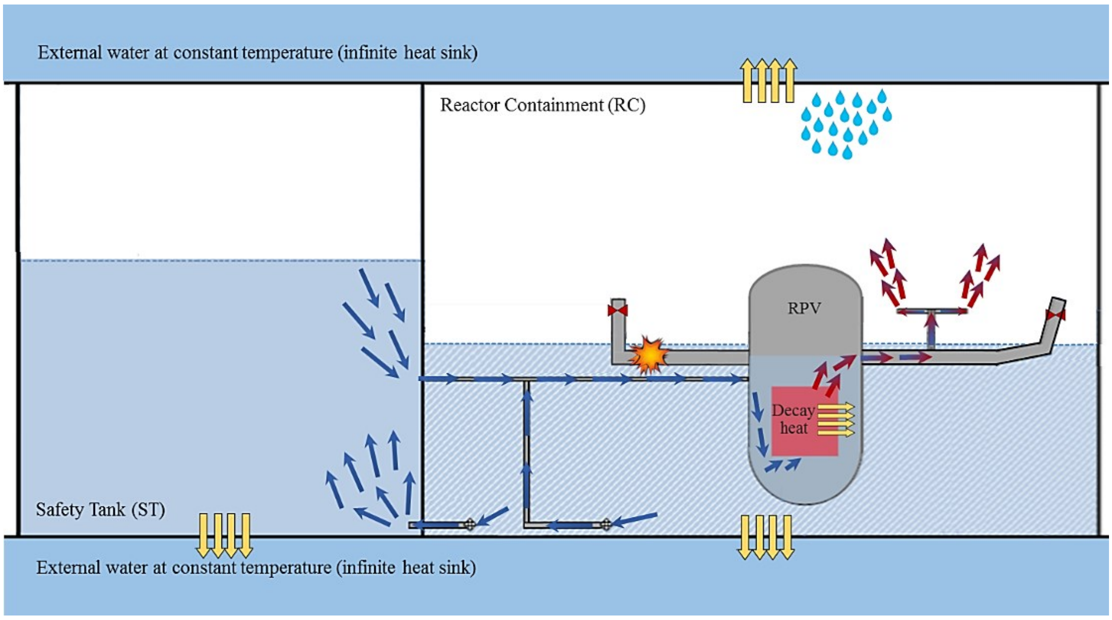


Figure 50 Sketch of the sump natural circulation flow at the end of the simulation (dimensions are not representative)

Table XV. Pressures and levels of liquid of the three simulation cases

		Pressure	Level of liquid
Standard curve	RPV	0.247 MPa	4.95 m *(5.90 m)*[1]
	RC	0.258 MPa	6.04 m *(5.56 m)*[2]
	ST	0.237 MPa	8.01 m *(2.11 m)*[3]
Accelerated curve	RPV	0.228 MPa	5.49 m *(6.42 m)*[1]
	RC	0.225 MPa	6.44 m *(6.06 m)*[2]
	ST	0.214 MPa	7.55 m *(1.65 m)*[3]
Low power	RPV	0.220 MPa	5.50 m *(6.43 m)*[1]
	RC	0.216 MPa	6.57 m *(6.19 m)*[2]
	ST	0.208 MPa	7.40 m *(1.50 m)*[3]

[1] *above the bottom of RC*   [2] *above the RL*   [3] *above the DVIs*

The simulation with the standard decay curve represents the evolution of the system for 25 hours after the starting point taken at 7h30' after the scram, given the initial conditions in Table XIV. The results confirm the good capability of the metal containment to reject decay power through the containment. Except for the first 1,000 s, which are affected by the conservative initial condition about non-condensable mass fraction, heat transfer rate to the exterior is almost always greater than decay power, as visible in Figure 51. The very large water inventory and the presence of external water acting as an infinite heat sink ensure that the feedwater injected into the RPV is always sufficiently cold. The largest part of the heat is removed through a liquid-liquid inner-to-outer heat transfer in the flooded part of the RC. At this step of the evolution of the accident scenario, when decay heat is less than 1% of the total power, steam production is quite low and wall condensation gives a lower contribution to the total heat transfer. However, the very conservative boundary conditions about the composition of the RC atmosphere is leading to underestimate this component.

The coolant in the core is always heated up until the saturation point: Figure 52a shows that the quality of the coolant at the outlet of the heated zone is always slightly greater than zero. To create the density gradient necessary to sustain natural convection, the system needs to produce a small amount of steam. This production is continuous and quite regular along all the simulated period. In general, fuel rods are always adequately cooled: the mass flow rate through the core allows the removal of the decay power, showing no critical issues for the thermo-dynamic conditions of the coolant and for the temperature of the fuel cladding. Decay heat is taken away both by the vapor and liquid phases, whose velocities are positive. (Figure 53a).

The large thermal capacity of the water inventory stored into the ST and the presence of the external water acting as an infinite heat sink provide great benefits to the core cooling process. Firstly, the amount of heat that can be stored into the safety tank is very high: even under conservative assumption, the temperature profile of the flooded zone of the RC always remains far below the saturation point. Then, the RC and ST are continuously cooled by the external water through the metal containment and the pool average temperature undertakes a decreasing trend after few hours of simulation (Figure 54a). Such liquid-to-liquid heat transfer is very efficient and represents the major way for heat transfer to the external water. This process ensures the presence of cold feed water for core cooling during the whole simulated transient. The steam generated condenses on the internal surface of the RC. Condensation heat transfer gives a lower contribution, but this process is likely to be affected by the very conservative initial conditions about the non-condensable mass fraction in the RC. Actually, the steam suppression system is supposed to operate immediately after the LOCA, thus removing a large part of non-condensable gases.

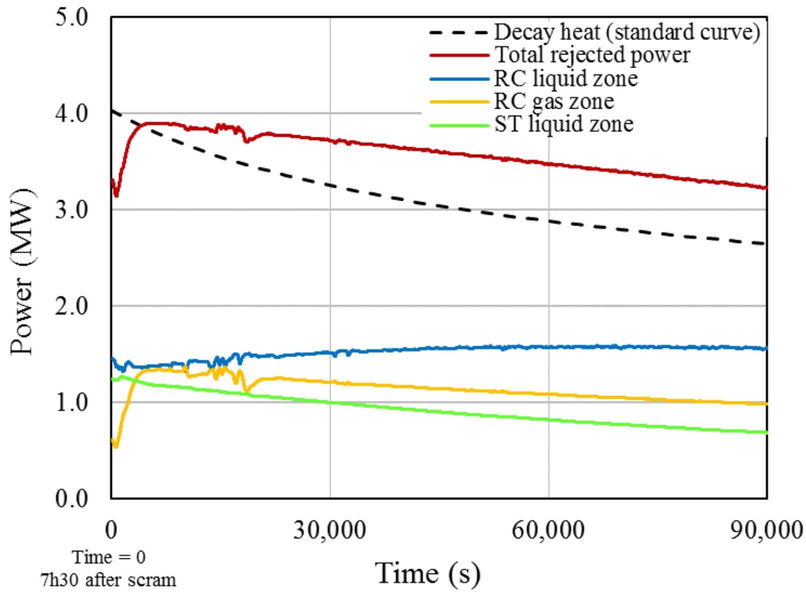
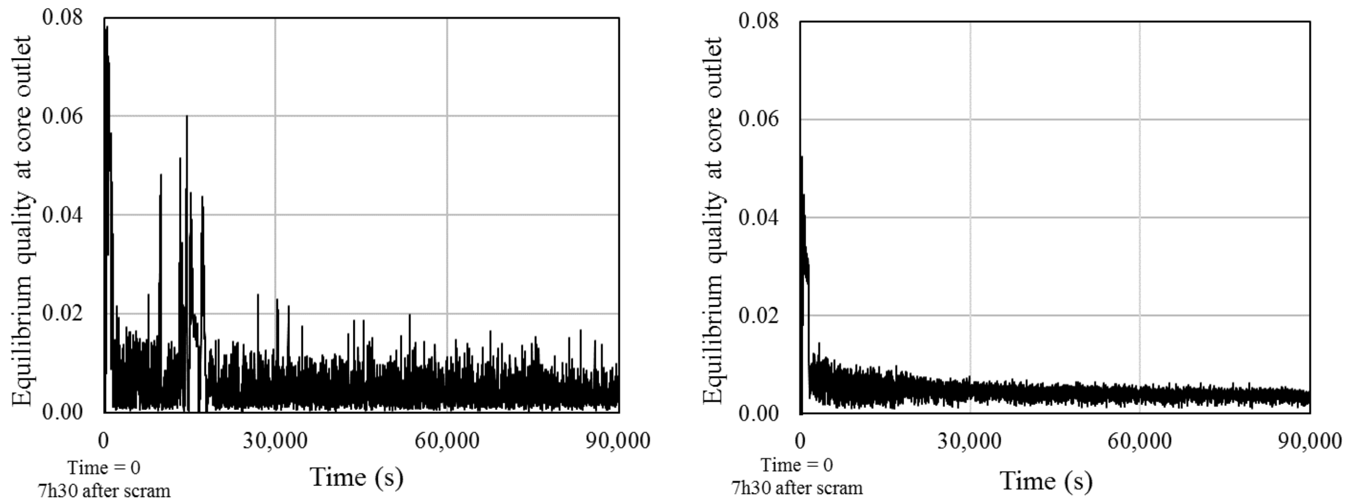
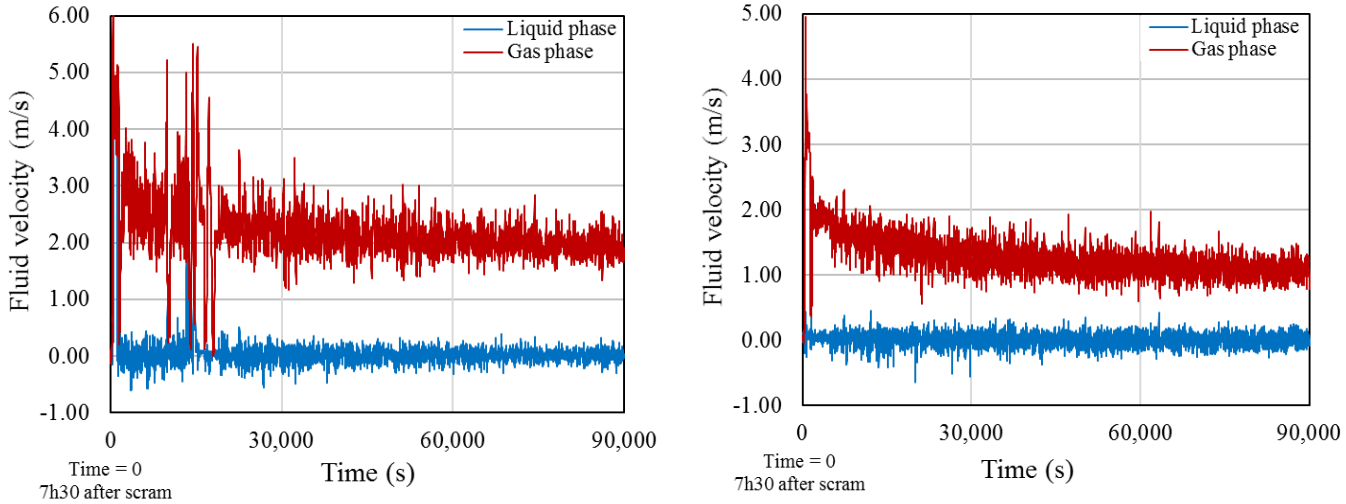


Figure 51 Heat transfer to the exterior compared with the decay power (standard curve)



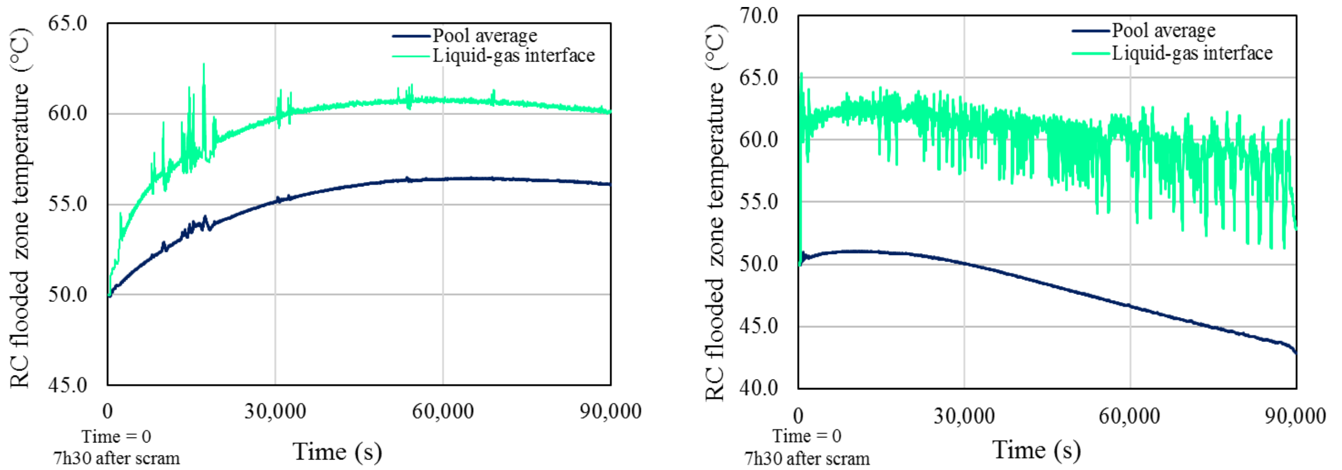
(a) (b)

Figure 52 Quality profile at core outlet for standard (a) and accelerated (b) curves



(a) (b)

Figure 53 Velocity profile in the hot legs for standard (a) and accelerated (b) curves



(a) (b)

Figure 54 Temperature profiles in the RC pool for standard (a) and accelerated (b) curves

Figure 52b – 53b – 54b regard the transient with the accelerated decay curve. At the end of the simulation time, the core power assumes the value that the standard decay curve reaches nearly 21 days after the scram. The results of the simulation assess the behavior of the passive safety systems under the given layout, water inventory and boundary/initial conditions. The long-term sump natural circulation flow is sustained also by a smaller power. This is noticed for decay power input ranging from 4 to 1 MW. During all the transient, a small production of steam in the core is necessary to maintain the density gradient and drive the natural circulation flow.

Similar conclusions can be carried out also from the low-power case. the low power simulation allows evaluating the behavior of the sump natural circulation flow when the core decay power, and consequently buoyancy force, has strongly decreased with respect to the simulation with the standard curve. According to the results, even with a low decay heat, i.e. 0.4 MW constant, the sump natural circulation flow is operating and requires the production of steam in the core (Figure 55). The outlet quality is however very low. In such circumstance, the capability of the containment to reject heat to the exterior is much higher than the decay power (Figure 56).

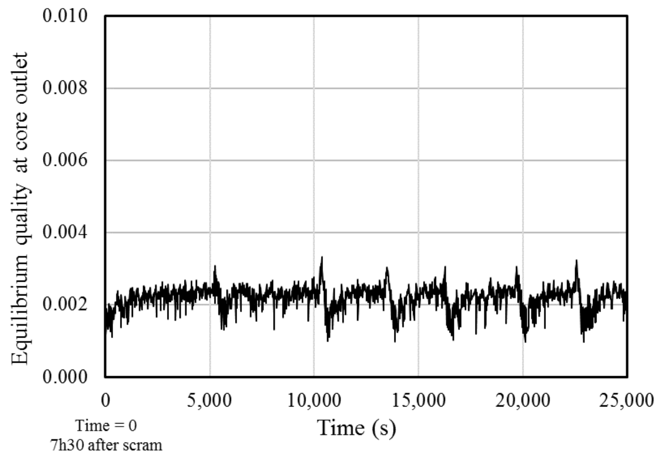


Figure 55 Quality profile at core outlet for low power simulation

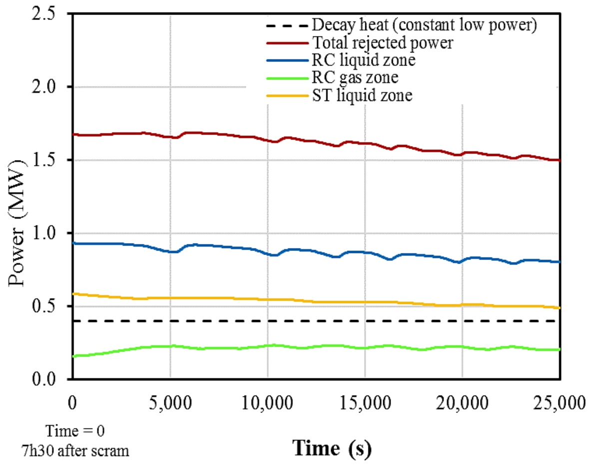


Figure 56 Heat transfer to the exterior for low power simulation

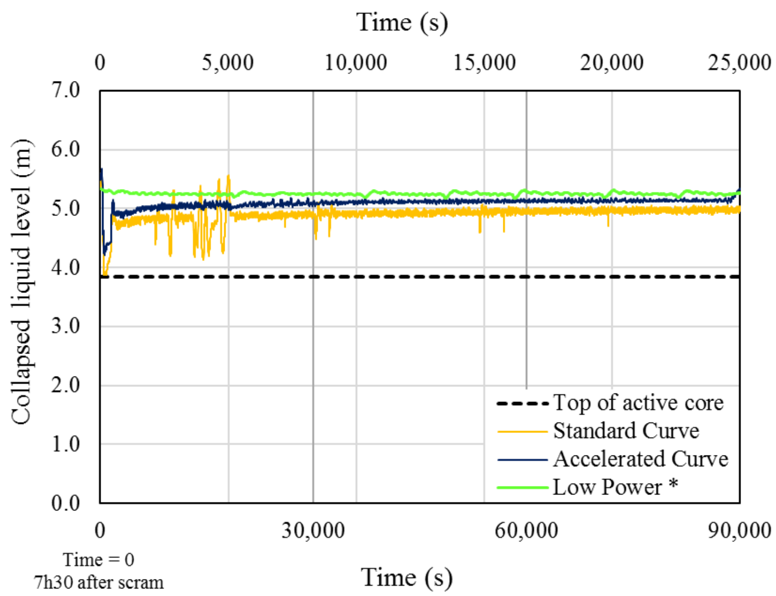


Figure 57 Collapsed liquid level: height from the bottom of RPV *(* time on the upper x-axis)*

In conclusion, a concept of passive safety systems based on a submerged containment and sump natural circulation flow can cool the fuel rods for a potentially unlimited period. The collapsed liquid level is always above the top of active core in the three cases (Figure 57). The temperature profile of the RC pool in Figure 54 is significant in order to verify the consistency of the approach based on the accelerated decay curve. It is observable that the RC pool temperature profile has a maximum and a then decreasing trend also in the simulation with the standard curve, i.e., within the first day after the scram. Therefore, it is sure that the simulation with the accelerated curve is not neglecting an accumulation of heat inside the RC pool using the accelerated curve.

3.4.6 Sensitivity on the nodalization of containments

Nodalization of containments (RC and ST) is a complex task, since Relap5 is a 1D system code and in general it is not considered the first choice when modeling of containments is required. Relap5 solves fluid balance equations considering the linear coordinate, while constitutive models account for phenomena with transverse gradients (e.g. friction and wall heat transfer). In principle, the application to a large volume with a mixture of steam and non-condensable gas in free convection goes beyond the limits of proper use of the code, even though with an accurate nodalization strategy acceptable results can be obtained (Papini, et al., 2011). This aspect represents the most critical issue of the spatial discretization and probably the largest source of numerical uncertainty. In the current study, the necessity of simulating the behavior of multiphase and multicomponent flow in large containments is addressed as explained in paragraph 3.4.3, i.e., with two pipes subdivided into 56 elementary volumes and with crossflow junctions between corresponding elements. To roughly evaluate the impact of the nodalization of RC and ST on the results of the simulations, a sensitivity analysis concerning the number of elementary volumes in each pipe has been performed. It varies from 28 to 70, i.e., 2x28 – 2x35 – 2x47 – 2x56 (ref.) – 2x70, and consequently the length of each element ranges between 0.5 and 0.2 meters. All cases employ the standard decay curve.

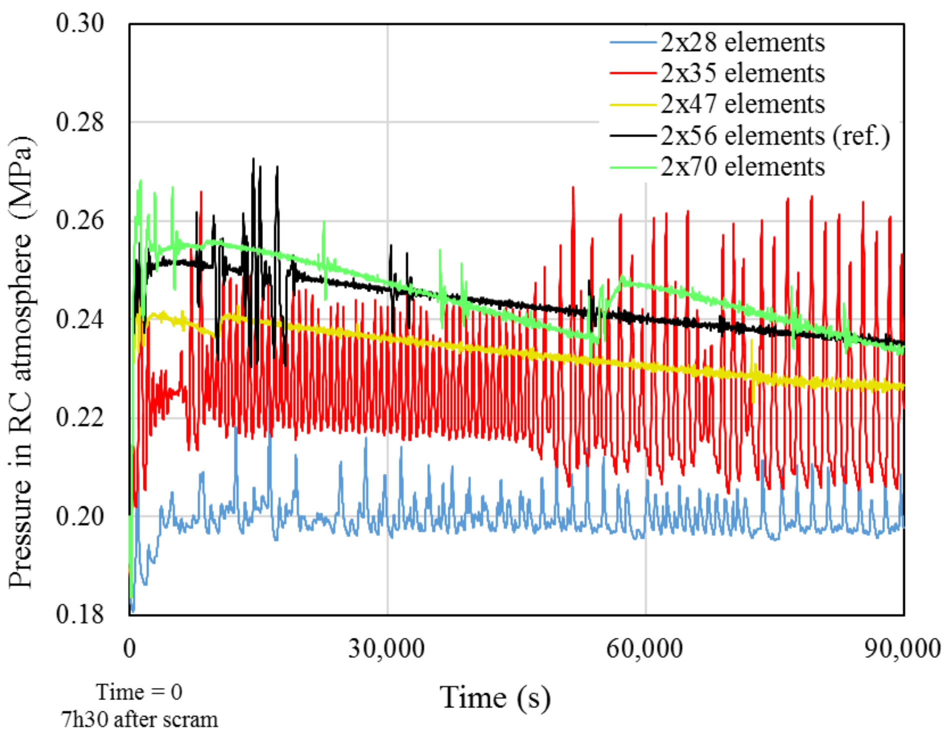


Figure 58 Pressure profiles in the RC for sensitivity cases, compared to the reference case

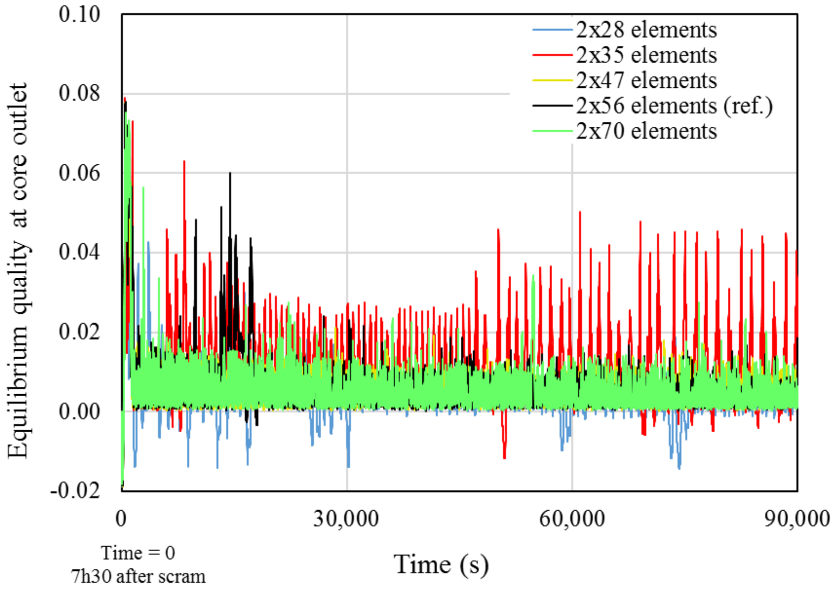


Figure 59 Quality profiles in the RC for sensitivity cases, compared to the reference case

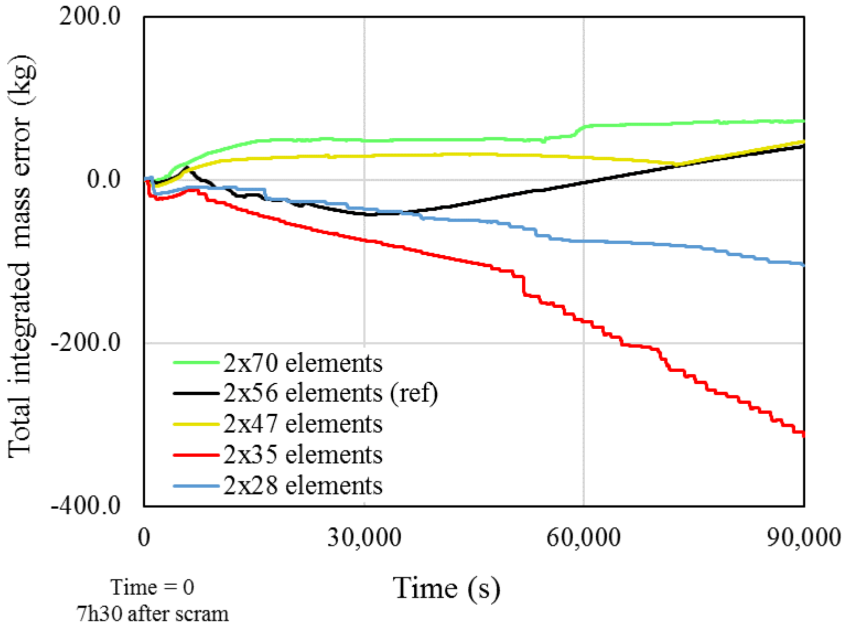


Figure 60 Integral mass error profiles for sensitivity cases, compared to the reference case

While all these cases agree that long-term core cooling is always ensured during the simulated transient, it is observable that nodalization of RC and ST has an important influence on the general behavior of the flow. From a qualitative perspective, the results of the study are consistent, but the discretization of the containments considerably affects the stability of the results. A coarse nodalization (cases 2x28 an 2x35) leads to the onset of even large oscillations in the profiles of RC pressure (Figure 58) and many other quantities (Figure 59). Such fluctuations have a clear numerical origin and they are dumped by reducing the length of the elementary volumes. However, with a very fine refinement of the grid there is not a convergence of the results, with the case 2x70 showing a discontinuity in the behavior (green curve in Figure 58). This analysis leaves some concerns about the sensitivity of the results to the nodalization of the containments. For cases with coarse nodalization, the elementary volumes might be too large, while in the case 2x70 there can be problems of distortion because of a high aspect ratio. Mass error analysis (Figure 60) reveals acceptable results for cases 2x47 and 2x56. Reported to the total mass inventory, the mass error of the reference case is in the order of 10^{-5}.

3.4.7 Final remarks

This activity has investigated the capability of a submerged reactor to passively cool the fuel rods following a break/depressurization of the primary loop and the operation of the safety injection systems. With a numerical approach, the sump natural circulation flow through the DVIs, the RPV, the ADSs, the RC, the recirculation lines and the ST has been analyzed. The calculations have been performed with Relap5-Mod3.3. The reference case has considered the time 7h30' after the scram as starting point, simulating the behavior of the passive safety systems for 25 hours. To allow exploring also the long-term period within a manageable computational load, two additional simulations, considering an accelerated decay curve and a constant low decay power, have been performed. These cases have investigated the conditions of the systems up to 21 days and several weeks after the scram, respectively. Reasonable and conservative assumptions for the initial conditions have been made.

The study has provided a numerical demonstration of the effectiveness of the sump natural circulation flow at the basis of the passive safety concept. Considering the long-term period, a successful core cooling process has been observed in the reference simulation and in the sensitivity cases. The system benefits of the thermal capacity of the large water inventory stored into the safety tank and of the excellent heat transfer capabilities of the external seawater, acting as an infinite heat sink. A warning about the non-conservativeness of the accelerated curve assumption has been identified, although its effects do not affect the qualitative results of the simulation and are balanced by many other conservative hypotheses. The sensitivity on the nodalization of large containments has presented some issues about the numerical stability of the simulations and the actual capability of Relap5 to simulate large containments. However, results are acceptable for a preliminary analysis. Validation of the numerical models toward specific experimental data will be necessary. In addition, given the importance of the RC pool temperature for the general behavior of the safety concept, a specific study about the flow circulation and heat transfer in the RC pool and in the ST should be performed, e.g., with CFD analysis.

3.5. Summary and next steps

A basic safety strategy for a submerged SMR has been defined, to face non-LOCA and LOCA accident scenarios. The fully passive safety systems exploit the surrounding water as a permanent and constant temperature heat sink. Contributions to the strength of the safety concept have been provided by the analysis of two reference accident scenarios, i.e. the SBO and the long-term cooling, with 1D system code Relap5-Mod3.3. The simulation of a SBO has revealed a good response of the EHRS: natural circulation loops ensure an efficient heat transfer from the core to the external water, keeping the level of liquid in the RPV far above safety margins. However, the failure of one EHRS train and the consequent reduction of the heat transfer area may lead to criticalities, which might require the intervention of other safety systems. Also, the long-term transient simulation after a primary break or depressurization has been performed. The sump natural circulation flow is efficient even when decay power has become very low. The cooling process can rely on a large water inventory and on the infinite heat sink. About the long-term scenario, the sensitivity analysis about the containment discretization has revealed some numerical issues, therefore further assessments, benchmarks and comparison with experimental data are recommended. Experimental campaigns are essential to prove the effectiveness of the safety strategy and to meet the requirements of the authorities. The two proposals in Section 5 would allow creating databases to validate both the SBO and the long-term cooling simulations.

