SAPIENZA
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# DEVELOPMENT OF INNOVATIVE SOLUTIONS SUITABLE FOR GENERATION III AND IV NUCLEAR REACTORS


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*Dottorato di Ricerca*

*in*

*ENERGETICA - XXV CICLO*

*To my father*

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# 1 INTRODUCTION

The present research project, carried out during a period of three years, has been especially focused on R&D issues related to innovative components and/or systems aimed at enhancing the safety features of nuclear reactor designs. Several reactor years' experience has been gained in operating hundreds of nuclear reactors since the 50s, but in the latest designs very important improvements have been introduced. Among these designs, some are especially focused on the intensive use of redundancy and physical separation of the safety related systems, others characterized by an intensive use of inherently safe or passive safety systems, prevent by design accidental conditions occurrence. Since the former possibility does not lead to the introduction of innovative systems or components, but only in a very limited level, the study and the development of safety systems, to be applied to nuclear reactors, have been mainly focused on systems characterized by an intensive use of passive devices that could foresee the use of innovative solutions.

The activation and operation of passive safety systems are generally characterized by limited driving forces (e.g. temperature difference, density gradient, pressure drop, etc.). As a consequence, the design of passive systems is more sensitive to the system parameters. An active system requires a simpler design since the presence of any unlikely force, opposed to the driving force, can be easily overcome by the higher intensity of the driving force itself. Concerning the passive systems, the design is much more complicated since the driving force, which allows the system to be activated and/or to operate, has a limited intensity, comparable with other natural forces present into the system itself (e.g. the pressure drop against the circulation pump head). The design of a passive system requires an optimization of the geometry and an accurate evaluation of the timing to be sure that the system complies with the needed safety requirements.

An active system can be used after having tested all the constituting components (e.g. pumps, valves, etc.) while a passive system has to be tested in a full-scale facility to be sure that the system performances comply with the requirements. Likewise, a valve shutter, the position of which is controlled by an electric actuated device (to open/close), can be considered almost unaffectable by fouling issues due to the high intensity force applicable to move the shutter. A passive valve has to overcome this problem by design since the applicable forces are of smaller intensity.

The main effort, that has been done in the present research work, has been mainly focused on the identification of design solutions able to improve the plant safety features. Studies have been mainly focused on passive safety systems characterized by special features that allow it to limit the possible initiating failure events that would make the system unavailable when requested. Better solutions are usually characterized by very simplified systems (number of components as low as possible) coupled with very robust designs in which the main purpose is the minimization of the forces that could counterbalance the main driving force.

The systems developed during these three years have safety features comparable with those of latest plant designs; in some cases, the safety features of the developed systems results to be even better. In particular, the present research work has been finalized to the development of special solutions optimized to guarantee the safety performance removing the decay heat from the primary coolant of nuclear reactors. As clearly visible below, the developed systems are especially optimized to be used on small sized nuclear reactors, where the required heat transfer surfaces are smaller, but an application on larger sized reactors could be also possible slightly reviewing the system design.

The research work has been initially focused on the analysis of existing nuclear reactors designs[1] in order to identify the most promising solutions and the possible areas of action to propose innovative solutions to apply. In this field, the most important issue, which has been identified to propose new solutions, is related to the heat removal function that must guarantee the decay heat removal, after the shutdown, also in emergency condition. Nuclear cores, after the shutdown, continues to produce thermal power (in different measure depending on the type) that in normal operating condition is removed by an

[1] The first step of the research work has been mainly focused on PWRs since they are the most widespread technology used to produce energy and hence characterized by the higher reactor experience.

active[2] system. In the unlikely condition in which electric power source of the active heat removal system is not available, backup systems have to be activated to guarantee this same function[3]. A very innovative solution, especially suitable for SMRs, has been developed and studied for a never ending heat sink.

Moreover, the growing interest in nuclear reactors during the last few years (nevertheless, affected by the Fukushima accident), led to study possible solutions to apply on fast nuclear reactors, in order to increase their safety features. The reactors here considered belong to the IV Generation of nuclear reactors, characterized by very strong safety requirements. In the present research work, an innovative system, originally studied for SFRs, allows to remove the decay heat from the primary coolant minimizing, thanks to a double wall barrier, the possibility of primary and secondary coolants interaction (especially concerning sodium-water reaction issues). The special features characterizing the system made it also suitable for application on LFRs. The different features of sodium and lead as coolants have been deeply analyzed to carry out a performance assessment of the DHX that is the key component of the innovative decay heat removal system studied.

The present study shows how, the proposed approach concerning the safety issues of nuclear reactors, thanks to the accidental condition prevention considered from the design stage, would make it possible to avoid or to minimize the occurrence of accidental conditions. The philosophy that has been followed in the present work is that of the "safety by design"; criterion according to which the extreme simplification of the systems together with the use of natural laws to guarantee the safety functions make the whole system cheaper and safer. Obviously it is true only if the design has been optimized to prevent any unlikely forces combination that would make the system unable to comply with the safety requirements.

Here below, a resume of some activities carried out during this research period are reported.

[2] With the term “active” is intended a system that requires a source of power to start and to operate.

[3] The backup systems can be both active and passive, depending on the whole plant design. In any case, the redundancy and physical separation criteria are applied.

# 2 NUCLEAR REACTORS SAFETY ASPECTS

Nuclear power plants have been always characterized by the highest industrial levels of safety, becoming, with time, more and more stringent. The different targets imposed to nuclear reactor plant designs to be licensed make it possible to classify nuclear reactors under different generations, starting from the first generation of prototypes of the 50s and 60s up to the Generation IV nuclear reactors that will be built in the coming decades.

## 2.1 Nuclear reactors' generations

Nuclear power plants have evolved over time exploiting the experience gained on the operation of existing plants. Unlike other technologies, nuclear reactors designs have never been based on the "trial and error" process but on many dedicated studies and experimental campaigns. In any case, nuclear reactor plants have evolved as distinct design generations, taking into account all the problems found in the operation of previous reactors. The different generations are the following:

- First generation: prototypes, and first realizations (~1950-1970)
- Second generation: current operating plants (~1970-2030)
- Third generation: deployable improvements to current reactors (~2000 and on).
- Fourth generation: advanced and new reactor systems ( 2030 and beyond)

Most of the operating nuclear power plants were built between the 70s and the 90s and belong to the second generation of nuclear reactor designs.

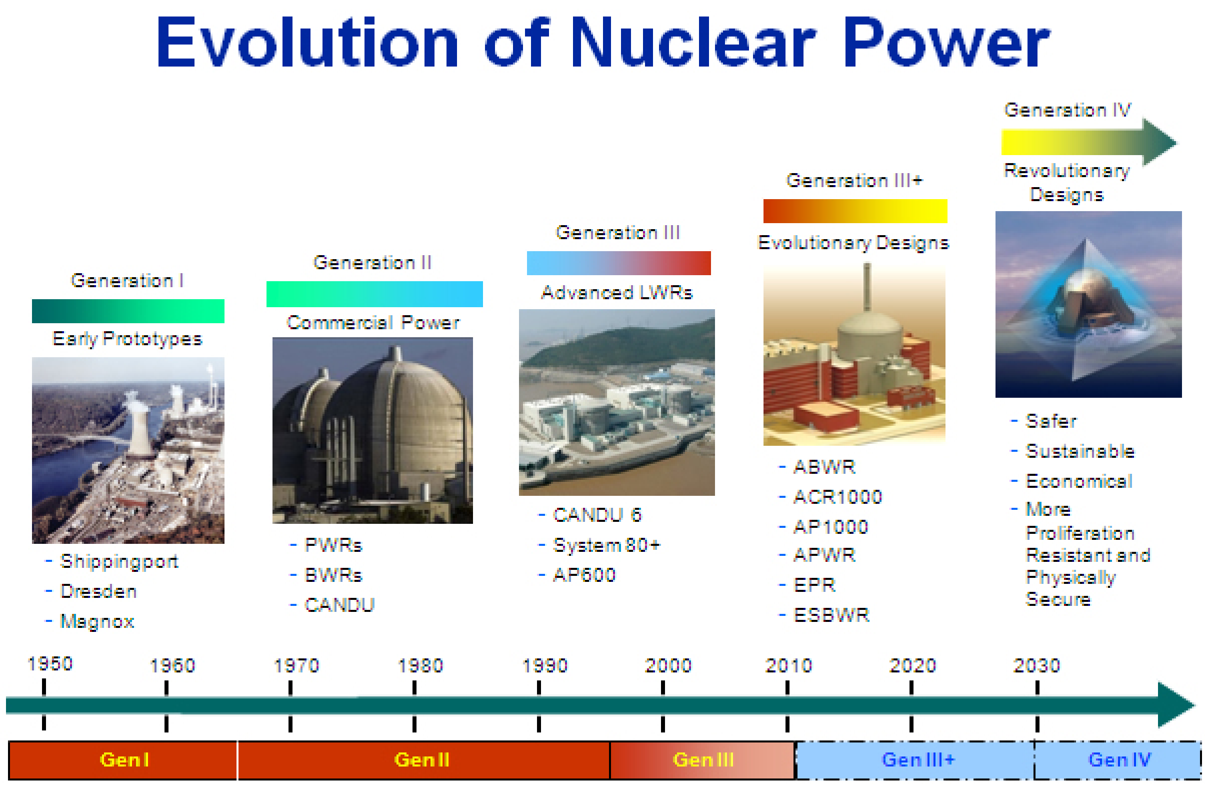


**Figure 1. Chronological development of nuclear reactors' generations**

To reach a widespread diffusion of nuclear reactors, future plants have to be designed with very enhanced safety and economic features. In particular, the Generation IV International Forum (GIF) defined some targets that Generation IV nuclear reactors have to comply with [1]:

**Sustainability-1:** Generation IV nuclear energy systems will provide sustainable energy generation that meets clean air objectives and provides long-term availability of systems and effective fuel utilization for worldwide energy production.

**Sustainability-2:** Generation IV nuclear energy systems will minimize and manage their nuclear waste and notably reduce the long-term stewardship burden, thereby improving protection for the public health and the environment.

**Economics-1:** Generation IV nuclear energy systems will have a clear life-cycle cost advantage over other energy sources.

**Economics-2:** Generation IV nuclear energy systems will have a level of financial risk comparable to other energy projects.

**Safety and Reliability-1:** Generation IV nuclear energy systems operations will excel in safety and reliability.

**Safety and Reliability-2:** Generation IV nuclear systems will have a very low likelihood and degree of reactor core damage.

**Safety and Reliability-3:** Generation IV nuclear energy systems will eliminate the need for offsite emergency response.

**Proliferation resistance and Physical Protection:** Generation IV nuclear energy systems will increase the assurance that they are very unattractive and the least desirable route for diversion or theft of weapons-usable materials, and provide increased physical protection against acts of terrorism.

## 2.2 Nuclear reactors' safety requirements

The abovementioned targets are very stringent and they are required for the GEN IV nuclear reactors. Even if the previous generations of nuclear reactors have not these requirements, three main safety function requirements have been always present. These safety functions are:

- The Chain Reaction stop (Core Shutdown);
- The Decay Heat Removal;
- The Containment Integrity.

