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Assessment of the safeguards to cope with the consequences of severe accidents in nuclear power plants

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ASSESSMENT OF THE SAFEGUARDS TO COPE WITH THE CONSEQUENCES OF SEVERE ACCIDENTS IN NUCLEAR POWER PLANTS

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Il presente documento descrive le attività di ricerca svolte all'interno dell'Accordo di collaborazione "Sviluppo competenze scientifiche nel campo della sicurezza nucleare e collaborazione ai programmi internazionali per i nucleare di IV Generazione"
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Sommario

This report presents the activities performed in the frame of LP1, Objective B (Accident consequences evaluation), task B2-1 of PAR 2016, ADP ENEA-MSE.
The analysis of the measures to reduce the risk due to severe accidents is reported.
It includes the safeguards designed to prevent and mitigate the consequences of severe accidents, such as the passive safety systems for the residual heat removal from the Containment.
The analysis is conducted by applying both deterministic and probabilistic approaches.

Note:

This document has been prepared with the following main contributors:

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List of Acronyms

AP 600/1000	Advanced Passive
CCF	Common Cause Failure
CMC	Ceramic Matrix Carbide
DW	Drywell
ENEA	Italian National Agency for New Technologies, Energy and Sustainable Economic Development
ESBWR	Economic Simplified Boiling Water Reactor
FAPCS	Fuel and Auxiliary Pools Cooling System
FMEA	Failure Mode and Effect Analysis
GDCS	Gravity Driven Cooling System
HEX	Heat Exchanger
IAEA	International Atomic Energy Agency
IC	Isolation Condenser
ICC	Internal Core Catcher
ICS	Isolation Condenser System
IVR	In-Vessel Retention
IVCR	In-Vessel Corium Retention
LOCA	Loss of Coolant Accidents
LWR	Light Water Reactor
MELCOR	Methods for Estimation of Leakages and COnsequences of Releases
NCF	Natural Circulation Failure
NPP	Nuclear Power Plant
PCC	Passive Cooling Condenser
PCCS	Passive Containment Cooling System
PCF	PCC Condenser Failure
PSA	Probabilistic Safety Assessment, Probabilistic Safety Analysis
PWR	Pressurized Water Reactor
RMPS	Reliability Methods for Passive Systems
RPV	Reactor Pressure Vessel
R-S	Resistance - Stress
SAM	Severe Accident Management
SBWR	Simplified Boiling Water Reactor
SMA	Safety Margin Assessment
SSC	Structure System and Components
T-H	Thermal-hydraulic

Executive Summary

As part of the implementation of strategies for the management of severe accidents (SAM, Severe Accident Management), the activity consists of the evaluation of performance in terms of reliability and availability of the safeguards to reduce the risk due to severe accidents, as a fundamental requirement resulting from the lessons learned from the accident in Fukushima.

These devices, such as systems for the retention of the corium inside the vessel, accumulators, heat exchangers for evaporation, condensation systems or for ventilation of the containment through filters for the removal of the pressure, etc. are designed for the prevention and mitigation of consequences of severe accidents.

The study proposes, on the one hand, the analysis of reliability of passive systems, for the purpose of maintaining the function of containment of radioactive products, as a follow on of a recent activity, see ENEA report ADP-FISS-LP1-072 “Analysis relating to the implementation of the safety safeguards for the severe accident management in nuclear reactors” issued in September 2016 by the same authors, which shows some specific aspects of passive systems for further evaluation of safety in probabilistic terms. On the other hand it is intended to study the behavior of the containment structures, such as the vessel, the containment system, etc., in order to assess from a deterministic point of view the safety margin (SMA) and propose engineering solutions/safeguards, based as well on the use of innovative materials for in-vessel retention, aimed at mitigating the effects of a severe accident scenarios.

Fukushima Daiichi accident emphasized that the complex phenomenology of In Vessel Corium Retention (IVCR) and Ex Vessel Corium Cooling/stabilization (EVCC) strategies has to have a highest priority to cope with the prevention of the failure of the bottom head of the vessel, first, and of the containment and/or the basemat melt-through, later.

Particularly the IVMR achieved by external reactor vessel cooling and/or in-vessel flooding is one of the most effective measures to prevent the progression of severe accidents at water-cooled reactors in order to reach a final stable state of the core and other sources of fission products, within prescribed limits (in the case of Anticipated Transient without Scram (ATWS), Station Blackout (SBO), and loss of all secondary feedwater (LOFW, for PWRs), etc.). In what follows we will investigate a new IVCR approach, based on an internal core catcher made of alumina.

To the aim, preliminary feasibility of the envisaged solution will be provided by means of analytical and/or simplified numerical models which allow to study the complexity of the transitional incidental and highlight the benefits of implementing the proposed strategies for the management of severe accidents.

1. Introduction and scope

The extensive use of passive safety systems, designed to extend the coping period in the event of an adverse situations such as occurred at the Fukushima Daiichi plant, is one of the most relevant strategies envisaged to meet the requirements to enhance the safety of nuclear power plants, as devised in ref.1, which provides a list of systems, consistent with the available knowledge as emerging, for the most, from a literature survey.

The study deals with the assessment of passive systems, implemented in reactor designs to exercise and ensure the safety functions such as removing heat from the containment and the control and abatement of the amount of hydrogen that could break free, following a severe accident involving the melting of the core.

In particular, considering the specificities of their operation, a careful analysis is required to evaluate their performance and ascertain the accomplishment of the required safety mission.

Within a risk informed approach, the study proposes:

- on the one hand, the assessment of the reliability of passive systems, conceived to maintain the function of containment of radioactive products, as a follow-on of ref.1, which shows some specific aspects of passive systems for further evaluation of safety in a reliability plane.

- on the other hand it is intended to study the behavior of the containment structures, such as the vessel, the containment system, etc., in order to assess from a deterministic point of view the safety margin (SMA) and propose engineering solutions/safeguards, based as well on the use of innovative materials for in-vessel retention, aimed at mitigating the effects of a severe accident scenarios.

  To these aims, preliminary proposal (feasibility) will employ analytical and/or simplified numerical models which allow to study the complexity of the transitional incidental and highlight the benefits of implementing the proposed strategies for the management of severe accidents.

2. Reliability analysis for passive systems – A case study on a passive containment cooling system

In ref.1 it is recognized that safety considerations and preliminary performance evaluations in probabilistic terms, are initially drawn mainly basing on operational aspects and expected performance/availability of the systems apportioned to prevent and mitigate the consequences of severe core damage.

In order to bolster the role of passive systems being implemented in reactor designs it is necessary to demonstrate their effectiveness in reliability terms to achieve the safety goal in severe accident management strategies.

This part is intended to deepen the topical issue related to passive system reliability and availability, which, in ref.1, is addressed in the form of preliminary analysis, to evaluate the fulfillment of their performance with respect to the safety requirements necessary to protect the health and safety of general public from radiological hazards.

Ref. 1 refers to passive safety systems relevant to severe accidents and Level 2 PSA as the systems required to prevent the containment modes of failure and the measures for Severe Accident Management (SAM) strategy, thus identifying two different groups of design features meant to improve the containment performance to cope with severe accidents.

The first category concerns the passive safety systems relevant to containment integrity and devoted to heat removal from the Containment, like e.g., Passive Containment Cooling System (PCCS) for AP1000 and PCCS condensers for ESBWR.

The other class refers more properly to systems tasked to cool the debris bed after the core melt, like In-Vessel Retention (IVR) of molten core via water cooling, and measures to cope with the risk of hydrogen explosion, like the passive autocatalytic recombiners.