The analysis proposed in this section is a starting point that define the principles of the safety strategy contributes to reinforce the potentiality owned by the submerged SMR concept. Nevertheless, it is not exhaustive and several important aspects still need to be addressed and investigated. Firstly, the transient in sub-section 3.4 starts from 7h30m after the scram, therefore the achievement of controlled state in case of a Small Break LOCA still needs to be defined. Operation of high pressure injection systems needs to be studied for an integral design in submerged containment, since the work by Haratyk & Gourmel (Haratyk & Gourmel, 2015) is referred to a VVER-type configuration. However, the approach of IRIS to face the small break LOCA described by Papini et al. (Papini, et al., 2011) may find quite similar application also for the analysis of the integral submerged SMR. Then, another key point is the efficacy of the EHRS to depressurize the system: simulations have investigated the possibility to reduce pressure only by the effect of the cold heat sink, i.e., without the opening of ADS in the SBO scenario. The results show a large pressure decrease due to the cooling only, but the case with only one SG module available has shown that the EHRS alone may not prevent from a dangerous lowering of the collapsed liquid level. Therefore, the analysis of actuation and operation of depressurization valves is a fundamental step of the safety strategy to manage the SBO transient. Moreover, the safety strategy must be completed in order to include other accident scenarios, such as the SG tube rupture, turbine trip etc.

The last important aspect of the safety strategy that must be addressed regards the Probabilistic Risk Assessment (PRA), derived from the complete definition of Design Basis Accidents (DBAs) the Design Extension Conditions (DEC). Given that the immersion offers inherent protection toward the LUHS, the basic safety strategy of this section has identified the SBO as the main DBA. Then, a parametric study about the halving of the SG heat transfer area has shown that such scenario goes beyond the DBA: to prevent significant fuel degradation, the EHRS only might not be sufficient and other safety feature should be exploited. The opening of ADS and activation of DVI and sump natural circulation represent the back-up strategy in case of failure of the EHRS (DEC-A, accident with no or very limited core damage). Anyway, this aspect needs to be examined in depth in the framework of the Level 1 PRA. The risk assessment of IRIS is an interesting basis for this purpose (Mizuno, et al., 2005), (Maioli, et al., 2005). Concerning the mitigation of situations with large core damage (DEC-B, severe accident, large core degradation), an accurate analysis of the flooded RC should be performed within the Level 2 PRA, in order to assess the efficacy of the in-vessel retention by studying the RC pool natural convection and heat transfer to the exterior. The aim of the analysis should be to prove that the very efficient cooling provided by the external water would ensure the quasi-zero probability of significant radioactivity releases even in case of severe accident.

Publications

The contents of sub-section 3.3 are published in:

Santinello, M., Ricotti, M.E., 2018. Preliminary analysis of an integral Small Modular Reactor operating in a submerged containment. *Progress in Nuclear Energy,* Volume 107, pp. 90-99

The contents of sub-section 3.4 are published in:

Santinello, M., Ricotti, M.E. & Gourmel, V., 2017. *Long-Term Sump Natural Circulation in a Submerged Small Modular Reactor.* Xi'an, China, Proc. of NURETH-17, 3-8 September 2017.

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Section 4

Relap5 - Apros comparison

The simulation of the long-term cooling scenario with the system code Apros 6 and the comparison with Relap5-Mod3.3 are described in this section. The activity has been performed in cooperation with VTT – Technical Research Center of Finland, the developer of Apros code and one of the owner of the commercial property (with the Finnish energy company "Fortum"). To address the topic, a preliminary analysis on a simple two-phase flow test case, i.e. the steam generator modeled in paragraph 2.3.7, has been done. The results have revealed large discrepancies between the two codes in the calculation of tube side HTC in nucleate boiling, transition and post-dryout regimes. Apros 6 presented very high HTC values during bulk boiling, up to four times higher than Relap5 results. This outcome seemed quite odd and has stimulated VTT to improve the two-phase modeling structure of Apros 6 code, fixing some issue in the HTC calculation. Predictions of the new version of Apros 6 show a better agreement with those of Relap5, although some differences remain about the position of the thermal crisis. The main test case, i.e., the long-term cooling scenario analyzed in sub-section 3.4, aims at obtaining a term of comparison for the safety analysis of the submerged SMR performed with Relap5. Like Relap5, also Apros 6 predicts the successful operation of the passive safety systems, without risk of coolant overheating. However, the thermal-hydraulic behavior identified by Apros 6 is quite different from the outcome of Relap5, especially considering the response to the initial conditions.

4.1. Overview

4.1.1 Motivation

The analysis of the long-term natural circulation cooling with Relap5, performed in sub-section 3.4, provided a numerical verification of the feasibility of the long-term cooling through sump natural circulation, but it also revealed some issues concerning the numerical stability of the results. The main reason is the high complexity of the physics under examination. Two physical reasons make the modeling and simulation particularly challenging: (i) the density gradient into the core, which drives the natural circulation flow, is due to boiling of a small amount of coolant, making the fluid working in a complex flow regime; (ii) the steam produced into the core condenses in contact with the RC wall, in an atmosphere full of non-condensable gas. In addition, the shape of the containments, a horizontal cylinder, is quite uncommon for nuclear systems, posing difficulties to optimize the discretization. Lacking specific experimental data for model validation, these modeling problems weaken the conclusions of sub-section 3.4. For this reason, comparing the results with those of another system code can reinforce the safety analysis in case of agreement of the predictions.

In the framework of a cooperation between Polimi and VTT/Fortum, an analysis of the safety scenario with the code Apros 6 has been conducted. Apros is a commercial simulation software product for full-scale modelling and dynamic simulation of industrial processes. The code has been created and developed by VTT Technical Research Center of Finland. The property is shared by VTT and Fortum, the Finnish leading energy company. A description about Apros features is given in Appendix A. The target of the investigation is to strengthen the safety analysis of the long-term cooling scenario by comparing the predictions of Relap5-Mod3.3 with the outcome of Apros 6. Performing a complete benchmark of the two codes is out of the scope of this activity, although several elements that could motivate the differences between the results have been evidenced and analyzed. However, VTT and Fortum are working to develop a simulation tool specific for SMRs, based on Apros. A comparison with Relap5, the reference code for nuclear safety analysis, is therefore very important and the present work can represent a good starting point.

4.1.2 Approach

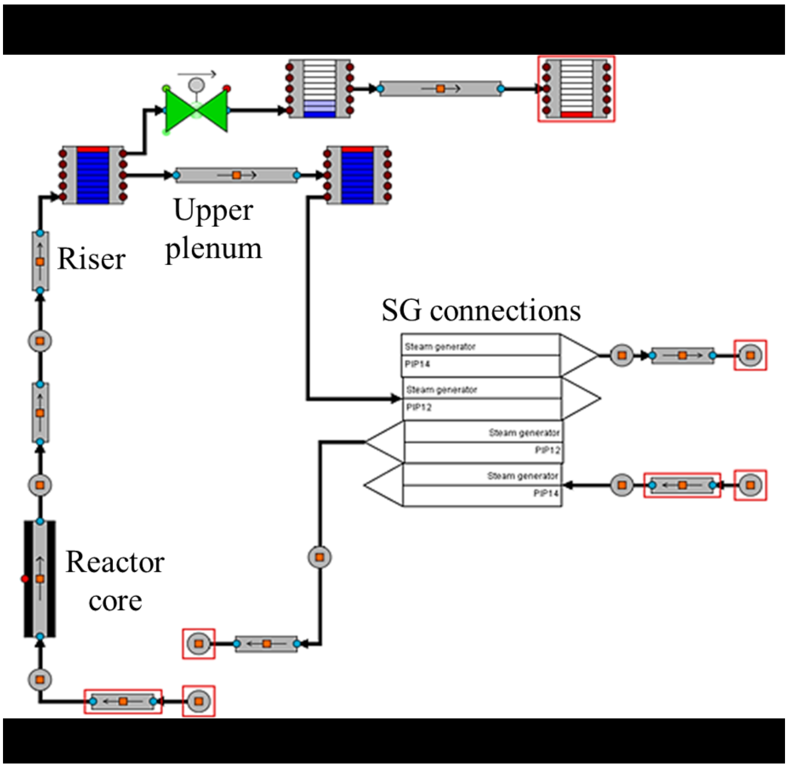
With no experimental data available, validating the predictions of the numerical analysis about the long-term cooling is not possible. However, the simulation of the analogous system with another system code offers a term of comparison and allows identifying the most important sources of uncertainty. The thermal-hydraulic problem under investigation has been approached with the system-code Apros 6, focusing the attention on the modeling of the RC and ST. The modeling and discretization of RPV and containments has been performed at the best performances of Apros 6 code in terms of nodalization capabilities and empirical correlations. This means that Apros 6 models have been optimized exploiting all the features available in the software. In other words, the nodalization in Apros 6 models do not exactly reproduce the Relap5 one, but it addresses the physical problem disregarding the assumptions and the limitations of the previous activity. This operation is necessary to compare Relap5-Mod3.3 and Apros 6 separately, as if the user tackles the same physical problem with the two codes independently. However, the component models available in Apros 6, e.g. for heat exchangers and containments, have not been used, employing elementary components only, i.e. nodes and branches. The model specific reactor containment has not been adapted to this case also because it calculates heat transfer from a vertical geometry only.

4.2. Preliminary analysis: a simple test case

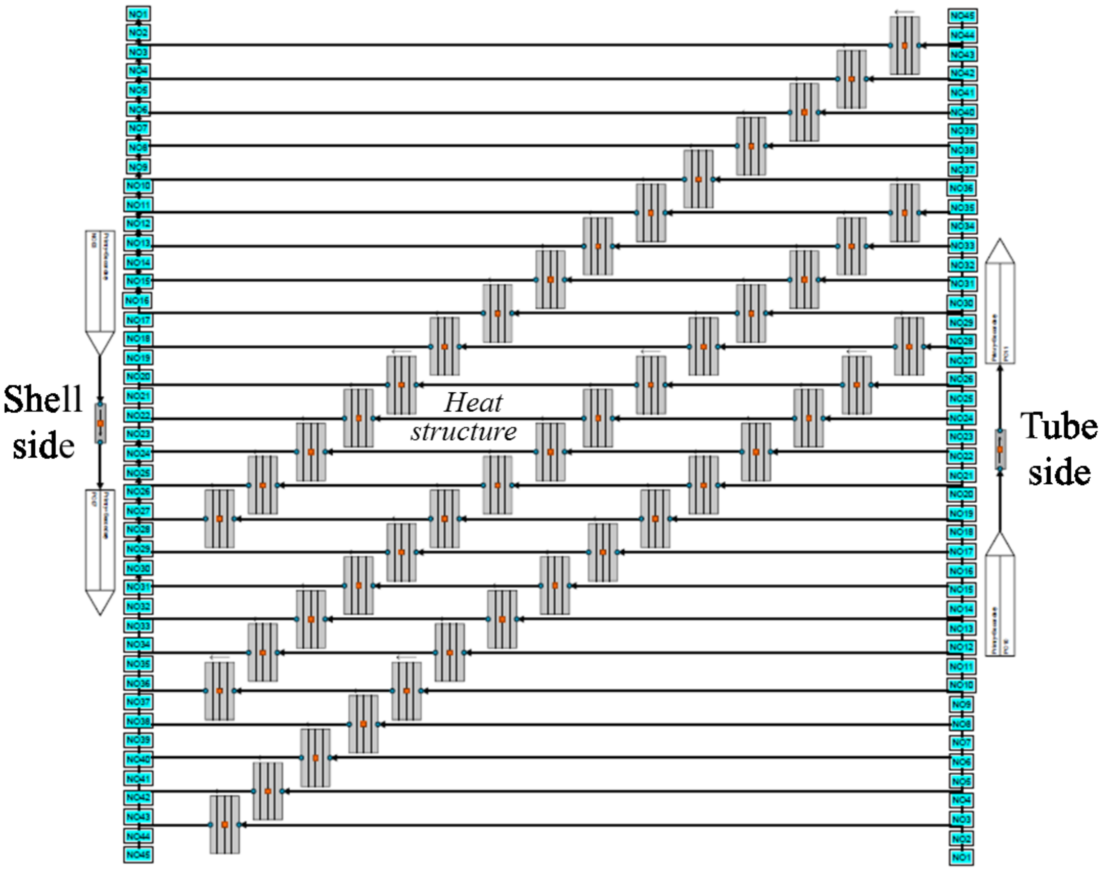
4.2.1 Model of IRIS-like steam generator in open circuits

Good practice suggests starting a comparison between two codes with a simple case and with simplified models. For this purpose, the IRIS-like steam generator studied in paragraph 2.3.7 is considered as reference case, while the fluid flow has been preliminarily modeled as a homogeneous mixture. The simulated system is made of two open circuits. In addition to the SG shown in Figure 21, the primary side include also a heated pipe, which simulates the reactor core, and another small circuit, which simulates the pressurizer and helps stabilizing the conditions at the inlet of the SG. The complete model is shown in Figure 61 (Apros graphics). System pressures, inlet temperatures, mass flow rates and core heating power have been set considering operating conditions similar to the SBO case in sub-section 3.3. Data are given in Table XVI. Also, the nodalization of the core, pressurizer and piping has been done like in the SBO simulation. Despite the fixed thermal power, the SG is working with imposed temperature regime, because the primary loop is open.

Based on the “best performance” approach described above, the nodalization of the SG is not equal for Relap5 and Apros models. In Apros, the SG tube has been subdivided into 45 elementary elements, instead of 160 as in Relap5. These values have been confirmed with a sensitivity analysis, showing the grid independency of the results. Each element of the shell side is thermally coupled to the tube side by the heat structure (Figure 61b).



(a) Circuit model



(b) Steam generator model

Figure 61 Apros 6 model of the IRIS-like preliminary case (Apros graphical interface)

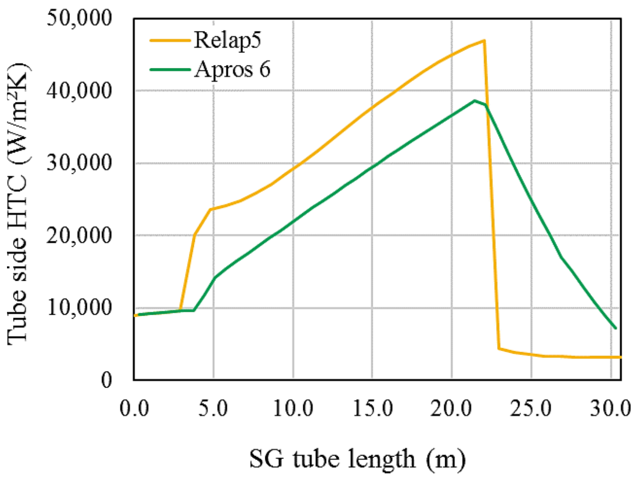
Table XVI. Operating conditions for IRIS-like preliminary case

	Primary	Secondary
Inlet temperature	297°C	212°C
Pressure	15.5 MPa	6.2 MPa
Mass flow rate	2250 kg/s	251.47 kg/s
Core heating power	500 MW	

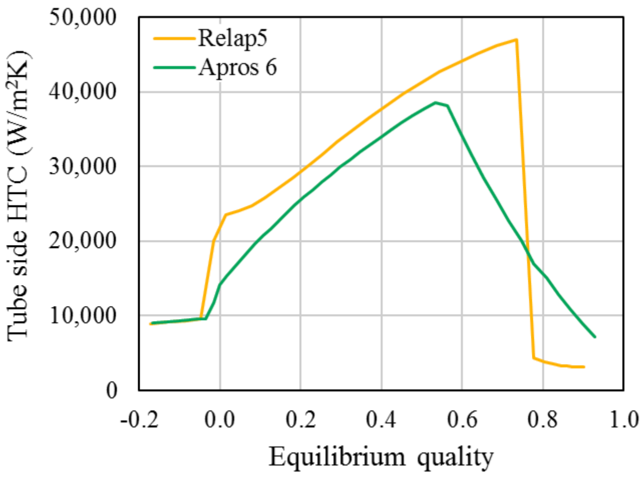
4.2.2 Homogeneous model

The Homogeneous Equilibrium Model (HEM) is a basic approach to deal with two-phase flow in both Relap5 and Apros. It is a three-equation model, where continuity, momentum and energy 1D equations are solved considering one fluid only. The fluid mixture has a single velocity and physical properties are obtained by average of the liquid and the vapor phases. Momentum transfer and friction between the phases are not considered. Phenomena with transvers gradients, such as friction and wall heat transfer, are modeled with empirical correlations. Currently, the homogeneous model is considered obsolete for simulation of a two-phase flow (Information Systems Laboratories, Inc., 2003). However, its simplicity can offer a good starting point for a code comparison.

The results of the simulations with the HEM already present considerable discrepancies. Focusing on the secondary side of the SG, the HTC profiles (Figure 62) show different qualitative trends. In the simulation with the homogeneous model, Relap5-Mod3.3 resolves the thermal crisis at quality x ≈ 0.75 with a sudden fall of the heat transfer, while Apros 6 has the peak at x ≈ 0.55 and then a decrease with a much lower slope. Discrepancies are observed also in the quality increase along the tube coordinate (Figure 63). The profile of Relap5 simulation clearly shows a change of the derivative in the point of the thermal crisis, while this is not visible in Apros 6. The inlet-outlet temperature difference in the SG shell side is greater for the Apros 6 case, because of higher average HTC (Figure 64). This results in a higher value of power transferred.



(a)



(b)

Figure 62 Tube side HTC profiles given as a function of the tube length (a) and steam quality (b) *Homogeneous model*

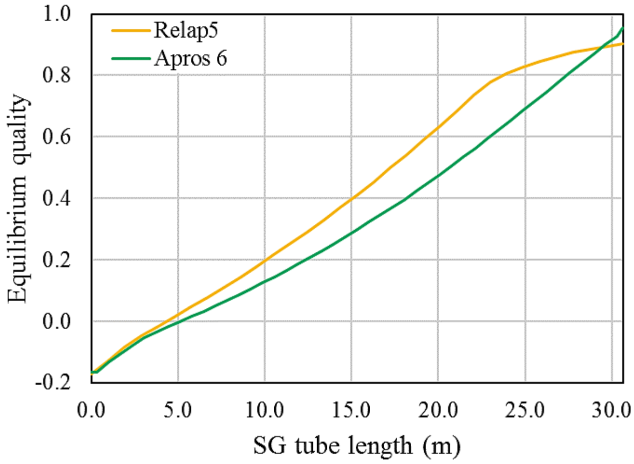


Figure 63 Equilibrium quality on SG tube side along the tube coordinate. *Homogeneous model*

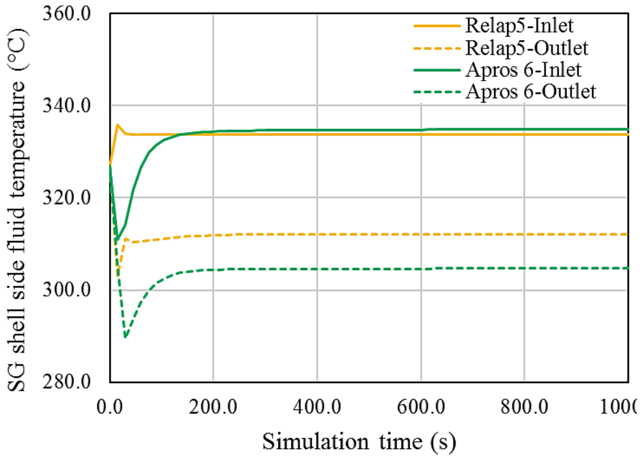


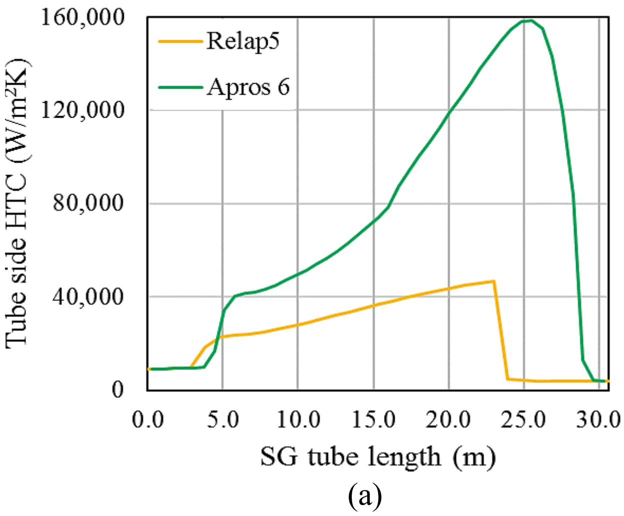
Figure 64 Inlet-outlet temperature of the fluid in SG shell side. *Homogeneous model*

The first consideration arising from these results is that Apros 6 and Relap5 may have quite large structural and modeling differences. Even on a simple test case with homogenous model, the prediction of heat transfer rate and thermal crisis shows discrepancies not only in the quantitative values, but also on the shape of the profiles. This means that, when applying more advanced models and simulating more complex physical phenomena, large differences between results are expectable.

4.2.3 Two-fluid model

The two-fluid model is based on the application of one-dimensional conservation equations of mass, momentum and energy for the liquid and gas phases, which are treated separately. Therefore, the model identifies different velocities for liquid and gas. Six partial differential equation are solved at each time step. The gas equations may contain also the terms that represent the non-condensable mass fraction. Coupling of conservation equations is made by constitutive relations that model friction and heat transfer at the wall and between phases. These relations are mainly based on empirical models. Based on flow conditions, the model identifies flow regimes, which determine the empirical terms.

The switch to the two-fluid model generates only small quantitative changes on the Relap5-Mod3.3 results. On the contrary, it has a large impact on the HTC profile (Figure 64) of Apros 6: the thermal crisis quality is now in agreement with the prediction of Relap5 at $x \approx 0.75$, but the quantitative value of the HTC reaches much higher values with respect to the homogeneous model and Relap5 prediction. Apros 6 HTC is up to nearly four times higher than Relap5 one in the bulk boiling region, just before the drop due to the thermal crisis. Such values are quite unrealistic. Differences are visible also in in quality and temperature profiles (Figure 65 and Figure 66). The origin of such large discrepancies is not clear, but it is likely that two key factors are: (i) the different heat transfer mode logic and (ii) the correlations that are included in the two codes and the way they are implemented/interpolated.



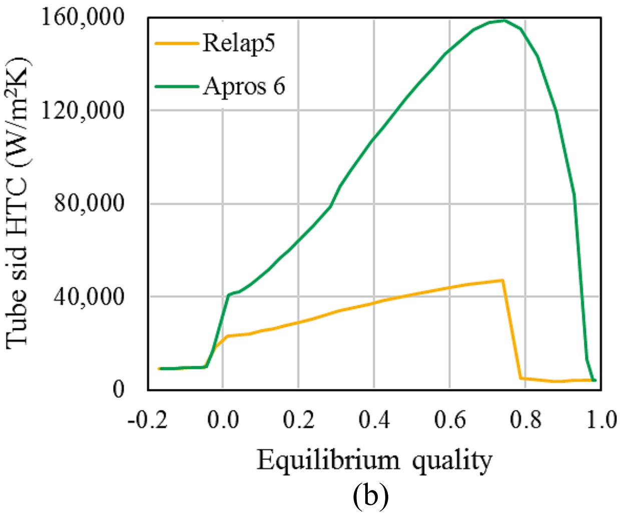


Figure 65 Tube side HTC profiles given as a function of the tube length (a) and steam quality (b) *Two-fluid model*

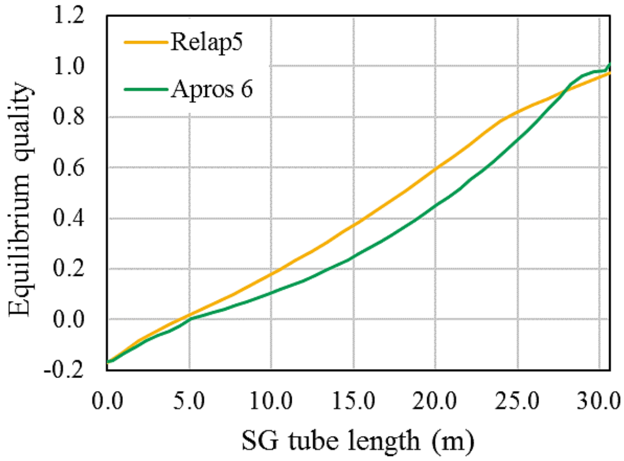


Figure 66 Equilibrium quality on SG tube side along the tube coordinate. *Two-fluid model*

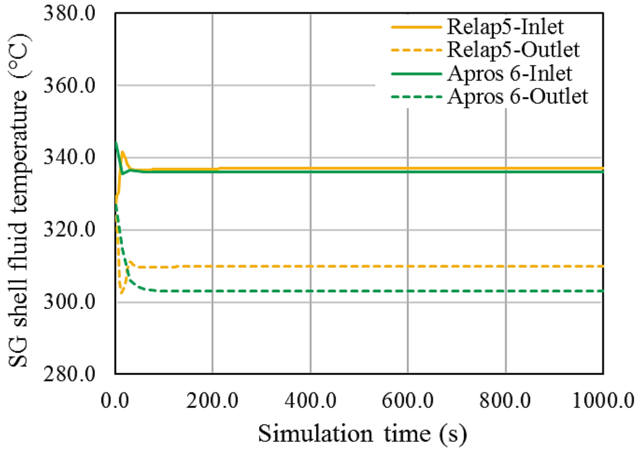
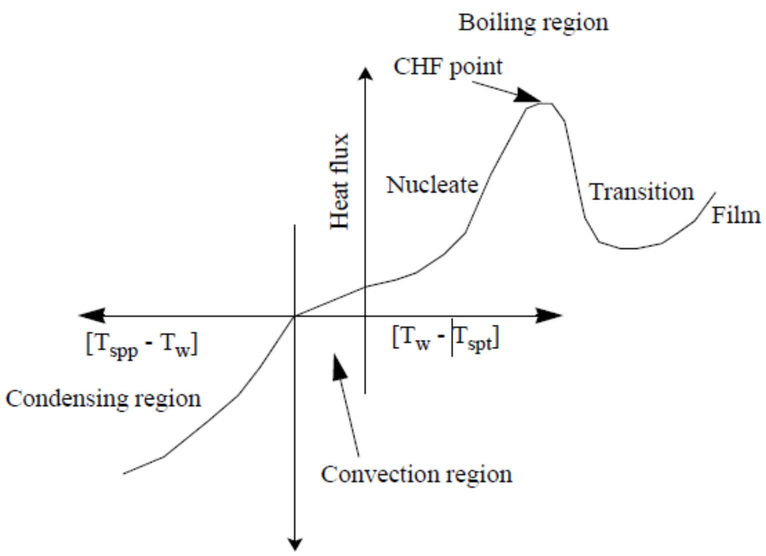


Figure 67 Inlet-outlet temperature of the fluid in SG shell side. *Two-fluid model*

Relap5 calculation of the two-phase flow HTC is based on heat transfer mode charts. Fundamentally, Relap5 uses values of the fluid and heat structure, such as pressure, void fraction, wall temperature etc., to distinguish among five basic heat transfer "modes": single-phase liquid, nucleate boiling, transition regime, film boiling and single-phase vapor. In addition, models for condensation, non-condensable convection and supercritical fluid are included. The selected mode calls a routine that implement the desired empirical correlation. On the other side, Apros identifies three heat transfer "zones": if wall temperature is lower than fluid saturation temperature, the wetted wall zone is identified, else, if the void fraction is very close to one and/or the calculated heat flux exceeds the CHF, a dry wall zone is assumed. A transition region is considered if wall temperature is higher than saturation and heat flux is below the CHF. The two logics are shown in Figure 68.



$$
\begin{aligned}
&\textit{if } \alpha > 0.99999,\ z = 3\\
&\textit{else if } T_w \le T_{sat},\ z = 1\\
&\textit{else if } z_{old} > 1 \,\&\, T_w \le T_{MFB},\ z = 2\\
&\textit{else if } z_{old} > 1 \,\&\, T_w > T_{MFB},\ z = 3\\
&\textit{else if } q_w > q_{cr} \,\&\, T_w \le T_{MFB},\ z = 2\\
&\textit{else if } q_w > q_{cr} \,\&\, T_w > T_{MFB},\ z = 3\\
&\textit{else},\ z = 1
\end{aligned}
$$

(a) Relap5
(Information Systems Laboratories, Inc., 2003)

(b) Apros 6
(Hänninen, et al., 2012)

Figure 68 Heat transfer mode selection logic in Relap5 (a) and Apros 6

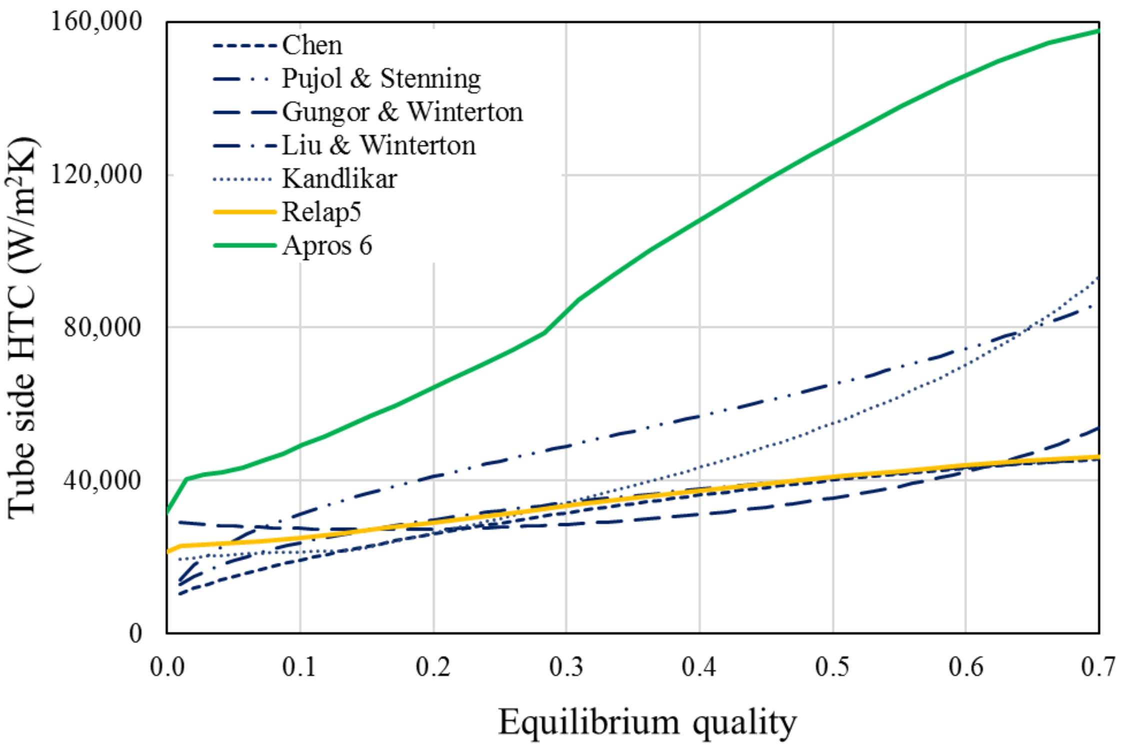


Figure 69 Comparison among the two-phase HTC predicted by Relap5, Apros 6 and some empirical correlations available in literature.
Correlations references: (Chen, 1966) (Wambsganss, et al., 1993) (Gungor & Winterton, 1986) (Liu & Winterton, 1991) (Kandlikar, 1990)

As far as heat transfer empirical correlation are concerned, once again the two codes behave differently. Relap5 uses a different correlation for HTC in each mode and, adopting the Chen correlation (Chen, 1966) for the nucleate boiling and the Chen-Sundaram-Ozkaynak correlation (Chen, et al., 1977) for transition boiling. Apros 6 relies on the Thom correlation (Thom, et al., 1966) for nucleate boiling, but for the transition zone, the approach to heat transfer calculation is different. It is based on an interpolation between the critical heat flux and the heat flux of the dry zone. The wall temperature corresponding with the critical heat flux is estimated assuming that the heat flux of the nucleate boiling is equal to the critical heat flux. Then, the heat flux is distributed to the phases (Hänninen, et al., 2012). Hence, Relap5 and Apros have very different ways to calculate the heat transfer rate, especially in the transition zone. In addition, Apros performs an interpolation and therefore the results depend also on the way it is done.