### 2.2.1 The Core Shutdown

The core shutdown function must be available in every reactor conditions since it guarantees the stop of the chain reaction. This requirement can be met by means of different systems but it implies specific requirements on the reactor design. The most commonly used method to obtain the reactor shutdown is the control rods' insertion, among or within the fuel assemblies. To guarantee, in every operative and accidental condition, the control rods insertion, the fuel assemblies must preserve their geometry; on the contrary, the control rod insertion would not be possible and thus the stop of the chain reaction. For this reason, in addition to the requirement of a robust core mechanical design, some backup solutions and criteria are foreseen to stop the reaction. First, the reactor shutdown must be guaranteed complying with the "single failure" criterion that means that the reactor shutdown is guaranteed even if a control element would remain outside the core. As backup solutions, referring to LWRs, additional water inventories with high concentration of soluble boron are foreseen; they are directly connected to the reactor vessel allowing to stop the chain reaction in case of failure of the control rods system.

### 2.2.2 The Decay Heat Removal

The Decay Heat Removal function has to be guaranteed since after the reactor shutdown, due to decay reactions of radioactive isotopes, the reactor core continues to produce thermal heat. At the time of the shutdown, the heat produced in the core is about 7% of the power before shutdown and then it decrease following an exponential decreasing trend.
In the LWR cores, in order to evaluate the decay heat produced after a $t_0$ period from the shutdown, the following correlations[4] are usually adopted:

$$\frac{P(t)}{P_0} = ANS(t) - ANS(t+t_0) \tag{2.1}$$

where:

- *P(t)* is the decay heat power [ W ];
- $P_0$ is the design thermal power of the reactor [ W ];
- *t* is the elapsed time from the reactor shutdown [ s ];
- $t_0$ is the equivalent full power time of the reactor core [ s ];

and the ANS dimensionless parameter has to be evaluated by the following set of empirical correlations:

$$ANS = \frac{0.0603}{t^{0.0639}} \qquad for\, 5\,s < t \le 10\,s \tag{2.2}$$

$$ANS = \frac{0.0766}{t^{0.181}} \qquad for\, 10\,s < t \le 150\,s \tag{2.3}$$

$$ANS = \frac{0.130}{t^{0.283}} \qquad for\, 150\,s < t \le 4\cdot 10^6\,s \tag{2.4}$$

$$ANS = \frac{0.266}{t^{0.335}} \qquad for\, t > 4\cdot 10^6\,s \tag{2.5}$$

[4] This set of equation has been defined by the ANS, neglecting the contribution of $^{239}U$ and $^{239}Np$ to the total produced power.

Usually, for the firsts five seconds, the decay heat is assumed to be equal to the core power before shutdown (in safety analyses, reference is made generally to the core rated power). More recent studies focused on FRs have defined the core decay heat power by means of the following correlations [2]:

$$\frac{P(t)}{P_0} = 0.2146 \cdot t^{-0.3239} \quad for\, t \leq 300\, s \tag{2.6}$$

$$\frac{P(t)}{P_0} = 0.1439 \cdot t^{-0.256} \quad for\, t > 300\, s \tag{2.7}$$

where:

*P(t)* is the decay heat power [ W ];

$P_0$ is the design thermal power of the reactor [ W ];

*t* is the elapsed time from the reactor shutdown [ s ].

The decay heat removal is one of the most important functions to absolve since the heat transfer mechanism takes place under the condition of imposed power. Under this condition, if the normal cooling system is not able to remove all the produced heat, the fuel temperature, as well as the cladding temperature, would rise up: the heat transfer capability of the safety decay heat removal system shall allow the guarantee heat removal at fuel and cladding temperatures compatible with the integrity of the core, with coolant medium in a thermodynamic state compatible with a reliable heat removal function.

$$P = U \cdot S \cdot \left(T_{core} - T_{coolant}\right) \tag{2.8}$$

where:

*P* is the decay heat power to be removed[5] [ W ];

*U* is the local heat transfer coefficient between the fuel and the coolant [ W $m^{-2}$ $K^{-1}$ ];

*S* is the reference heat transfer surface [ $m^2$ ];

[5] It is defined by the elapsed time from the shutdown and by the time that the fuel elements had spent within the core.

$T_{core}$ is the local fuel temperature [ K ];

$T_{coolant}$ is the local reference temperature of the coolant [ K ];

If the fuel temperature ($T_{core}$) locally becomes higher than the fuel melting point over the full diameter of the fuel pellet, a severe accident with core melt would occur. To prevent any unlikely accidental condition that can lead the core to greatly increase its temperature, the availability and the reliability of the decay heat removal system are safety issues of primary prominence.

When a reactor shutdown occurs, in normal plant condition the decay heat is removed by the secondary coolant, of by the normal Residual Heat Removal System, an active system that requires a powered component to operate (e.g. circulating pumps are needed). Since the decay heat removal function is one of the most important safety related issues, it is necessary to foresee backup systems (i.e. the DHRSs), able to operate in case of the first system failure.

The most recent reactor designs are characterized by an extensive use of passive systems [3], especially related to safety functions, as the DHRS. Passive systems allow reducing the number of components and systems required and increasing hence the system availability. Moreover, systems that do not require any external intervention to operate are especially suitable to comply with the safety requirements.

A large part of the present work has been focused on the research and development of safety heat removal systems and components, especially passive, able to remove, for an unlimited period of time, the thermal heat produced by the core in any accidental condition.

### 2.2.3 The containment integrity

The last safety function that has always to be guaranteed is the containment integrity. In the unlikely event of any accidental release of radioactivity from the reactor vessel, the containment has to assure that the external environment will not be affected by the accident with radioactive releases overcoming specified allowable limits.

If one of the two previous functions fails (or also in accidents releasing the primary coolant, always radioactive, into the containment, even without a core melt), the containment integrity would protect the surrounding environment from excessive radioactive release. The NPP protection systems act from the core (shutdown) to the environment (containment) passing through the heat removal that brings the system to a controlled condition.

# 3 NEED FOR PASSIVE SAFETY SYSTEMS

## 3.1 Passive system classification

According to the IAEA definition [3], the passive systems are defined as follows:

For components and systems (but not structures) having safety functions, there must be at least two states corresponding to the normal function and to the safety function. Then, to change from the normal to the safety state:

- there must be "intelligence" such as a signal or parametric change to initiate action;
- there must be power and potential difference or motive force to change states; and
- there must be the means to continue to operate in the second state.

A component or system can be called passive when all three of these considerations are satisfied in a self-contained manner. Conversely, it is considered active if external inputs are needed.

Since the passive systems can have different characteristics, the IAEA TECDOC-626 foresees a subdivision in four different classes described here below (excerpt from literature [3]):

<u>Category A</u>

This category is characterized by:

- no signal inputs of "intelligence", no external power sources or forces,
- no moving mechanical parts,
- no moving working fluid.

(The no-motion requirement does not extend to unavoidable changes in geometry such as thermal expansion.)

Examples of safety features included in this category are:
- physical barriers against the release of fission products, such as nuclear fuel cladding and pressure boundary systems;
- hardened building structures for the protection of a plant against seismic and or other external events;
- core cooling systems relying only on heat radiation and/or conduction from nuclear fuel to outer structural parts, with the reactor in hot shutdown; and
- static components of safety related passive systems (e.g., tubes, pressurizers, accumulators, surge tanks), as well as structural parts (e.g., supports, shields).

<u>Category B</u>

This category is characterized by:

- no signal inputs of "intelligence", no external power sources or forces,
- no moving mechanical parts, but
- moving working fluids.

The fluid movement is only due to thermal-hydraulic conditions occurring when the safety function is activated. No distinction is made among fluids of different nature (e.g., borated water and air) although the nature of the moving fluid may be significant for the availability of the function performed within this category.

Examples of safety features included in this category are:
- reactor shutdown/emergency cooling systems based on injection of borated water produced by the disturbance of a hydrostatic equilibrium between the pressure boundary and an external water pool;
- reactor emergency cooling systems based on air or water natural circulation in heat exchangers immersed in water pools (inside containment) to which the decay heat is directly transferred;
- containment cooling systems based on natural circulation of air flowing around the containment walls, with intake and exhaust through a stack or in tubes covering the inner walls of silos of underground reactors; and
- fluidic gates between process systems, such as "surge lines" of PWRs.

<u>Category C</u>

This category is characterized by:

- no signal inputs of "intelligence", no external power sources or forces; but
- moving mechanical parts, whether or not moving working fluids are also present.

The fluid motion is characterized as in category B; mechanical movements are due to imbalances within the system (e.g., static pressure in check and relief valves, hydrostatic pressure in accumulators) and forces directly exerted by the process. Examples of safety features included in this category are:

- emergency injection systems consisting of accumulators or storage tanks and discharge lines equipped with check valves;
- overpressure protection and/or emergency cooling devices of pressure boundary systems based on fluid release through relief valves ;
- filtered venting systems of containments activated by rupture disks; and
- mechanical actuators, such as check valves and spring-loaded relief valves, as well as some trip mechanisms (e.g., temperature, pressure and level actuators).

<u>Category D</u>

This category addresses the intermediary zone between active and passive where the execution of the safety function is made through passive methods as described in the previous categories except that internal intelligence is not available to initiate the process. In these cases an external signal is permitted to trigger the passive process. To recognize this departure, this category is referred to as <u>"passive execution/active initiation"</u>.

Since some desirable characteristics usually associated with passive systems (such as freedom from external sources of supply and from required human actuation) are still to be ensured, additional criteria such as the following are generally imposed on the initiation process:

- Energy must only be obtained from stored sources such as batteries or compressed or elevated fluids, excluding continuously generated power such as normal AC power from continuously rotating or reciprocating machinery;
- Active components are limited to controls, instrumentation and valves, but valves used to initiate safety system operation must be single-action relying on stored energy; and

- manual initiation is excluded.

Example of safety systems which may be included in this category are:

- emergency core cooling/injection systems, based on gravity driven or compressed nitrogen driven fluid circulation, initiated by fail-safe logic actuating battery-powered electric or electro-pneumatic valves;
- emergency core cooling systems, based on gravity-driven flow of water, activated by valves which break open on demand (if a suitable qualification process of the actuators can be identified); and
- emergency reactor shutdown systems based on gravity driven, or static

## 3.2 Paper on Passive Safety Systems for SMRs

| | |
|---|---|
| Conference: | Proceedings of the XXIX UIT –Heat Transfer Conference |
| Date: | June 20-22, 2012 |
| Title: | Analysis of Passive Safety Systems in a Small/Medium Size Nuclear Reactor |
| Authors: | D. Vitale Di Maio, F. Giannetti, A. Naviglio, L. Gramiccia, L. Ferroni |


## Abstract

The possibility of having a worldwide spread of nuclear energy, in next years, will be strongly affected by safety and management features of the new reactors. Projects in which passive systems are intensively used can guarantee the plant management without any external intervention in any plant condition.

Beyond the traditional large sized nuclear reactors, can be possible a development of a parallel market, characterized by small/medium sized nuclear reactors which, thanks to their special features, allow developing special safety systems. The choice prefer a small/medium sized nuclear reactor to a traditional size may be due to several factors including the possibility to use special safety systems developed ad hoc or the need of a small capital for the initial investment.

Specifically will be analysed safety systems related with the decay heat removal, and passive systems which, in a different way, can guarantee the safety of the entire system.

Several safety systems, some already developed for the MARS reactor while others derived from existing solutions will be critically analysed to identify what benefits can be effectively achieved, in terms of safety and economy.