Thus these systems are of possible interest, as proposed in ref.1.:

Passive safety systems for heat removal from the containment

- PCCS, Passive Containment Cooling System designed for AP600/AP1000
- PCCS, Passive Containment Cooling System condensers designed for SBWR/ESBWR

Passive safety systems as specific measures for SAM

- IVR (In-vessel Retention) System for AP1000

- Autocatalytic Re combiners

According to severe accident terminology, as defined in ref.2 for PWRs, these safety systems refer to specific representative containment failure mechanisms which are depicted by the so-called α, β, γ, δ, ε -modes:

- α-mode corresponds in general to the steam explosion mechanism in water-cooled reactors: in terms of consequences, the missile generation threatening the containment and SSC is dreaded;
- β-mode corresponds to the containment isolation failure:
  - trough interfacing systems as a result of induced rupture of heat exchanger walls following the core degradation onset;
  - by containment penetrations to auxiliary buildings (failure to isolate the containment);
- γ-mode corresponds to combustion phenomena (mostly H_2 combustion in PWRs) in the containment building potentially leading to its early failure;
- δ-mode corresponds to the late failure of the containment due to slow over-pressurization;
- ε-mode corresponds to the containment failure through the base mat, generally following the corium spreading in the containment building.

As underlined in ref.1, performance assessment of passive safety systems pertinent to severe accident, will represent a new challenge owing to the large amount of uncertainties, mainly because of the lack of significant operational and experimental data, as well as the modeling of the involved phenomena, like e.g., the condensation and boiling heat transfer coefficients or the heat transfer coefficients under the presence of non-condensable gases. Consequently difficulties arise to achieve a qualified reliability figure, since the scarcity of data and the little experience. This is a crucial issue to be resolved for their extensive use in future nuclear power plants.
Therefore the present analysis is performed according to the level of definition of the design, based on the information available mostly from the open literature, with the objective to provide any significant reliability/availability figures of merit, as a measure of risk management.

Finally, due to the actual limited resources, the analysis is limited to Passive Containment Cooling System condensers designed for SBWR/ESBWR, referred to as T-H passive system.
At first a brief overview on passive system reliability approaches is provided.

3. Reliability assessment for a passive system: generics

A number of methodologies have been developed in order to investigate the reliability of T-H passive systems. These include approaches based on independent failure modes, failure modes of passive system hardware components, functional failure, and the reliability methods for passive safety functions (RMPS), as reported in ref.3.
In the approach based on independent failure modes, the reliability is seen from the perspectives of system/component reliability and physical phenomena reliability. The first contribution calls for engineered passive components and it is treated in the classical way, i.e. in terms of failures of components. The latter is concerned with the way the natural physical phenomena operate. The failure probability is evaluated as the probability of occurrence of the different failure modes which are considered independent. Failure causes are seen in terms of critical parameters for the natural circulation performance or stability. Difficulties arise for example when identifying probability density functions for the states of important parameters.
Difficulties in independent failure modes approach can be overcome by associating each physical failure mode to a failure mode of a hardware component designed to ensure the conditions for successful system performance. Thus the probabilities of physical failures are reduced to unreliabilities of the components whose failures complicate the successful passive system operation. For example some problems in a heat transfer process can simply be seen as a failure of heat exchangers.
The functional failure approach exploits the concept of functional failure to define the probability of failing to carry out a given safety function. The idea is adopted from the resistance – stress (*R*-*S*) interference model from fracture mechanics. For T-H passive systems reliability assessment, *R* expresses safety functional requirement on a physical parameter and *S* expresses system state. *R* can be for example a minimum required value for water mass flow, whereas *S* could represent the actual value of mass flow. Probability distributions are assigned to both *R* and *S* and failure probability is computed as the probability that *S* is less than *R*, with reference to the flow rate case. Hence the states of the system are divided into the failed and the safe states.

The reliability methods for passive safety (RMPS) functions is a research and development framework programme supported by the European Union. The RMPS functions addresses issues such as

- Definition of failure criteria of the passive system.
- Identification and quantification of the sources of uncertainty and determination of the most important ones.
- Propagation of the uncertainties through T-H modelling.
- Evaluation of the passive system unreliability.

The RMPS methodology consists of several steps, starting with the identification and characterization of the accident scenario and ends up with the reliability evaluation of the system in question.

The reliance of passive systems on inherent physical principles makes the reliability assessment quite difficult to accomplish in comparison to classical system reliability analysis. The current knowledge of passive reliability contains large uncertainties, especially in the area of thermal-hydraulics. The assessment of reliability of the passive T-H systems is a crucial issue because they are increasingly used in future NPPs. By developing solid methods to beat uncertainties relating to reliability assessment of T-H passive systems, also the public acceptance for future reactor systems may increase.

It is essential to be aware of the fundamental differences between the analysis of passive and active safety systems. Reliability assessment for an active system can even be regarded as a somewhat mechanical and straightforward procedure whereas issues relating passive systems are of higher complexity containing more uncertainties. In active systems the number of components is usually higher than in their passive counterparts, and in active systems dependencies within the system can be considered mainly functional, whilst passive systems rely on physical or phenomenal principles and are more difficult to model.

For active systems it is often easy to identify the failure modes, which are typically discrete and few in number. The system either works or does not work. For passive systems the measures which determine failure states can be continuous instead, and the failure criterion can be defined to be e.g. some percentage value of the performance in nominal conditions.

Also there is normally more information and reliability data available on active systems. Knowledge of passive systems rely more on expert judgements and simulations.

4. Passive Containment Cooling System Operation and Description (ESBWR-like configuration)

The PCCS is required to maintain the containment within its pressure limits for DBAs. The system is designed as a passive system with no components that must actively function, and it is also designed for conditions that equal or exceed the upper limits of containment severe accident capability. The PCCS consists of six, low-pressure, totally independent loops, each containing a steam condenser (Passive Containment Cooling Condenser), as shown in Figure 1 (Ref. 4). Each PCCS condenser loop is designed for 11 MWt capacity and is made of two identical modules. Together with the pressure suppression containment, the PCCS condensers limit containment pressure to less than its design pressure for at least 72 hours after a LOCA without makeup to the IC/PCC pool, and beyond 72 hours with pool makeup. The PCCS condensers are located in a large pool (IC/PCC pool) positioned above, and outside, the ESBWR containment (DW).

Each PCCS condenser loop is configured as follows. A central steam supply pipe is provided which is open to the containment at its lower end, and it feeds two horizontal headers through two branch pipes at its upper end. Steam is condensed inside vertical tubes and the condensate is collected in two lower headers. The vent and drain lines from each lower header are routed to the DW through a single containment penetration per condenser module as shown on the diagram. The condensate drains into an annular duct around the vent pipe and then flows in a line that connects to a large common drain line, which also receives flow from the other header, ending in a GDCS pool.

The non-condensable vent line is the pathway by which drywell non condensables are transferred to the wet well. This ensures a low non condensable concentration in the steam in the condenser, necessary for good heat transfer. During periods in which PCCS heat removal is less than decay heat, excess steam also flows to the suppression pool via this pathway.

The PCCS loops receive a steam-gas mixture supply directly from the DW. The PCCS loops are initially driven by the pressure between the containment DW and the suppression pool during a LOCA and then by gravity drainage of steam condensed in the tubes, so they require no sensing, control, logic or power-actuated devices to function. The PCCS loops are an extension of the safety-related containment and do not have isolation valves.

Spectacle flanges are included in the drain line and in the vent line to conduct post-maintenance leakage tests separately from Type A containment leakage tests. Located on the drain line and submerged in the GDCS pool, just upstream of the discharge point, is a loop seal. It prevents backflow of steam and gas mixture from the DW to the vent line, which would otherwise short circuit the flow through the PCCS condenser to the vent line. It also provides long-term operational assurance that the PCCS condenser is fed via the steam supply line.