Given all these differences in the approaches to heat transfer calculation, it is expectable that the results of the two codes present some discrepancy. In general, two-phase heat transfer problems have to deal with quite large uncertainties, since the theoretical knowledge of such physical phenomena is still incomplete and experimental measuring often encounters technical difficulties, which increase the uncertainty of final data. However, in this case the discrepancy is huge: the HTC profile predicted by Apros 6 reaches very high values, up to 160,000 W/m^{2}K, which seem unrealistic with respect to typical experimental data. The comparison with the HTC predicted by some correlations available in literature is shown in Figure 69. While Relap5- Mod3.3 profile is aligned with most empirical correlations, Apros 6 profile is clearly out of range. Such results have been presented to Apros developers in VTT, stimulating a deep analysis on the routines of the two-phase flow. VTT found an issue in the calculation of the HTC under some conditions and provided a preliminary corrected version of Apros 6. Results of the new version are shown in Figure 70 and Figure 71: HTC predictions are more similar to those of Relap5. However, differences remain about the position of the thermal crisis (Figure 70a), which seems to occur near the outlet of the tube, at x ≈ 0.92.

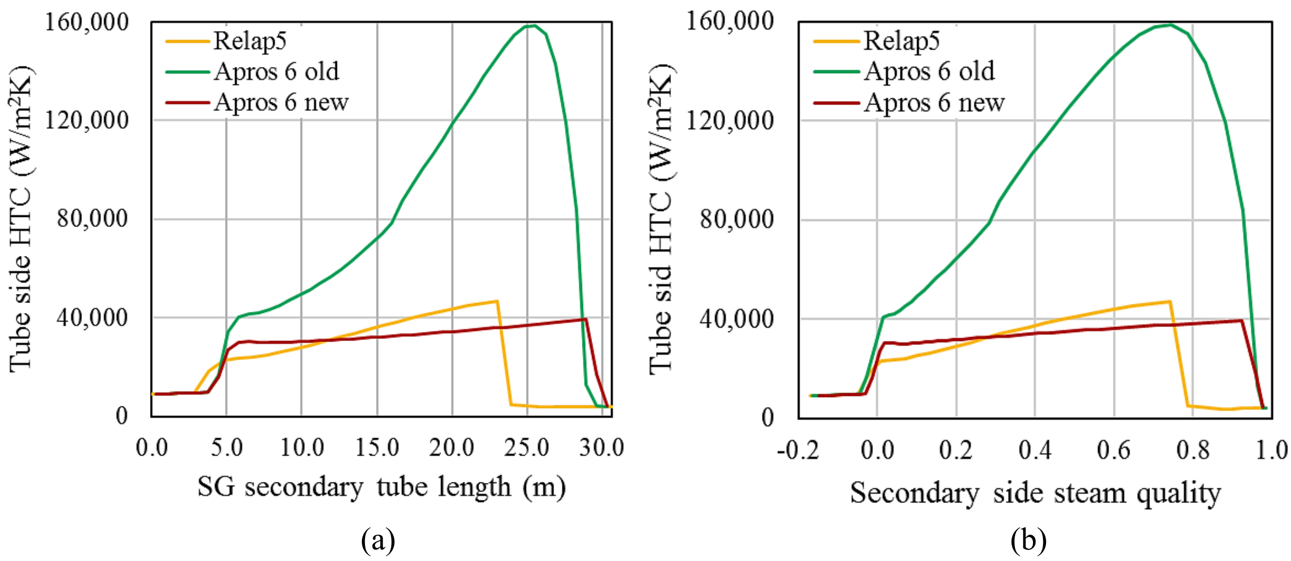


Figure 70 Tube side HTC profiles given as a function of the tube length (a) and steam quality (b) *Two-fluid model – old and new Apros 6 versions*

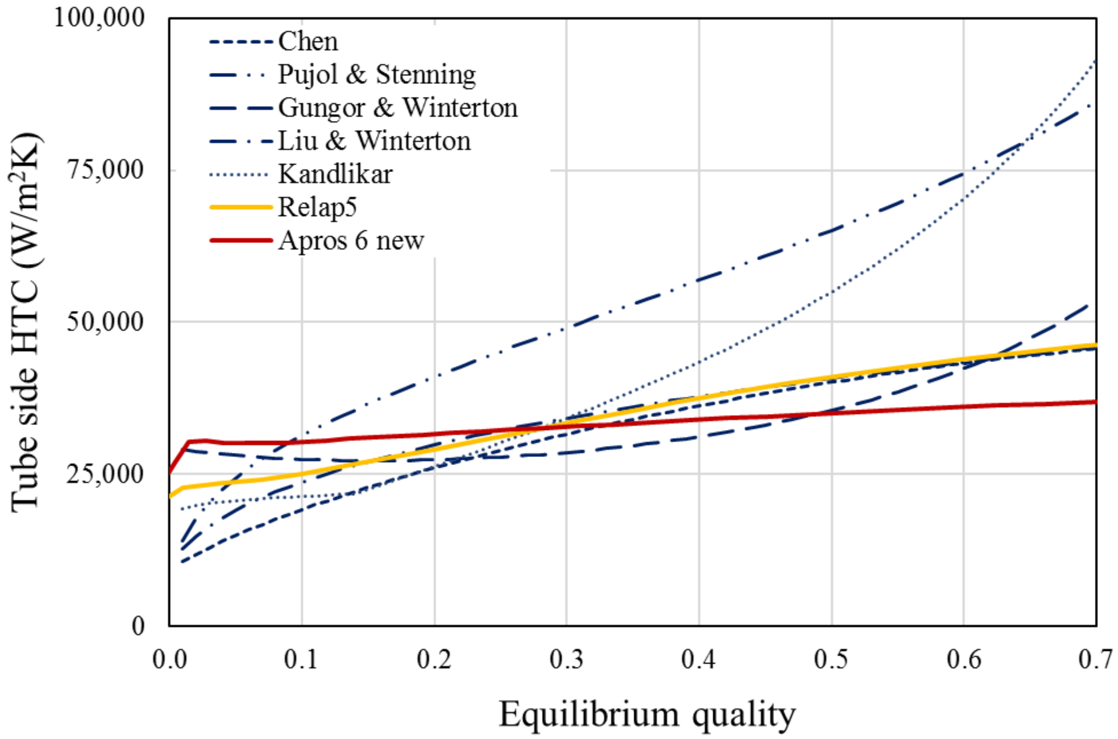


Figure 71 Comparison among the two-phase HTC predicted by Relap5, Apros 6 new version and some empirical correlations available in literature (ref. as in Figure 69)

In conclusion, this preliminary benchmark on a simple test case has evidenced remarkable discrepancies between Relap5 and Apros 6. Even deducting the issue found on HTC calculation and considering the results of the new Apros version, differences remain in the prediction of the behavior of the steam generator. In light of these results, it is expectable a priori that the simulation of a much complex scenario, involving also natural circulation and wall condensation in presence of a non-condensable gas, would evidence even larger discrepancies between the results.

4.3. Analysis of the long-term cooling case with Apros 6

4.3.1 Modeling

Although Apros 6 contains several models for heat exchangers and specific components, the nodalization has been done considering elementary components only. Apros 6 has a model created to simulate large containments, but it has not been used since the wall heat transfer is implemented for vertical plate/cylinder only. Although this step of the investigation is independent of the previous activity with Relap5, some commons criteria for the spatial discretization of the containments have been followed. Firstly, the RPV nodalization is analogous to that made with Relap5 (Figure 72). Then, the modeling of the containment with Apros 6 elementary components relies on the same sliced approach used in Relap5, as explained in paragraph 3.4.3. The reference case has 56 nodes for the RC and for the ST, both subdivided into two columns, i.e. 2x28 nodes (Figure 73). The height of each node is 0.5 m. Nodes are connected by branches in cross flow and vertical directions: this configuration is required to simulate flow recirculation. Unlike Relap5-Mod3.3, where the shape of a node can be defined by only length, flow area and hydraulic diameter, Apros 6 allows specifying the flow area or the volume of the node as a function of the vertical coordinate. Thanks to this feature, it is possible to obtain a much more detailed definition of the horizontal cylindrical geometry of the containment, even with a lower number of calculation nodes. Each node is connected also to its heat structure, an inclined plate that simulates the convective and conductive heat transfer from the internal fluid to the external heat sink. It represents the 6-cm steel containment. Heat structure geometry is characterized with length, breadth, inclination and thickness.

Thermal and hydraulic boundary conditions in Apros 6 modeling have been taken identical to those of the Relap5 activity. The real decay curve has been considered for the volumetric power source in the core. The heat transfer coefficient profile on the external side of the containment is discretized into seven zones, as shown in Figure 49. Initial conditions are the same as those listed in Table XIV, except for the initial non-condensable mass fraction. That parameter has been set to 97.5% instead of 100%, since the convergence of a simulation starting with totally dry air was quite problematic with Apros 6. This assumption is believed to have very low impact from a physical viewpoint, but the difficulty to find convergence with non-condensable gas only in the RC atmosphere poses a warning about the numerical aspect of the simulation.

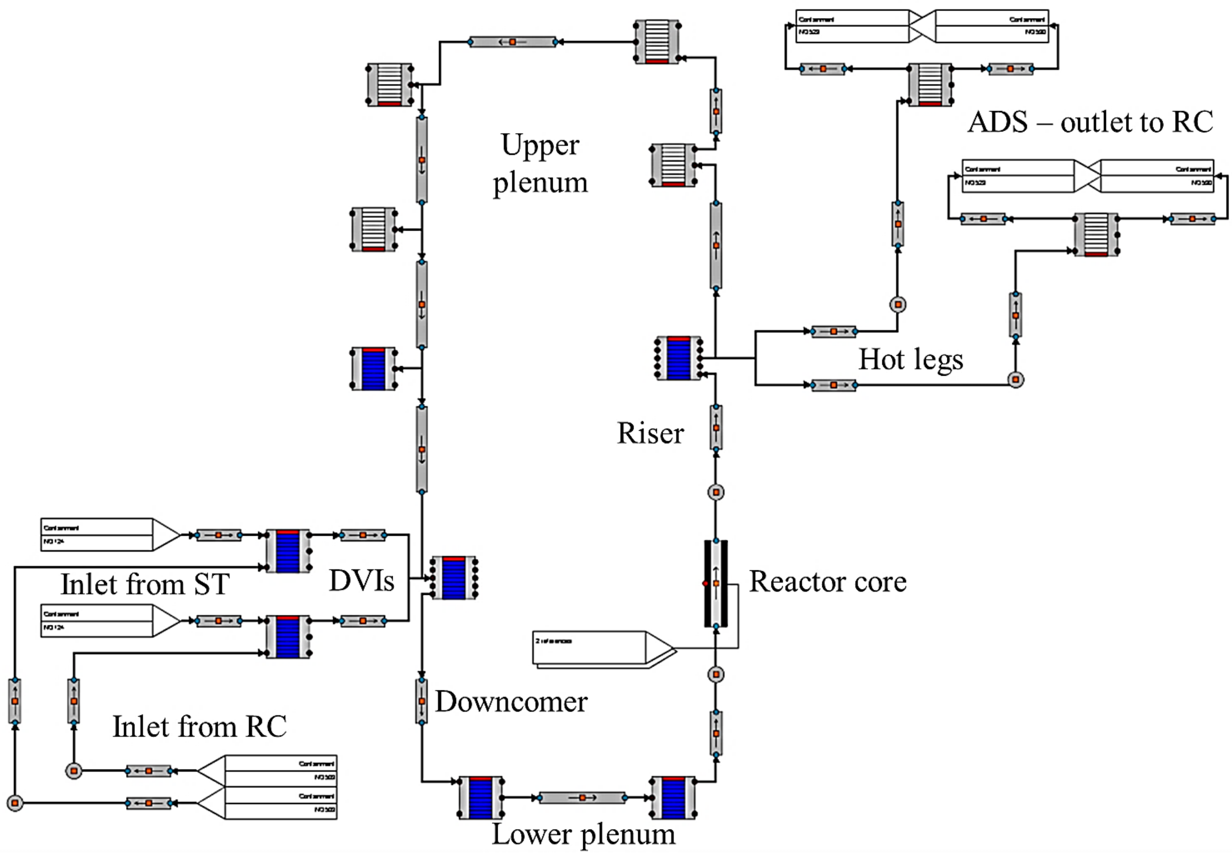
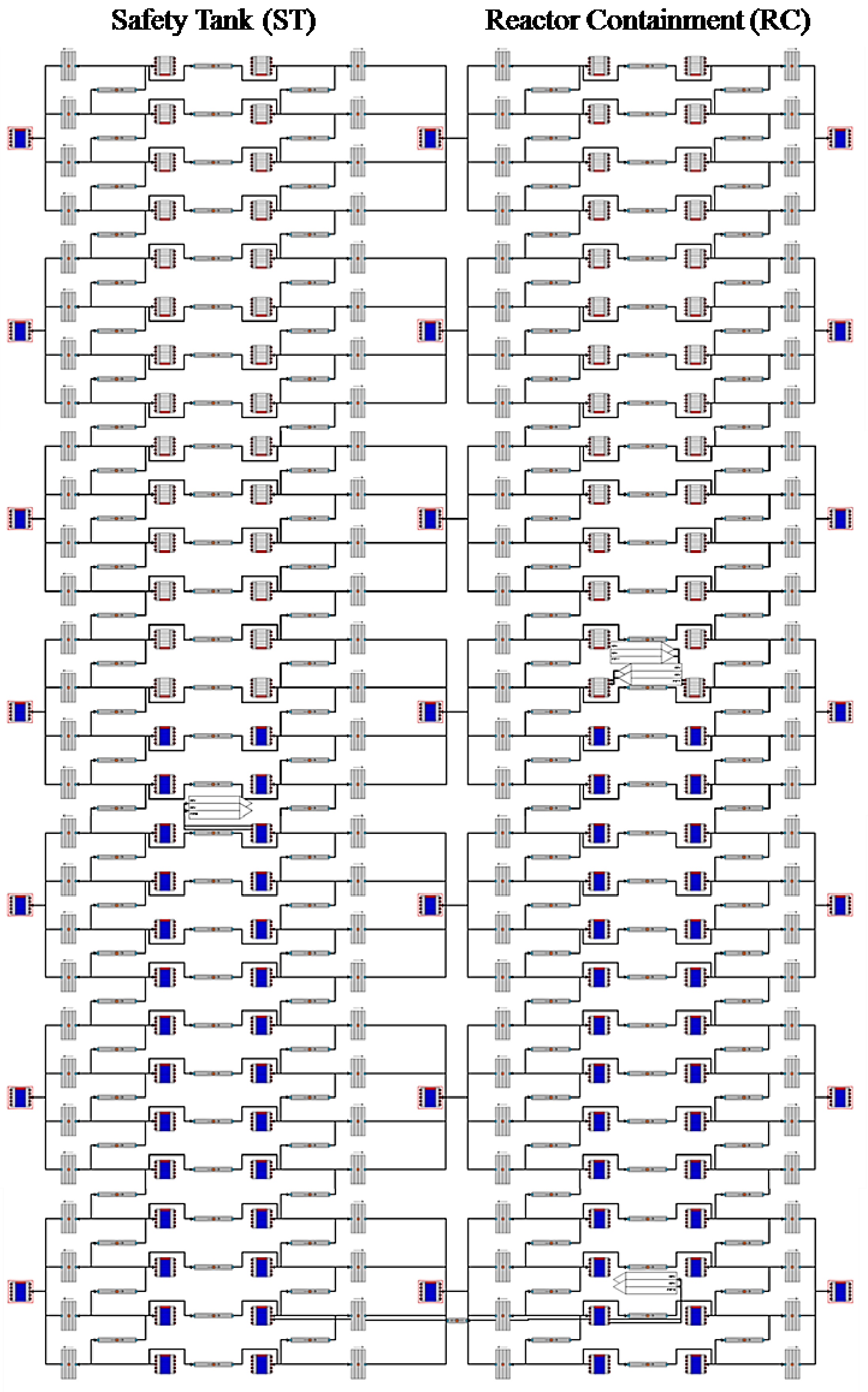
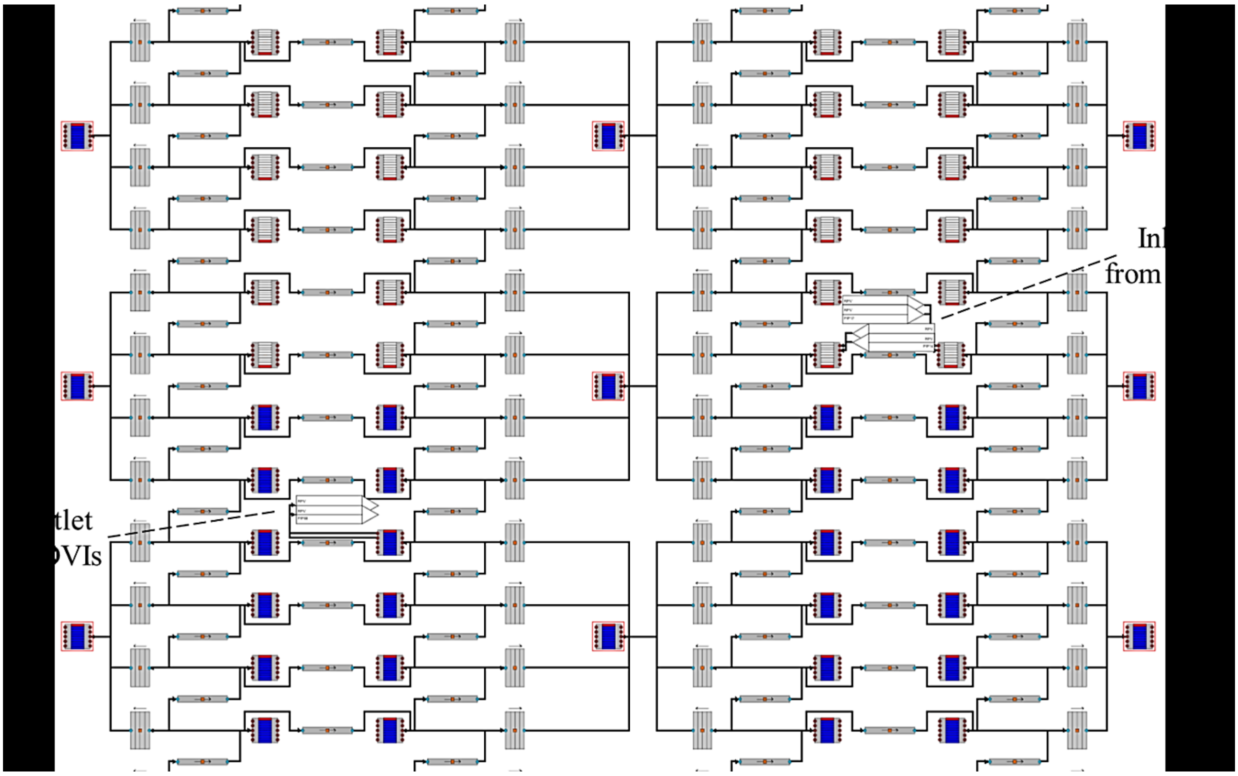


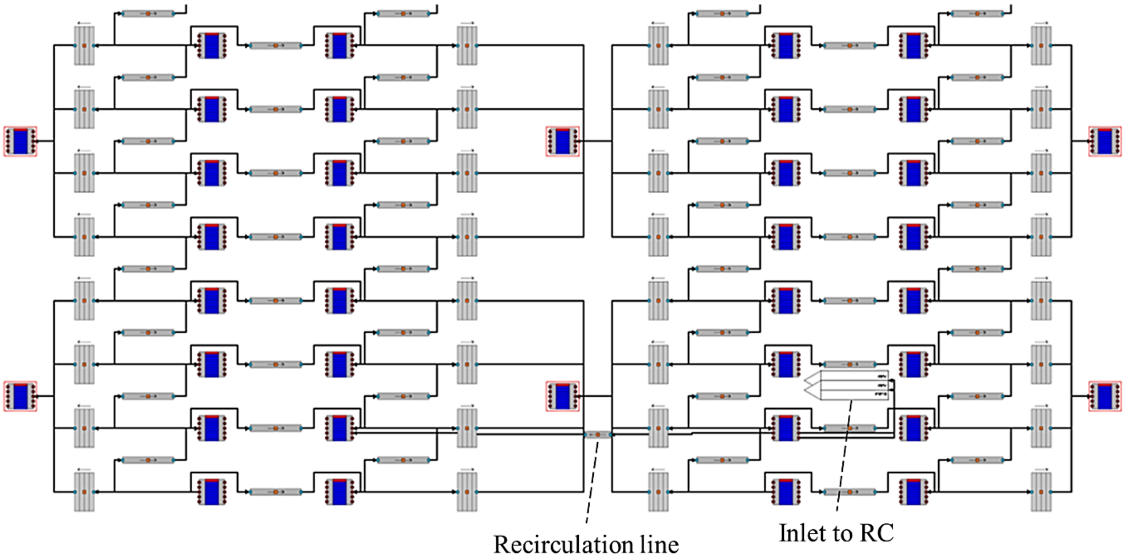
Figure 72 RPV model with Apros 6



(a) Full view



(b) Outlet to DVIs and inlet from ADSs



(c) Recirculation line and inlet to RC

Figure 73 Reactor Containment and Safety Tank model with Apros 6, full view (a) and details of connections with RPV (b) - (c)

4.3.2 Results

The results shown in this paragraph have been obtained with the fixed (new) version of Apros 6. The transient has been extended to 180,000 s, keeping the decay power constant in the second part of the simulation, since during the simulation it has been observed that after 90,000 s the behavior of the natural circulation flow was still evolving. The constant decay power has been determined by software needs, but anyway it is a conservative assumption.

The Apros simulation of the long-term sump natural circulation evidences that, under the given conditions, the system is able to remove the decay heat through natural convection: the power transferred to the exterior becomes greater than decay power after roughly 70,000 s (Figure 74). The void fraction profile at the outlet of the core is always quite low (Figure 75), indicating that there is a small production of steam in the core. Basically, the behavior of the flow during the transient can be subdivided into three stages and is driven by the temperature of the RC pool (Figure 76).

stage i. Up to nearly one third of the total simulated time, the production of steam into the reactor core is very small, the fluid is in nucleate boiling and void fraction is very low. In the meantime, the temperature of the RC pool is rapidly increasing, sign that the system is accumulating heat. This is verifiable by observing that in this period the total rejected power (red curve in Figure 74) is lower than the decay heat produced by the core. The temperature of the flow at the RPV outlet is few degrees below the saturation point (Figure 77). Under these conditions, nucleate boiling is possible, therefore the void fraction is around 1-2%.

stage ii. A much more consistent production of steam begins. The cause is probably a sort threshold in the temperature of the coolant at the inlet of the RPV. The water temperature in the DVIs depends on the mixing between the flows coming from the RC and ST. When this temperature is sufficiently high, the flow at the outlet of the core is two-phase. The void fraction profile is irregular up to roughly 100,000 s. The large amount of steam generated flows into the RC and increases its pressure. This fact originates a general increase of recirculation flowrates (Figure 78), which increase also the heat transfer to the exterior and the mixing inside and between RC/ST pools.

stage iii. The production of steam stabilizes: the void fraction profile reaches a constant value, at the net of fluctuations. The pool temperature also tends to stabilize and the heat transfer to the exterior fins in equilibrium. The flow at the outlet of the RPV is a mixture of steam and liquid at saturation point. Steam is released in the containment and condenses in contact with the wall, although the presence of a large amount of non-condensable gas makes condensation quite slow. The condensate is collected by the RC pool. This third stage is similar to the conditions observed in Relap5.