## Main nuclear reactors safety features

Nowadays there are a lot of different technologies of nuclear reactors, in this study the attention is especially focused on nuclear reactors cooled by water and, among them, on the PWR technology. All water cooled nuclear reactor technologies have in common some safety functions that must guarantee their functionality in each plant condition. In particularly, the three main safety functions of nuclear reactors are:

- Fission products containment;
- Reactor shutdown;
- Decay heat removal.

Each safety feature will be shortly developed below, before discussing possible solutions applicable to small medium sized nuclear reactor rather than to large sized ones.
In order to simplify the paper comprehension each safety features will be analysed with reference to standard solutions used in GEN II PWRs, solutions adopted in newest PWR of GEN III and solutions used in the MARS[6] reactor design. Moreover, some studies on a small solution of the MARS reactor, mini-MARS, have been carrying out, during which some alternative safety features are developed and will be discussed below.

## Fission products containment

The containment of fission products is accomplished by the multi-barriers strategy; several elements in series are foreseen between the fuel and the environment. First of all, and common to all designs, the barriers are the ceramic composition of the fuel pellets, the fuel pin cladding and the pressure boundary of the primary circuit. Moreover, the reactor containment building is the last barrier before the environment.
Concerning on the reactor building, some differences can be identified among different reactor designs, even if, according to the GDC16[7], reactor containments must be designed with some specific features. The reactor containment shall be provided to guarantee a leak-tight barrier against uncontrolled release of radioactivity to the environment. Beyond the issues concerning the features, the functions can be reached with different designs.
Usually, a solution with large steel or concrete containment building, composed by one or more layers, is adopted; since the large size of the reactor vessel and auxiliary systems do not allow any alternative solutions. Concerning the Small and Medium size Reactors (SMR), many solutions can be applied to guarantee the fourth barrier from the leak of fission products.

[6] The MARS is a small size pressurized water reactor (PWR) with enhanced safety features. The MARS design was developed at the University of Rome "La Sapienza" and it shows many features that would have been taken from modern designs.
[7] 10 CFR 50 Appendix A, General Design Criteria 16

The newest nuclear reactors of large size, as the EPR and the AP1000, foresee a two layers containment even if with significant differences. The EPR has an inner containment (Reactor Building), made up of post tensioned concrete with a steel liner to ensure the tightness and an outer containment (Shield Building), made up of reinforced concrete to ensure protection to the reactor building from external events. The AP1000 is characterized by a double layer containment even if the outer one does not have tightness features. The whole system is composed by the containment building, made up of steel in order to have a sealed volume, and the shield building, made up of concrete and with special design features, to ensure the containment cooling only by natural circulation of an air flow, but with direct connections to the environment. The difference between the two projects is that the EPR design has a double containment with both mechanical and tightness features, the AP1000 design has only a double mechanical containment but a single tightness layer.
A SMR allows foreseeing different solutions to have a double containment, both mechanical and physical, as in the MARS design solution: a double layer reactor pressure vessel rather than a double containment building. In the MARS design solution, outside the reactor pressure vessel (RPV), there is a secondary vessel creating an annulus around the RPV (see Fig. 1).

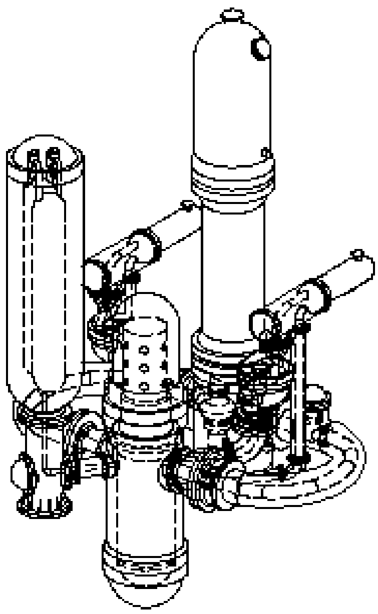

*Fig. 1. Double pressure vessel of the MARS reactor*

This solution allows to have only a one layer containment building, since the design is provided by two pressure vessels, and the possibility to have economic advantages, depending on the RPV and containment sizes. Also safety features of project would result enhanced, according to the defence in depth strategy, since with a double pressure vessels the LOCAs can be considered almost impossible. The RPV operates almost without any pressure difference between the two wall sides, hence without any mechanical stresses due to inner pressure. The annular volume between the two vessels is maintained fill of pressurized cold water and only the outer pressure vessel is subjected to a pressure gradient. To maintain a low temperature water inside of the annulus, it is required a heat removal system during normal operating condition to compensate the heat losses from the RPV; the thermal losses are, in any case, limited by an insulating layer provided within the RPV. The presence of a dedicated heat removal system allows, thanks to special design solutions, to use the same heat exchanger to remove heat from the RPV during severe accident condition. Further studies are carried out on the possibility to apply the double vessel solution to a smaller reactor with lower thermal power than the MARS also because it is characterized by an integrated primary design: the core, the steam generators and the primary pumps are located within the RPV. It is noticed that, in a single unit plant, a reduction in the reactor size would mean an increase of the kWh cost. Things change when the reference power plant is equipped by several SMRs; since many systems can be used in common, by more than one nuclear reactor, the kWh cost would reduce, since the smaller size of the SMRs means the possibility to build systems or structures outside the plant. Moreover, special features can be developed and used to obtain enhanced safety passive systems, whose operation is based on natural laws.

## Reactor Shutdown

In PWR design the First Shutdown System (FSS) is provided by the fast insertion of some clusters of absorber rods while, as backup solution, the Second Shutdown System (SSS) is made up by an independent system that should also be physically different. The traditional Control Rod Drive Mechanism (CRDM) will not be analysed but only some innovative solutions will be take into account. Usually, as SSS is used water with a high boron concentration able to shutdown the fission reaction.

SMRs, in this case the focus is on PWRs, have smaller core size and could not be necessary using soluble chemical neutron absorber in the reactor coolant (boric acid), during normal operating condition, to flat the flux profile and/or compensate fuel consumption, rising in fission products production and burnable absorber depletion. SMRs can have the FSS with further systems/components to guarantee the reactor shutdown and also SSS based on different physical phenomena. In any case some systems or components, developed for SMRs, can be applied to larger reactor, since SMR usually have enhanced safety features compared with large size nuclear reactors.

During this study, several SMR designs were analysed and some interesting solutions were identified. Some examples will be shown below: the PSRD, a small size nuclear reactor conceptually developed by Japan, has a special latch system on the extension of the control rod shaft. The latch system, placed into the primary coolant above the core outlet, consists of a permanent magnet and a control magnet with the magnetic saturation flux characteristic, function of temperature. When, due to abnormal conditions, the primary coolant temperature raises the latch mechanism increases its temperature causing a reduction of the magnetic force and the absorber rods fall, automatically thanks to gravity force, into the core stopping the fission reaction.

Another system to assure redundancy in the reactivity control is optionally provided to the MARS reactor. Some absorber rods, dedicated only to shutdown, are maintained outside the core thanks to a special mechanical system based on differential thermal expansion of different metallic materials. An increasing in the primary coolant temperature causes a thermal expansion of mechanical structure and system in contact with the fluid. Choosing the right materials, depending on the specific operating conditions and on the foreseeable transients, the special mechanism developed allows control rods falling, only thanks to natural laws, within the core stopping the fission reaction (see Fig. 2).

Since the two systems just analysed are applicable also to traditional nuclear reactors, the attention is now shifted to systems applicable only, or preferably, to SMRs. In this case also some flashes on possible fast SMRs reactivity control systems are provided.

Is important to be noticed that, usually, SMRs have a once through core cycle since their cores, composed by a very limited number of fuel assemblies, have a small diameter. The absorber rod, especially in small fast nuclear reactors[8], can be completely provided outside the core region. A fast reactor has two important features that make it especially interesting for this kind of solution rather than thermal reactors:

[8] In fast reactor this solution can be easily applied since the reactivity control can be strongly based on neutrons leaks because of the mean free path of fast neutron compared with thermal one.

- Higher neutrons mean free path;
- Higher core power density (small diameter with the same power).

A special simplified system constitute of some rotatable elements made up of a half volume of reflector and the other half volume of absorber materials (see Fig. 3). This solution allows to have not any mechanical element above the core, guaranteeing more reliable actuating systems and shorter refuelling periods. After the reactor startup, when there is the higher reactivity excess, the half absorber volume of the control elements is rotated towards the core; when the design burnup is almost reached the half absorber volume of the control elements is rotated toward the outside while the reflector volume is directed toward the core.

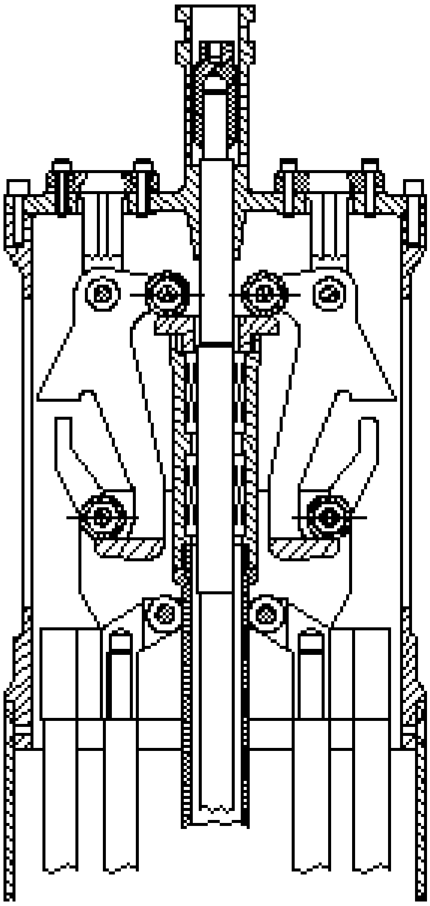

*Fig. 2. MARS passive safety mechanism for shutdown backup solution*

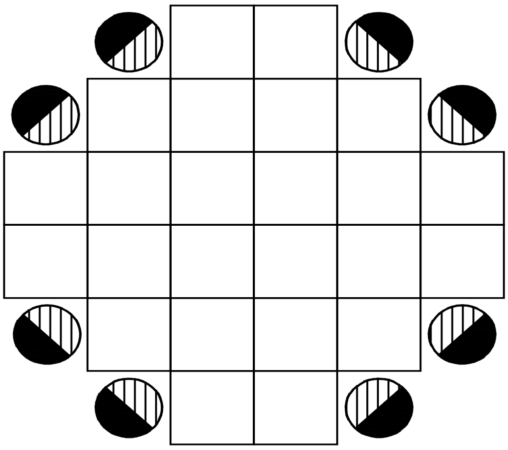

*Fig. 3. Core with external rotatable control rods*

The SMR design is generally characterized by an integrated structure of the primary system, a smaller core compared with traditional size PWR, lower pressure, absence of soluble boron in normal condition, higher primary coolant inventory and lower secondary coolant inventory. These features allow to use, as backup shutdown system, small volume water tanks with high boron[9] concentration; the scheme below (see Fig. 4) shows an active system, but also a similar system with passive features could be obtained. The main components/features of the system are:

- One tank with a very high boron concentration in water, up to 40000 ppm. To avoid the possibility to have boron deposit, the tank is heated, eventually pressurized, and mixed continuously;
- A pump to guarantee the boron injection;
- Since the whole operating time of the pump, required to obtain the complete injection of the boric acid and reach a stable subcritical condition, is shorter than 2 minutes, the pump can be powered by a dedicated batteries system.