Each PCCS condenser is located in a subcompartment of the IC/PCC pool, and all pool subcompartments communicate at their lower ends to enable full use of the collective water inventory independent of the operational status of any given IC/PCCS sub-loop. A valve is provided at the bottom of each PCC subcompartment that can be closed so the subcompartment can be emptied of water to allow PCCS condenser maintenance.

Pool water can heat up to about 101°C (214°F); steam formed, being non-radioactive and having a slight positive pressure relative to station ambient, vents from the steam space above each PCCS condenser where it is released to the atmosphere through large-diameter discharge vents. A moisture separator is installed at the entrance to the discharge vent lines to preclude excessive moisture carryover and loss of IC/PCC pool water. IC/PCC pool makeup clean water supply for replenishing level is normally provided from the Makeup Water System.

Level control is accomplished by using an air-operated valve in the makeup water supply line. The valve opening and closing is controlled by water level signal sent by a level transmitter sensing water level in the IC/PCC pool. Cooling and cleanup of IC/PCC pool water is performed by the Fuel and Auxiliary Pools Cooling System (FAPCS). The FAPCS provides safety-related dedicated makeup piping, independent of any other piping, which provides an attachment connection at grade elevation in the station yard outside the reactor building, whereby a post-LOCA water supply can be connected. There have been extensive qualification tests of the PCCS, including full-scale component tests and full height scaled integral tests.

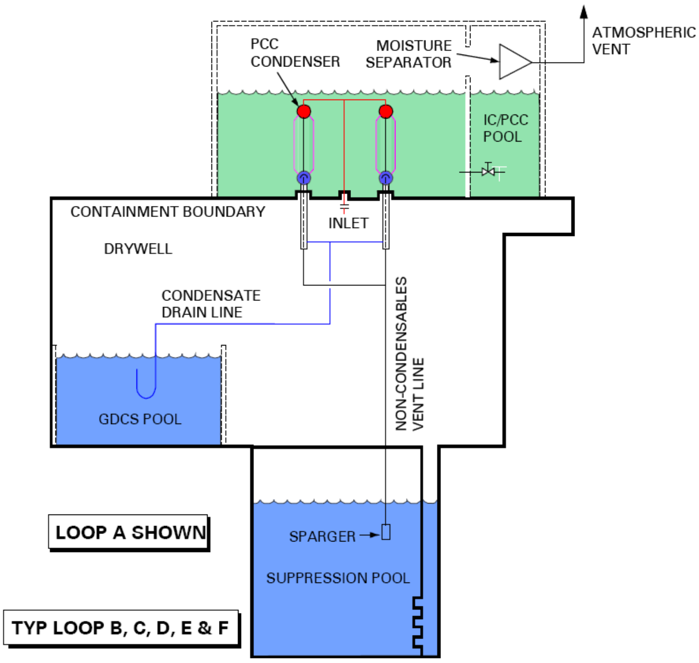

Figure 1: ESBWR Passive Containment Cooling System (redundancy degree not reported)

5. Reliability assessment of Passive Containment Cooling System (ESBWR-like configuration)

Ref. 1 lays the foundation for the system reliability assessment: the paucity of research material addressing the system assessment on the probabilistic viewpoint, i.e., reliability, rather than the performance evaluation, represents a serious concern. In accordance with ref. 5, PCCS is classified as a type B passive system, i.e. it relies on natural circulation: its configuration and operation exhibits many analogies and similarities with the Isolation Condenser System (ICS) (ref. 6), utilised for decay heat removal function during an isolation transient.

As previously pointed out, the reliability assessment of the PCCS, or any other passive thermal-hydraulic system, necessarily differs from classical component reliability based approach. The significant uncertainty related to thermal-hydraulic system operation is complicating reliability analysis.

However, we have seen that also methods which apply classical reliability assessment techniques have been developed.

Many other techniques rely on computer programs, which simulate the thermal hydraulic physical phenomena related to the system under investigation, in our case the MELCOR code. When the failure criteria have been defined, it is possible to provide a reliability estimate for the passive safety system according to the simulations.

The thermal hydraulic analysis, conducted on a set of cases defined by design and critical parameters, randomly selected from the probability distributions (Monte Carlo simulation), gives rise to a set of results, in terms of system performance, duly elaborated to achieve a proper reliability figure.

However, given the limited resources and for the purpose of the case study, we'll limit our investigation to commonly adopted methodologies for reliability analysis, thus overlooking the thermal hydraulic assessment by code simulation.

The assessment may also be purely qualitative without giving any numerical values to evaluate the reliability.

Consistently with the reliability-oriented methodology proposed in ref.1 and resumed in the previous paragraph, the approach for the reliability assessment of such systems consists of two parts: the first part includes the classical reliability analysis of components and the second part concerns the passive function, in this case, natural circulation.

Quantification of a thermal-hydraulic system is challenging because of the large number of uncertainties, and because of this, one has to sometimes settle for qualitative analysis. It is often beneficial to begin reliability studies with failure modes and effects analysis (FMEA), which helps to identify potential failure modes and effects related to them.

Natural circulation reliability is then evaluated through reliability analysis of the components designed to assure the best conditions for the function of it. It calls for identification of the mechanisms which maintain the intrinsic phenomena.

Thus the reliability assessment can be conducted according to the classical procedure using for example fault trees.

System unavailability can be a result of either a defect of some component or a failure of natural circulation, referred to as functional failure, identifying a unique failure mechanism, i.e the failure of the system to complete its desired role due to a deviation from the expected conditions rather than the failure of a physiscal component (ref. 3).

5.1 Reliability data and failure modes and effects analysis

In PCCS, the most critical components are obviously the heat exchangers and in addition to that, piping is of great significance. In Table 1 the reliability data, used for the analysis, are reported, basing mainly on ref. 6, where a similar study was performed for the isolation condenser system and component reliability data were taken from sources 2 and 7. In addition expert judgements were of great benefit. The absence of the triggering valve on the condensate line and on the vent lines marks the difference with respect to the ICS. This adds simplicity to the present study.

Note that CCFs (Common Cause Failures) are not addressed in this study, acknowledging the compliance of the redundant systems with the criteria of independence, separation and diversification. Unplanned demands and spurious operations of the system are disregarded, as well as the associated recovery actions from failure to operate.

Component	Failure mode	Failure rate
HEX	Single pipe rupture	$3.0\ 10^{-10}$/h
HEX	Multiple pipe rupture	$3.0\ 10^{-11}$/h
HEX	Single pipe plugging	$3.0\ 10^{-10}$/h
HEX	Multiple pipe plugging	$3.0\ 10^{-11}$/h
Piping	Rupture	$2.4\ 10^{-8}$/h

Table 1: PCC component reliability data

The failure probability of the natural circulation upon which the system operation is based implies the identification of the corresponding failure modes and the evaluation of the uncertainties associated with their evaluation. Three main failure modes can be identified to be a loss of heat transfer, large amount of noncondensible gases and envelope failure, i.e. loss of primary boundary.

Loss of heat transfer to an external source (PCCS pool) can be due to insufficient water in the pool or due to heat exchanger pipe excessive fouling. The lack of water can occur because of leakages or malfunction of devices responsible for maintaining sufficient water level. Presence of non condensable gases can result from a fouling in vent line. In that occasion, the system is unable to purge the non condensables into the suppression pool in the wetwell. The envelope failure is in principle present also in the component based analysis as a piping rupture.

The reliability data for natural circulation failure modes are once again from ref.6, except for the insufficient water, for which the failure rate evaluation is given by the author, thus containing uncertainty of the highest kind (we consider $1.0\ 10^{-7}$, since the function must be assured for 72 hours of mission time without make up: thus the value refers to pool leakage/rupture): the envelope failure (i.e., loss of primary boundary) failure mode is given as the failure rate relative to piping rupture; excessive pipe fouling failure mode is assigned the failure rate relative to multiple pipe plugging for the heat exchanger, while insufficient pool inventory is represented by the makeup valve fault.