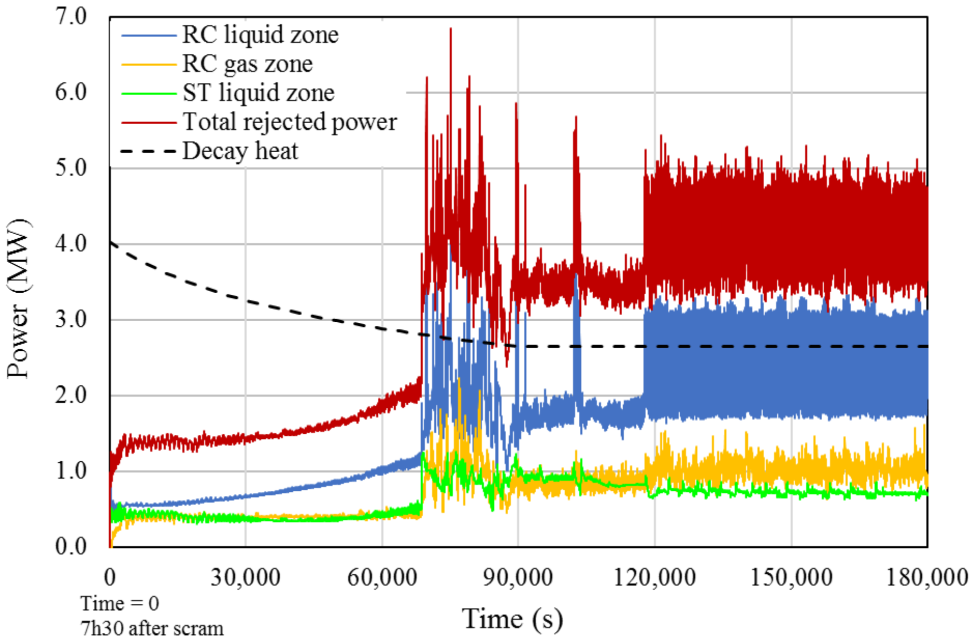


Figure 74 Heat transfer to the exterior compared with the decay power

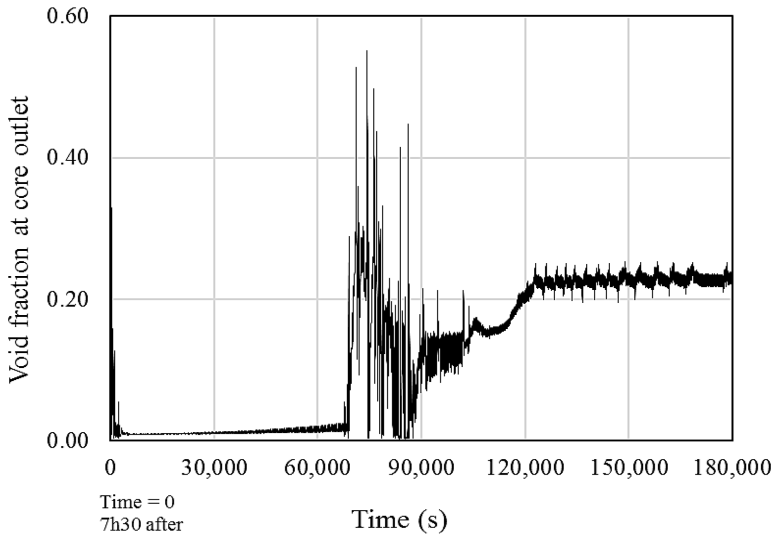


Figure 75 Void fraction profile at core outlet

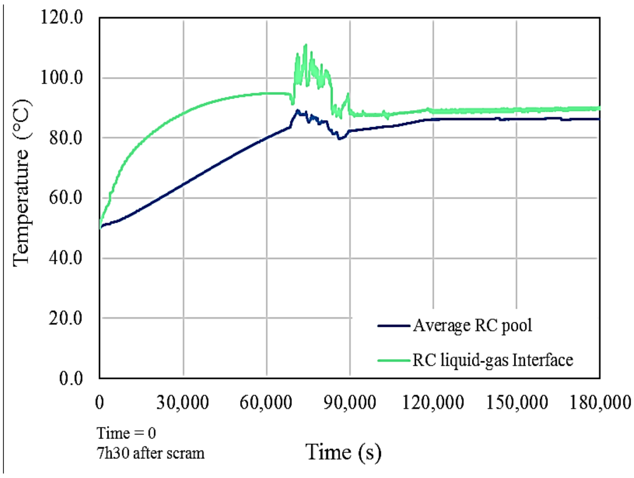


Figure 76 RC pool temperature profiles

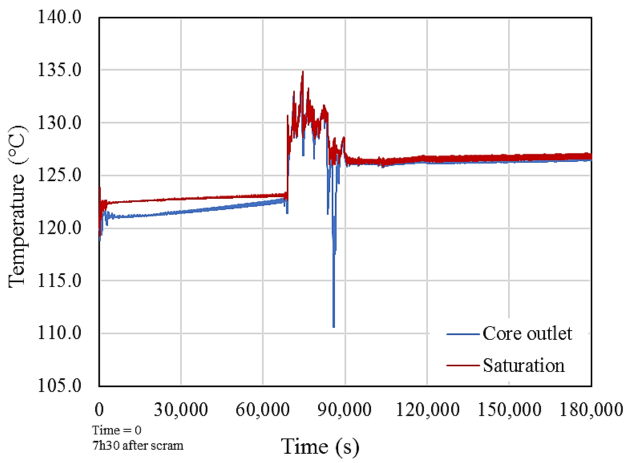
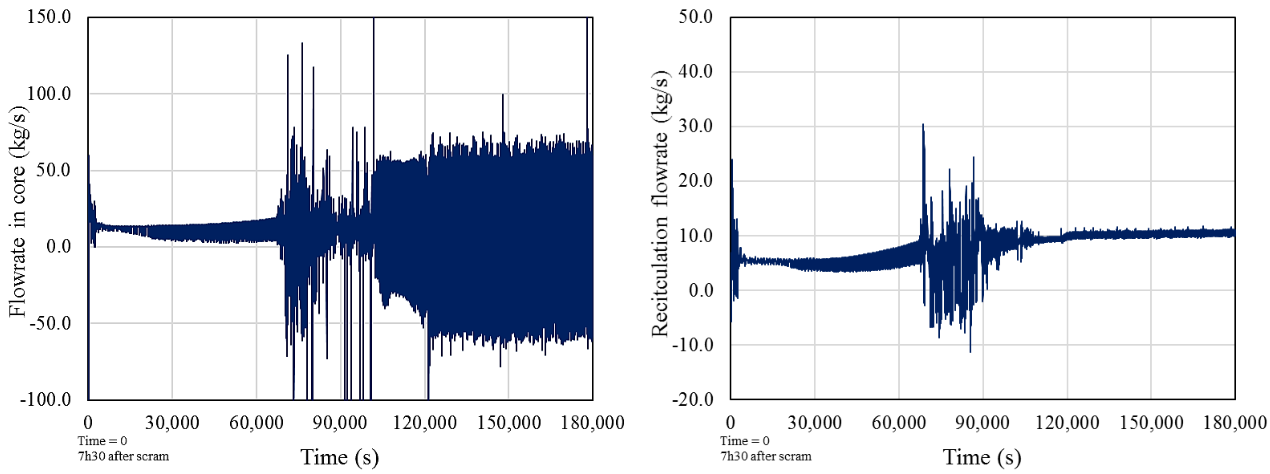


Figure 77 Coolant temperature at the outlet of the core, compared with saturation



(a) (b)

Figure 78 Flowrates profiles in the core (a) and from the RC to the ST (b)

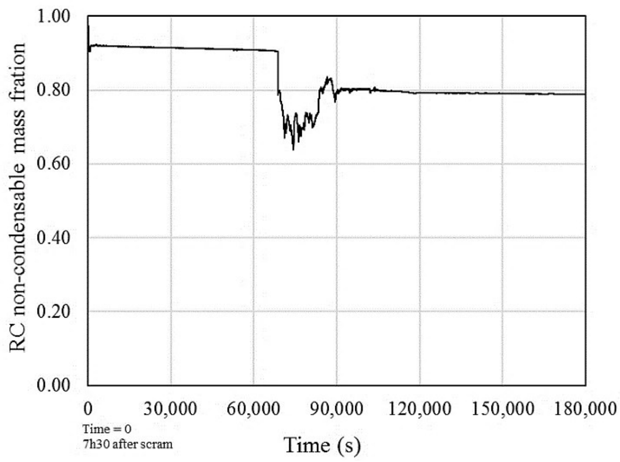


Figure 79 Profile of non-condensable mass fraction in RC

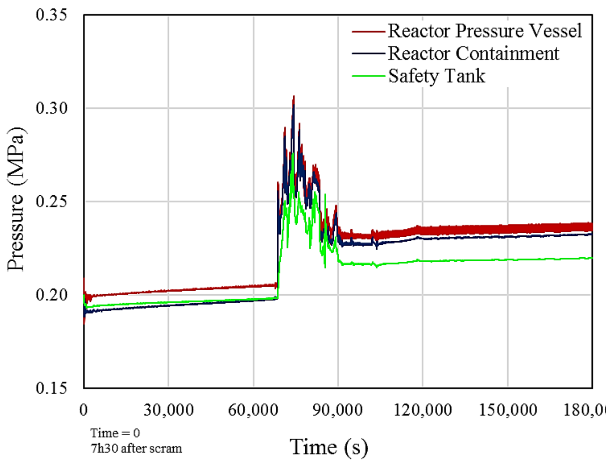


Figure 80 Pressure profiles in RPV, RC and ST

Heat transfer to the exterior occurs through wall condensation and liquid-liquid heat transfer. The latter results much more efficient and gives the largest contribution (Figure 74), especially considering that the external heat transfer coefficient is higher on the upper part of the containment. This is due to the conservative initial condition of the RC atmosphere, which is full of nitrogen. The high mass fraction of non-condensable gas (Figure 79) is a strong limit the heat transfer, since even small amounts can strongly hinder the condensation. Pressure profiles in RPV, RC and ST (Figure 80) reflect the presence of three different behavior of the natural circulation flow and justify the direction of the recirculation flow.

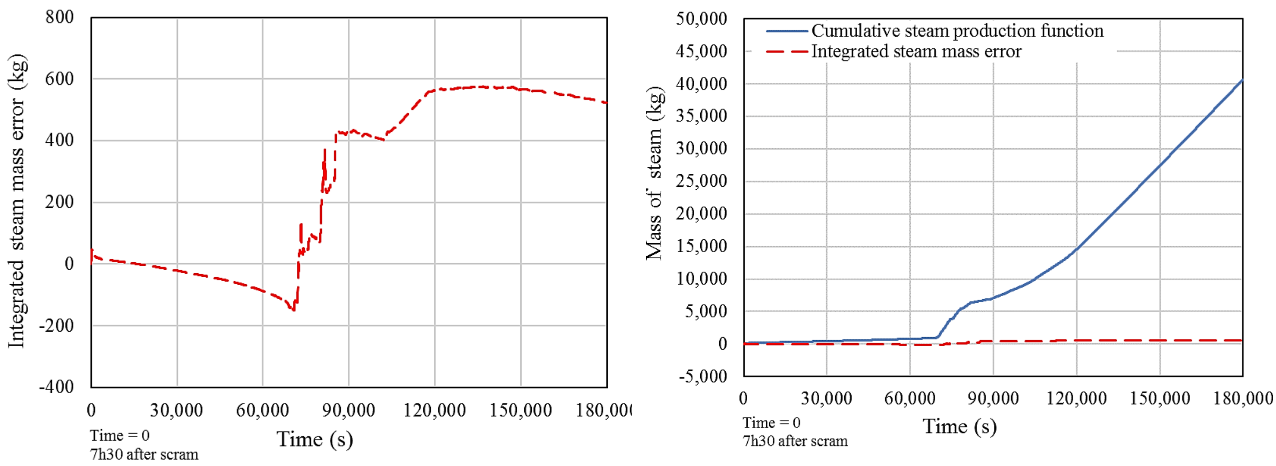
The RC pool works as a first heat sink and it is continuously cooled through the metal containment. As already observed in the Relap5 simulations, the very large water inventory in the RC and ST is the key aspect of the passive safety strategy: in-hull water can absorb a very large amount of heat without reaching the saturation point and then providing coolant for the RPV. The RC pool works as a primary heat sink, which is continuously cooled through the metal containment. The large heat transfer surface of the hull allows finding an acceptable thermal equilibrium between the internal and the external sides. Eq. (10) in paragraph 3.4.4 ensures that the RC temperature set as initial condition is conservative. In the third stage, the RC temperature become almost constant and close to the interface temperature. There is little stratification and the temperature distribution is quite uniform.

4.3.3 Mass error

Apros 6 evaluates the mass error separately for liquid phase, steam and non-condensable gas. In the current analysis, the latter has resulted very low and it is neglected in the current analysis. Table XVII reports the integrated mass errors at the end of the transient. Although the parameters for mass error check and time step reduction have been reduced with respect to Apros default values, i.e. bringing the relative error per time step from 10^{-4} to $1.5x10^{-5}$ for the total mass and from 10^{-4} to 10^{-5} for the steam phase error only, the final integrated mass errors are much higher than Relap5 outcome. Given the large mass inventory, i.e. 2.6 x 10^{6} kg, the relative error on the liquid phase is two orders of magnitude higher than the Relap5 case, where the sum of liquid and steam integrated errors, divided by the total inventory, returns roughly 10^{-5}. The integrated error on steam is 1.28% the steam flowrate at the outlet of the core integrated throughout the total transient. Figure 81 evidences alto that the sign of the mass error and the slope of the curve is negative during the first stage, where the production of steam is very low. Then, the derivative of the mass error curve becomes strongly positive in the irregular second stage, when large steam generation begins, and it finds a stabilization when also the natural circulation flow tends to stabilize in the third stage. This behavior leaves some concern about the reliability of calculations in the central stage, since the overall mass error is very high.

Table XVII. Results of mass error analysis

	Liquid	**Steam**
Liquid inventory / Steam production	2.6 x 10^6 kg	40,699.1 kg
Integrated error at the end of simulation	+ 9,571.8 kg	+ 522.9 kg
Relative error at the end of simulation	+ 3.7 x 10^{-3}	+ 1.28 x 10^{-2}



(a) (b)

Figure 81 Mass error profile (a) and comparison with the generation of steam (b)

4.3.4 Final remarks

The Apros 6 case here presented has simulated the successful operation of the natural convection cooling of the fuel rods following a LOCA, up to more than 50 hours after the scram. The behavior of the flow resolved by Apros is variegated and three different stages have been identified. Some concerns remain about the quite high mass error. The current case represents a first optimized simulation of the scenario with Apros 6, where the nodalization of the system has been calibrated with reasonable operations to obtain convergence. Given the necessity to fix some issues of Apros software, as explained in paragraph 4.2.3, and the time necessary to this operation made by VTT, the present work does not include a sensitivity and parametric analysis, required to assess the goodness of the current results. This activity should be the next step of the investigation.

4.4. Comparison: Relap5 vs. Apros 6

4.4.1 Comparison of general features

Relap5 and Apros have been created for different purposes. Relap5 specific application is calculating the behavior of the coolant in LWR during safety and operational transients, but it can be used for simulation of a wide variety of hydraulic and thermal transients also in nonnuclear field. Conversely, Apros is a software for modeling and dynamic simulation of several different industrial processes, which has been adapted also to nuclear applications. However, the system codes own a similar modeling philosophy, based on the subdivision of the system into volumes connected by junctions. Conservation equations calculate the thermo-dynamic state of the fluid in each volume and are solved considering one coordinate, which describes the main flow direction. Although simpler hydrodynamic models are available, e.g. homogeneous and equilibrium models, in both Relap5-Mod3.3 (Information Systems Laboratories, Inc., 2003) and Apros 6 (Hänninen, et al., 2012) the most advanced formulation of the governing equations is based on a two-fluid model for flow of a two-phase steam-water mixture that can contain non-condensable components. The codes solve continuity, momentum and energy equation for both the liquid and gas phase. Transverse gradients, e.g., wall/interfacial friction and heat transfer, are accounted with empirical models based on constitutive relations.

As far as the numerical methods are concerned, both codes employ the finite difference method to discretize space and time derivatives. Both codes also use implicit schemes, although Relap5 has a wider variety of numerical schemes that can be selected. Given that the spatial averaging process of the two-fluid differential equation originates an ill-posed problem, Apros and Relap5 show remarkable discrepancies to deal with the non-linear terms of the equations and obtain a well-posed problem. In general, Apros 6 tends to simplify the solving algorithms by using mainly linearization techniques, with an advantage in terms of speed of the calculation (Siikonen, 1987). On the other side, Relap5-Mod3.3 approach is oriented to find stable numerical algorithm by supporting linearization processes with several stabilizing techniques, thus gaining in accuracy of the solution. This difference is originated by the diverse focuses of the two codes. The main target of Apros is the simulation of plant processes, to study the control of industrial systems and to bring the simulation speed up to real time. Conversely, the perspective of Relap5 is more centered on the physics of the two-phase steam-water flow, since resolving the complexities of the liquid-gas interface is a fundamental need for nuclear safety analysis.

Concerning the constitutive relations to close the governing system of equations, both codes defines flow regimes and heat transfer logics to determine the proper empirical model for the current flow conditions. In general, Relap5 approach is much more detailed than Apros ones. The flow map implemented in Apros 6 includes four regimes: bubbly, annular, droplets and stratified. Relap5 adopts a more extensive approach: the flow map is modeled as nine regimes, i.e., four pre-CHF heat transfer, four post-CHF heat transfer, and one for stratification. The maps are also diversified for vertical, horizontal and other specific conditions of the flow. About wall heat transfer, the description given in paragraph 4.2.3 has evidenced large differences in algorithms and empirical correlations used to calculate the heat transfer rate. In addition, to model the effect on non-condensable gases on condensation, Apros 6 employs the correction factor presented by Vierow and Schrock (Vierow & Schrock, 1991) to the local heat transfer coefficient based on the Nusselt theory, while the default model in Relap5 is the Colburn-Hougen diffusion method (Colburn & Hougen, 1934), which determines the liquid/gas interface temperature.

In summary, discrepancies between the predictions of the two codes can originate from the numerical methods used for the solution of the governing equations and from the constitutive relations and empirical models employed to model the physics of the problem. To these elements, one should add the nodalization process, whose features can be different in the two codes and can be another source of uncertainty. Table XVIII summarizes the similarities and the differences between Relap5 and Apros 6 evidenced in this paragraph.

Table XVIII. Overview of differences between Relap5-Mod3.3 and Apros 6

Feature	Relap5-Mod3.3	Apros 6
Main scope of the code	Calculating the behavior of a LWR reactor coolant system during a transient (safety and operational analysis)	Modeling and dynamic simulation of several different industrial processes
Basic modeling philosophy	Subdivision of the system into elementary nodes connected by junctions. Solution of 1D conservation equations for liquid and steam. Empirical modeling of transverse gradients (friction, HT, two-phase interface, etc.).	
Formulation of governing equations	Two-fluid model, steam-water mixture w/o non-condensable gases (other simplified formulations available)	
Numerical schemes	Finite differences (semi-implicit, nearly-implicit or fully-implicit coupling)	Finite differences (fully implicit)
Treatment of non-linearities	Oriented to resolve complex two-phase flow behaviors	Oriented to gain computational speed at acceptable accuracy
Two-phase flow maps	9 main flow regimes: 4 pre-CHF, 4 post-CHF and stratified. Diversification for horizontal/vertical flow	4 flow regimes: bubbly, annular, droplets, stratified
Two-phase wall heat transfer	5 basic two-phase heat transfer modes and flow chart to determine the most appropriate	Three heat transfer zones: wetted wall, dry wall and transition zone, the latter calculated by interpolation
Non-condensable gases modeling	Colburn-Hougen method (def.) Vierow and Schrock (opt.)	Vierow and Schrock correction to Nusselt theory

4.4.2 Comparison of long-term cooling case

Relap5-Mod3.3 and Apros 6 have been used independently to simulate the case under examination, adopting an approach based on elementary components. Both codes predict that, for the entire duration of the simulation, the sump natural circulation flow is capable to cool the fuel rods and reject the decay heat to the surrounding water. In both cases the coolant into the core never reaches critical conditions and the total power transferred to the exterior becomes higher than the decay power within the end of the simulated transient. This fact represents an important element in order to assess the feasibility of the safety concept: two different simulation software, well-validated and widely adopted in nuclear industry, both predict the successful performance of the passive safety systems in a long-term cooling scenario through natural circulation, given almost the same conservative initial and boundary conditions.

Nevertheless, remarkable discrepancies in the behavior of the coolant can be observed: the two codes respond differently to the given conditions. After a small period, where the codes need to find a stabilization with respect to the initial conditions, Relap5-Mod3.3 sees only one flow behavior, where at the outlet of the reactor core there is a small steam quality and void fraction fluctuates around mean values roughly from 0.25 to 0.35. This behavior is constant throughout all the transient and no apparent reasons for sudden changes can be noticed. On the contrary, Apros transient takes nearly 120,000 s to see a Relap5-like behavior and the "stable" value of the core outlet void fraction is a little bit lower (Figure 82a). A similar argument can be applied to the RPV flowrate (Figure 83a): Apros "stable" value is higher than Relap5, thus compensating the lower void fraction.

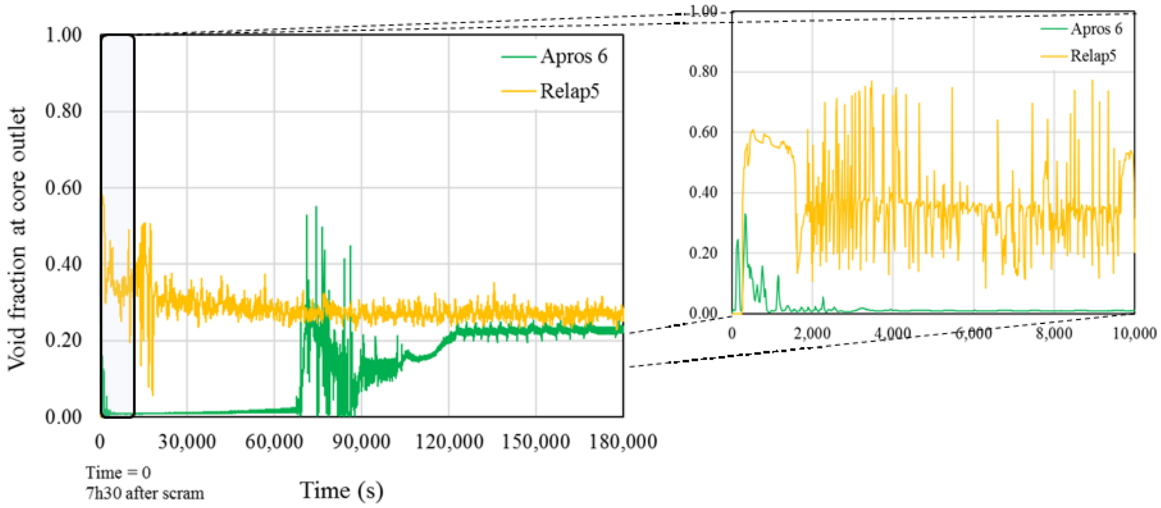


Figure 82 Comparison between Relap5-Mod3.3 and Apros 6 void fraction at core outlet: full transient and detail of the initial part

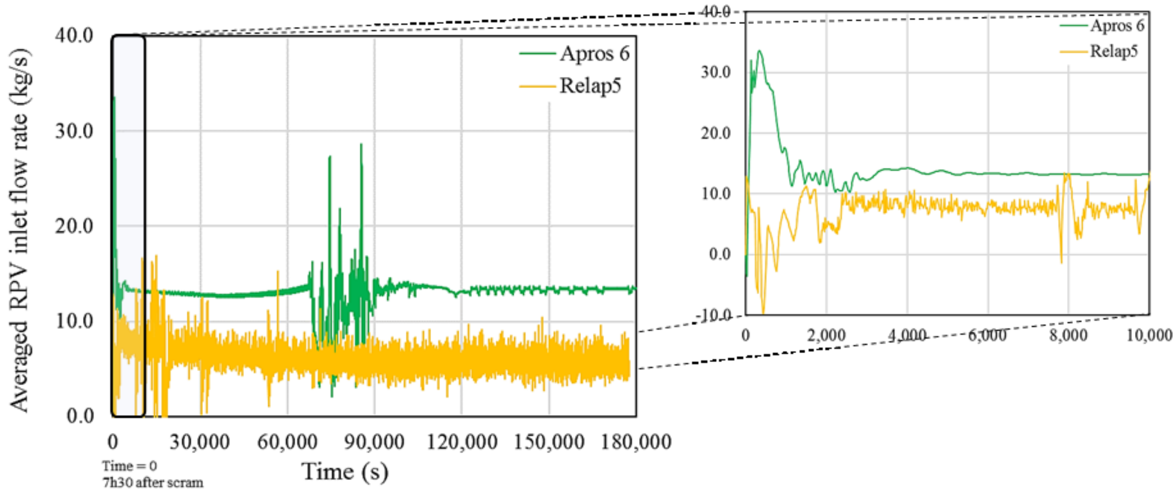
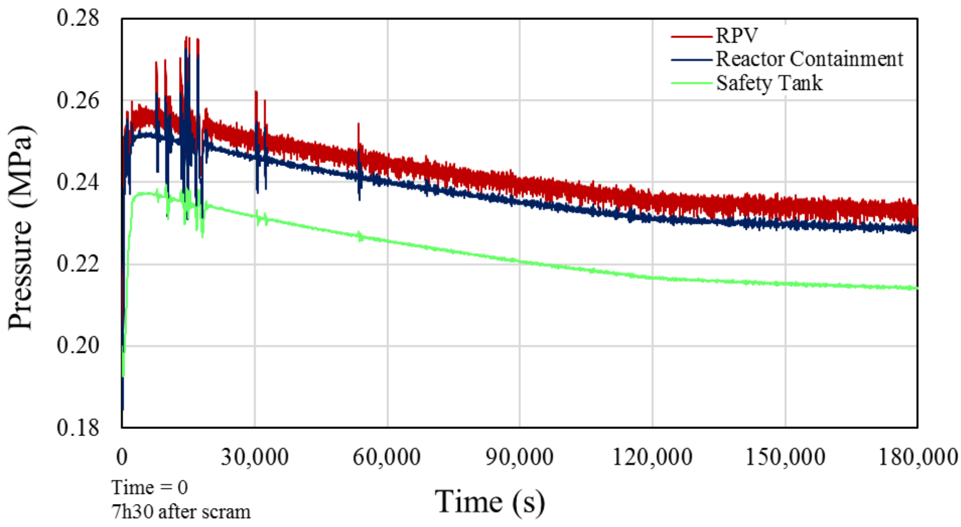
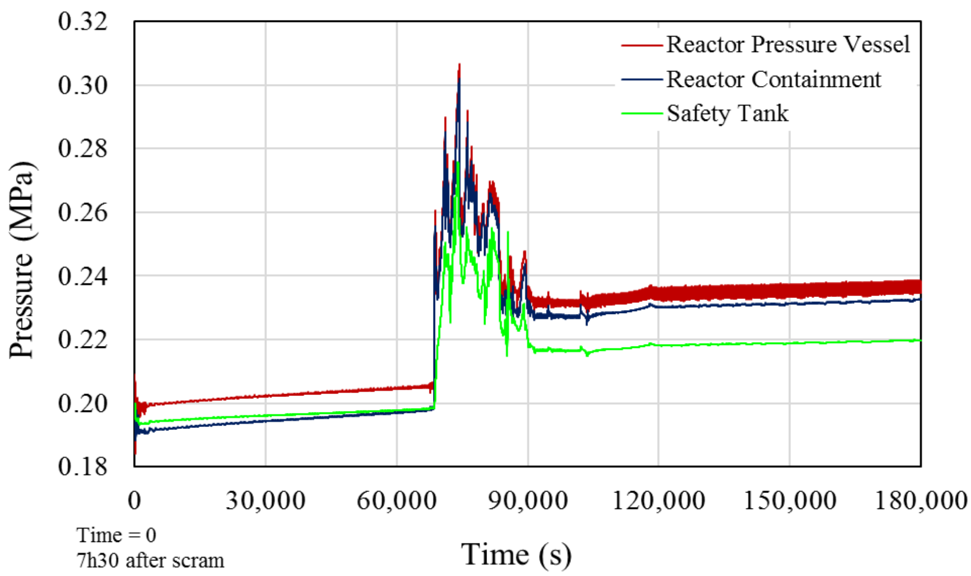


Figure 83 Comparison of Relap5-Mod3.3 and Apros 6 RPV flowrates: full transient and detail of the initial part

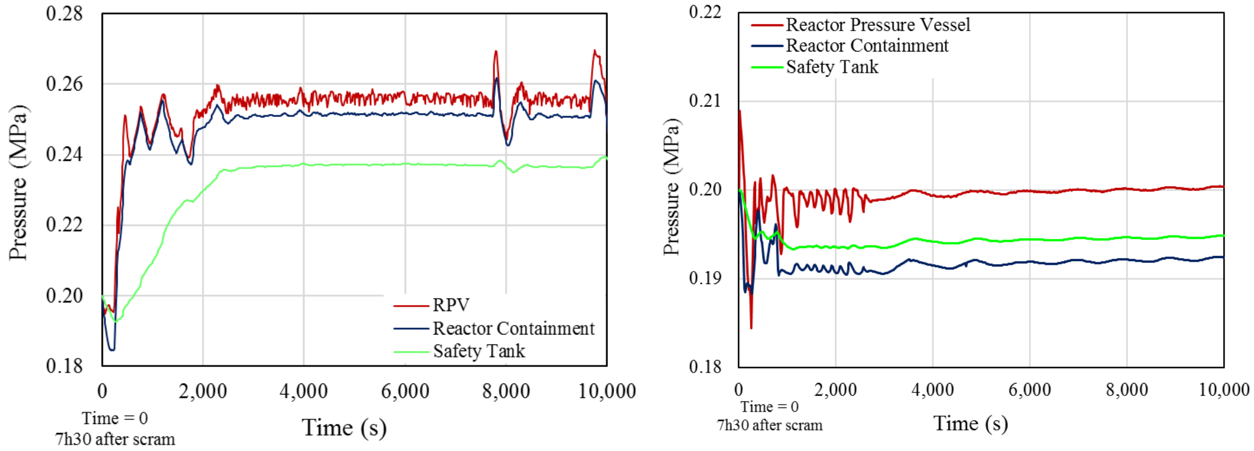
Given that the different behavior is shown immediately after the beginning of the simulation, it is worth analyzing what happens in the first period of the transient, when natural circulation begins. The different production of steam in the very first phases of the transient causes different pressure relations among RPV, RC and ST (Figure 84), thus generating different levels of liquid and different flowrates in the RPV and in the recirculation lines. Figure 84 evidences that pressures in RPV, RC and ST are resolved differently by Relap5 and Apros. The Relap5-Mod3.3 simulation shows a sudden increase of the three pressures as soon as steam is produced and released in the RC (Figure 82b and Figure 84c). On the contrary, the beginning of the natural circulation in Apros still produces a certain amount of steam, but the amount is much lower than Relap5 case (Figure 85). Therefore, pressure tends to decrease (Figure 84d) because dominant effect is not the injection of steam in a dry atmosphere, but the cooling the non-condensable gas in contact with the RC wall, whose initial temperature is 120°C. The non-condensable gas reduces its specific volume and consequently also the overall pressure inside the RC. Such different mechanisms generate different relations among pressures and liquid levels and has effects on the circulating flowrate. In natural circulation problems, the flowrate is inherently correlated to the density gradient: these two aspects are coupled and the system finds the working point given the initial conditions. Equilibriums are in general quite fragile, since the driving forces are weak, therefore even small changes in the conditions can generates divergences of the behavior. The low production of steam in Apros 6 during the initial part of the transient force the system to reject the heat almost only through the liquid zone of the RC. The consequence is an increase of the RC pool temperature, which become much higher than in the Relap5 case (Figure 86).