In case of an accident in which the boron injection system is required, the pump between the boron tank and the RPV, automatically actuate the reactor shutdown. The boron injection system is designed as a high pressure system since, if necessary, the boron injection has to be carried out in a short time and wait for the primary depressurization is not possible.

[9] Boron enriched in boron 10 isotopes.

Concerning a hypothetical system configuration for the mini-MARS reactor, characterized by a thermal power of 100 $MW_{th}$ and by an integrated structure, a simplified calculation to evaluate the requested time to lead the reactor in a subcritical condition is carried out. The value of about 1300 ppm, requested to reach a subcritical condition and evaluated through neutronic calculation, is reached in less than 2 minutes.

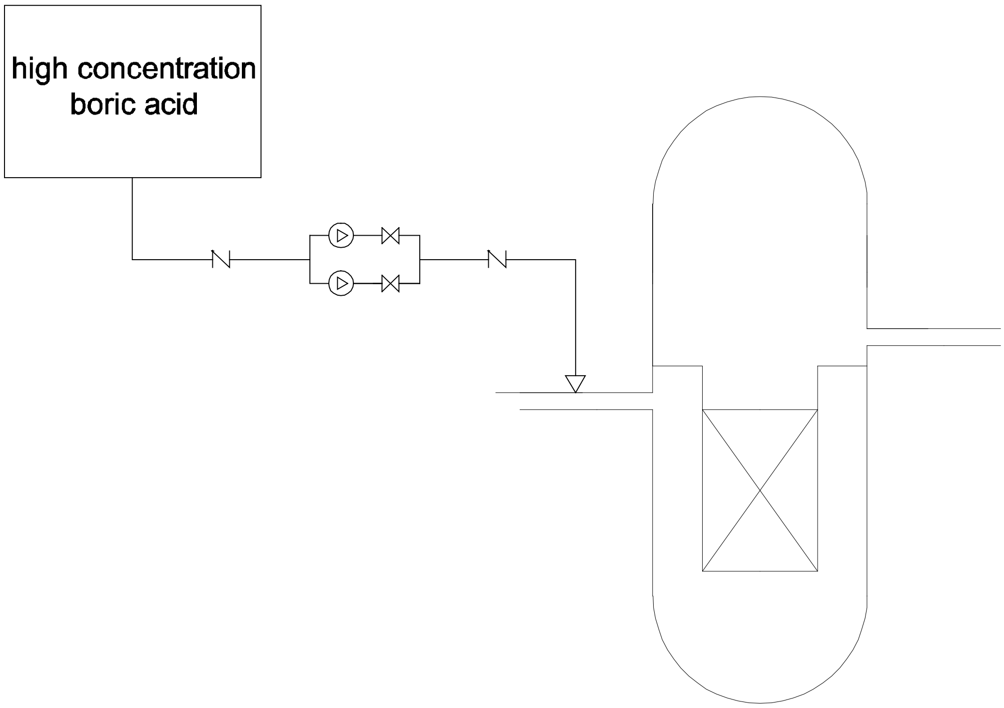


*Fig. 4. Scheme of high concentration boron injection system*

## Core cooling system and decay heat removal

The decay heat removal function, especially in emergency conditions, is one of the main safety related features that must be continuously maintained. Generally this function can be actively guaranteed by high, medium and low injection systems that allow having enough coolant inventory to control the core temperature and by active circulating systems, equipped by pumps that accomplish the heat release to an external heat sink. According to the Utility Requirements, the decay heat removal function has to be guaranteed at least for 72 hours without any external intervention. To meet safety requirements, the circulating pumps and all the energized components, must be connected to the plant emergency power system; generally supplied by the emergency diesel generators. After the Fukushima accident, we shall consider also the possibility that the plant emergency power system fails. This accident realized a great differentiation between completely passive actuated and operated systems and active actuated and operated systems.

Below, a simplified treatment of the Emergency Core Cooling System (ECCS), used in traditional large size PWR, is carried out. In this context is clearly visible the advantages of SMRs, since they must release a lower thermal power to the external environment taking advantage of the coolants natural circulation even though velocity and heat transfer coefficients result lower.

**Standard GEN II PWRs solutions**

The ECCS has two main functions: to ensure the core coolability even after primary coolant losses, and the maintenance of subcritical conditions following the cooldown due to a Main Steam Line Break Accident (MSLBA). To ensure these functions the ECCS uses the same water source: the Refuelling Water Storage Tank (RWST).

To perform the injection functions, four separate systems are foreseen: the high pressure injection system, the medium pressure injection system, the cold leg accumulators and the low pressure injection system. Each system is equipped by two redundant pumps, in parallel, each capable of providing the 100% of the required flow. These pumps are powered by the plant emergency power system; the only system among them that does not require power supply to operate is the cold leg accumulators. They operate with a pressurization made by nitrogen gas and when the primary pressure drops below the prefixed value, the nitrogen leads borated water into the RPV.

The long term coolability of the core is guaranteed by the recirculation alignment of the ECCS. It takes water from the containment sump; thanks to the pumps, water flows through the residual heat removal system heat exchanger to be cooled, then the cooled water is sent back to the RPV.

**GEN III PWRs solutions**

Two different philosophies can be identified in the newest PWR reactor designs:

1. Safety enhancement according to redundancy (four 100% systems), physical separation (each train is powered by the corresponding diesel generator and is completely independent from the others) and diversification criteria (different power supply systems can be powered the trains), based on active systems and actuating components.

2. Safety enhancement according to the minimization of the systems and components together with an intensive use of passive systems (core cooling function is ensured without any active system or component even if the actuation of the system requires an initial alignment of valves).

Even if the two design philosophies are completely different, the main functions of both are the same:

- Safety water injection to ensure the core cooling;
- Decay heat removal from the core when the steam generators are no longer available.

**SMRs possible solutions**

Hereafter, two possible solutions applicable to SMRs will be analysed. The first solution is taken from the MARS reactor design while the second is a similar solution, developed for the mini-MARS project, with enhanced safety features concerning the unlimited operating time without any external intervention.

The MARS ECCS is made up of three loops in series, a first loop where the primary coolant flows, a secondary loop to guarantee a sufficient number of barriers between the primary circuit and the external environment and, downstream of the secondary loop, there is a tertiary loop that allows system to release heat to the environment (see Fig. 5). The tertiary loop is composed by an atmospheric pressure pool, a condenser and connecting pipes. The condenser is a special atmospheric condenser able to condensate steam using atmospheric air as heat sink. Once the liquid water flows outside the condenser, thanks to the density difference between steam and liquid water, it is lead back to the pool to remove again the core decay heat from the upstream loops.

The startup and operating conditions of the Safety Core Cooling System (SCCS) are completely passive; the system starts when a special check valve opens due to loss of primary coolant flow. The theoretical “unlimited” operating time of the MARS SCCS results in practice limited by the presence of uncondensable gases inside the atmospheric condenser that requires a system to remove them.

An alternative solution, in order to overcome the problem related to uncondensable gases, has been developed, and some components patented, during studies carried out on the mini-MARS solution (see Fig. 6). The innovative solution, especially suitable for small size nuclear reactors, is characterized by three loops in series, all operating with coolant in natural circulation, and by a pool. The three loops thermally connect the core and the pool, used as heat sink, guaranteeing the decay heat removal. The main difference, between the MARS and the mini-MARS solutions, is the absence of the atmospheric condenser and the consequent possibility to obtain a never ending heat sink.

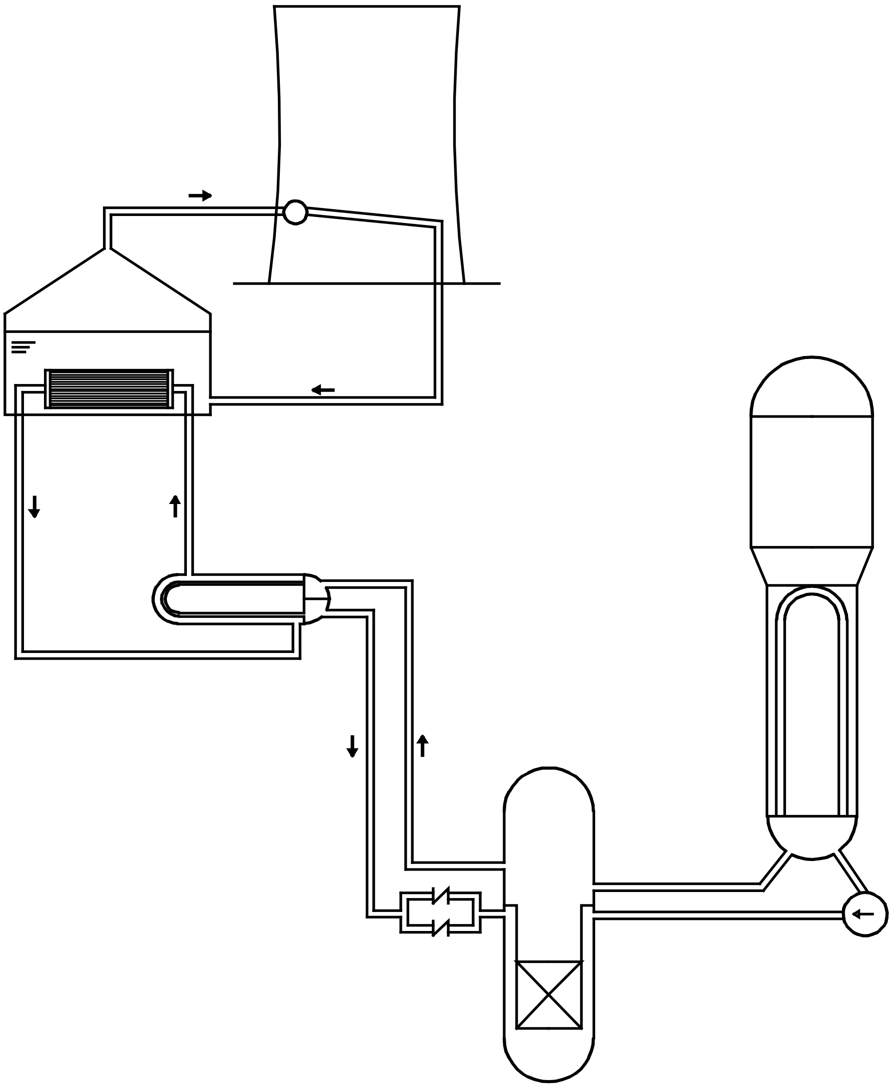

*Fig. 5. Scheme of the MARS ECCS solution*

The pool, fill of water, guarantees a higher heat transfer coefficient during initial transient phase, when the thermal power to remove is higher. Afterwards, when the pool water inventory decreases, due to the water vaporization, the heat sink slowly and automatically shifts from water heat sink to an air only heat sink but, at the same time, the decay heat to be removed is decreased.