The values are in Table 2.

Component	Failure mode	Failure rate
Heat exchanger	Excessive pipe fouling	$3.0\ 10^{-11}$/h
PCCS pool	Insufficient water	$1.0\ 10^{-7}$/h
Vent line	Excessive pipe fouling	$3.0\ 10^{-11}$/h
Primary boundary	Rupture	$2.4\ 10^{-8}$/h

Table 2: PCC natural circulation reliability data

Failure modes and effects analysis (FMEA) is a step-by-step approach to discover the possible ways in which a system of interest might fail, while identifying also the effects and consequences resulting from different failure modes and , finally, classifying them according to the seriousness of consequences and failure frequency. The purpose of the FMEA is identify reliability critical areas, to discover which actions are most essential to take in order to eliminate or reduce failures.

In Table 3 is a simple outline of a qualitative FMEA for PCCS, not committing itself on the quantitative evaluation of severity of failures, including, in addition to the mechanical components reported in Table 1, the natural circulation function represented in Table 2.

Component/Function	Failure Mode	Failure Cause	Consequences
Heat Exchanger	Pipe rupture; Pipe plugging		System not operational
Piping	Pipe rupture		System not operational
Natural circulation	Insufficient heat transfer	Insufficient water in the PCCS pool; Pipe fouling	Decreased heat transfer capability
	Envelope failure	Piping rupture	System not operational
	High concentration of non condensable gases	Vent line fouling	No vent line flow

Table 3: Simplified FMEA for PCCS

5.2 Fault tree failure probability

In this paragraph the failure probability of the system is evaluated adopting the approach based on classical fault tree, which is constructed in Figure 2.

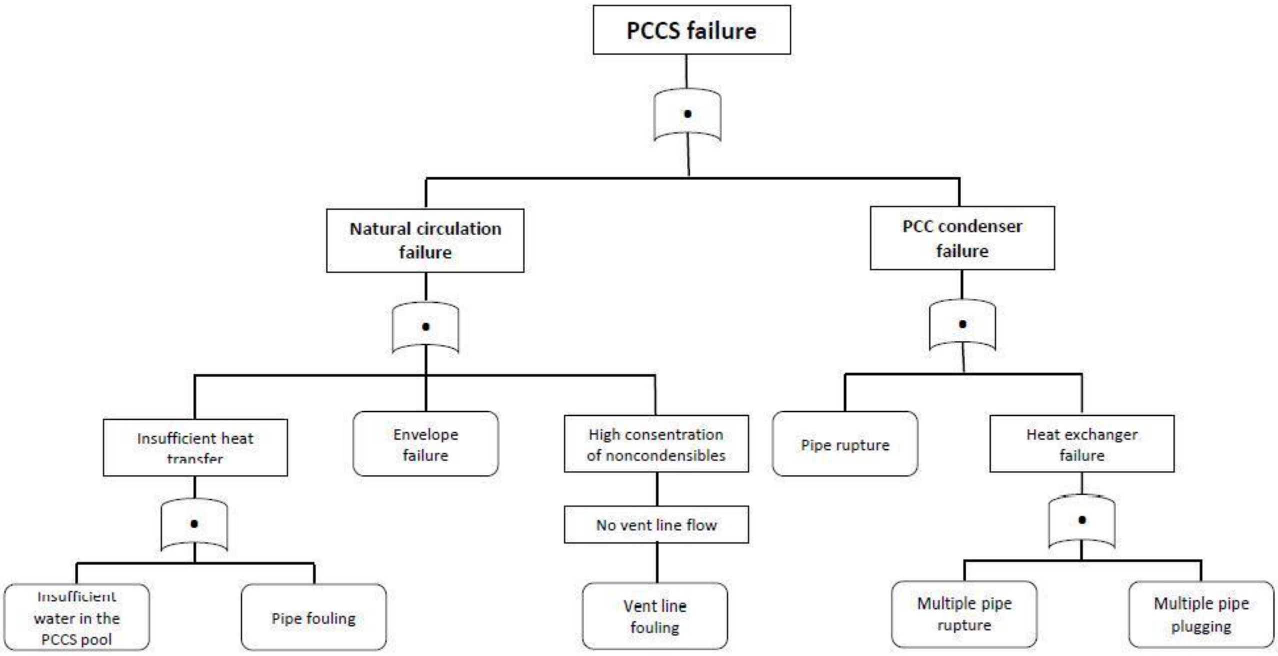

Figure 2: Fault tree for PCCS

It is consistent with FMEA in Table 3and it consists of two branches, one for failures due to natural circulation and the other for PCC component failures. PCCS failure rate can be quite easily calculated with help of this tree and the component failure rates are given in Table 1 and Table 2. It is worthwhile to consider failure probabilities of each branch separately, because an indication of relative importance can thus be acquired. Because there are only or gates present in the tree, exact probabilities are straightforward to calculate by using using Boolean algebra. The events in the tree are denoted as follows:

NCF	Natural circulation failure
A	Insufficient water in the PCCS pool
B	Heat exchanger pipe fouling
C	Envelope failure
D	Vent line fouling

PCF	PCC condenser failure
E	Pipe rupture
F	Multiple heat exchanger pipe rupture
G	Multiple heat exchanger pipe plugging

NCF = A+B+C+D

PCF = E+F+G

Failure probabilities are assessed taking a 72 hours value for the mission time T_m of the system consistently with the passive system requirement. Note that single component failure probabilities are evaluated in the form of $\lambda^* T_m$, which is acceptable for $\lambda^* T_m <0,1$ like in the present case.

In Table 4 failure probabilities for both fault tree branches and the total failure probability are reported.

It has to be noted that, despite the analysis is carried out on the basis of quite generic numerical values assigned to an oversimplfied model of the system (for instance the two-unit heat exchanger is modelled as a single component), one infers that the natural circulation failure probability, which is evaluated with regard to the failure of specific system components, is the main contributor to the total system unavailability. This outcome is consistent with ref. 6, where a companion analysis relative to the IC is provided.

This is not surprising, seen the similarities and the features the two systems share, both in terms of components and structures and modes of operation.

System	Failure probability	Contributor (%)
Natural circulation	$8.932\ 10^{-6}$	83,76%
PCC condenser	$1.732\ 10^{-6}$	16,24%
PCCS Total	$1.066\ 10^{-5}$	100%

Table 4: Failure probabilities for the PCCS

Moreover, it has to be pointed out that the aforementioned degree of redundancy of the units (for example, the system foreseen for the SBWR consists of six redundant units, each unit made of two identical modules that act as heat exchangers) leads to a reduction in failure probability values.

The minimal-cut-set analysis identifies the failure of heat transfer to the external source due to insufficient water in the PCCS pool (in the study this failure is represented by the basic event consisting of the pool leakage/rupture) as the most important contribution to the natural circulation reliability together with the loss of primary boundary, which is the “major” responsible for the failure probability of the PCC.

Conversely HX modes of failure impairing the heat exchange process and preventing the system from accomplishing its function contribute very little to the reliability figure.

5.3 Interface with T-H analysis

The connection of MELCOR simulations to reliability assessment and probabilistic safety assessment (PSA) provides both an opportunity and a challenge.

In fact it has to be noticed, that the simulations by the MELCOR code, do not relate directly to the reliability assessment of PCCS and the results are more useful to illustrate PCCS operation and performance. However, the information is valuable among other things for the determination of failure criteria for further reliability analysis and also builds up the understanding of the phenomena involved.