(a)
Relap5-Mod3.3



(b)
Apros 6



(c) Relap5-Mod3.3 (d) Apros 6

Figure 84 Pressure profiles for Relap5-Mod3.3 (a) and Apros 6 (b).
Detail of the initial part of the curves: Realp5-Mod3.3 (c) and Apros 6 (d)

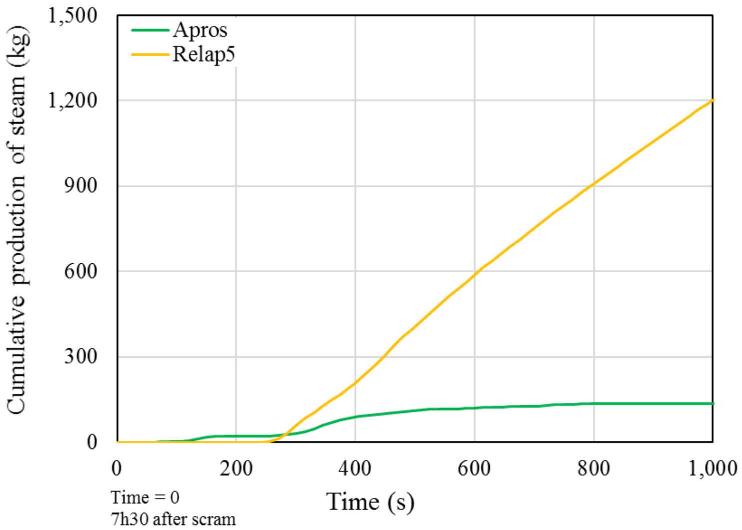
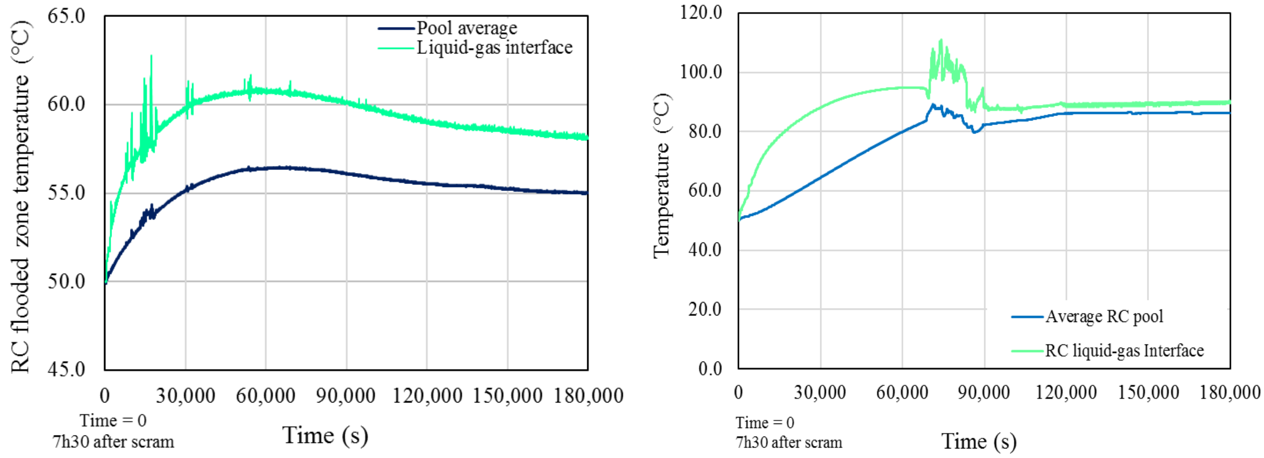


Figure 85 Cumulative production of steam in the first 1,000 s for Relap5-Mod 3.3 and Apros 6



(a) (b)

Figure 86 Comparison of RC pool temperature: Relap5-Mod3.3 (a) and Apros 6 (b)

Another element that may contribute to originate different behaviors is the evaluation of pressure drop and heat transfer into the core. The wall friction of the coolant in the core, the interphase friction and the wall heating depend on the flow regime and heat transfer modes recognized by the software. At the outlet of the core, Relap5-Mod3.3 sees a slug flow throughout the whole transient. Apros 6 does not include the slug flow in its flow map and it identifies a bubbly flow at the beginning, switching to an annular flow in the final part of the simulation. In principle, the use of different correlations to evaluate friction and wall heating can have a certain impact on the natural circulation equilibrium. Finally, it should be considered that different numerical methods can influence the behavior of the system. In particular, the higher mass error in Apros 6 can be an important aspect, especially to determine the boiling at the beginning of the transient. Also, nodalization methods have been oriented at exploiting the best modeling performances of each code. Therefore, the discretization of the system is not perfectly equal.

In the final part of the transient, fluid conditions at the outlet core are similar: disregarding small fluctuations, the void fraction profiles are almost constant around close values. Also, pressure profiles show comparable values and analogous relations among RPV, RC and ST. Therefore, the general behaviors are in agreement, but flowrate and RC pool temperature are higher in Apros 6 than in Relap5Mod3.3. Consequently, in Apros 6 there is much production of steam, which leads to the increase of pressure observable in Figure 84b. This fact is a warning, since it signifies that Apros 6 has not reached a “steady” state yet. The higher temperature of the RC pool is and indicator of more critical conditions. While for the Relap5 case it is possible to conclude that no apparent reasons for core overheating can be observed, the high RC pool temperature of the Apros 6 case and the slightly increasing pressures put some concerns in this sense and stimulate further analysis.

4.5. Summary and next steps

The use of Apros 6 software for the analysis of the submerged SMR concept has reinforced the safety analysis done in sub-section 3.4 regarding the long-term cooling scenario. Starting from very similar initial conditions, i.e. 7h30m after the reactor scram due to a LOCA, the Apros simulation has predicted the good operation of the sump natural convection to cool the fuel rods. A natural circulation motion flows through the RPV and generating a small steam quality, which is condensed in contact with the cold RC wall. The safety systems operate successfully for 50 hours. This positive outcome, jointed with the Relap5-Mod3.3 results, allows concluding that the working principle of the passive safety systems in long-term cooling scenario is verified by two different system codes adopted in nuclear industry. However, a warning remains about the conditions of the systems at the end of the simulation. In addition, the reference test case about the long-term cooling and the preliminary analysis of the heat exchanger have evidenced remarkable differences between the two codes. Apros and Relap5 own similar modeling philosophies, but different primary scope of the code, which translates to different approaches to model transverse gradients and to numerically solve the governing equations. The analysis of a simple test case has allowed VTT to discover an issue in the calculation of the two-phase HTC under certain condition.

Apros and Relap5 are two valid tools that can support the safety analysis of a submerged SMR. In principle, they are able to simulate the reference safety scenarios providing credible results, although some critical aspects concerning modeling capability of large horizontal containments have been evidenced for both codes. The next steps of this activity should include the investigation of many other accident situations that can occur in a submerged SMR. In addition, given that, unlike Relap5, the main target of Apros is not the nuclear safety analysis, the development of Apros nuclear features would benefit from an accurate benchmark on test cases with experimental data available.

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Section 5

Experimental activities to support submerged SMRs development

In 2016, Flexblue has been proposed as the European reference SMR in the framework of an R&D project submitted to a H2020 Euratom call (project INSPIRE- INtegration of Smr's PotentIal Role in EU framework), led by ENEA (Ente Nazionale Energia e Ambiente) and supported by a consortium involving 13 organizations among universities, R&D centers and industries from 6 EU countries. This section presents a summary of the candidate's collaboration with the SIET (Società Informazioni Esperienze Termoidrauliche) laboratories in the definition of the experimental part of the INSPIRE proposal. The activity includes the design of the experimental facility and corresponding campaign aimed at studying the safety potentialities of a submerged SMR, covering the gaps of knowledge. Two proposals have been defined, regarding both the Emergency Heat Removal System (EHRS) and the heat transfer through the submerged containment. In addition, another experimental activity concerning an alternative configuration for the steam generator, i.e., the bayonet tube, is included in this section.

5.1. Design of experimental facilities for INSPIRE proposal

5.1.1 The INSPIRE proposal

INSPIRE (INtegration of Smr's PotentIal Role in EU framework) is a proposal submitted in 2016 to the Euratom Work Programme "NFRP-4: research on the safety of small modular reactors" (INSPIRE Consortium, 2016). It was led by a by ENEA (Ente Nazionale Energia e Ambiente) and supported by a consortium involving 13 organizations among universities, R&D centers and industries from 6 EU countries. The list of the participants is shown in Table XIX. Its objective was to improve the standards of nuclear safety in EU member states, developing the SMR technology, providing European nuclear community and stakeholders:

(i) specific knowledge on SMRs potential safety features, with respect to the EU Safety Directive;
(ii) a credible methodology for the demonstration of the essential safety features and how to implement it in the licensing process;
(iii) a critical analysis of the key aspects of a possible SMR integration in the EU framework.

INSPIRE selected the Integrated Pressurized Water Small Modular Reactor (iPW-SMR) as the technology for the R&D investigation, among the different SMR concepts under consideration in the world. The Flexblue concept of submerged SMR was the most important reference design considered in the proposal.

Table XIX. List of participants to INSPIRE proposal

Participant Organization Name	Acronym	Role	Country
Agenzia Nazionale per le Nuove Tecnologie, l'Energia e lo Sviluppo Economico Sostenibile *(Leader)*	ENEA	R&D Centre	Italy
Commissariat à l'Energie Atomique et aux Energies Alternatives	CEA	R&D Centre	France
Consorzio Interuniversitario per la Ricerca Tecnologica Nucleare	CIRTEN	University	Italy
DCNS SA	DCNS	Company	France
Fondazione Politecnico di Milano	FPM	Foundation	Italy
Fortum Corporation	FORTUM	Utility	Finland
Institut de Radioprotection et de Sureté Nucléaire	IRSN	Technical Safety Organisation (TSO)	France
National Nuclear Laboratory	NNL	R&D Centre	UK
NucAdvisor	NUCADVISOR	SME (Consultancy)	France
Società Informazioni Esperienze Termoidrauliche S.p.A.	SIET	Company	Italy
SRS Engineering Design srl	SRS	SME (Engineering)	Italy
Tecnatom SA	TECNATOM	Company	Spain
TRACTEBEL Engineering S.A.	TRACTEBEL	Company	Belgium

5.1.2 Gaps of knowledge

The INSPIRE activity has been targeted at identifying the gaps of knowledge on thermal-hydraulics about the submerged SMR concept and, consequently, to design an experimental campaign to fill these gaps. The proposed experimental activity is aimed at assessing the feasibility of the safety strategy for an iPW-SMR operating in a submerged containment. The reference scenarios to be investigated experimentally trace the classification given in sub-section 3.2., i.e. intact or broken primary system. Two experimental facilities have been designed, regarding the operation of the EHRS during a SBO and the condensation heat transfer through the submerged containment. The importance of performing such experiments is given by lack of databases available in open literature. Basically, integral facilities to test passive safety circuits have been performed, but data are not open, while the main issue about natural convection through a submerged containment is the very large size of the cylinder. The purpose is to generate an experimental data basis for the development and validation of condensation models given the geometrical particularities and respective boundary conditions.

Integral experiments were made for licensing of NuScale (NIST facility, US), CAREM25 (Argentina) and ACP100 (China) reactors. Databases are property of respective customers and such results are not available for EU. However, some testing regarding only the pool heat sink or the helical coil steam generator can be found. Pool experimental campaigns were performed in the facilities NOKO at the Forschungszentrum Jülich (Palavecino, et al., 1995) and PANDA at Paul Scherrer Institut (PSI) (Roelefs & Taralunga, 1997). Also, several helical coil SG tests were performed in SIET for IRIS project, e.g. (Santini, et al., 2008). However, the gap of knowledge concerns also the behavior of the integral system exploiting a constant temperature heat sink: this is a novel concept in nuclear industry and no experiment exists about it.

Outer containment cooling is currently being investigated e.g. for the Large Scale Test Facility (LST) for AP1000 (Broxtermann & Allelein, 2001) or in the Passive Containment cooling System (PCS) water distribution test bench for the CAP1400 (Chang, et al., 2011). In these facilities, the influence of water flowing from the gravity drain water tank is investigated. However, the physical phenomena for INSPIRE reference case are different and the results are not transferable. In a submerged reactor the key phenomenon is natural convection heat transfer from a large horizontal cylinder. Currently, no codes are available with validated models for this application. Several empirical correlations can be found in literature: they are expressed as a function of the Rayleigh number and they are validated up to $Ra=10^{12}$. However, the Rayleigh number depends on the third power of the characteristic length, then for the Flexblue case, with a 14-m diameter hull, it is in the order of 10^{15} (Santinello, et al., 2017-a). The plot Nusselt versus Rayleigh in Figure 87 shows a good agreement for $Ra < 10^{12}$ and a divergence beyond that value. Therefore, new experimental data are required for model and correlation development. The challenge is to design a test section for a large hull model, in order to reach natural convection conditions with the maximum achievable Rayleigh number. It is important to find a compromise between containment size, spatial and time resolution of the measurements, and the economic constraints. Other particularities to be investigated are the difference in heat flux distribution around the circumference (e.g. due to the condensation inside the hull and the drainage of the condensate layer at the inner surface of the hull), the influence of the paint coating and the roughness (due to biofouling) on the development of the boundary layers.

Another aspect with gaps of knowledge is the condensation in presence of non-condensable gas. Its investigation is very challenging, because the phenomenon is affected by a very large number of scale dependent and independent factors. In contrast to other thermal hydraulic phenomena, the development in modelling of wall condensation without or with non-condensable gases was more theoretical than experimental. Well-known theoretical models developed by Uchida, Tagami, Dehbi, Kataoka, Murase and Liu, described and compared by IAEA (IAEA, 2005), are only valid for defined scopes. In order to obtain conservative results, a robust and manageable approach based on Uchida's correlation was developed in some nuclear regulations. Experimental investigations performed in various scales have been described by many authors (Choi, et al., 2011) (Funke, et al., 2012) (IAEA, 2012). The large surface of the containment requires either a scaling or the examination of the section (IAEA, 2005). For both approaches, the setting of representative boundary conditions is extremely difficult. Steam condensation is largely affected by conditions that can be split into two groups. The scale independent factors are, e.g., pressure, fraction of non-condensables or gas composition. The scale dependent factors are phenomena related to the size of the hull, which need to be investigated in original or scaled geometries. The coupled phenomenon, made of internal condensation plus external natural convection, also needs to be tested, to observe the mutual interaction.

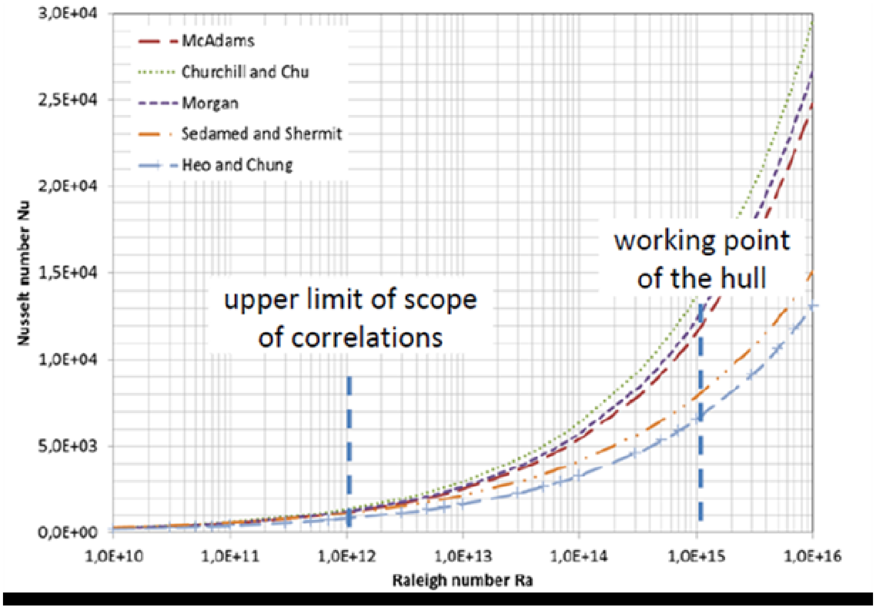
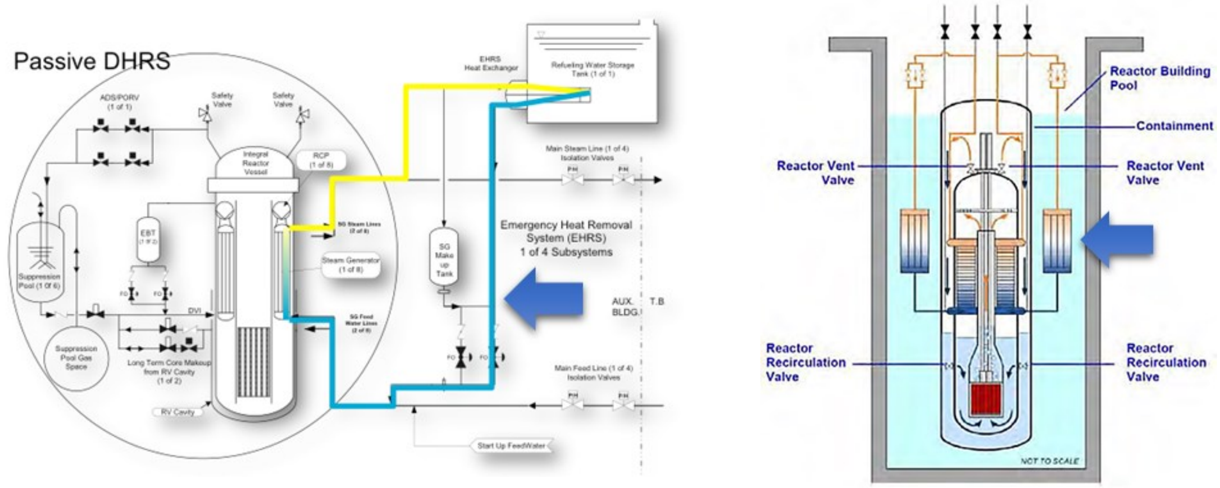


Figure 87 Nusselt number versus Rayleigh number of existing correlations (INSPIRE Consortium, 2016)

5.1.3 Passive EHRS facility

The EHRS is widely adopted in several LWR and iPW-SMR concepts. The examples of IRIS and NuScale are shown in Figure 88. The typical decay heat removal systems of iPW-SMR is based on natural circulation flow between the secondary side of SG and the external heat sink. In case of core scram due to accident scenario in which there is no rupture of primary system, e.g. a SBO event, the core cooling relies on the operation on the passive ERHS. The natural circulation flow starts and aims at keeping the temperature of the fuel rods below safety limits. Nowadays, such system is considered the most effective passive system able to safely manage the SMR incidental and accidental situations and to accomplish the long-term decay heat removal without any needs for electricity or external input. The proposed experiment will offer an integral overview of the EHRS, testing both and simultaneously the secondary side of the SG and the pool heat exchanger in a closed loop, as sketched in Figure 89 for the case seawater heat exchanger. The pool is maintained at constant temperature to simulate an "infinite" heat sink. Experiments will observe the evolution of two-phase natural circulation flow, steam generation, condensation and pool behavior.

During the development of IRIS project, SPES-3 facility (Carelli, et al., 2009) was conceived to support IRIS licensing. SPES-3 (Figure 90) is a large-scale facility capable to simulate the primary, secondary and containment systems of an iPW-SMR. It was designed to be 1:100 volume scale, full elevation, and prototypical thermal-hydraulic conditions with respect to IRIS reactor. SPES-3 is located in SIET and construction is currently at 60% completion. In full operation, SPES-3 will allow creating an open access European database about integral testing, it will increase the knowledge and the development of iPWRs and it can be used for educational & training purposes.



(a) IRIS (Carelli, et al., 2004) (b) NuScale (Reyes Jr., 2012)

Figure 88 Example of key passive safety system (EHRS) in IRIS (a) and NuScale (b)

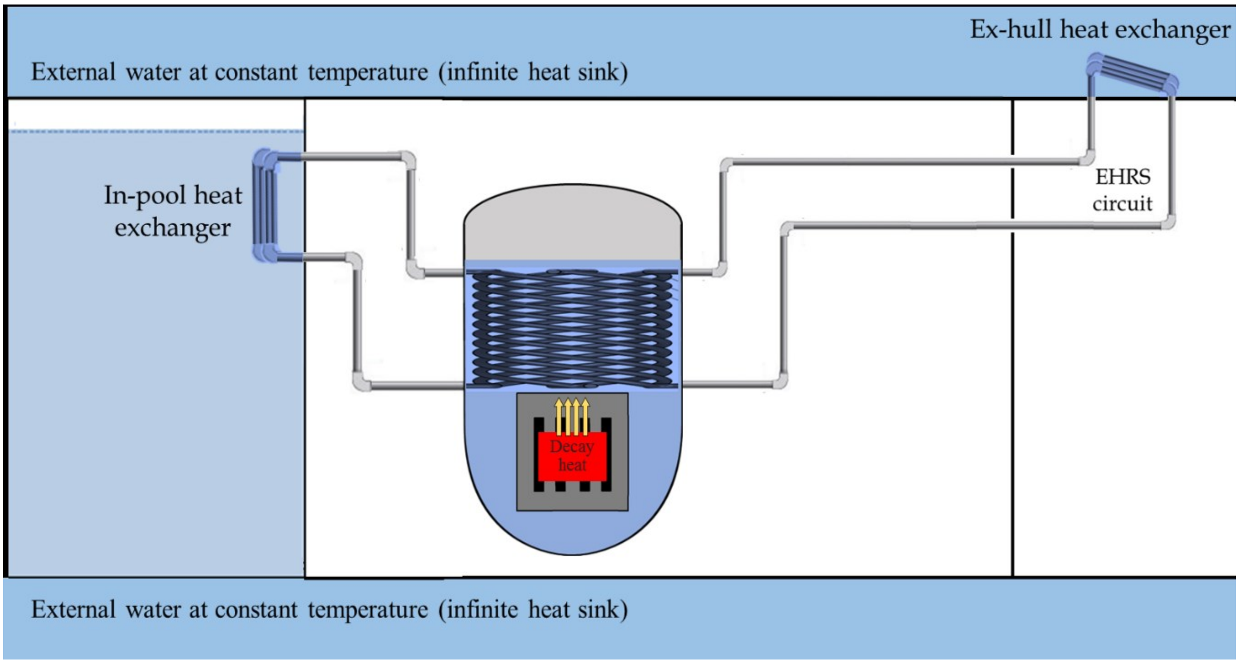


Figure 89 Sketch of an EHRS for a submerged SMR (as in Figure 28a)

The test section (conceptual scheme in Figure 91) is designed to exploit the SPES3 existing facility, thus realizing a project left incomplete by the shutdown of IRIS project. It simulates the typical EHRS of a iPW-SMR. The heat exchanger is dipped in a water pool at room temperature, in configurations able to simulate both a finite pool and the sea, located at higher position than the helical coil SG. The upper and lower headers of the heat exchangers are connected to the main steam line and feed water line of the SG. The in-pool, vertical tube heat exchanger of SPES-3 integral test simulator is connected to a helical coil tube bundle, simulating the steam generator module. The primary side is not simulated: the SG bundle is electrically heated and the power provided to the fluid through the tube walls allows producing steam at prototypical pressure and temperature conditions.

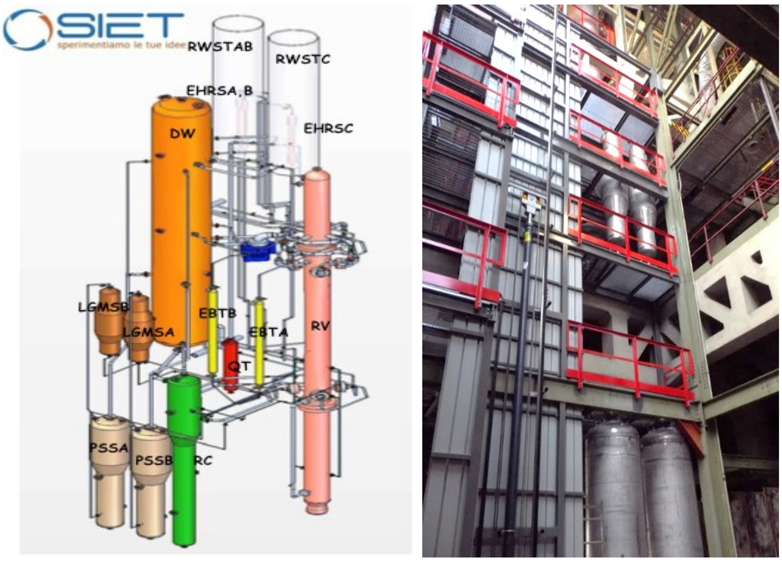


Figure 90 Sketch and side view of SPES-3 facility (Carelli, et al., 2009)

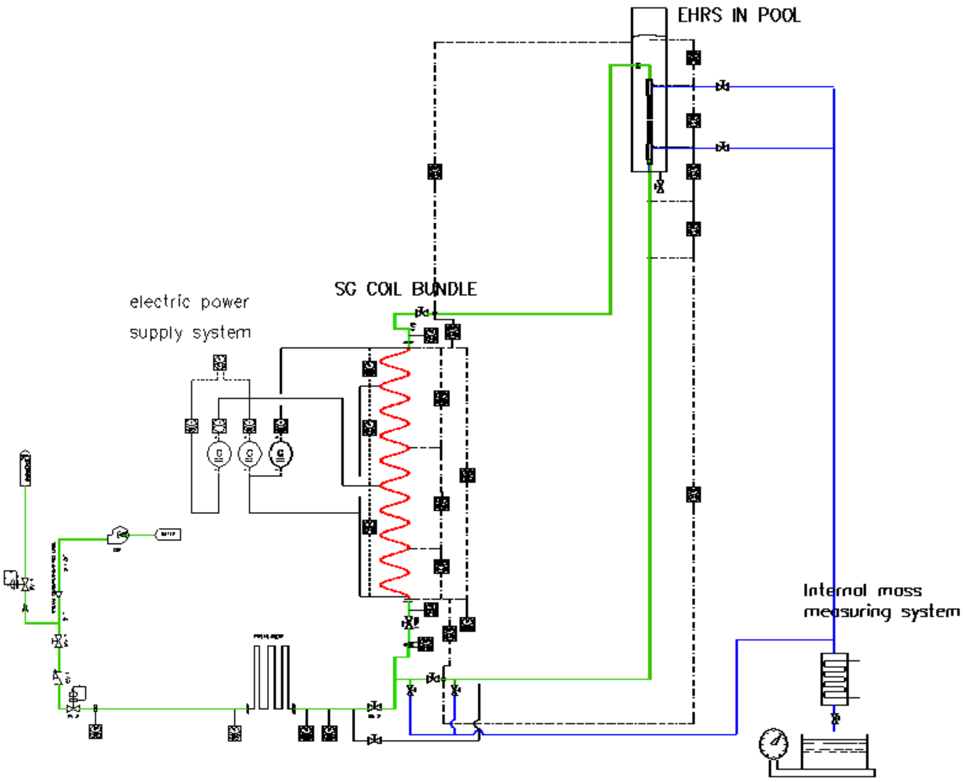


Figure 91 Conceptual scheme of the EHRS Test Facility

Testing activity regards the operation of the EHRS from the starting of natural convection regime to the reaching of steady state. Water pool temperature monitoring and cooling allow simulating both the finite pool and the seawater. For the in-seawater EHRS case, pool temperature is controlled and kept constant through dedicated heat exchangers and/or water refilling. The two other main parameters driving the behavior of the EHRS during an accident scenario are the decay power and the initial water inventory into the loop. Both parameters can be monitored and controlled. Natural circulation is also affected by the hydraulic resistance of the loop: its effect can be investigated with valves. Heat transfer can be also strongly influenced if a non-condensable gas is present into the loop, e.g. due to a malfunction of the feed water pumping system. Different percentages of gas can be injected into the loop, to observe heat transfer degradation. The facility is designed to operate at full scale pressure and temperature conditions, and at large scale about elevation, power and volume, exploiting SPES-3 components and systems (SPES-3 original scale was 1:100 for IRIS reactor). The experimental results are expected to provide data on:

- two-phase flow natural circulation (e.g. loop mass flow rate, pressure drops and flow stability);
- evolution of operating pressure for both in-pool and in-seawater configurations;
- specific single- and two-phase heat transfer features for helical coil steam generator (e.g. critical heat flux) and for heat exchanger;
- grace period for the in-tank EHRS.