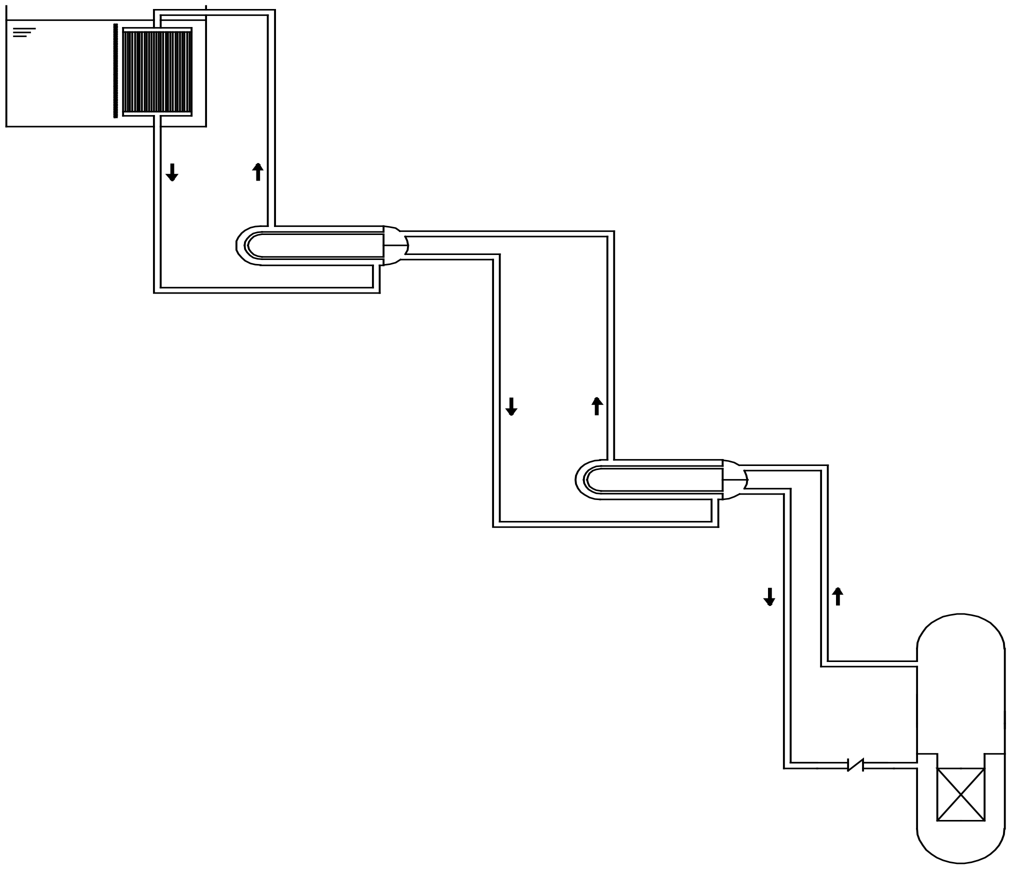

*Fig. 6. Scheme of the mini-MARS ECCS solution*

In order to obtain a sufficient delay in the air only heat sink occurrence, making compatible the decay heat to remove with the heat transfer to a natural circulation air flow, the initial pool water inventory should be chosen large enough.

## Conclusions

As conclusion of this analysis on engineering safety systems, many points, on which the attention has to be focused, are emerged:

1. The SMRs, thanks to their small dimensions, can be easily managed, both concerning the construction and operability of the reactor (i.e. the possibility to avoid soluble neutron absorber during operating condition or the possibility to preassemble structures or systems before arriving on the plant site).
2. The SMRs, by definition, have a limited core thermal power; this means a smaller decay heat to remove during shutdown condition. This feature allows using air as

heat sink to guarantee a never ending heat removal during reactor emergency conditions. The low heat transfer capability of the air and the low thermal power to release in the environment allow using finned heat exchanger with dimension small enough to be considered constructively and economic practicable.

3. After Fukushima accident, it is predictable that emergency systems, also if their operation is required with a probability close to zero, should be based on passive concepts. A power plant where safety systems are passive and characterized by an unlimited operating period, without any external intervention, ensures a greater reliability also in the condition of loss of on and off site power.

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# 4 SAFETY SYSTEMS SUITABLE FOR ANY NUCLEAR REACTOR, ESPECIALLY SMRS

According to the main nuclear reactors' safety requirements (see also chapter 3), some systems to guarantee the decay heat removal in any accidental condition have been studied. In the present chapter, an activity focused on the development of a never ending heat sink, especially suitable for Small and Medium sized nuclear Reactors[10] (SMRs), is presented. The heat sink is a special pool, whose operation is based on natural laws and special solutions that make it able to operate without any external intervention and for an unlimited period of time.

## 4.1 Background of the invention

The University of Rome "La Sapienza" (Department of Nuclear Engineering and Energy Conversion) developed, since the 1984, the design of an innovative SMR, PWR type: the MARS. The design of this reactor is characterized by many innovative solutions; the most important ones are:

- the CPP – Containment for Primary loop Protection;
- the ATSS – Additional Temperature-actuated Shutdown System;
- the SCCS – Safety Core Cooling System.

The first solution is aimed at avoiding the reactor vessel to be operate under a pressure difference between the inner volume (all primary coolant pressure boundary) and the outer zone. To meet this requirement, a second circuit, made of piping and vessels (larger than the piping and vessels of the primary coolant

[10] According to the IAEA definition, small nuclear reactors are characterized by an electric power less than 300MW while medium nuclear reactors by an electric power less than 700 MW.

pressure boundary), filled with pressurized cold water, encloses all primary coolant components, including the primary reactor vessel. This solution, guaranteeing an extreme reduction of any stress on the primary coolant pressure boundary components, including reactor vessel, practically eliminate any unlikely possibility of failure. Since the CPP encloses also the primary coolant loop, any LOCA occurrence is automatically avoided by design.

The ATSS is an additional safety system, finalized to passively reach the condition of reactor shutdown, based on the differential thermal expansion of two concentric pipes made of metals characterized by different thermal expansion coefficients.

The last system, the SCCS, is the most important one in relation to the present work. The SCCS is a completely passive safety system aimed at removing the decay heat from the reactor core without any external intervention (e.g. energized systems or components, plant operators, etc.) and for an unlimited period of time, exploiting the density difference between hot and cold legs to allow the coolant circulation. The SCCS system is characterized by three circuits in series (Figure 2):

- the Primary Safety Circuit (PSC) through which the primary coolant flows;
- the Intermediate Safety Circuit (ISC) though which subcooled water flows;
- the Tertiary Safety Circuit (TSC) through which water and steam flow.

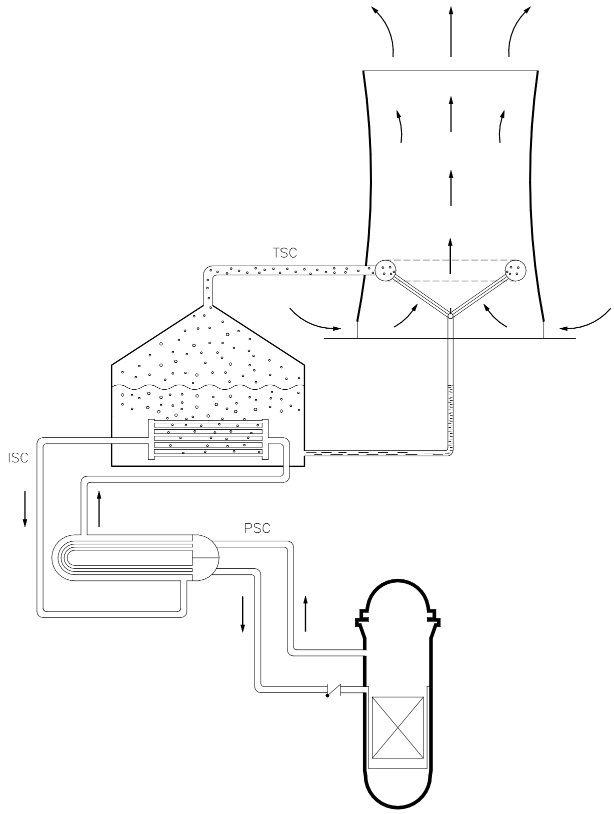


**Figure 2. MARS SCCS circuits**

The tertiary circuit is equipped with a water pool at atmospheric pressure (at the beginning filled with cold water), a condenser and connection piping. During reactor normal operating condition, the pool is filled with subcooled water at ambient temperature. When the SCCS becomes operative, the heat removed from the primary coolant causes the water pool temperature increase. This temperature rises up to the saturation temperature (at a pressure slightly higher of the external atmospheric value) reaching the heat transfer condition of pool boiling; hence this sets the TSC temperature at about 100 °C. The produced steam, flowing through the TSC, releases the heat to the atmospheric air, condensing in a special condenser with an innovative design and turning back to the pool, in a closed two-phase loop (Figure 3).

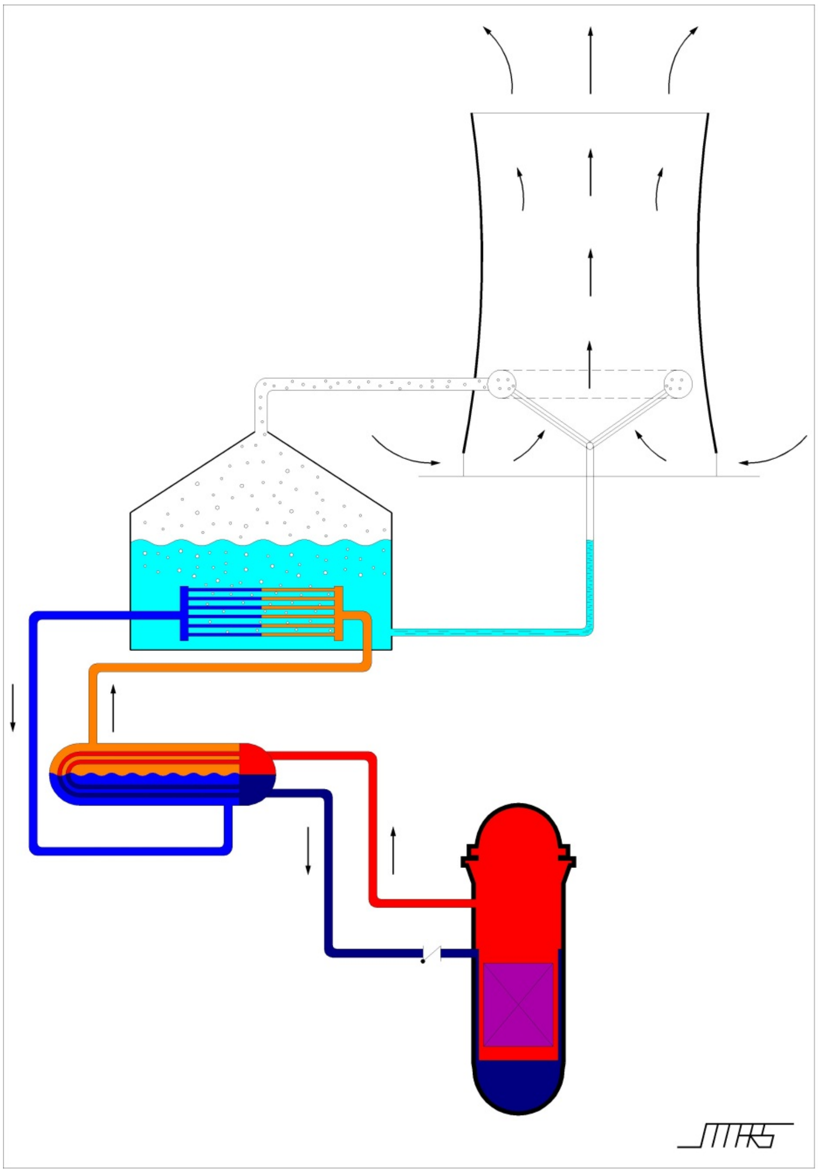

**Figure 3. MARS SCCS in operation**

The system can be automatically actuated by means of a special completely passive check valve (to comply with the redundancy criterion, a standard check valve is installed in parallel to this innovative component). The innovative check valve is kept in closed position exploiting the pressure difference between the reactor vessel inlet and outlet, corresponding respectively to the discharge and to the suction of the primary pump. The pressure difference is approximately proportional to the square of the mass flow rate through the reactor core. When the primary flow rate decreases and hence the pressure difference, the valve automatically opens allowing the primary coolant to flow through the SCCS.