For instance, one example of a candidate for a failure criterion is the total energy transferred to the PCCS pool. Other quite obvious alternatives are the drywell pressure and temperature. The criterion can be chosen to be some percentage value of a performance measure in nominal conditions.

The reasons of this statement lies in the fact that MELCOR is intended primarily to model the progression of severe accidents. It is quite simple, approachable and flexible through its block based nature, but it may not be very effective in conducting analysis of performance or reliability for a single safety system such as PCCS. If intention is to obtain a numeric reliability estimate for PCCS, some more specific thermal-hydraulic codes could potentially be more practical for this purpose of use. With MELCOR it is difficult to introduce probabilistic aspects into the analysis. It would be beneficial to conduct reliability assessment with a tool with which it is possible to sample system parameter values from given distributions, once the appropriate ones are determined. However, information gained via simulations can be exploited e.g. for determination of failure criteria. The simulation information can also be used in a dynamic approach to CET modelling, thus linking up with level 2 PSA. The result of the CET is then a probability distribution for the source term (consequence) of the CET.

If the objective of the analysis is to obtain a reliability estimate for PCCS, the reliability assessment would probably be more profitable with some other, more specific simulation tool, able to deal more effectively with some factors affecting PCCS function, like, for instance parameters such as pipe inclination and surface oxidation.

Obviously also the deterministic nature of the code poses some challenges, for probabilistic purposes. Maybe the most practical approach to simulation based reliability analysis would be to give system parameters some distributions from which to sample the values for them. The failure criteria would have been determined beforehand. This kind of Monte Carlo method would require quite many simulation runs and the system reliability estimate would be determined according to the fraction of runs which do not exceed the chosen failure criteria. This is quite prohibitive to be performed with MELCOR, but as pointed out above, the MELCOR simulations can be advantageous for example in determination of reasonable failure criteria and in dynamic approach to CET modelling.

The depicted approach is consistent with the RMPS method described in more details in ref. 9.

6. Reliability analysis discussion

Natural circulation failure evaluation is included in the present study proposed for PCCS reliability assessment: this is attained through an approach aimed at the thermal-hydraulic performance assessment on the probability standpoint.

According to the previous section, some relevant methodologies for passive system reliability assessment prompted so far suggest propagation of important system uncertainties through Monte-Carlo simulation of a detailed best estimate mechanistic system, like MELCOR.

The main drawback consists in the requirement of a large number of system analyses using best estimate system code. Typically mechanistic thermal hydraulic codes of complex nuclear safety systems are computationally expensive and MCS using such models require considerable and often prohibitive computational effort to achieve acceptable accuracy.

One common and straightforward approach to reduce computational effort is adopting approximate solutions which require less computational effort for evaluating system response.

In the present treatment the approach based on failure modes of passive systems hardware components (ref.3) is applied.

Thus, the issues pertaining to the natural circulation unavailability estimation suggest to approach the problem at the component level overriding the failure modes related to the onset of thermal-hydraulic phenomena that would impair the passive function of the system. Nevertheless, failures that defeat or degrade the natural mechanism upon which the passive system relies are treated as an unavailability of components, which challenges the boundary conditions or mechanisms needed for assuring the passive operation.

In the natural convection analysis, this concept is this concept is applied, for instance, to the presence of the noncondensables failure mode.

If, on the one hand, this approach may in theoryrepresent a viable way to address the matter, on the other hand,some critical issues arise with respect to the effectiveness and completeness of the performance assessment over the entire range ofpossible failure modes that the system may potentially undergo andtheir association to corresponding hardware failures. In this simplified methodology, degradation of the natural circulation processis always related to failures of active and passive components, notacknowledging, for instance, any possibility of failure just becauseof unfavorable initial or boundary conditions.

Consequently, as previously underlined, the outcome of the analysis represents a preliminary appraisal as regards the unavailability of a thermal-hydraulic passive system relying on free convection and under the assumptions taken in the study, the results show the important weight of the natural circulation failure probability on the overall unavailability of the system.

At last we can state that the probability of a loss of containment heat removal is significantly reduced because the PCCS is highly reliable, due to the redundant heat exchangers and passive component design.

The importance of research on passive systems seems to be increasing and future reactor types will most likely exploit more of these safety features. Thus the need for proper methods for reliability analysis of such systems becomes more urgent. The passive containment cooling system represents passive systems with thermal-hydraulic properties and therefore poses further challenges. This study on PCCS reliability should be regarded as a preliminary analysis, laying the basis for further analysis to conduct the assessment with suitable tools, by leveraging the thermal-hydraulic simulations, and to yield any possible reliability estimate.

The features of suitable codes have to be examined in order to evaluate their appropriateness to effectively handle the issue for the given purpose.

7. In-vessel corium retention strategy

The severe accident in a LWR assumes the occurrence of a meltdown of the core, whose corium can breach the safety barriers (in a short time if exothermic reactions occur) in absence of coolability, with consequently release of radioactivity to the environment. Therefore, the prevention of the failure of the bottom head of the vessel, first, and of the containment and/or the basemat melt-through, later, is felt of meaningful importance for the severe accident management (SAM).

It is commonly recognized that the IVR strategy achieved by external reactor vessel cooling and/or in-vessel flooding is one of the most effective measures to prevent the progression of severe accidents at water-cooled reactors. Several operating nuclear power reactors (e.g. VVER-440, VVER-1000, Indian PHWR), or new ones (e.g. AP1000, APR1400, CAP1400, KERENA, HPR1000, ACP 100) use IVR strategy (Figure 3) and have dedicated systems.

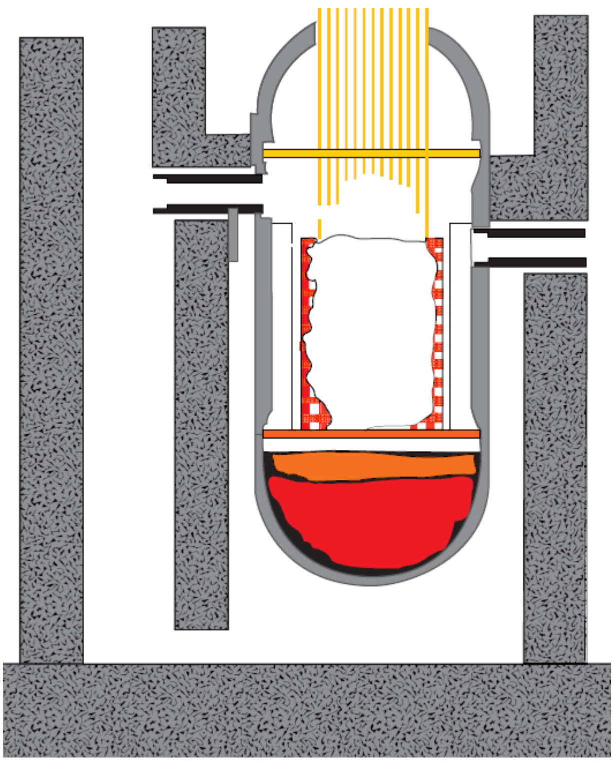

Figure 3: IVCR - Configuration of the vessel with a melt pool in the bottom head and the cooling of the outer wall of the vessel with two-phase flow along the vessel [10].

A lot of R&D has been done and are still on-going to develop IVR strategy and technologies, based also on new findings oriented at maximizing the reactor safety.

Indeed, Fukushima Daiichi accident pointed out the need of a substantially reduction of the heat flux from the metal layer on the vessel in order to preserve a margin to the CHF (additional time grace to manage plant emergency).

Water injection into the RPV appears the most obvious way to mitigate a severe accident. Nevertheless, core damage occur because water injection was not available for a prolonged period of time and the prolonged core heating phase may result in the oxidation of the fuel cladding (plant damage state: oxidized), and subsequently melting and relocation to the lower plenum. Ultimately it may result in a melt-through of the RPV. The heat to be removed is stored decay heat, and heat generated by the metal-water reaction.