5.1.4 Submerged containment facility

The safety concept of a submerged SMR relies widely on its capability to remove the residual power from the core to the ultimate heat sink. During a LOCA, steam is released inside the containment and condenses on the containment surface. Hereby, the hull heats up, the temperature at the exterior surface increases and a thermal and velocity boundary layer develops along the curved surface as shown in Figure 92. The proposed experimental activity investigates both the external and the internal natural circulation flow.

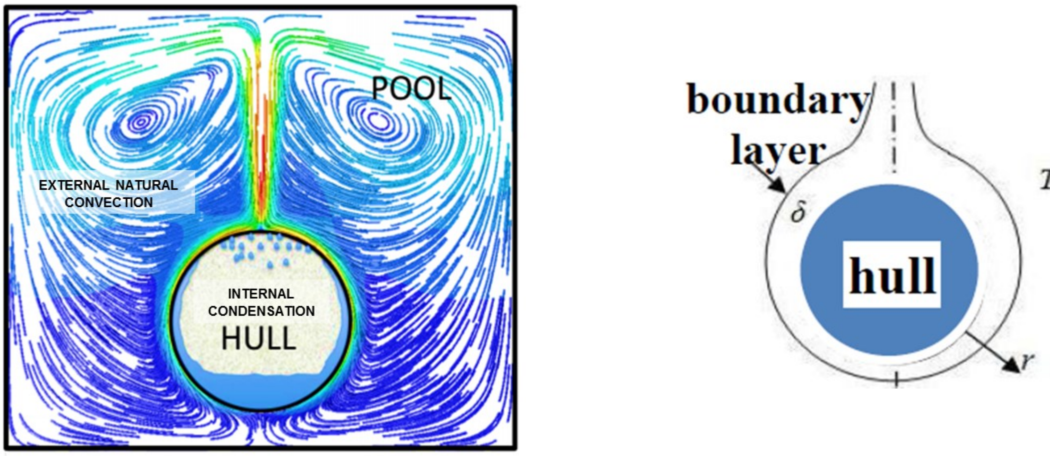


Figure 92 Concept of decay heat removal through the submerged containment

The facility (conceptual scheme in Figure 93) is a large-scale model, i.e. around 1:5, of the submerged containment of a sea-based SMR. The reference design adopted for the investigation is the sea-based Flexblue SMR, but the layout is representative also of the Chinese ACP100 design. The facility allows both the separate effect investigation, i.e. external cooling/fluid dynamics only, and the integral effect, i.e. external cooling and internal condensation in presence of non-condensable gases. A horizontal cylinder is representative of the containment and is immersed in a water pool, representing the sea. For the separate effect investigation, electrical heaters are placed in contact with the internal surface of the hull. The heating distribution can be controlled, to manage the heat flux distribution. For the integral effects investigation, external auxiliary systems inject steam and non-condensable gas. Steam condenses on the inner surface of the cylinder and heat is rejected to the pool. A control system on the external water temperature is used to simulate the infinite heat sink.

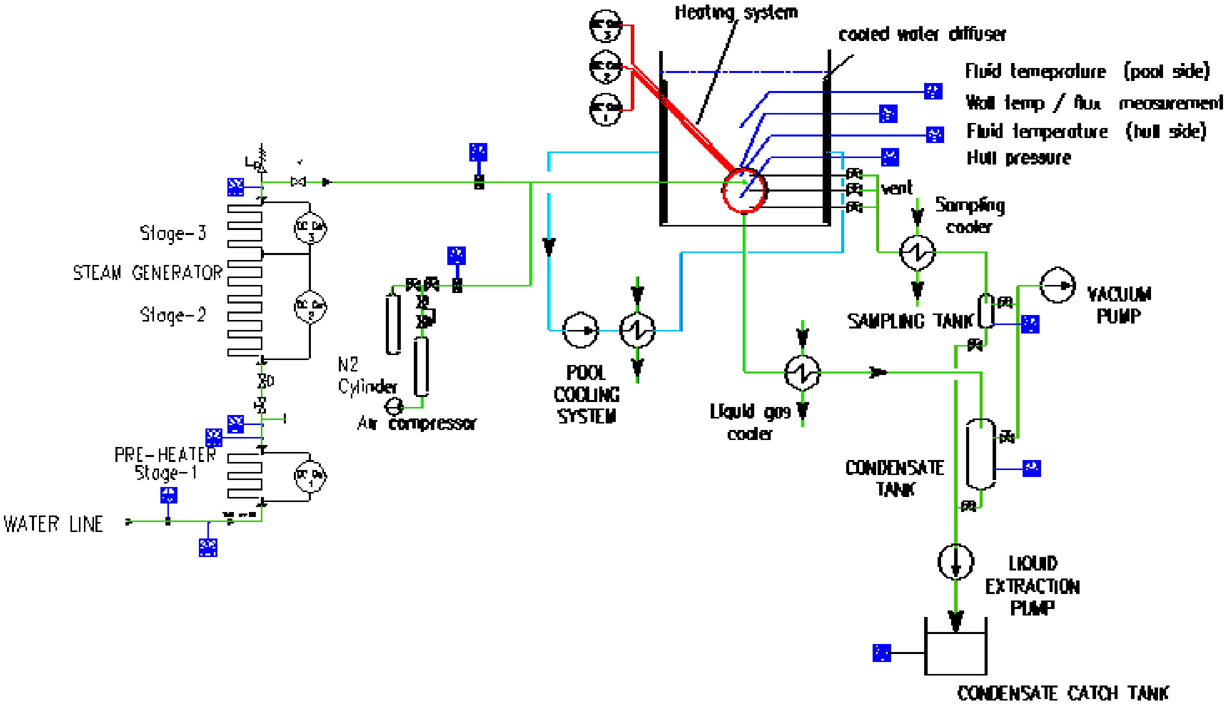


Figure 93 Conceptual scheme of the test facility for the submerged containment

The facility simulates a slice of the reactor hull with a vertical scale for the horizontal cylinder of about 1:5, basing on the preliminary design value of Flexblue hull diameter, equal to 14 m. The derivation of the scaling concept is of fundamental importance, to extend the validity limit of empirical correlations for external natural convection heat transfer from a horizontal cylinder beyond $Ra = 10^{12}$. In addition, the size of the pool should be as large and deep as possible, in order to avoid significant influences of the pool walls to the external natural circulation flow. The experimental results are expected to provide data on:

- natural convection and condensation in a large horizontal cylinder;
- pool side natural convection;
- single-phase heat transfer coefficient on the external surface of the cylinder;
- condensation heat transfer coefficient on the internal surface of the cylinder.

5.1.5 Concluding remarks

The two facilities here described, presented in the framework of the INSPIRE proposal, allow creating specific databases regarding the two reference scenarios of the safety strategy presented in this thesis. Performing a critical analysis of the results of the experimental and modelling activities would be fundamental to determine the goodness of modeling strategies and codes use. Post-test analysis of such experiments would be fundamental to validate the numerical models about the SBO scenario in sub-section 3.3 (Relap5-Mod3.3) and the long-term cooling in sub-section 3.4 (Relap5-Mod3.3) and 4.3 (Apros 6). In addition, the proposed experimental campaign would play a key role in order to identify future needs and recommendations for further experimental and modeling activities, referring, in particular, to possible shortcomings of code capabilities to simulate relevant phenomena, requirements for experimental proofs for passive systems operations and needs for a possible harmonized SMR licensing process.

5.2. Testing and analysis of a bayonet tube heat exchanger

5.2.1 Overview

The simplest design of a bayonet tube consists of two concentric tubes vertically placed in the firebox or in the boiler. The tubes operate completely immersed in the heating fluid or exposed to combustion gases. The main advantage of using bayonet tubes is the low rupture frequency. Being welded or rigidly coupled to the tube sheet only at one extremity, they do not suffer from thermal expansion stresses, thus reducing the formations of cracks and improving reliability with the respect of single tube boilers. Moreover, they can be easily extracted from the component and replaced, if failed. The major issue is the accumulation of dirt at the closed end, which is hard to wash out and can obstruct the passage of the fluid. In applications where long lifetime of the component and design simplicity are fundamental requirements, bayonet tubes can become an interesting alternative to the helical geometry. Thanks to their peculiar design (Figure 94), bayonet tube heat exchangers require welding/coupling to the sheet only at one extremity, thus halving the number of penetrations and avoiding the occurrence of thermal expansion stresses. This feature reduces the formation of cracks and provides a faster access to the component for maintenance operations.

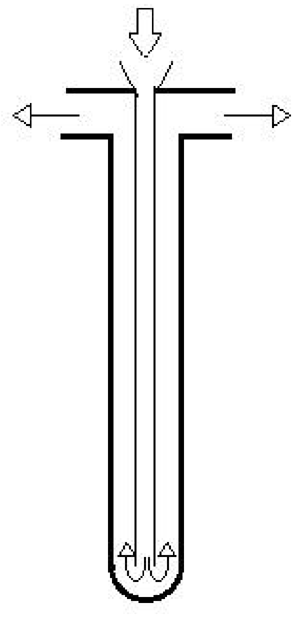


Figure 94 Basic layout of a bayonet tube: process fluid flows downward into the central tube and heats up while rising through the annulus

Disregarding some patents dated back to the '70s, mainly related to gas and sodium reactors, in modern times the most important application of bayonet tubes concerns the Advanced Lead Fast Reactor European Demonstrator (ALFRED). Ansaldo Nucleare (Damiani, et al., 2013) has proposed a conceptual design of bayonet tube steam generator made of eight modules and immersed into the liquid lead. The basic tube consists of four co-axial tubes, which include a safety outermost tube and an internal insulating layer. As far as LW-SMR are concerned, bayonet tubes can be suitable for the use as in-vessel steam generators in an integrated layout, due to their easy assembly and reduced number of penetrations. This solution was considered in the design of the steam generator for the IRIS reactor (Cinotti, et al., 2002) and in recent times a study about the implementation in an integrated layout has been made by ENEA (Caramello, et al., 2015-a). Generally, bayonet tube geometrical features find very good application in pool-type geometry and can represent an important technological solution also for decay heat removal systems. An important example is the heat exchanger-pump module adopted by the SCOR integral design (Gautier & De Masi, 2005): the emergency circuit is made of bayonet tubes placed inside the RPV and immersed into the primary fluid, while secondary water is flowing inside.

In this context, ENEA is leading an important campaign about bayonet tubes heat exchangers in collaboration with POLIMI and SIET S.p.A, within the framework of National Research Program funded by the Italian Minister of Economic Development. This activity responds to the necessity of creating ex novo a database, which characterizes the thermal-hydraulic behavior of boiling water in terms of single- and two-phase pressure drops, critical heat flux conditions and dynamic instabilities. ENEA commissioned two test sections, which are currently object of investigation. The first one has been designed to study the steam generator in a Lead Fast Reactor (LFR) within the ALFRED project. The second one is dedicated to the analysis of heat removal system and steam generator in LWRs, with particular attention to SMR. POLIMI has given a contribution to the development of the pre-test/post-test analysis, and to a preliminary study regarding this second test section. The study explores the feasibility of an alternative option for the SG module for the SMR, based on bayonet tube technology instead of helical-coil tube.

5.2.1 Experimental campaign on the HERO-2 facility

HERO-2 (Heavy liquid mEtal pRessurized water-cOoled tube – 2), a heat exchanger unit designed by ENEA and supplied to SIET, which dealt with assembling instrumentation, test section preparing and testing. HERO–2 is composed of two bayonet tubes manufactured with commercial parts of AISI 304 Stainless Steel coupled together by an upper plenum. Each bayonet tube is composed by three concentric tubes, as shown in Figure 95, and the external diameter is 1 inch. The cold fluid at the inlet flows down the slave tube and then it rises in the annular gap between slave tube and outer tube, where it is heated by the heat provided to the external surface of the outer tube. To avoid large heat transfer between ascending and descending fluid, an insulation with air or inert gas is made between the inner and the slave tube. 110 band-type electrical heaters placed on each tube provide thermal power to the test section.

Polidori et al. (Polidori, et al., 2016) have discussed the execution of the experimental campaign on the HERO-2 component. The tests have allowed the creation of a valuable database for the characterization of a heated steam-water mixture flowing in a bayonet tube. Tested pressure varies between 2.0 and 7.0 MPa and mass flowrate is included between 7.0 and 50.0 g/s per tube. The maximum average heat flux supplied by electric resistor is around 38 kW/m^2, considering thermal losses, with maximum allowed temperature of the heaters equal to 350°C. However, each tube of the test section is subdivided into two different power-controlled zones, in order to provide different heating in saturation and super-heated conditions.

The experiment also focused on the detection and quantification of thermal-hydraulic instabilities of the tubes under specific operating conditions. The measurements of several instruments have shown an oscillatory behavior even operating only one tube. The oscillations impact those steady-states characterized by lower mass flowrate and higher power. The interpretation of this behavior given by the authors is the onset of Density Wave Oscillation (DWO) induced by a valve downstream the test section, which acts as an orifice. DWOs have been observed also on tests with parallel tubes, for cases at low pressure, low flowrate and high heat flux.

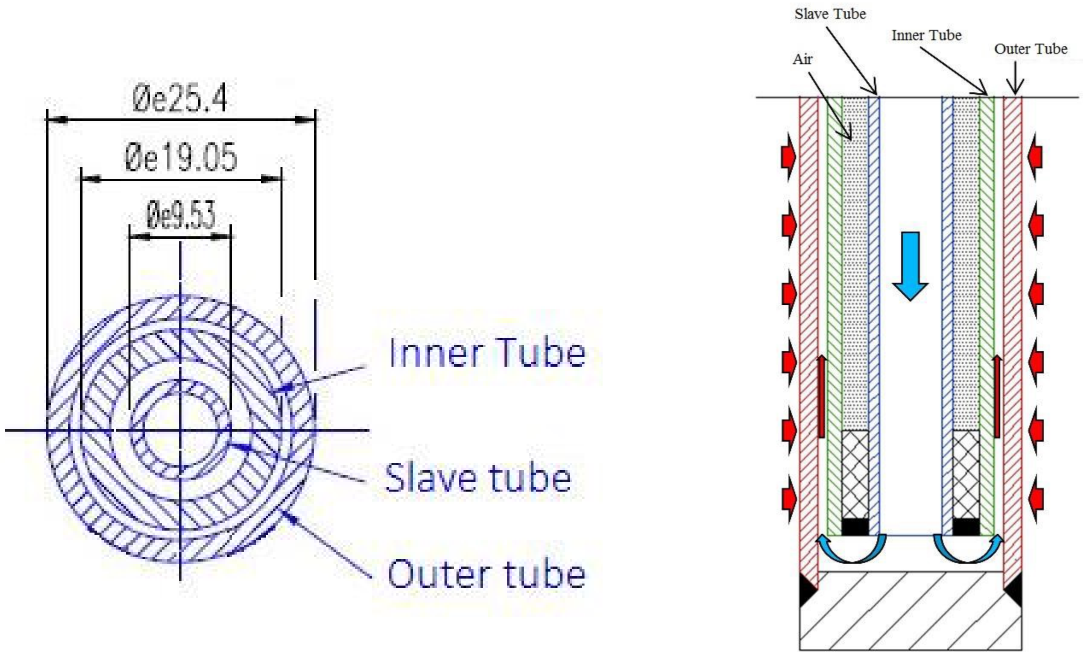
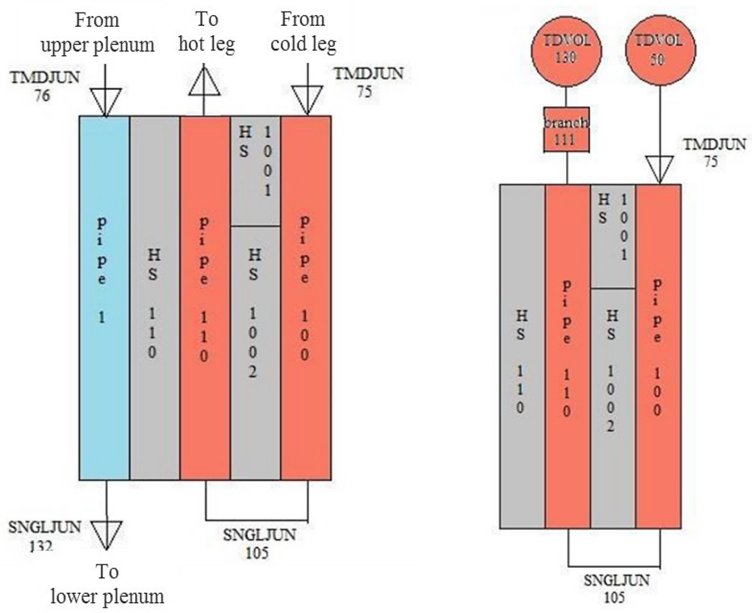


Figure 95 HERO-2 bayonet tube section (Polidori, et al., 2016)

Figure 96 View of the HERO-2 facility at SIET labs, Piacenza, Italy

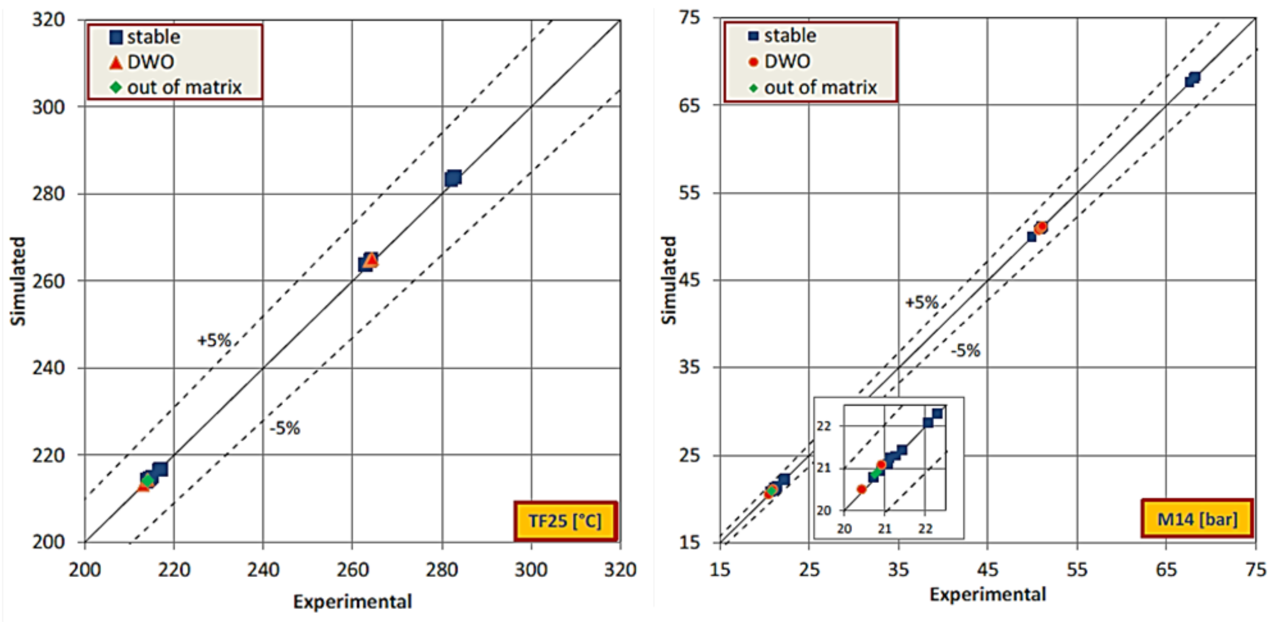
5.2.2 Relap5 model and validation

Caramello et al. (Caramello, et al., 2015-b) and Polidori et al. (Polidori, et al., 2016) designed several Relap5 models to simulate the behavior of a bayonet tube and compared it to the experimental data. The reference model is a single bayonet tube operating in a steam generator, coupled with a single tube which has geometrical dimensions equal to the equivalent hydraulic diameter of the flow channel of the primary fluid. This single-tube model (Figure 97a) represents heat transfer phenomena, temperature and pressure trends inside the steam generator unit. Another RELAP5 model was built to better match the experimental condition of test, in which no primary fluid exists but thermal power is provided by an imposed heat flux (Figure 97b). Both these models are not sufficient to identify possible parallel channel instabilities due to asymmetrical behavior of the fluid. To do that, a third model simulating both parallel pipes of HERO-2 facility was created (Figure 97c).



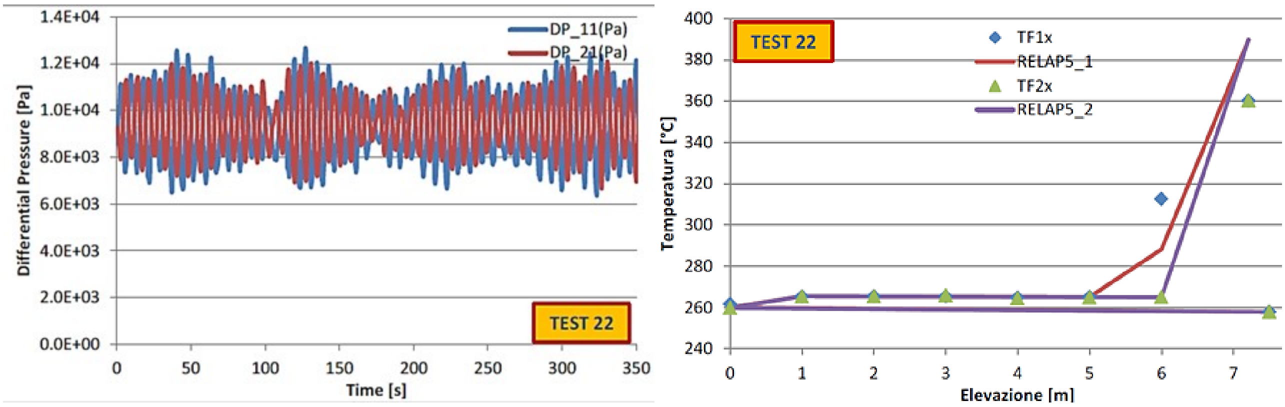
(a) (b) (c)

Figure 97 Relap5 bayonet tube models



(a) (b)

Figure 98 Examples of comparison of experimental vs. simulated temperatures (a) and pressures (b) inside the annulus of the bayonet tube (Polidori, et al., 2016)



(a) (b)

Figure 99 Validation of Relap5 model for a test case affected by DWO – pressure 5.0 MPa, flowrate 10 g/s, heating power 20 kW per tube (Polidori, et al., 2016)

The post-test analysis with Relap5 allowed the validation of the models through the experimental data. Good agreement was found between model predictions and results of the experiments, regarding both thermal-hydraulic conditions and instabilities. The model is capable to predict the unstable behavior of the flow at low pressure and high heat flux. As an example, some comparisons between model predictions and experimental points are shown in Figure 98 and Figure 99

5.2.3 Simulation of bayonet tube SG integrated in an EHRS

The Relap5 model of a bayonet tube SG in Figure 97a, validated against experimental data, has been integrated in the EHRS model illustrated in sub-section 3.3. The purpose is to compare the performances between the helical and the bayonet configurations during a SBO accident scenario. Unlike the helical coil SG case, a preliminary design for a bayonet tube SG to be placed inside an integral SMR does not exist yet. The total number of tubes necessary to transfer the total thermal power can be determined by keeping similar heat transfer surface with respect to the helical coil SG case. The case tested considers equal heat transfer surface between the two technological solutions: 16847 tubes are necessary, assuming that the header may reduce the usable surface of the SG from 4 m to 3.5 m. This is done to consider the headers at the top of the steam generator. Inner tube and annulus diameters are taken equal to those of HERO-2 facility. Operating and boundary conditions are kept equal to the helical case. The resulting Relap5 model is given in Figure 100.

During the SBO simulation, the general behavior of the secondary is successful. The power exchanged in the bayonet tube steam generator and in the EHRS is always greater than the decay heat. The profiles of heat transferred as functions of simulation time are observable in Figure 101. Unlike the helical SG case, the condition of the flow at the outlet of the bayonet SG reveals that the fluid, after an initial transient, is single-phase (Figure 102). This effect happens because the length of the SG tubes is much smaller (3.5 versus 30.64 m) and the fluid cannot reach the saturation point. The hydraulic resistance of the bayonet SG is lower, with respect to the helical configuration, therefore the lack of steam generation is balanced by the higher mass flowrate. The calculation of the collapsed liquid level ensures that, for the reference case, no risks of core uncovering are taken during the 5 hours of simulated transient. However, also in this case the situation becomes critical in case half of the heat transfer surface of the SG becomes unavailable (Figure 103).

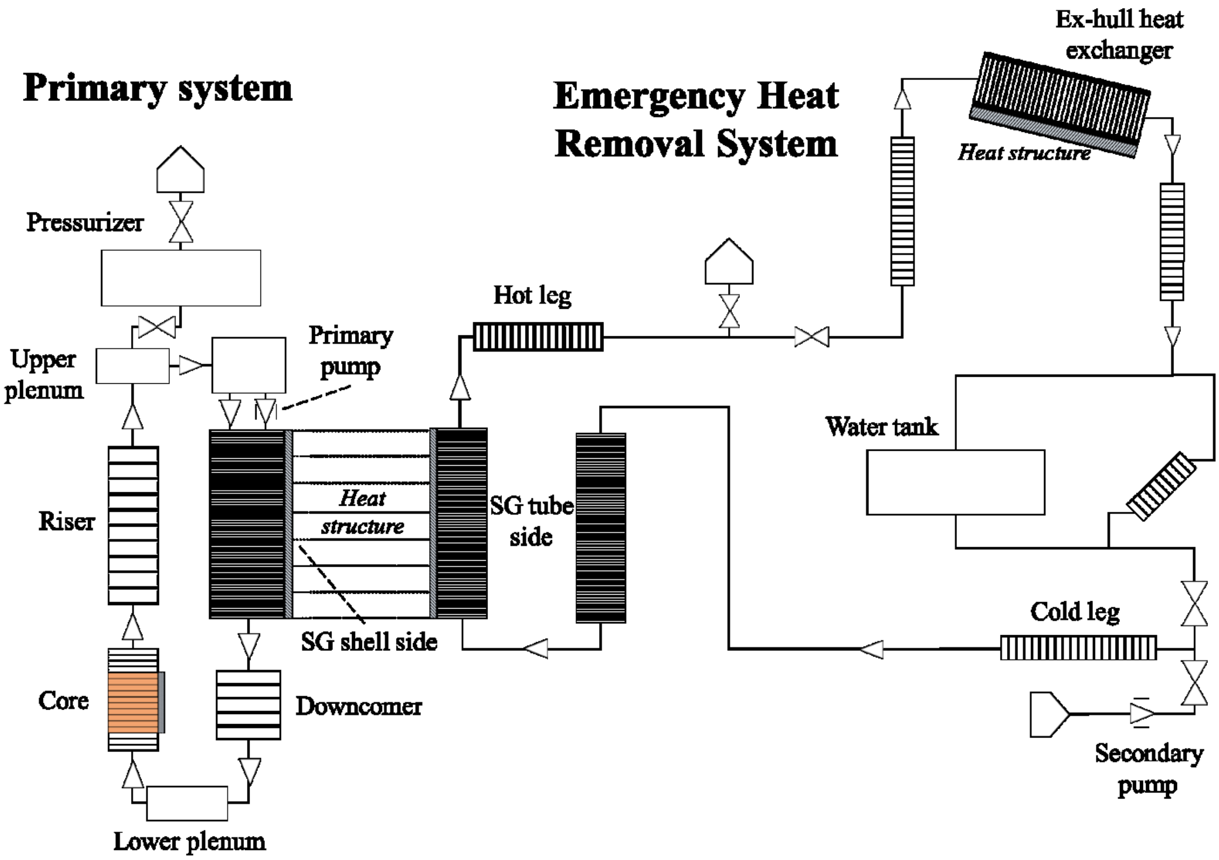


Figure 100 Relap5 nodalization of the primary system + EHRS with bayonet tube SG

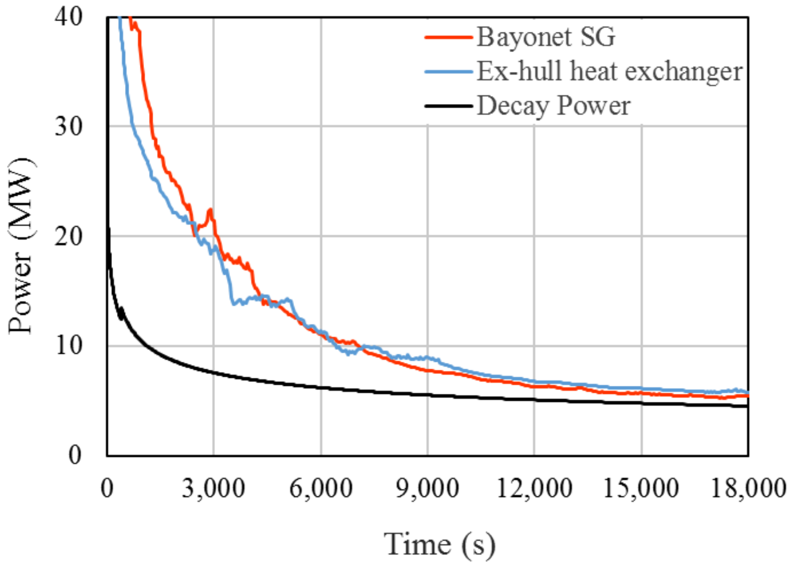


Figure 101 Comparison among decay power (black curve), power exchanged through the SG (red curve) and power exchanged through the ex-hull heat exchanger (blue curve)

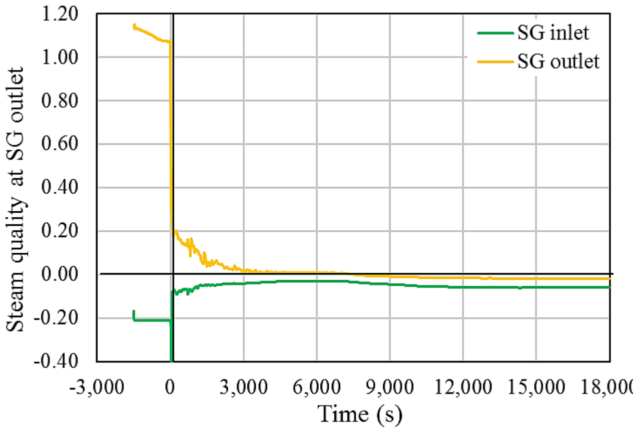


Figure 102 Steam quality at the bayonet SG secondary side, inlet and outlet

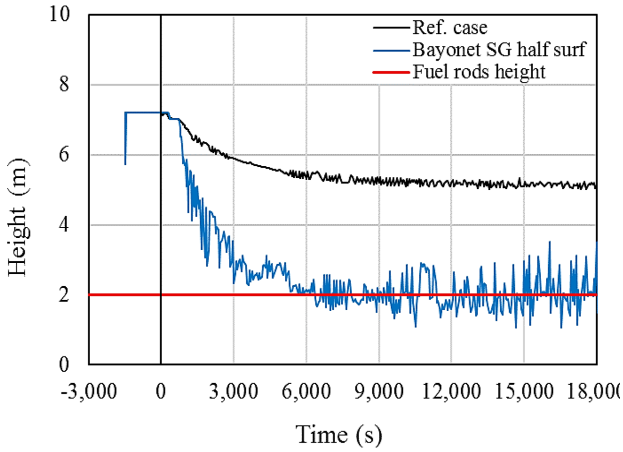


Figure 103 Collapsed liquid level in core barrel (zero is the base of active core)

5.2.4 Concluding remarks

In the design of innovative steam generator for SMR, bayonet tubes can offer an interesting alternative to conventional layouts. Thanks to a specific experimental campaign, ENEA has created a database on the two-phase flow in bayonet tubes, focusing on SMRs applications. The experiments have explored the typical operating conditions of a once through SG integrated in the RPV of a SMR, identifying also some conditions for onset of DWO instabilities. The data have allowed the validation of Relap5 models. These have been used to simulate the behavior of a bayonet tube SG integrated in a EHRS, during a SBO scenario. The comparison with the helical coil SG analyzed in sub-section 3.3 shows that both configurations behave well if all the SG heat transfer surface is available. In case of failure of one SG module, criticalities may arise.