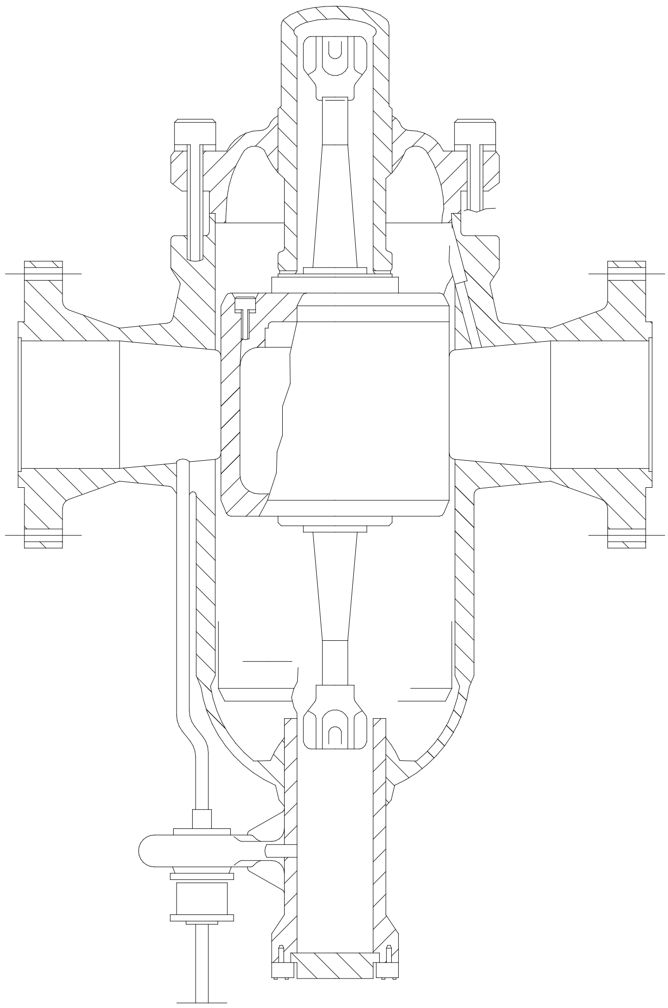

**Figure 4. MARS special check valve**

In the scheme of the innovative check valve (see Figure 4) the valve plug is represented in its upper position (valve closed); when the pressure difference between inlet and outlet of the reactor vessel is no longer sufficient to maintain the valve closed (due to a reduced coolant flow rate), the valve plug falls, allowing the primary coolant to flow through the primary loop of the SCCS.

This system has very innovative features and its design introduced very important safety issues in conceiving new nuclear reactors designs. More recent projects have been characterized by enhanced safety features, never seen before the MARS project.

Nevertheless, the SCCS shows some limits in a long period operation since no information related to the presence of noncondensable gases in the TSC are predictable. A very limited amount of noncondensable gases within the steam can strongly reduce the heat transfer performance of the system, making it not able to release the decay heat to the atmosphere [5].

The presence of noncondensable gases strongly affects the global heat transfer performance since gases represent a thermal resistance at the vapor/condensate interface. Due to condensation phenomenon, vapor moves towards the heat transfer surface and condenses; noncondensable gases move with the vapor up to the cold heat transfer surface where they are accumulated. As direct consequence, the condensable partial pressure near the cold wall results to be reduced and hence the “driving force” for condensation is also reduced. Furthermore, the concentrated layer of noncondensable gases partially prevents the vapor from reaching the wall. For these two reasons the condensation rate is reduced even at a relatively low concentration of noncondensable gases.

The presence of noncondensable gases would reduce the whole SCCS heat transfer performance making the system no-longer capable to completely remove the decay heat from the reactor core. Due to this not negligible problem, an off-gas system was studied to guarantee the noncondensable gases removal from the steam.

An innovative system, which takes into account the problem of the noncondensable gases overcoming it, has been studied and is presented in the paragraphs below.

## 4.2 Pool description

The whole new DHR system has been conceived with a design similar to the MARS SCCS. The system[11] proposed and here below described is the final heat sink and it is equipped with a special heat exchanger (PHX) that allows to release the decay heat to the atmosphere. In addition to the heat removal function, this top-reliable heat sink is characterized by special safety features and by a very high simplicity. The special design of the system allows good heat transfer capability, especially characterized by a functional trend that can easily be defined in order to follow the decay heat trend, and unlimited operating time without any external intervention.

[11] According to the IAEA definition, this system is a passive system that belongs to the Category “C” (see paragraph 3.1).

The PHX is made by the following main components: a pool, a separating septum and a heat exchanger (Figure 5).

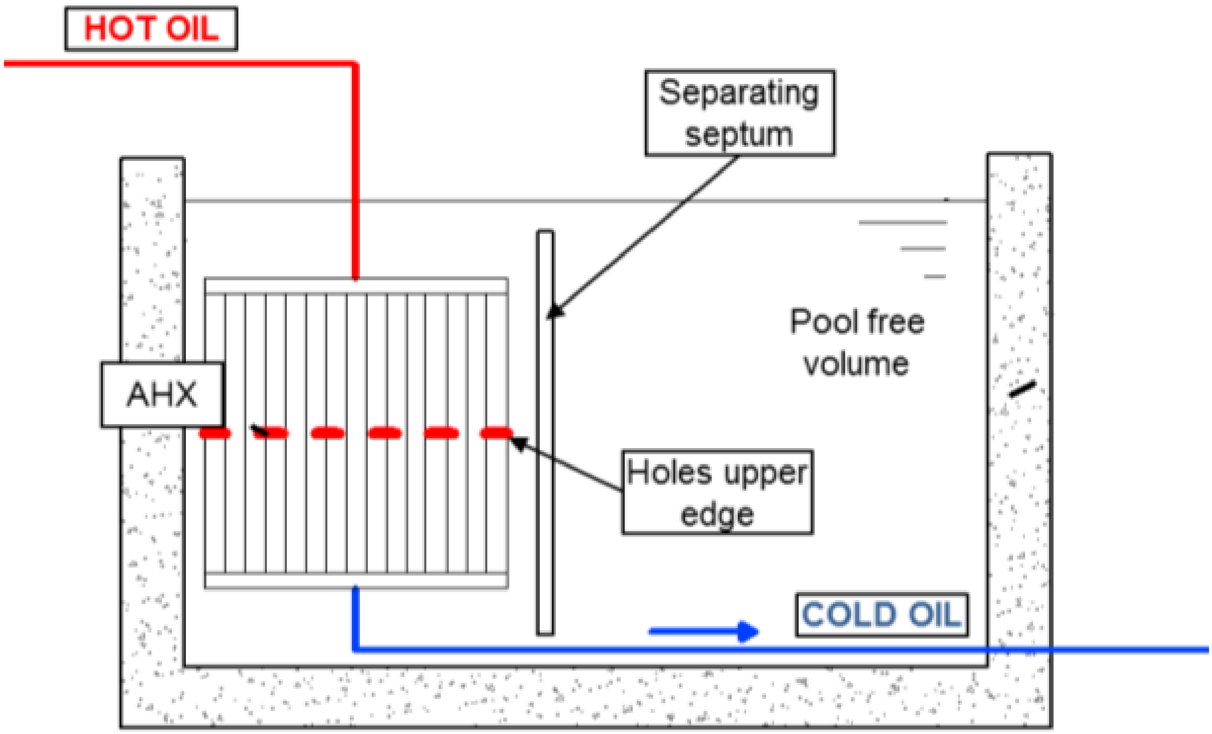


**Figure 5. Scheme of the PHX components**

In order to achieve a never-ending heat sink capability, the pool has been divided into two interconnected volumes separated by a special septum. In one of the two regions, there is the heat exchanger, while the other region does not contain any mechanical component. The innovative components, expressly developed for the special purposes of this system, are the heat exchanger and the separating septum, both characterized by a special design.

The heat exchanger is made up of vertical tubes equipped with four special fins and arranged in a square matrix. Each fin is provided, in its bottom region, by vertical slits (Figure 6). Fins of adjacent tubes are linked together constituting about as many squared sub-channels as the tubes' number, creating a vertical tube heat exchanger (Figure 7).