The In-Vessel Corium Retantion (IVCR) seems, therefore, very attractive because it intervenes by flooding the reactor cavity before relocation of the corium into the pressure vessel lower head occurs. Figure 4 provides an overview of melt-pool composition, and of the heat transfer processes during the melt progression; in addition it is shown how the melt relocates within the bottom head [11-12].

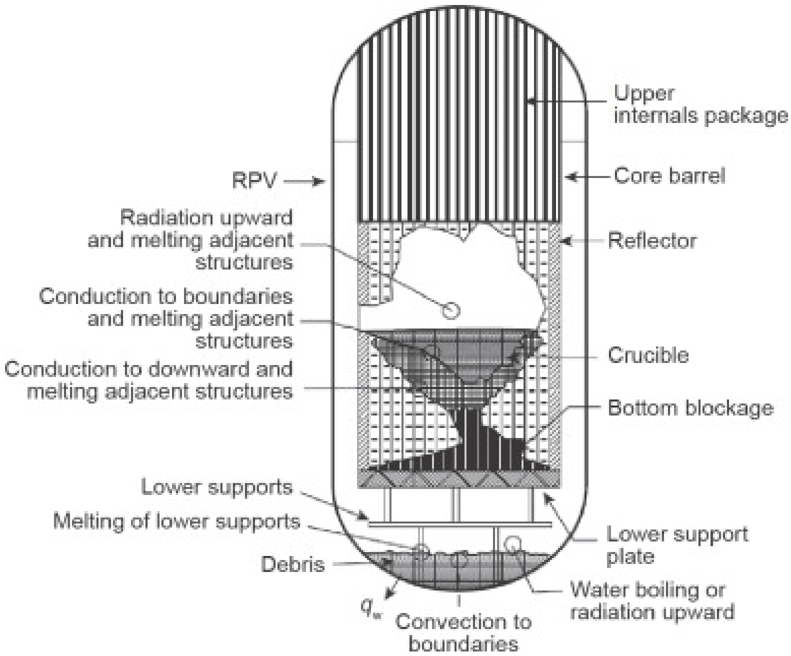

Figure 4: Core meltdown and melt-relocation (to the bottom head) scenarios as during TMI-2 accident: material configuration and heat transfer processes at an intermediate state of melt progression, after initial major relocation and before final steady state [13].

7.1 Requirements on the IVMR

It was pointed out that the behaviour of molten pools is important to determine local heat flux values which may result in larger threat to the integrity of the RPV. In fact a postulated severe accident in a LWR with the meltdown of the core, breaching the first barrier of the clad proceeds with the molten corium moves to the bottom head of the vessel. The bottom head will fail if the corium melt remains uncooled,as well as the containment in a short time if some energetic reactions, for example, hydrogen burn (explosion), steam explosion, or direct containment heating occurs.

In this study we will propose a new IVCR approach, based on an internal core catcher made of alumina (Figure 5) and on the ‘thermal criterion’. This latter is one of the two success criteria of the IVMR strategy to make sure the heat flux from in-vessel molten pool less than the CHF at the outer surface of the RPV lower head that is determined by external cooling conditions with water flooded in the reactor cavity. The main factors affecting the thermal criterion are:

1) stability of the natural circulation;
2) geometry of the flow path;
3) surface conditions of RV;
4) water subcooling at the inlet of the flow path.

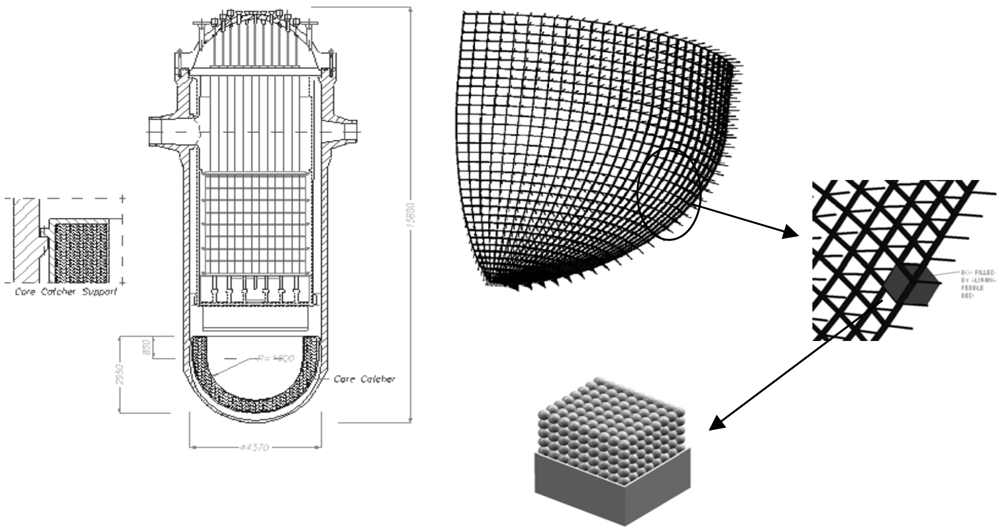

Figure 5: Overview of the proposed internal core catcher (left figure) and of the frame structure made of SiC-SiC ceramic matrix composite with box filled by alumina pebble bed (right figure) for a RPV of 1000 MWe PWR [11].

As for the new alumina made internal core-catcher concept, developed at the DICI-University of Pisa, it consists of the alumina pebble-bed, that are positioned within alumina boxes, covering the bottom head of the reactor vessel, (Figure 5).
The selection of the alumina is because of its high refractoriness and capability to accommodate thermal expansion without high thermal stresses. The boxes are assembled like bricks of masonry, supported by means of a CMC (SiC-SiC) framework. The internal and external liners of core catcher are made of high alloy steel.

8. Molten Pool Behaviours and Structural Integrity of RV

The proposed internal core catcher approach is based on the successful coolability of the core melt (temperature higher than 2000°C) at the bottom head, allowing the stabilization of the severe accident. The instauration of heat transfer processes will ensure that at least the bottom layers of the RPV wall thickness are not undergoing “heat flux focusing effects” [14-15] and keep unaltered structural properties.
The internal core catcher (ICC) has to be properly designed in order to account for the thermal effects (and material degradation occurring during the core melting and relocation).
It is understood from the past research that molten pools separate into 2 layers (i.e. lower oxidic layer and upper liquid metal layer) or 3 layers (heavy metal, oxidic and light metal layers), and there forms an oxidic crust between the oxidic and light metal pools.
ICC issues are due to the heat-up of vessel thickness from the inside because of the metal melt layer, riding on the top of the crusted upper boundary of the oxidic melt pool, and to the heat removal capacity.
In consideration of that, we proposed an ICC concept characterised by the inner surface of the RPV bottom head coated with an internal layer made of alumina pebbles bed. In this way, we will increase the overall thickness of bottom head facing heat flux and delay the heating-up of RPV wall profiting of the lower thermal conductivity of alumina [16]. The main goal to achieve is so to avoid the “boiling crisis” of the bottom head.

8.1 Preliminary analytical analysis

The adoption of this ceramic layer, resting on the inner bottom head, will increase the area of the metal layer in touch with the vessel so to keep the imposed heat flux below ~1.5MW/m^2

(corresponding to 1.3 MW/m^2 maximum thermal load for a 1000 MWe PWR). This value was measured in ULPU-2000 experimental facility without any special shaping of the flow.

The proper thickness of pebble bed (acting like "insulator") is evaluated by taking into account some basic design assumptions:

- Primary system depressurized.
- Lower head fully submerged in cavity water before that core debris reaches the lower head.
- Water level in the cavity would be maintained indefinitely.