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Section 6

Critical issues to be addressed for the deployment of submerged SMRs

This section examines the critical issues that still require to be addresses in order to achieve the final design, licensing and commercialization of submerged SMRs. These issues regard both engineering and non-engineering aspects and include: (i) design of a boron free core, (ii) remote operating and control, (iii) maintenance and refueling, (iv) seismic assessment, (v) licensing procedures, (vi) international regulation, (vii) economic competitiveness, (viii) public acceptance. A critical discussion of these aspects is proposed.

6.1. Overview

Whilst the passive safety strategy described in Section 3, based on a permanent and infinite heat sink, owns a "ultimate" potential to solve the most challenging scenarios about nuclear safety, the deployment of a submerged SMR presents several other critical issues. To achieve the final design, licensing and commercialization of submerged SMRs, many aspects still require to be seriously addressed. Lee et al. (Lee, et al., 2015) provided a brief review about the engineering challenges of offshore nuclear power plants. Many of the proposed issues are in common with the submerged SMR concept, while some topics that can be matter of concern for floating reactors, such as tsunamis and marine collisions, find inherent protection with a reactor moored on the seafloor or also operating in an artificial lake.

The aspects identified and discussed in this section, which are given by the peculiar location of the underwater SMR and are not present in conventional reactors, are: (i) design of a boron free core, (ii) remote operating and control, (iii) maintenance and refueling, (iv) seafloor seismic assessment, (v) licensing procedures, (vi) international regulation, (vii) economic competitiveness, (viii) public acceptance. Some of them are not merely engineering problems, but they regard also other fields, such as economics, law, regulation and society. Strict connections among these subjects can be noticed, e.g., between the maintenance procedures of a submerged SMR and the economic sustainability or international regulation about transportation of spent nuclear fuel. Addressing all these issues would require not only a separate approach, but also an overall picture of the deployment strategy.

6.2. Engineering issues

6.2.1 Design of a boron free core

Currently, commercial PWRs operate with boric acid diluted in the primary coolant basically for three reasons (Ingremeau & Cordiez, 2015): (i) to achieve cold shutdown, (ii) to control reactivity swing during the cycle, reducing reactivity of fresh fuel and enhancing it when fuel is depleting, and (iii) to allow load following only by control on the Chemical Volume Control System (CVCS). The use of soluble boron eases the management of reactivity and it gives advantages in terms of enhanced burnup and, in general, fuel performances. However, it carries also several drawbacks, concerning both operational and safety aspects. In the last 30 years, many studies have been performed to investigate the feasibility of boron-free PWRs. A pioneering study was performed by EPRI (Sugnet & Yedidia, 1989), while a large contribution has been given by the CEA - Commissariat à l'Energie Atomique - Franc (Fiorini, et al., 1999) e (Thomet , 1999) (Deffain, et al., 1999). Also, a Korean study proposing the conceptual design of a boron-free large PWR core is available in literature (Kim & Kim, 1999). All these works have been motivated by the research of simplified design, operation and maintenance. The presence of boric acid in the primary circuit is cause of corrosion and generates a large contribution on effluent volume. This effect makes the design of CVCS much complex and voluminous. From the safety viewpoint, the presence of boron as neutrons absorber lessens the negative moderator coefficient in case of a sudden increase of the coolant temperature. In addition, wrong boron dilution in the coolant is a specific scenario that need to be addressed by the safety analysis. A comprehensive analysis of advantages and disadvantages of soluble boron-free core has been provided by Fiorini et al. (Fiorini, et al., 1999). The authors considered three cases: maintaining the soluble boron during cold shutdown; soluble boron only for accident conditions; total elimination of soluble boron. In general, the design of reliable boron-free PWR systems is considered a breakthrough of the nuclear technology, especially in relation to the use of mixed oxides and new types of fuel.

The use of soluble boron in a submerged SMR has been discussed by Ingremeau and Cordiez for the Flexblue case (Ingremeau & Cordiez, 2015). They observed that the recycling system of borated water, which is voluminous and requires frequent maintenance, cannot be suitable for an underwater reactor, where the available space is limited, and maintenance cycles must be quite long. They noticed also that a design based on boron control can lead to criticality in case of a severe scenario with seawater flooding the reactor compartment, if boron is used to maintain cold shutdown. Therefore, a submerged PW-SMR cannot rely on boron for reactivity control and boron-free core is a must for this technology. The design of soluble boron free core in a submerged SMR needs to define an accurate strategy to safely manage the cold shutdown, the control rod ejection and other type of reactivity accidents, the Xenon stability, the load following and the reactivity swing. The mentioned study by Ingremeau and Cordiez used a numerical approach and focused on two aspects: the achievement of cold shutdown and the optimization of core design, extending the fuel cycle and maximizing the availability of the reactor. In conclusion, it proposed a preliminary core design for Flexblue, relying also on an innovative heterogeneous configuration of gadolinium distribution. In addition to a more difficult management of neutronic transients, boron-free core poses other critical points regarding the design of the control rods driving mechanism. Integration in the RPV layout is more complex with no boron, because each assembly requires more control rods. Also, the diversification of the insertion mechanism is fundamental aspect, since there is not another system for reactivity control and shut down.

6.2.2 Remote operating and control

A sea-based SMR operating few kilometers far from the shore would need a remote-control system. The distance between the reactor site and the control room is much larger than in conventional power plants and the control system would require more components and cables. There are more variables that can cause the damage/failure of the system and disturb the control operations. To ensure the reliability of remote control operations, ad-hoc I&C systems need to be developed and tested. An important work has been performed by DCNS for the Flexblue reactor (IAEA, 2017). DCNS has designed a robust Instrumentation and Control (I&C) system based on its experience in the naval industry, especially nuclear submarines. The I&C system is designed to be operated from a remote control room. Thanks to redundancies of trains, different levels of defense especially for communication systems, technological diversification and the principle of reactor autonomy, the I&C system is considered highly reliable. A single control room can operate different power units, thus reducing costs.

6.2.3 Maintenance and refueling

Concerning maintenance, the two key challenges of a submerged SMR are the limitation of human interventions and the increase of reactor availability. During operation, Flexblue has no human presence on board, but periodic access of operators for light maintenance can be planned. The ongoing suited are focused on the automation of maintenance operations, remote control maintenance or reengineering of the equipment to eliminate the need for maintenance. (IAEA, 2017) These factors can reduce the need of human access to the reactor and therefore increase the availability.

Figure 104 Artist views of a Flexblue module lifecycle: a) the subsea production site and the onshore control center; b) the ship transport; c) the support facility for refueling and maintenance; d) the transport back to the production site. (Gourmel, et al., 2016)

Refueling and large maintenance operations would need to be done in factory, moving the reactor from the site and then extending the stop period. The refueling process shown in Figure 104. In addition, DCNS has planned that each Flexblue power unit would undergo major maintenance and control every ten years, i.e., every three fuel cycles. These operations can be cause of a long period of unavailability, considerably larger than the stopping time of conventional reactors. The issue is not problematic if the power plant has 4/5 or more units, since the unavailability of one module would represent a reduction of less than 25% of the electric output of the plant. In contrast, if only the plant has a unique module, the refueling period is a critical aspect. In principle, it could be solved by replacing the whole reactor module with another unit. However, the replacement is not feasible at the beginning of the deployment of the concept, when few units have been produced and are operating.

6.2.4 Seismic assessment

In general, the prevention from the effects of an earthquake in a submerged SMR moored on the seafloor has several common features with the seismic protection of land-based reactors. The reactor needs to be isolated from the ground and isolation systems specific for a marine application must be designed. Kim et al. (Kim, et al., 2014-b) studied the case of an offshore reactor operating on a Gravity-Based Structure (GBS) and introduced a base isolation system to reduce acceleration by adjusting the total weight of the GBS. The study is very interesting, since it addresses the seismic issue of an offshore reactor. However, the case of a submerged SMR moored on the seabed, like Flexblue, has peculiar features that require not only structural investigations of the reactor and isolation from the ground, but also geological analyses of the site. One of the main concern in case of a submarine earthquake is the stability of the seabed. The choice of the site would require the accurate analysis of the composition of the soil and its response in case of seismic event.

6.3. Non-engineering issues

6.3.1 Licensing procedures

In almost all countries, licensing regulation has been developed for large power plants, therefore procedures still need to be adapted to SMRs (Ramana, et al., 2013). Reference works on SMR licensing issues were authored by Soderhom (Soderholm, 2012) and Soderholm et al. (Söderholm, et al., 2014). These works identify the main features that make the licensing of a SMR different from a large size plant, e.g. the standardization of the design, the modularity concept, the mass production and the serial construction. The works also propose a conceptual methodology to approach the licensing of SMRs. Another issue is that the licensing process must be included into a variegated international framework, where requirements are often variable from one country to another. The gravity of licensing schedule is highly critical in the process of SMR deployment, since the benefits of short construction time would be weakened, or even lost, if the long licensing process prolongs the commissioning and approach to full-power operation. Therefore, strong actions concerning the simplification and harmonization of licensing models are strictly required, especially in Europe. On this path, an important effort is under way at IAEA level: the SMR Regulator's Forum has been established in 2015 (IAEA, 2015). Moreover, within the World Nuclear Association, the CORDEL Working Group in 2013 established the Small Modular Reactor Ad-hoc Group (SMRAG), to elaborate a path towards harmonized and well-regulated global SMR deployment (CORDEL, 2015).

Currently, little or no experience about the licensing of offshore SMRs is owned by the nuclear industry. The main reference on this field is the floating barge KLT-40, which is under construction in Russian Federation (Kuznetsov, 2012) and received final approval from Russian authority in late 2017 (World Nuclear News, 2018). Issues specific of a nuclear reactor operating on the seafloor still need to be addressed.

6.3.2 International regulation

Whilst land-based reactors are typically managed by a single State and there is no need to involve other countries, transportable nuclear reactors pose some questions concerning the international regulation and agreements among countries. At date, no regulatory frameworks relevant for transportation of fueled Transportable Nuclear Power Plants (TNPPs) exist, but only general regulations for nuclear ships and for the transportation of dangerous goods and nuclear materials. Two aspects can be identified in this field: (i) the transportation of the nuclear power plant containing fissile material and irradiated fuel and (ii) the relations between the host State of the TNPP and a foreign supplier/operator. These challenges have not been fully addressed yet in international regulation. A work group by IAEA (IAEA, 2013-a) observed that the development of these standards is a kind of chicken/egg discussion, i.e. how the TNPP should be designed so that it can meet the transportation requirements, or how the requirements should be written so it can meet the design demands. Therefore, the work group argued that requirements must be developed in parallel with the technological development. In practice, the main aspects are safety-related. Cooperation between states (governments, industry, regulators, etc.) would be essential to apply rules with an international consensus.

In recent years, IAEA (IAEA, 2013-b) developed a preliminary study about this topic. The analysis addressed several challenges of the deployment of transportable nuclear power plant from the viewpoints of legal issues. These challenges include: nuclear safety, radioprotection, security, safeguards, liability. The work is based on two cases of TNPP, i.e., (i) completely assembled, fueled tested maintained, re-fueled and decommissioned in factory or (ii) assembled and pre-tested in factory, tested, maintained, re-fueled and decommissioned on site. In addition, from a legal and institutional viewpoint the work also considers two scenarios: (i) the supplier provides, operates and takes back the entire reactor, including the spent fuel, while the host State is the regulator and licenses the TNPP; (ii) the TNPP is not operated by the supplier, but by an entity established by the host State, with all the other aspects equal to scenario (i).

6.3.3 Economic competitiveness

The economic competitiveness of submerged SMR concept is not differential with respect to on-shore SMR designs, since it relies as well on modular investment, series production and possibility to build the reactor in factory and not on site. Economic advantages of SMR in terms of easier financing and accelerated return on investment, besides criticalities due to the economy of scale, are applicable also to the underwater reactor case. However, refueling and O&M for submerged SMRs could be much more expansive than for conventional on-shore nuclear power plants. Such operations may largely affect the reactor availability, especially for the first power units deployed, for which it is not possible to replace the reactor under refueling with a spare unit to reduce the impact on the utilization factor. Therefore, the most critical aspect of the economic sustainability of a submerged SMR is to start the market: the energy cost of the first-of-a-kind would be considerably larger than the n-th unit.

Haratyk et al. (Haratyk, et al., 2014) estimated 100 €/MWh as targeted energy cost for Flexblue. This price is high if compared with larger nuclear and fossil fuel plants but can become competitive in some niches of the energy market, e.g. in zones where there are energy needs but the land is scarce, isolated or not suitable for the construction of a power plant. Efforts should be done to reduce this cost, acting on fuel cycle and standardization of maintenance operations.

6.4. Public acceptance

6.4.1 Overview

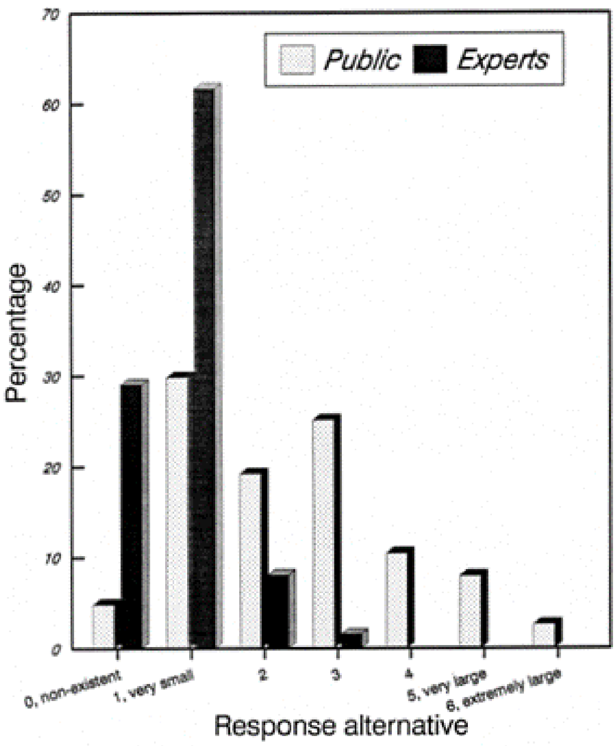
Today, the public acceptance of nuclear power is a fundamental aspect, perhaps being in Western countries the most relevant and complicated issue that can determine the deployment of a new nuclear technology. The presence of strong emotional and ethical concerns in non-expert population has always characterized the debate about the peaceful uses of nuclear power. During the last 60 years, environmental associations all around the world have opposed harsh resistances to those governments that decided to adopt nuclear technology for electricity production, often gaining large consents among the public opinion. Such protestations have focused on several ethical implications posed by the unavoidable presence of potentially dangerous scenarios of many nuclear activities and by the operations acted for their preventions. The two main concerns regard the safety of facilities, since a nuclear accident has in general very low probability to occur but potentially very large consequences, and the security against malevolent actions and proliferation of fissile material. Shrader-Frechette (Shrader-Frechette, 1980) analyzed the controversies raised by the use of nuclear technology for energy production and recognized a critical aspect of the public debate. In the author's opinion, environmentalists' positions are often overly emotive and question begging and ethical conclusions are carried out in an intuitive, rather than a closely reasoned, manner. On the other hand, nuclear proponents often ignore the social and ethical aspects of energy decision-making, focusing instead only on a purely scientific assessment of fission generation of electricity. The brief analysis presented in this sub-section aims to provide a general overlook on the issue of public acceptance in Western countries and to point out some key aspects of a good communication strategy.

6.4.2 A "non-rational" debate

Laypeople often show a non-rational fear of ionizing radiations. Psychometric studies, in which expert and non-expert categories are demanded to express their risk perception about nuclear power and several other human activities, allow evidencing public perceptions of radiation risk differ from the assessments of most experts. Figure 105 show the results of two studies by Slovic et al. - Canada, 1980 - (Slovic, et al., 1987) and by Sjöberg and Drottz-Sjöberg - Sweden, 1994 - (Sjöberg & Drottz-Sjöberg, 1999). This uncharacteristic difference in the perception of nuclear risk is due to the different approach to the analysis of the matter. To express their judgment, experts employ technical knowledges that in general are not owned by laypeople. Therefore, many researchers agree that a considerably large part of the population tends to fill the lack of information with moral and emotional elements and that attitude formation is not, in the technologists' sense, a "rational" process (Otway, et al., 1978). The study by Slovic et al. is particularly significant, considering that interviews have been done before Chernobyl nuclear disaster. The Fukushima event has further changed public attitudes toward nuclear energy and the level of acceptance of nuclear power has diminished after the accident (Kim, et al., 2013). the result of this general situation is that the debates around nuclear power are often heated and emotional, overlooking the rational aspects. Policy makers typically respond to these emotions in two ways: either they ignore the emotions of the public or they take them as a reason to prohibit or restrict nuclear technology (Taebi, et al., 2012). Roeser (Roeser, 2011) called these two options the "technocratic pitfall" and the "populist pitfall". Country such as US, Russian Federation, China and (in the past) France have followed the first solution, while Germany, Italy and Switzerland have decided to quit nuclear power on the populist wave following Chernobyl or Fukushima accidents. In both cases, a genuine debate about nuclear energy has always been avoided and nowadays the subject is still controversial.

Activity or technology	League of Women Voters	College students	Active club members	Experts
Nuclear power	1	1	8	20
Motor vehicles	2	5	3	1
Handguns	3	2	1	4
Smoking	4	3	4	2
Motorcycles	5	6	2	6
Alcoholic beverages	6	7	5	3
General (private) aviation	7	15	11	12
Police work	8	8	7	17
Pesticides	9	4	15	8
Surgery	10	11	9	5
Fire fighting	11	10	6	18
Large construction	12	14	13	13
Hunting	13	18	10	23
Spray cans	14	13	23	26
Mountain climbing	15	22	12	29
Bicycles	16	24	14	15
Commercial aviation	17	16	18	16
Electric power (non-nuclear)	18	19	19	9
Swimming	19	30	17	10
Contraceptives	20	9	22	11
Skiing	21	25	16	30
X-rays	22	17	24	7
High school and college football	23	26	21	27
Railroads	24	23	29	19
Food preservatives	25	12	28	14
Food coloring	26	20	30	21
Power mowers	27	28	25	28
Prescription antibiotics	28	21	26	24
Home appliances	29	27	27	22
Vaccinations	30	29	29	25

Ordering of perceived risk for 30 activities and technologies (Slovic, et al., 1987). The ordering is based on the geometric mean risk ratings within each group. Rank 1 represents the riskiest activity or technology. (Interviews made in 1980)

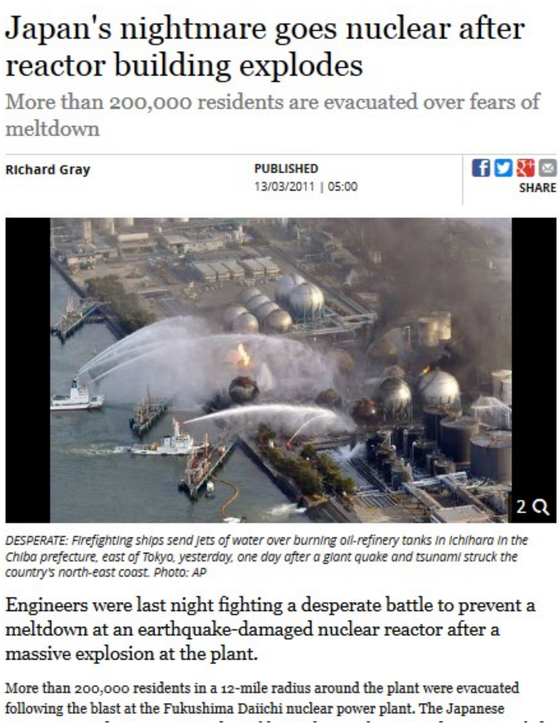


Judgments of the perceived risk of domestic nuclear power to people in general. Data from experts and from the public (Sjöberg & Drottz-Sjöberg, 1999)

Figure 105 Examples of psychometric studies about the perception of nuclear risk

6.4.3 Transparency and communication

In a framework where experts have to relate with people basing their opinion of non-rational elements, transparency and communication assume a great importance. It is common opinion that authorities did not properly managed risk communication during 2011 Fukushima-Daiichi crisis (Ng & Lean, 2012) (Figueroa, 2013) (Funabashi & Kitazawa, 2012). The Japanese and global public had received very little timely and authoritative information, often written in highly technical language. This fact lowered the trust in public administration and resulted in a sense of fear and anxiety, which led people giving consideration to questionable sources found on social media (e.g., Facebook, Twitter, but also online newspaper) where the truth was often exaggerated or misinterpreted. Consequences reflected also on the public acceptance of nuclear technology: the mentioned study by Kim et al. (Kim, et al., 2013) concluded that the reduction of consent to nuclear power after Fukushima is highly correlated with the government's political decision-making. A countercheck to this fact is given by the virtuous case of UK nuclear plan for the next decades. Taylor (Taylor, 2016) has noticed that the national policy included a large extent of public consultation, which has brought nuclear power from being the most secretive, furtive and distrusted part of the public sector to an emblem of open government, in which every decision is carefully described, explained and justified. Good transparency policies and efficient communication strategies are both a moral and ethical requirement to increase the public consent toward the nuclear option.



(a) *www.independent.ie*

(b) *www.ibtimes.com*

(c) *www.telegraph.co.uk*

(d) *www.huffingtonpost.com*

Figure 106 Examples of misleading communication adopted by some online newspapers. [7]

[7] (a) https://www.independent.ie/world-news/asia-pacific/japans-nightmare-goes-nuclear-after-reactor-building-explodes-26713055.html
(b) http://www.ibtimes.com/breaking-news-explosion-heard-japans-nuclear-plant-275295
(c) https://www.telegraph.co.uk/news/worldnews/asia/japan/8377497/Japan-earthquake-desperate-race-to-cool-atomic-reactor-before-radioactive-leak-or-worse.html
(d) https://www.huffingtonpost.com/elliott-negin/fukushima-the-story-of-a_b_4869476.html

However, also mass media share an important part of responsibility. Following a catastrophic event, it is natural that a common sense of fear and anxiety tends to arise among citizens. Therefore, people cannot bear the hole of knowledge left by the fallacies in authorities' communication strategy and, if only scarce and unclear official information is available, environmentalists and attacker newspapers have an opportunity to exploit the worries of population for political purposes or just for enlarging their audience. During the days of Fukushima nuclear crisis, a huge amount of false, misleading and even subliminal information circulated all around the world, leading non-expert people to believe that the circumstances were much more catastrophic than what they actually were. An emblematic case was the widespread use of photographs of the flaming refinery in Chiba prefecture, a facility located near Tokyo that was strongly damaged by the earthquake, to present newspaper articles about the nuclear emergency, since these images full of fire and black smoke were more upsetting than the real ones. Examples are shown in Figure 106. Thus, transparency policies of nuclear authorities and industry are fundamental also to "fight" against opponent communicators.

6.4.4 The case of the submerged SMR

Although the "submerged concept" represents an undeniable advantage for safety strategy and mitigation of severe accident consequences, the perception of the public opinion could be different, since the nuclear debate is characterized by non-rational elements. The concern that the undersea deployment is a way to escape control (Haratyk, et al., 2014) and the fear of an "irreversible" sea contamination could prevent non-experts from appreciating the safety improvements brought by the submerged SMR concept, thus keeping unchanged or even reducing the level of public acceptance. Haratyk et al. also pointed out some emotional factors, like the belief that the ocean is the cradle for life on Earth or that deep waters are the "last frontier", still protected from human activities.

6.4.5 Final remarks

Emotional and irrational elements exist in global public opinion about nuclear power and they should not be ignored. At the current evolution of nuclear debate, the question of public acceptance of submerged SMRs and, in general, of nuclear technology has some chance to be successful only if suppliers, operators, stakeholders and regulatory authorities adopt optimum transparency policies and communication strategies. In this sense, the focus should not be the explanation of technical concepts in a way accessible also to non-experts: this aspect is important, but it is not sufficient because it is not addressing the irrational part of the debate. The focus should be gaining the trust of civil population, showing that the nuclear community has a real interest to pursue the good of the society, to find the best solutions to preserve the environment and to provide easy and cheap access to electricity to the entire population. In conclusion, (i) taking on ethical implications of nuclear energy with carefulness and accuracy, (ii) avoiding underrating the emotive sensations of population and (iii) spending the necessary efforts for communication to non-experts and obtaining their trust are three key-points of a good strategy to handle the public acceptance of nuclear power. whose paramount importance will determine not only the possible deployment of the submerged concept, but also the future of nuclear energy in Western countries.

Publications

The contents of Section 6.1, 6.2 and 6.3 are published in:

Santinello, M., Ricotti, M.E., 2018. Preliminary analysis of an integral Small Modular Reactor operating in a submerged containment. *Progress in Nuclear Energy,* Volume 107, pp. 90-99

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Concluding remarks

The thesis work has addressed the study of a SMR operating in a large metallic horizontal cylinder, submerged into the sea or in an artificial lake. The research developed during the three years of the PhD activity represents a proposal and a critical, albeit non-exhaustive, analysis of reactor design and potential safety features of a submerged SMR. The outcomes of the thesis can be summarized with the following key-points.

a. The submerged SMR owns safety features that inherently protect from some of the most challenging accident scenarios. The safety strategy can rely on the presence of a heat sink, i.e. the water surrounding the reactor containment, which is permanent and infinite if the reactor is immersed into the sea or in an artificial lake. The development and deployment of this concept will represent a sort of "ultimate" solution to Fukushima-like scenarios.

b. The reactor to be placed inside the horizontal cylindrical containment must satisfy the constraint given by the limited hull diameter (14 m) and the heat transfer capability to the external water. To exploit the potentialities of the submerged concept, the reactor design must be appropriate for a fully passive safety strategy. The analysis of three proposed alternatives, i.e. VVER-type, SCOR-F and IRIS-like, has identified in the IRIS concept the most suitable design to gain compactness and to implement passive safety. A preliminary configuration has been proposed: it is a scaled version of IRIS with roughly half thermal output, where the design of primary components has been revisited to reduce the total height below 14 m.

c. A fully passive safety strategy for emergency reactor cooling has been identified, exploiting the external water as infinite heat sink. It is based on two reference processes for decay heat removal: (i) the natural circulation in the emergency heat removal system, if the primary circuit is intact; (ii) the long-term sump natural convection through the submerged containment, in case of a break in the primary circuit. This second process represents the ultimate solution for decay heat removal in case of failure of all the other safety systems.

d. The simulation of a station black-out scenario (intact primary circuit) and long-term cooling following a loss of coolant accident (broken primary circuit) has provided positive outcomes, predicting a successful and effective decay heat removal capability in all the cases simulated. Simulations performed with the system code Relap5-Mod.3.3 have shown that well-designed passive system can ensure an unlimited grace period, substantially without the need of electrical inputs or human intervention.

e. A comparison of Relap5 simulation on the long-term cooling scenario with the commercial system code Apros, developed by VTT Technical Research Center of Finland, has shown the same positive outcome about the effectiveness of the sump natural circulation. However, the behavior of the flow resolved by Apros is remarkably different from the Relap5 case. The analysis has provided useful elements to understand the differences at the basis of the two system codes. In particular, the preliminary investigation on a two-phase flow test case has revealed that Apros tends to overestimate the two-phase heat transfer coefficient, with respect to Relap5 and empirical correlations. This outcome seemed quite odd and has stimulated VTT to improve the two-phase modeling structure of Apros 6 code, fixing some issues in the HTC calculation.

f. An experimental testing is fundamental to assess the working principle of passive safety systems and to fill the gap of knowledge about several complex thermal-hydraulic phenomena. Two facilities have been designed in the framework of a proposal, namely INSPIRE, for the H2020 Euratom call on SMRs. The testing regards both the emergency heat Removal system and the heat transfer through the submerged containment. In addition, another experimental activity concerning an alternative configuration for the steam generator, i.e., the bayonet tube, is described.

g. Besides safety capabilities, the most relevant criticalities for the deployment of submerged SMRs have been identified and briefly discussed. The issues include (i) design of boron-free core, (ii) remote operation & control, (iii) maintenance and refueling, (iv) seismic assessment, (v) licensing procedures, (vi) international regulation, (vii) economic sustainability, (viii) public acceptance.