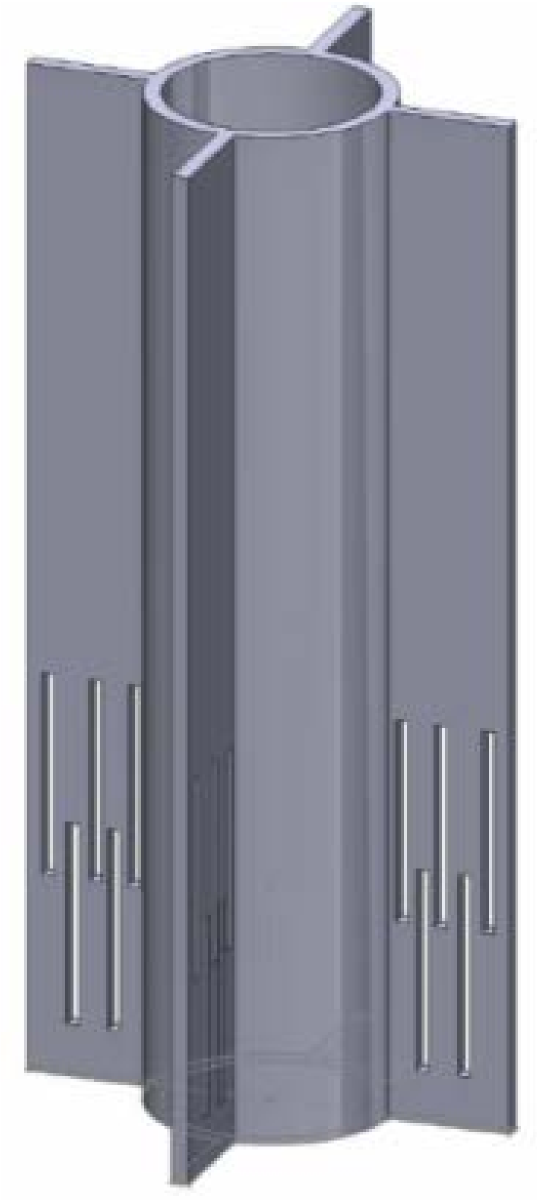

**Figure 6. Single tube of the PHX**

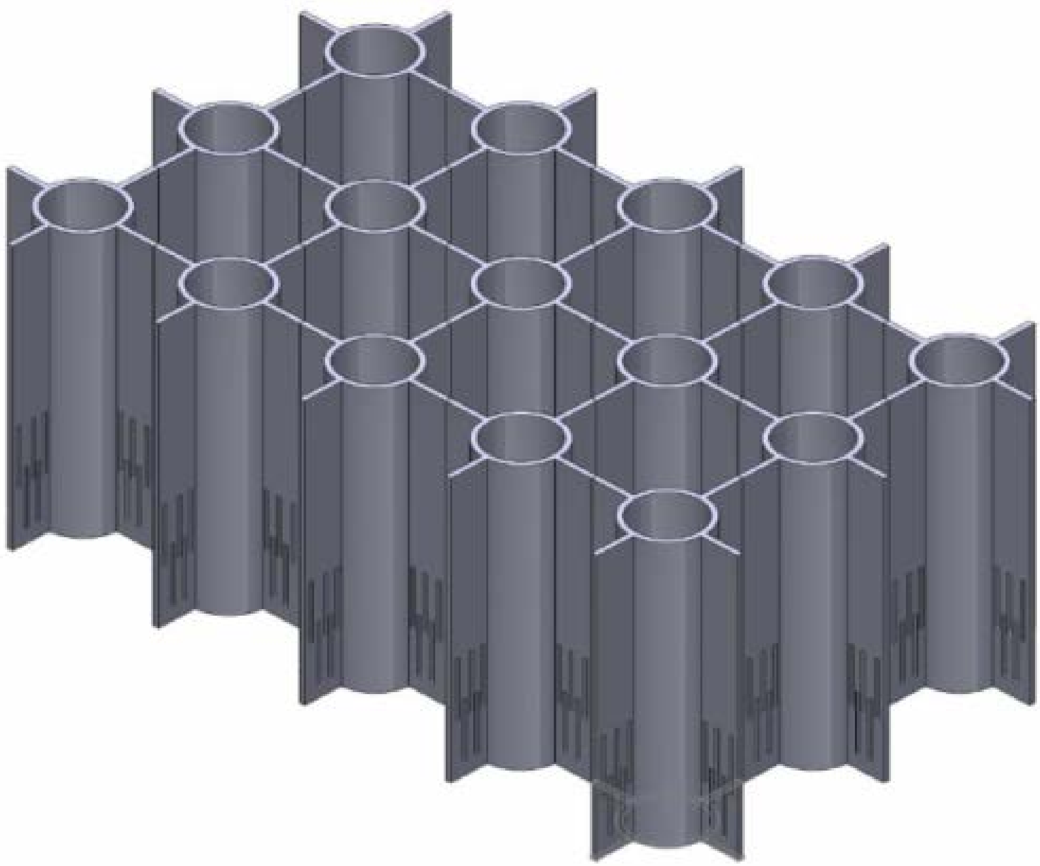

**Figure 7. Layout of PHX (out of scale)**

The special design of the opened fins allows mixing the water throughout the pool crossing different adjacent sub-channels. Later, when the pool water level is significantly reduced (due to boiling depending on the decay heat removed), the vertical slits allow air to flow into the heat exchanger before the complete vaporization of the pool water, preventing any possibility to have flooding occurrence at the heat exchanger bottom edge.

Thanks to the opened fins, the two bottom volumes of the pool are in communication, while in the upper region of the heat exchanger the two volumes are independent. At the beginning, in the stand-by condition, the water level should be maintained over the fins' upper edge in order to facilitate the pool water heating, exploiting the whole heat capacity of the pool. Maintaining the water level higher than the fins allows natural circulation to easily establish within the pool. As for the fins, also the separating septum is provided by vertical openings in the bottom region to allow mixing water among adjacent sub-channels and to allow, later, an atmospheric air flow through the heat exchanger sub-channels from the empty volume before the complete vaporization of the pool water. In Figure 7 an isometric view of the PHX is shown. In Figure 9 a detailed view of the PHX sub-channels with the vertical slits is shown.

**Figure 8. View of the PHX**

**Figure 9. PHX detail of the sub-channels and the fins**

## 4.3 Pool operating condition

The special feature of the PHX system is its capability to release the heat to the water pool at the beginning and, after the complete water vaporization, to the atmospheric air. This shifting from the first to the last condition happens without any external intervention nor active systems/components nor plant operators. An accurate description of the AHX operating conditions is reported here below. The different phases of the pool operating condition are the following:

- stand-by condition;
- heating phase;
- pool boiling condition;
- mixed heat transfer condition (pool boiling in the lower part and air + steam convection in the upper one);
- convection with air only, as heat sink.

### 4.3.1 Stand-by Condition

During the reactor normal operating condition, the whole DHR system, including the PHX (heat sink), is switched off by means of a passive check valve (Figure 10). The pool water level has to be maintained high enough to guarantee a sufficient system heat capacity.

Upstream of the pool more than a single loop can be foreseen to provide a multi-barrier protection between the primary coolant and the external atmosphere. For simplicity, in the Figure 10, the check valve has been represented on the last loop (for scheme simplification purposes only) despite the best solution foresees the check valve on the first loop. The solution that foresees the check valve on the first loop is preferable since the passive opening should be directly based on the primary circuit behavior. The check valve to isolate the circuit is required to minimize the thermal losses during reactor operating conditions preventing the coolant flow within the loops of the safety decay heat removal system.

The passive check valve has not been redesigned since the solution already developed for the MARS reactor could be easily applied to this system (only small changing to fit with the reactor size should be foreseen).

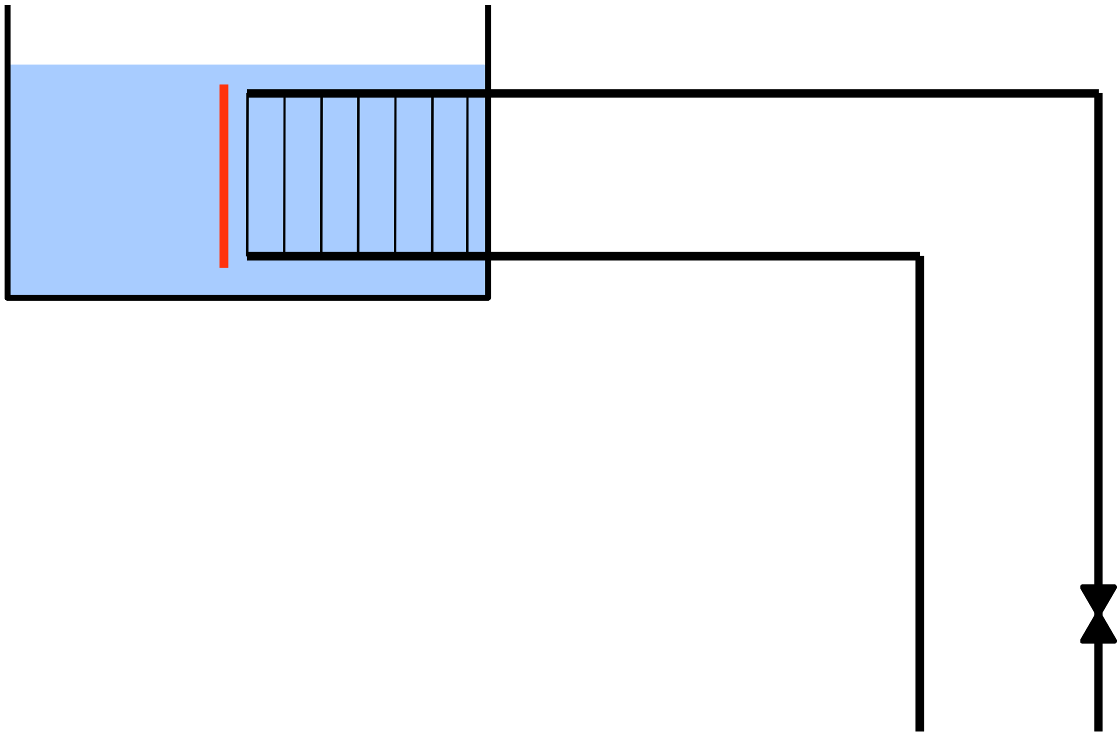

**Figure 10. PHX Stand-by condition**

### 4.3.2 Heating Phase

When the DHR has to operate, the check valve is automatically opened allowing the coolant circulation. As already said, between the pool and the primary coolant, at least another loop should be foreseen, equipped with its own heat exchanger. If the check valve is installed on the primary loop of the DHR (where primary coolant flows), the coolant within the last loop would start to flow after the temperature has increased enough to guarantee a sufficient density difference between hot and cold legs. Once the coolant flows by natural circulation, the pool water, subcooled at the beginning, starts to increase its temperature up to the saturation point (Figure 11) thanks to internal mixing flows due to the density difference.

Known the decay heat that has to be removed, and thanks to the free choice in the pool dimensions, the plant designer can easily define the time at which the saturation temperature will be reached. From that time on, the pool boiling condition occurs and the water level starts to continually decrease. The right definition of pool operating condition timeline is of primary prominence since following the decay heat power trend, the pool dimensions can be optimized on the specific plant in which it has to be installed.

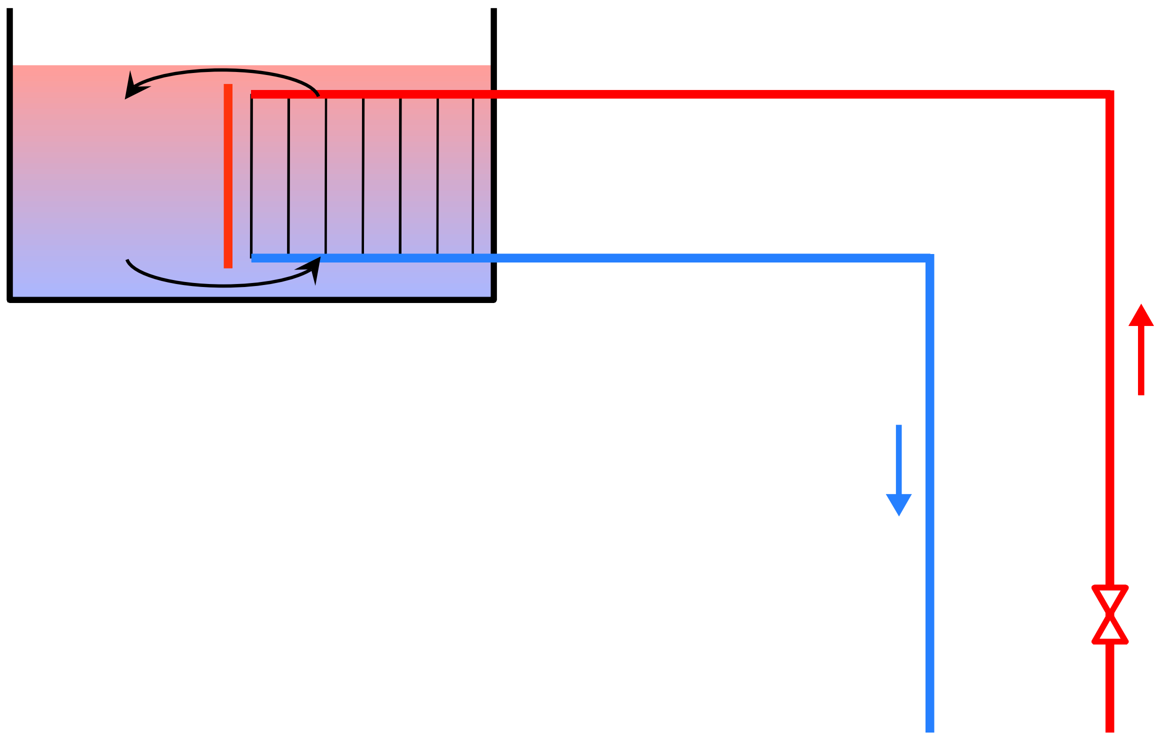

**Figure 11. PHX Heating Phase**

As an example, a comparison between two different solutions is here represented in order to understand the importance of the pool dimension on the transient timeline. Only the water heating period is here analyzed. The comparison has been carried out taking into account a large size nuclear reactor (design thermal power of 3600 MW) and a small nuclear reactor (design thermal power of 100 MW).