In the setting up of the methodology [17] to determine the suitable alumina thickness, we first start analysing a simple static model, solving the heat transfer equation, in spherical coordinates, by assuming a constant thermal conductivity (k) and q'''= 0.

No chemical reactions were included in that assessment.

In Figure 6 is represented the RPV bottom head axial section along which Fourier problem is solved: moving outwards through the thickness we encounter the internal core catcher (layer A), the steel (layer B), the outer insulator (layer C) and the water (layer D).

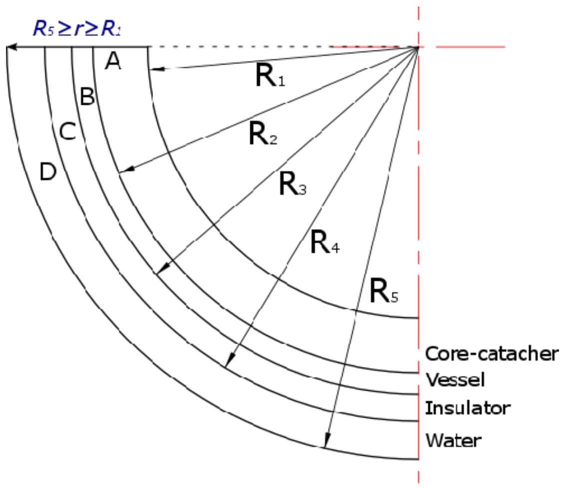

Figure 6: Analytical model with thermal insulator (instead of adiabatic boundary condition) [17]. The boundary conditions are $T(R_1)=T_{corium}$ and $T(R_5)=T_{water}$.

This steady state problem in one dimensional system, exhibiting azimuthal and poloidal symmetry, is only dependent on the radius, and is solved by direct integration of the Laplace equation:

$$T(r) = C_1 + \frac{C_2}{r} \tag{1}$$

For the water (because of LB LOCA), layer D in the scheme of Figure 6, the thermodynamics conditions are: atmospheric pressure and an unknown temperature distribution that tends to a cavity-wall fixed temperature. The thermal insulation (layer C) has the same low thermal conductivity (0.15 W/m/K) and thickness (0.25 m), of the real one in nuclear reactor. The boundary conditions are respectively the melt temperature and the water temperature (40 °C). Geometrical and material properties of the simplified model are given in the below Table 5.

Table 5: Geometrical and material properties of the simplified model

Region	Radius r [m]	Thickness t [m]	Thermal conductivity k [W/m/K]
Core-catcher	2.50÷3.00	0.5	0.50
Vessel	3.00÷3.20	0.2	18.00
Insulator	3.20÷3.45/3.50	0.25/0.30	0.15
Water	3.45/3.50÷3.80	0.30/0.35	0.60

9. RPV pool behaviour analysis discussion

By varying the Tcorium from 1000 °C to 2000 °C and assuming the worst heat transfer condition between the thermal insulator and the water (no convection), we determined the steady state loss of heat through the section [17]. Figure 7 represents the heat removal by only conduction along the vessel wall thickness: from these results, it was observed that the water temperature in contact with thermal insulator is higher than 100 °C.

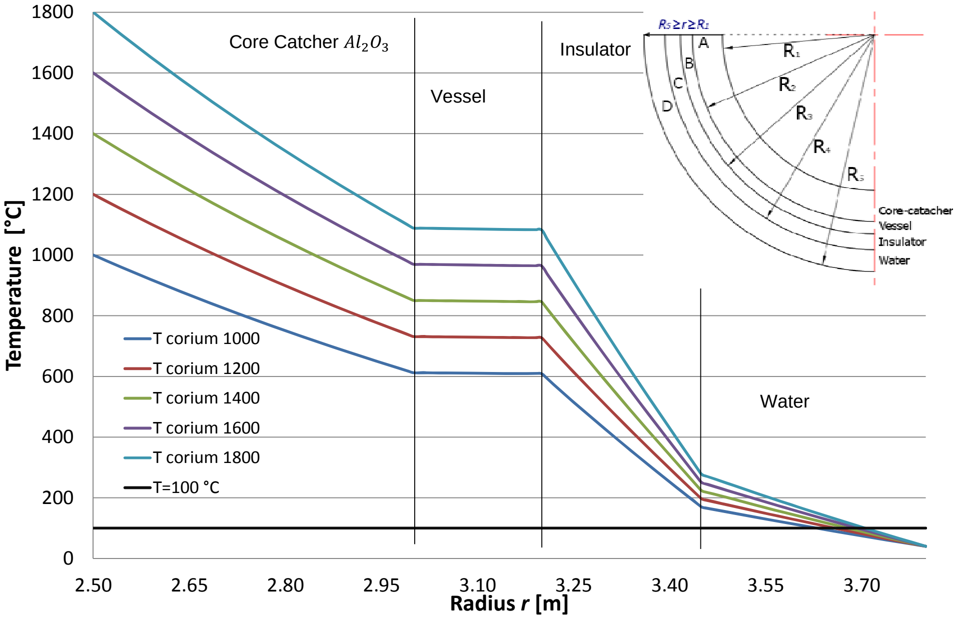

Figure 7: Radial temperature trend in the hemispherical bottom head.

Furthermore a parametric study was carried out in order to identify the correct value of R5 (below 0.15 m) corresponding to which the temperature in the layer D (R_4) is less than 100 °C (at t∞). After that a second parametric analysis was performed varying the in core-catcher thickness and keeping R_5=3.55 m. Fig. 5 shows the reduction of the average vessel temperature (diamond shape) and the insulator-water interface temperature (squared shape) with variation (reduction) of the alumina thickness from 60 cm down to 10 cm.

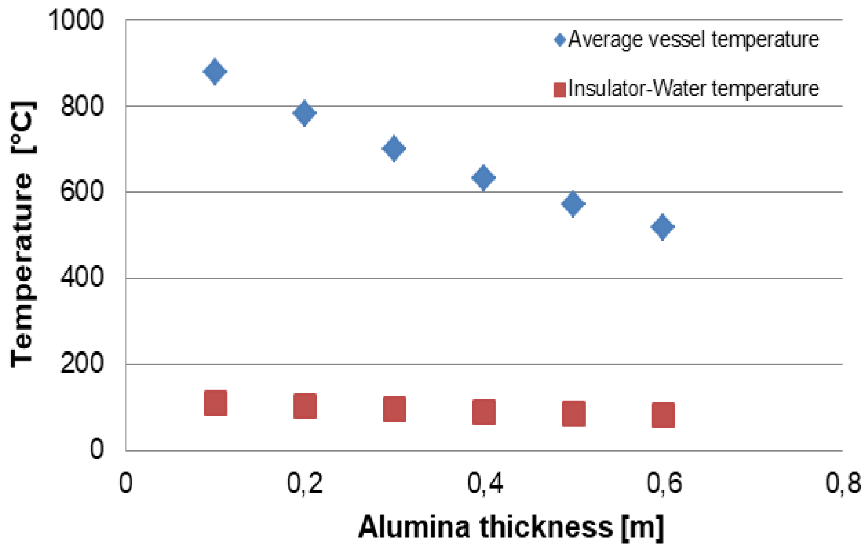

Figure 8: Temperature trend for different core-catcher thickness

Finally, we repeat the previous calculation preserving the RPV geometrical domain (except for the water having T(R_4) = 100 °C) in order to determine loss of heat along the wall thickness.

Figure 9 provide overview of the heat transfer across the overall thickness: it is possible to observe that it is mainly due only to the conduction mode.