The findings of the thesis work about decay heat removal potentialities are very interesting and encourage further investigations to continue the development of the submerged SMR concept. Reactor design needs additional analysis to reinforce the proposal of an IRIS-like configuration and to define a more accurate layout. The boron-free core represents the most challenging technical issue of the reactor design. Concerning passive safety, the next step of the investigation cannot disregard the experimental testing. The submerged concept could represent a "European" SMR proposal, in competition with US, Russian and Chinese designs. Future developments should be conducted in the framework of EU collaborations. Passive safety is a very strong point, whose benefits justify addressing the critical issues identified in Section 6. Topics such as the licensing and the international regulation require joint efforts among industry, research centers/universities and Regulatory bodies (especially Technical Safety Organizations-TSOs). At the date of writing, a French consortium (CEA, EDF, Framatome, Naval Group) is strongly considering the SMR option and in the next future it will lead a EU research proposal about the topic. In this framework, the submerged concept will be accurately evaluated. Eventually, the issue about public acceptance is still the most uncertain aspect, even if not only for the submerged SMR concept, but for the nuclear option in general. It should be addressed with a suitable strategy, involving citizens, media and NGOs, targeted at obtaining the trust of the public opinion.

Appendix A

Description of simulation software: Relap5 and Apros

A.1 Relap5

A.1.1 Code description

The RELAP5 (Reactor Excursion and Leak Analysis Program) computer code is a light water reactor transient analysis code developed for the U.S. Nuclear Regulatory Commission (NRC) for use in rulemaking, licensing audit calculations, evaluation of operator guidelines, and as a basis for a nuclear plant analyzer (Information Systems Laboratories, Inc., 2003). Specific applications of this capability have included simulations of transients in LWR systems, such as loss of coolant, anticipated transients without scram, and operational transients such as loss of feedwater, loss of offsite power, station blackout, and turbine trip. RELAP5 is a highly generic code that, in addition to calculating the behavior of a reactor coolant system during a transient, can be used for simulation of a wide variety of hydraulic and thermal transients in both nuclear and nonnuclear systems involving mixtures of steam, water, non-condensable, and solute.

The specific version that used for the calculation of this work is RELAP5/MOD3.3. It has been developed jointly by the NRC (US Nuclear Regulatory Commission) and a consortium consisting of several countries and domestic organizations. The mission of the RELAP5/MOD3.3 development program was to develop a code version suitable for the analysis of all transients and postulated accidents in LWR systems, including both large and small-break loss-of-coolant accidents (LOCAs) as well as the full range of operational transients.

RELAP5 uses a two-fluid, non-equilibrium, and non-homogeneous, hydrodynamic model for transient simulation of the two-phase system behavior. The hydrodynamic model of the code is a one-dimensional, transient, two-fluid model for flow of single-phase and two-phase steam-water mixture that can contain non-condensable components in the steam phase and/or a soluble component in the water phase. Thus, simulations solve 1D-balance equations for liquid and vapor phases, which are formulated in terms of volume and time-averaged parameters of the flow. Phenomena that depend upon transverse gradients, such as friction and heat transfer, are formulated in terms of the bulk properties using empirical transfer coefficient formulations. In situations where transverse gradients cannot be represented within the framework of empirical transfer coefficients, such as subcooled boiling, additional models specially developed for the particular situation are employed. The code contains specific constitutive relations for defining flow regimes and flow-regime-related models for interphase drag and shear, the coefficient of virtual mass, wall friction, wall heat transfer, interphase heat and mass transfer, and direct (sensible) heat transfer. RELAP5 user's manuals (Information Systems Laboratories, Inc., 2003) accurately describe the strategies for physical modeling.

The code employs a control volume approach: components are subdivided into volumes connected by junctions. Each volume is characterized with a series of parameters that represent its geometry, e.g. cross section, length in the direction of the flow, hydraulic diameter, roughness. Junctions are characterized by the flow area and the form losses coefficients. Averaged hydraulic properties are evaluated in the center of the cell, which is the control volume for mass and energy quantities, except for velocities, which are evaluated in junctions, namely momentum quantities control volumes. Heat structures permit the calculation of heat transfer across solid boundaries of hydraulic volumes. Heat flow path are once again modeled in a one-dimensional sense, using a finite difference mesh to calculate temperatures and heat flux vectors. To simulate the heat flow path normal to the fluid flow path these structures can be connected to hydro-dynamic volumes. Coupling is made by solving heat transfer equations in a lumped parameter approach.

Relap5/Mod3.3 is written in FORTRAN 77 for a variety of 64-bit and 32-bit computers.

A.1.2 Code validation status

While equations and physical models used in the code have their own validation, RELAP5 developers need to provide a validation of the mathematical basis for the numerical method of solution used, in order to demonstrate the accuracy, stability, and convergence of the calculations to a solution of the original equations. This is done for Relap5/Mod3.3 through the numerical comparisons to analytical solution and separate effects experiments of four simple problems, i.e., the water faucet problem, the Edwards pipe problem, the Christensen's subcooled boiling problem, and the Oak Ridge void profile problem. The main results obtained in this work regarding the accuracy, stability, and convergence of the calculations are as follows.

- The computational mesh size used should be compatible with the modeling scale.
- The restriction of the analytic problem to a compatible mesh constitutes a well-posed problem.
- The numerical solutions are stable and convergent to a solution of the restriction of the analytic problem to a compatible computational mesh.

The complete validation procedure is accurately described in Relap5 manual (Information Systems Laboratories, Inc., 2003).

A.2 Apros

A.2.1 Code description

Apros is a software for modeling and dynamic simulation of several different industrial processes. It has been jointly developed by VTT and Fortum since 1986 (Hänninen & Ylijoki, 2008). Apros combines process modelling with automation modelling and is capable of simulate and study long term steady-conditions as well as fast transients with different degrees of details. It can be used in a wide range of applications, from deep-in-detail safety analysis to real-time training simulators. While the source code is written in Fortran, the latest version (Apros 6) presents a graphical user interface based on Simantics platform. Apros Nuclear package includes models for most of the commercial reactor designs, including PWR, BWR and VVER. Thermo-fluid-dynamics can be studied with different approaches, namely homogenous, drift-flux and separate flows.

The most advanced flow model included in the code is a two-fluid approach named "six-equation model" (Hänninen, et al., 2012). It is employed for detailed simulations and analysis of one-dimensional two-phase flow. It adopts a separate flow model, where the phase velocities may be different and a thermal non-equilibrium condition may subsist. The solution of the six partial differential equations governing the system, in terms of pressures, void fractions, phasic velocities and phasic enthalpies, is achieved through a discretization in space and time, following the principles presented by Siikonen (Siikonen, 1987). The calculation of the one-dimensional two-phase system is based on the conservation of mass, momentum and total energy; applying the balances to each phase. A further mass balance equation is computed to take into account the presence of non-condensable gas. The equations are solved with the help of empirical friction and heat transfer correlations, which strongly influence the solution.

The space discretization adopts a staggered scheme, i.e. the mass and energy equations are computed in the same control volume (node), while the momentum equation is calculated in a control volume between two nodes (branch). For the enthalpies the first order upwind scheme is used. The solution is reached through an iterative procedure, where the linear equation groups of pressures, void fractions and phasic enthalpies are solved one after the other. The material properties of each phase are calculated as a function of pressure and enthalpy. Eventually the convergence is reached when mass errors for both phases are lower than a maximum allowed value. Apros uses an adaptive time step strategy, i.e., halving the time step if convergence is not reached.

Friction and heat transfer processes, between phases and between the wall and each phase, require a separate description through empirical correlations. Various correlations are needed varying the flow regimes, the six-equation model treats bubbly, annular, droplet and stratified flows. The selection of the flow is based on void fraction, rate of stratification and rate of entrainment, the same parameters are used to determine weighting coefficients for the transition between flow regimes. Correlations and flow regimes are discussed in Apros manual (Hänninen, et al., 2012).

A.2.2 Code validation status

Apros two-phase model has been tested by calculating several well-known test cases. The validation is repeated every time new version of APROS is released. Four reference validation cases are described in detail by Hänninen & Ylijoki (Hänninen & Ylijoki, 2008). These are typical thermal-hydraulic problems:

(i) the Edwards pipe, which tests the flow in a horizontal pipe at high pressure;
(ii) a top blowdown experiment, which simulates the rise of water level in a boiling water reactor in case when the pressure is suddenly decreased;
(iii) the Becker experiment, which tests test the heat transfer correlations for dry-out and post dry-out at steady state conditions;
(iv) the rewetting on a vertical tube initially filled with steam at 0.3 MPa.

References

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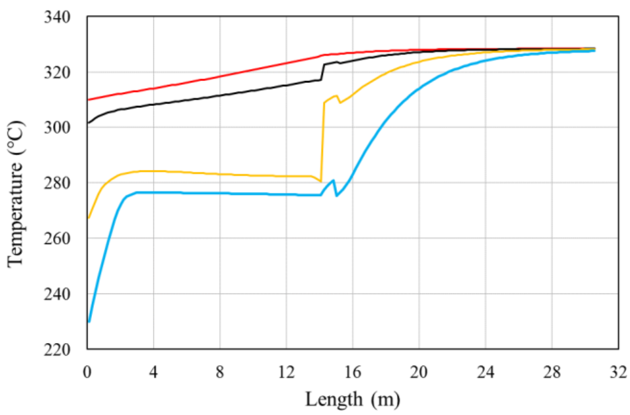
Appendix B

Comparison between Mod3.2 and Mod3.3 versions of Relap5

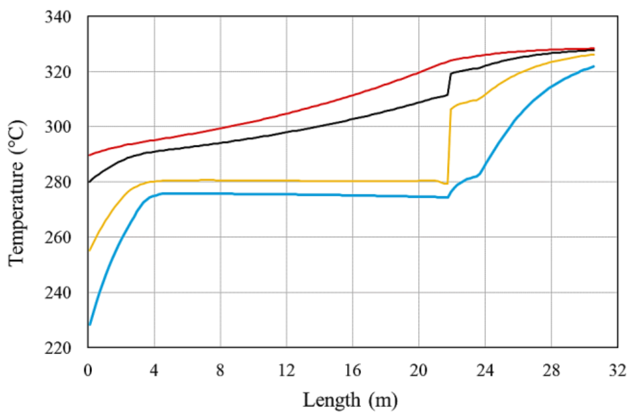
Analysis of results of SG sizing with Relap5-Mod3.2 and Relap5-Mod3.3

To complete the analysis and preliminary sizing of IRIS-160 steam generator in Section 2, this appendix provides a more detailed analysis of the discrepancies between the predictions of Mod3.2 and Mod3.3 versions of Relap5, observed in paragraph 2.3.7. As visible in Figure B1 and B2, both Mod3.2 and Mod3.3 cases show behaviors that are unclear or even unphysical.

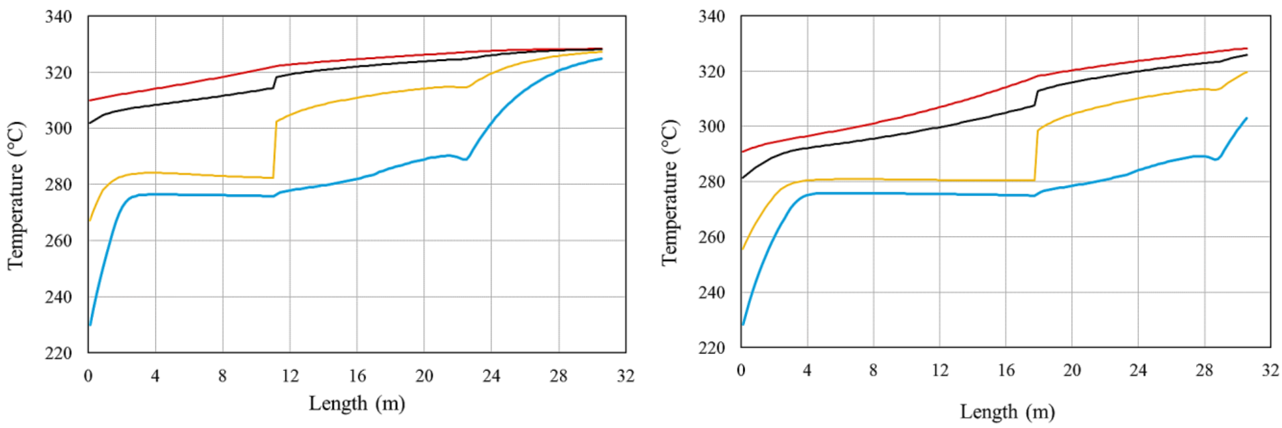
Dryout location. The two Relap5 versions predict the onset of the thermal crisis at different position of the SG tubes. Table BI reports the positions of the HTC drop from the tube inlet, corresponding to a sudden increase of the wall temperature. While the beginning of boiling is resolved at similar position in both versions, Mod3.3 tends to anticipate the dry-out. The differences between predictions of crisis point and dryout quality are quite big. This fact is uncomfortable, since the two versions are not much dissimilar and use the same correlations for HTC. The reason of such divergence is ascribable to different ways to determine flow regimes and heat transfer mode. Nevertheless, the quantitative impact on the simulation results is quite limited. The largest contribution the total thermal resistance across the tube the total thermal resistance across the tube, around 65%, comes from the conductive resistance of the metal. The in-tube HTC contributes for not more than 25%.



***(a)** Mod3.2 - $\dot{m}^I$ = 4500 kg/s*

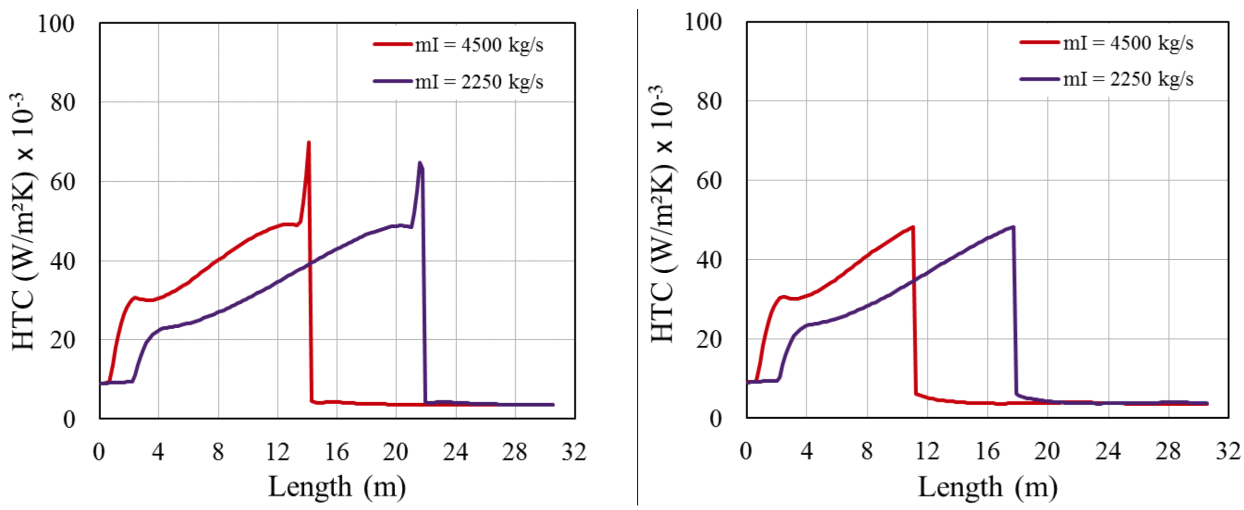


***(b)** Mod3.2 - $\dot{m}^I$ = 2250 kg/s*



(c) Mod3.3 - $\dot{m}^I$ = 4500 kg/s *(d) Mod3.3 - $\dot{m}^I$ = 2250 kg/s*

Figure B1 Temperature profiles calculated with Relap5 Mod3.2 and Mod3.3



(a) Mod3.2 *(b) Mod3.3*

Figure B2 Tube side heat transfer coefficient profiles predicted by Relap5

Table BI Comparison between Mod3.2 and Mod3.3

	Mod3.2			Mod3.3			*Difference*		
	Boiling point	**Crisis point**	**Dryout quality**	**Boiling point**	**Crisis point**	**Dryout quality**	*Boiling point*	*Crisis point*	*Dryout quality*
$\dot{m}^I$ 4500 kg/s	2.20 m	14.27 m	0.955	2.20 m	11.20 m	0.679	0 %	24.10%	28.90%
$\dot{m}^I$ 2250 kg/s	3.73 m	21.93 m	0.962	3.54 m	17.91 m	0.716	5.22%	20.18%	25.57%

Unphysical steam cooling. Past the thermal crisis, the tube bulk temperature profile exhibits a local maximum and then a slight decrease, before increasing again. This is visible in both cases with Mod3.3 and also in the low flowrate case with Mod3.2 (Figure B1a), although the location and the shape of the anomaly are different. Relap5 is predicting that saturated/superheated steam absorbs power and cools down. This is not physically consistent, since the secondary fluid is continuously being heated. This problem, which is likely to be determined by software issues, is extremely uncomfortable from a theoretical viewpoint, but has very little effects on quantitative results.

Heat transfer coefficient peak. Mod3.2 predicts a sharp peak in the heat transfer coefficient of the boiling mixture just before the dry-out, possibly too sharp (Figure B2). This jump probably related to flow regime transition and/or interpolation between different correlations. However, a similar peak has been observed by Bertoletti et al. inside of vertical straight tubes (Bertoletti, et al., 1964): just before the thermal crisis the authors observed the wet layer on the wall becoming thinner and thinner while the HTC suddenly arises. In fact, while the film thin down, the water channels into rivulets overlapped to the wet layer and this behavior cause the HTC even to double its value. When the film breaks, the thermal crisis occurs, but the rivulets continue to wet the wall causing oscillations in wall temperature that gradually decrease. Clearly, this effect could not be considered by Relap5 code.

Reference

Bertoletti, S., Lombardi, C. & Silvestri, M., 1964. *Heat transfer to steam-water mixtures,* Technical report CISE.

Acknowledgments

I have just arrived at the end of this road. This is the right time for the acknowledgements. During the three years and eight months since the beginning of the PhD path I have met several people that helped me to develop the research activity, as well as to gain knowledge, competence and professional maturity.

The first and most important thanks are for my PhD supervisor prof. **Marco Ricotti.** Despite the tons of tasks that make him probably one of the busiest man in the world, he has always found the time to take care of my doctoral research and to provide the right orientation for carrying out a good research work, organic and targeted on the challenges of the worldwide research on SMRs. I have learned a lot from his experience, his vision, his ability to manage national and international projects, his perseverance to pursue the wanted results in every situation. I am grateful also to the professors of Polimi-NRG, **Antonio Cammi**, **Lelio Luzzi** and **Hisashi Ninokata**, because, even if we did not have specific cooperation, they often showed interest in my activity and PhD path.

I would like to express my gratitude to the French friends of **DCNS** (now Naval Group), who supported the long-term safety analysis providing both technical guidelines and economic resources. In particular, I want to thank **Vincent Gourmel** for being always available to discuss our work and organize meetings in the hot summer of Paris and in the cold winter of Milan. Thanks also to the Finnish friends of **VTT** and **Fortum** for giving me the possibility to work with Apros and supporting my training course. This cooperation is the added value of my PhD thesis. A special thanks to **Ismo Karppinen** for having the perseverance to read my reports and to answer my e-mail, even during summer holidays. His suggestion about Apros and his interpretation of the results has been fundamental to reach the targets of the activity.

(now it's better if I switch to Italian…)

Era il mese di luglio 2006, l'Italia aveva appena vinto i mondiali ed io ero un diciottenne inconsapevole ed un pochino incosciente che attendeva con ansia il primo giorno utile per immatricolarsi al Politecnico di Milano. Ora invece è cambiato il mondo. Attendo il 13 luglio 2018 per discutere tre anni lavoro davanti alla commissione, con l'Italia che ai mondiali nemmeno si è qualificata e con una consapevolezza di aver fatto un cammino formativo e professionale magari non eccelso, ma sicuramente dignitoso e gratificante. Dodici anni (dodici) di Poli, dai primi tempi abbastanza spensierati di laurea triennale alle mazzate degli esami della laurea specialistica, poi il periodo da assegnista che diventa dottorato perché "già che siam qui a far questo lavoro, almeno portiamoci a casa il titolo di dottore… quello vero però, non quello dei pischelli che fanno i cori «dottore dottore» alla laurea triennale". Di tutto questo percorso sono grato al Politecnico di Milano per avermi dato la forma mentis dell'ingegnere, quella capacità di affrontare problemi complessi scomponendoli in problemi più piccoli e risolvendoli con pazienza uno alla volta… cosa che si applica non solo all'ingegneria, ma a tutti i problemi della vita.

In questi sei anni al Polimi Nuclear Reactor Group ho trovato un ambiente proprio bello (lo so, potrei usare qualche termine più forbito, ma in questo caso "bello" è la parola giusta). Ricordo i primi tempi da tesista ed assegnista in cui i dottorandi più grandi, **Manu, Ponci,** il **Colombo, Stefano** e gli

altri, mi facevano capire che qui in questo gruppo si lavora tutti insieme per costruire un clima positivo, familiare, accogliente, in cui non ti devi far problemi a chiedere un consiglio, un aiuto anche al di fuori delle collaborazioni effettive, o pure di provare una presentazione perché è la prima volta che vai ad una conferenza internazionale in Giappone e non hai mai presentato in inglese. A questo servono il pranzo tutti insieme, la briscola chiamata (la nobile arte della briscola chiamata con la sua ricerca estetica), la grigliata, la cena di Natale, le pause collettive etc. E non è un caso se ci sono tesisti di altri gruppi di ricerca che comunque a pranzo vengono da noi, perché da noi "si sta meglio". Ho quindi imparato questa cosa preziosa e ne ho "approfittato" (specialmente con il carissimo **Matteo Zanetti** che mi ha sopportato per due anni buoni), avendone io un gran bisogno dato che nei miei anni al Poli non ho mai avuto modo di collaborare direttamente con altri dottorandi. Così sono rimasto in piedi anche quando ero solo a dover impostare la mia attività di ricerca e prendere le decisioni. E per questo che, capita l'importanza di questo spirito, una volta divenuto uno dei senatori del gruppo ho cercato, con tutti i miei limiti personali, di trasmetterla un poco alle generazioni di dottorandi dopo la mia. Certo, ovviamente il modo migliore con cui "insegnavo" questa cosa era continuando a fare ciò che avevo sempre fatto, ossia rompere le balle questa volta al buon **Marco Tudor Cauzzi** per ogni dubbio termo-idraulico o al buon **Tommaso Barani** per tutte le mie lamentele e tutti gli sviluppi della mia tesi. E così questi anni sono volati, velocemente, passando da essere un dottorando inesperto fra menti super brillanti a uno di quelli con la maggiore anzianità, che cerca faticosamente di fare qualcosa si spera dignitoso nel proprio ambito. A camminare al mio passo (solo in termini di percorso di dottorato, perché sul piano della ricerca e delle competenze sta un bel pezzo avanti) un certo **Davide Pizzocri**, conosciuto per caso all'esame di Zio nel lontano 2012 e poi compagno di tutti gli altri esami del dottorato, oltre che dei passaggi d'anno e della notiziona della riduzione retroattiva dei crediti necessari per il dottorato che, tradotto, significava poter balzare l'esame di fuel cell (che momento euforico quello!). Sono onorato di poter condividere con lui anche il giorno della defense e la proclamazione. Sono davvero grato a tutti i ragazzi del Polimi-NRG, PhD passati, presenti e futuri, per questo fantastico ambiente di lavoro di cui facciamo parte, per i pranzi, per le partite a briscola (quelle memorabili e quelle inguardabili), per le "letture illuminate", per le chiacchierate e le risate, tutte cose che mi mancheranno tremendamente una volta fuori dal Poli. In ordine cronologico: **Marco** (il **Colombo**), **Manu, Ponci, Alberto, Stefano, Matteo, Pini, Davide, Sara, Cauzz, Tommy, Eric** (compagno di avventure cinesi), **Christian, Carolina** (compagna di avventure americane), **Luana, Andrea,** oltre a tutti i tesisti che sono passati per il nostro gruppo.

Certo, poi il dottorato è anche quel periodo dove il tuo capo ti dà dei tesisti per il tuo progetto di ricerca e ti dice "bon, occupatene tu". E così devo improvvisarmi mentore per qualcuno, per poi ritrovare il mio nome sulle tesi di **Giorgia Baldocchi** e **Luigi Iacopini**, persone che sostanzialmente si sono messe a lavorare per me ed hanno avuto la pazienza di ascoltarmi. Invece per qualche strano motivo il mio nome è finito anche sulla tesi di **Roberto Pettini,** anche se il mio contributo al suo egregio lavoro è stato davvero minimo. Però sono contento di questa cosa, perché lascia un segno delle nostre avventure finlandesi.

So già che a settembre avrò una sensazione stranissima quando per la prima volta dopo 12 anni non dovrò più andare in Bovisa tutti i giorni. Anche se, come condannato al grigiore della periferia milanese, ho sempre invidiato chi frequenta università in location top-level (tipo Como e Lecco dove in pausa si fa la passeggiata sul lungo lago, o la statale di Milano dove tutte le mattine si passa davanti al Duomo), alla fine mi sono affezionato a questi luoghi, alla vista del gasometro e dell'aereo, alle strutture gialle sempre più sbiadite del grande capannone, alle chiacchiere da bar con quelli del **Marvy** dopo la giornata di serie A e la Champions League. So già che la Bovisa mi mancherà.

Se torno indietro al 2014, quando iniziai il dottorato, il 2018 mi sembrava lontanissimo. Questi sono stati gli anni in cui ho visto tanti amici e parenti dare la direzione alle loro vite: chi trova lavoro fisso, chi prende casa, si trasferisce all'estero, chi si sposa e mette su famiglia. Il mio futuro invece è quanto mai incerto. Dopo la **"Reazione GorlaMexicana"** (ecco, omettiamo il nome vero che è meglio), non ho ancora idea di cosa sarà della mia vita. Certo, papà e mamma mi stanno aiutando a ristrutturare una casetta mia e questo è già un passo importante. Ma devo ancora decidere cosa voglio fare da grande, o meglio, chi voglio essere da grande. Sono fiducioso che tutto quello che ho ricevuto in questi anni mi permetterà di cavarmela dignitosamente anche questa volta. La preghiera più grande che rivolgo a Dio è che del mio futuro facciano parte tutte le persone belle del mio presente. Per gruppi WhatsApp storici e meno storici: gli amici da tanto del gruppo **"Budapest"**, gli scappati di casa della **"Benste"**, il gruppo Fantacalcio (che anche qui è meglio non scrivere il nome vero) e i **"The kings of the mountains"** che un giorno non lontano arriveranno in vetta al Kilimanjaro; i **"Giovani Catechismo"** ed i sacerdoti punti di riferimento spirituale con cui sono cresciuto e sto crescendo nella fede, fra cui il **"Il magnifico coro UPG"** che sopporta un direttore decisamente stressante e tutti gli educatori con cui condivido l'educazione nella fede dei ragazzi più piccoli; **"L'altra ensemble"** che ancora non ha trovato un nome ma non importa perché ciò che conta è stare bene suonando insieme, la **"Banda Gorla Maggiore"** perché ha nome Corpo Musicale Santa Cecilia da 118 anni e da 16 ne faccio parte stando bene e suonando insieme; i ragazzi della **"Vita comune Gruppo Samuele"** che, anche se non ci sentiamo da un po', hanno decisamente lasciato un segno importante durante gli anni del mio dottorato.

Si chiude quindi questo grande capitolo della mia vita. Lascio il Politecnico felice per quanto ho dato e ricevuto, consapevole delle mie forze e speranzoso per il futuro.

Milano, 6 giugno 2018