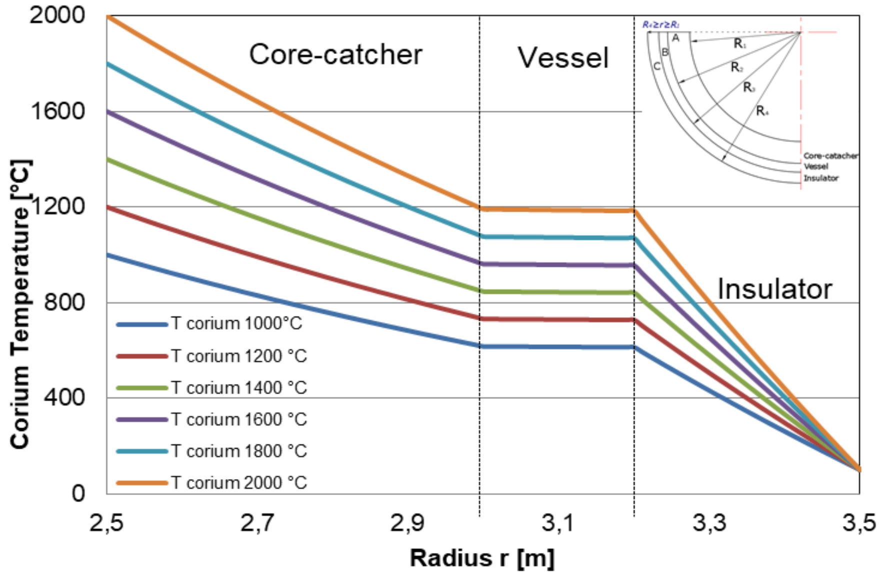

Figure 9: Radial temperature variation in the hemispherical bottom head for T(R_4) = 100°C.

10. Final summary

Passive nuclear safety, concentrating on the reliability aspect has been examined in the first part of the study. An overview on the methodology utilized in reliability assessment of such entities was given, and also uncertainties related to passive systems were handled. Special emphasis was given on the passive containment cooling system, a thermal-hydraulic passive safety system, and the function of the system was introduced quite thoroughly. Tools such as failure modes and effects analysis (FMEA) and fault tree analysis were used to conduct reliability assessment for PCCS. Component reliability data was taken from literature. Estimate for failure probability was obtained, but it must be regarded only as a suggestive approximation, as the main point was to show how to use classical methods in reliability assessment of passive systems.

The study shows the reduction in risk of containment heat removal failure, because the PCCS is highly reliable, due to the redundant heat exchangers and passive component design.

Additional effort is required aimed both at a deeper investigation on the thermal-hydraulic performance of the system and at the assessment of the relative uncertainties, with help of a suitable tool. This is in order to add credibility and validate the passive system reliability analysis.

In this report a new IVCR strategy to face the issues due to the flooding of the reactor cavity before the relocation of the corium in the pressure vessel lower head is presented. This new technological system is based on an original design of a core catcher made of batch of ceramic multi-layered pebble (by profiting of the low thermal conductivity).

To evaluate the feasibility of this system and the positive effects in terms of safety management of severe accident scenario, thermal criterion' is taken into account. Therefore potential ranges of loads, appropriate failure modes and the correspondent material characteristics are considered to represent the cooling of the corium, performed by means natural circulation of the water, in inside the reactor cavity in regimen of nucleate boiling.

The minimum thickness which assures the structural integrity of the RPV pool is analytically calculated according to the assumption of maximum heat flux.

Results show that the core catcher, under the most conservative scenario, could resist indefinitely if coolability from the upper surface is guaranteed. Additional effort is required for a deeper investigation of the thermo-mechanical performance of the IVCR system, with help of a suitable numerical modelling tool capable to investigate the core catcher at the final state of corium relocation (including characterization of material properties at severe accident conditions). This is in order to add credibility and validate the passive system reliability analysis.

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Dr Rosa Lo Frano (F) has focused her research studies on the safety behaviour of NPP plants structure and components subjected to external/internal accident event. She is Researcher at the DICI University of Pisa and won over the past 7 years several research award and appreciations, like the 2009 ASME PVP mention award. Since 2007 she is teaching assistant at the course of 1) "Design Techniques of Mechanical, Chemical and Nuclear Constructions" for the Nuclear and Mechanical Engineering Degree; 2) "Complex Plant design" for the Nuclear Engineering Master Degree at Pisa University; 3) Nuclear power plants, and 4) Structural Mechanics for the Nuclear Engineering Master Degree at Pisa University.

Her research activity, documented by more than 50 publications on Intl. Journals and Proceedings of Intl. Conference-Congress and participation in several relevant European and Italian projects (link http://arp.unipi.it/listedoc.php?lista=ALL&ide=11443&ord=C), refers mainly to the safety of nuclear plants and components design, the structural integrity of NPP, safety of RWs transport, safety analysis of structures in normal and accident conditions.

As part of her research, R. Lo Frano was and still is one of the scientific referees of projects and contracts at national and international levels, such as the AdP MSE-ENEA, FP6 ELSY, FP7 LEADER, FP7 ESNII+, PETRUS III, etc.

She was/is track-co-chair at ICONE 24 and ICONE25, etc, member of ICONE and NENE conference organizing commeette and of thecnical body like ASME and UNI.

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Daniele Del Serra got a PHD in Nuclear and Industrial Safety Engineering at University of Pisa, in July 6, 2015, discussing a thesis, developed in collaboration with EDF (France), on noble gas adsorption properties of activated charcoals, used in nuclear power plant off-gas systems. During the PHD course, Dr. Del Serra designed and built a computer-controlled laboratory scale testing facility for the charcoal qualification by means of the determination of the adsorption coefficient in a wide range of pressure, temperature and relative humidity conditions. During the PhD, Dr. Del Serra attended the following courses Analogue and digital Electronics, Mathematics for engineering, Thermodynamics for chemical engineering, Characteristics and applications of lasers.

Since 2015 Dr.Del Serra is a fellow researcher at University of Pisa, performing research activity in the nuclear fusion field, in the framework of collaboration with ITER Organization. The main subject is the investigation of the steam condensation phenomena in water tank at sub-atmospheric pressure, with the purpose to study the accidental scenarios in case of overpressure of the ITER machine vacuum vessel. Dr. Del Serra was charged of the design, supervision of construction and commissioning of the experimental apparatus. He also performed the tests and collaborated to the elaboration of experimental data.

During these years he acquired skills in the fields of nuclear thermohydraulics, radioprotection and nuclear measurements, as well as in mechanics and data acquisition systems.

He is competent with most Microsoft Office programmes, 2D and 3D CAD software, MCNP Montecarlo code for neutron and particle transport, Matlab and ANSYS.

CV A. Facchini

Alberto Facchini got the MSc degree on Nuclear and Industrial Safety Engineering Engineering the 22nd of February 2016, at University of Pisa. He is currntly PhD student at KAEST (KR). The work experience are:

• European Commission, Joint Research Centre - Institute for Energy and Transport (JRC - IET), Petten (Netherlands); I worked with the MCNP code on the model of the European Sodium Fast Reactor looking for the most stressed fuel pin at end of cycle during an accidental scenario (Unprotected transient of overpower UTOP) [01/09/2015 to 31/01/2016].

• University of Pisa, Research Fellow; in this period I worked principally as support to the research group of Prof. Aquaro that is now involved in verification for ITER components. In parallel I am working also in a project for in-vessel retention; I am studying a possible internal core-catcher for PWR designed in alumina pebble-bed [01/04/2016 to 28/02/2017].

Current occupation: PhD student at Seoul National University, Seoul, Republic of Korea [01/03/2017 to present].

During these years he acquired skills in the fields of nuclear physics, fusion/fission safety design as well as in mechanics. He is competent with most Microsoft Office programmes, 2D and 3D CAD software, MCNP Montecarlo code for neutron and particle transport, Matlab etc.