The reference dimensions of the pool, for the comparison, are shown in Figure 12 and Table 1. Since to be passively operative (exploiting the natural circulation) the pool has to be installed at higher level compared to the heat source, the pool should be installed on the top of the containment building or in an elevated area at the plant site (choice depending also on criteria of protection from external events). Obviously, larger reactor size means larger pool and hence more critical design issues related to the large amount of water in the pool itself.

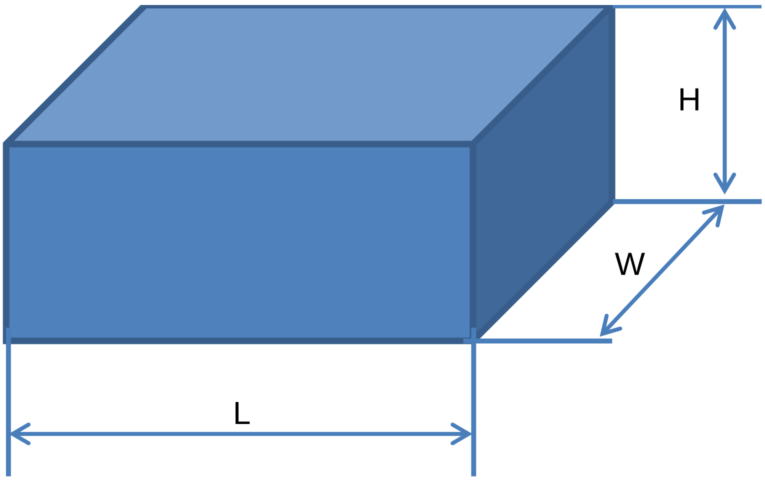


**Figure 12. Scheme of the pool dimensions**

**Table 1. Pool reference dimensions**

| | | |
|---|---|---|
| Large (L) | [ m ] | 3.00 |
| Width (W) | [ m ] | 3.00 |
| Height (H) | [ m ] | 5.00 |
| Volume | [ $m^3$ ] | 45.00 |

The pool water heating is assumed to be from the ambient temperature (20 °C) up to the saturation point at atmospheric pressure (100 °C). In the geometrical condition above defined (see Table 1), , the energy (*E*) to be released to the water pool in order to reach the saturation point is about 1.5E+04 MJ. Since the decay heat produced by the core decreases with time, the required time to reach the saturation conditions (*t**) in the pool has been evaluated by the equation (2.4) as follow:

$$E = P_0 \int_{150}^{t^*} \frac{0.13}{t^{-0.283}} dt \tag{4.1}$$

The correlation here considered for the evaluation is the relation corresponding to the time range between 150 s and 4.0E+06 s, the correlations relating to the time before 150 s have not been considered since at those times it is not credible that the natural circulation has been already established in all the circuits, including the water pool volume[12] (see Table 2).

**Table 2. Comparison between two different core sizes**

| | | | |
|---|---|---|---|
| $P_0$ | [ MWth ] | 3600 | 100 |
| time | [ s ] | 150 | 12240 |
| $P_{average}$ | [ MWth ] | 102 | 1.22 |
| $P/P_0$ | [ % ] | 2.82 | 1.22 |

The very limited period of time such as that evaluated for the large sized nuclear reactor, is not acceptable because the temperature of loops would increase too much, leading to unacceptable conditions.

What has been said means that this system could be efficiently installed to systems characterized by a limited thermal power. For larger nuclear reactors, this system could be evenly installed but the large dimensions that the pool would have, make interesting a plant solution based on the redundance criterion. Indeed, a large nuclear reactor with a thermal rate power of about 3600 MWth requires a DHRS that could be sized on approximately 30 MWth (less than 1% of the design power). In order to reduce the thermal power to be removed by a single PHX system, four pools with a 50% capability (about 15 MWth each) can be foreseen. In this condition, only two operating pools (2x50%) are required to completely remove the decay heat from the primary circuit, guaranteeing at the same time a limited size for each single pool.

### 4.3.3 Pool Boiling Condition

Once the saturation temperature has been reached, the pool boiling condition within the pool guarantees higher heat transfer coefficient and hence a higher heat removal capability.

[12] During first transient period, before the natural circulation is established, the whole system temperature increases and later, when the DHRS is able to remove more heat than the decay power, the system temperature starts to decrease.

The pool boiling condition at atmospheric pressure sets the temperature at 100 °C and the high heat transfer coefficient characterizing this heat transfer mechanism, limits at about the same temperature, the lower temperature of the last loop of the DHR. Water vaporization from the pool causes the water level reduction during time and hence a reduction of the “active” heat transfer surface of the PHX.

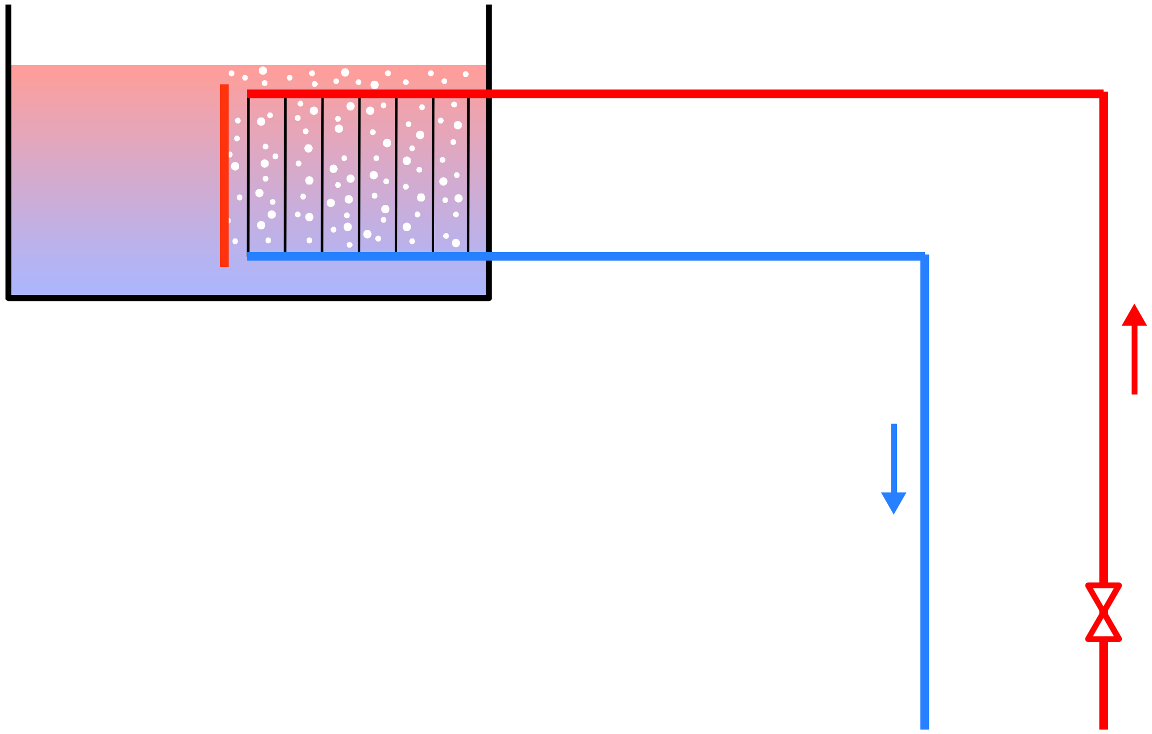

**Figure 13. PHX Pool Boiling Condition**

## 4.3.4 Mixed Heat Transfer Condition

As the water level decreases, the reduction of the heat transfer surface is acceptable up to a certain value; beyond that value, unacceptable conditions would be reached if a special solution, to increase the heat removal capability of the system, is not foreseen (the loss of water up to the bottom region of the pool would prevent the PHX from cooling).

In order to compensate the heat transfer surface reduction, due to the level variation, the system has been especially designed to allow air entering and flowing through the PHX before the complete vaporization of the pool water. Thanks to the specially designed openings on the fins and on the separating septum, an atmospheric air flow can contribute to the heat removal before the water complete vaporization.

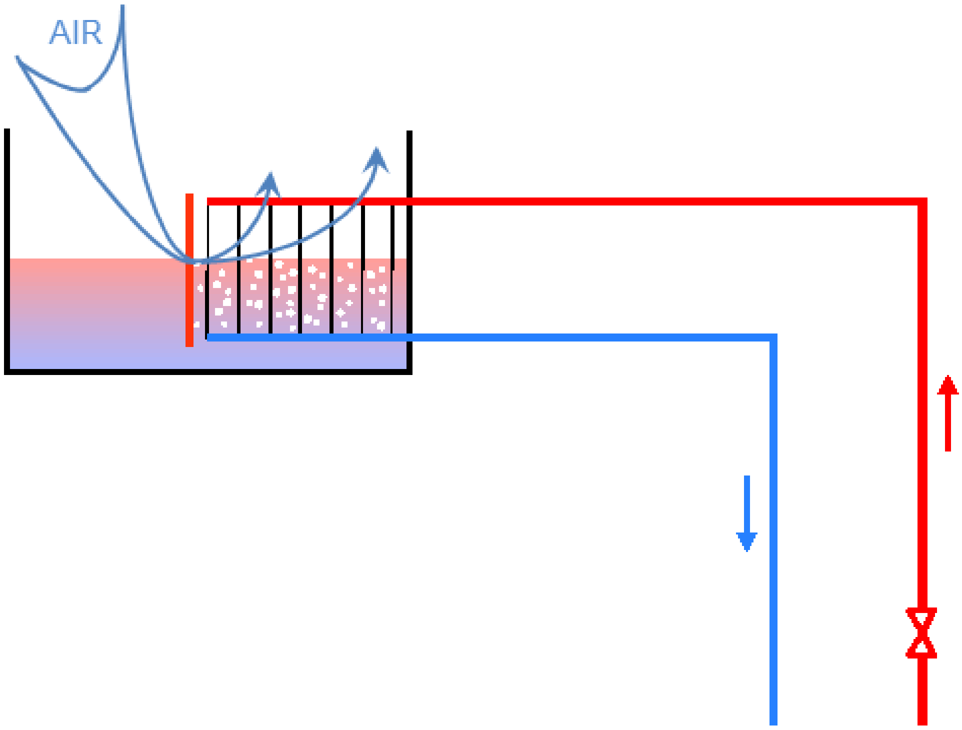


**Figure 14. PHX Mixed Heat Transfer Condition**

The height of the fins is defined by the designer in order to control the maximum temperature that will be reached, without any criticality, within the primary loop. This condition is present until the water complete vaporization.

### 4.3.5 Air Only Heat Sink Condition

Once water is completely vaporized, the heat removal is guaranteed by air only. The most critical condition during the whole transient period is usually reached at the moment in which the complete water vaporization occurs.

This last operating condition guarantees a never-ending heat removal capability. The heat transfer coefficient and performance reduction, due to the shift from water to air heat sink, is compensated by the reduction, during the transient time, of the decay heat produced by the core.

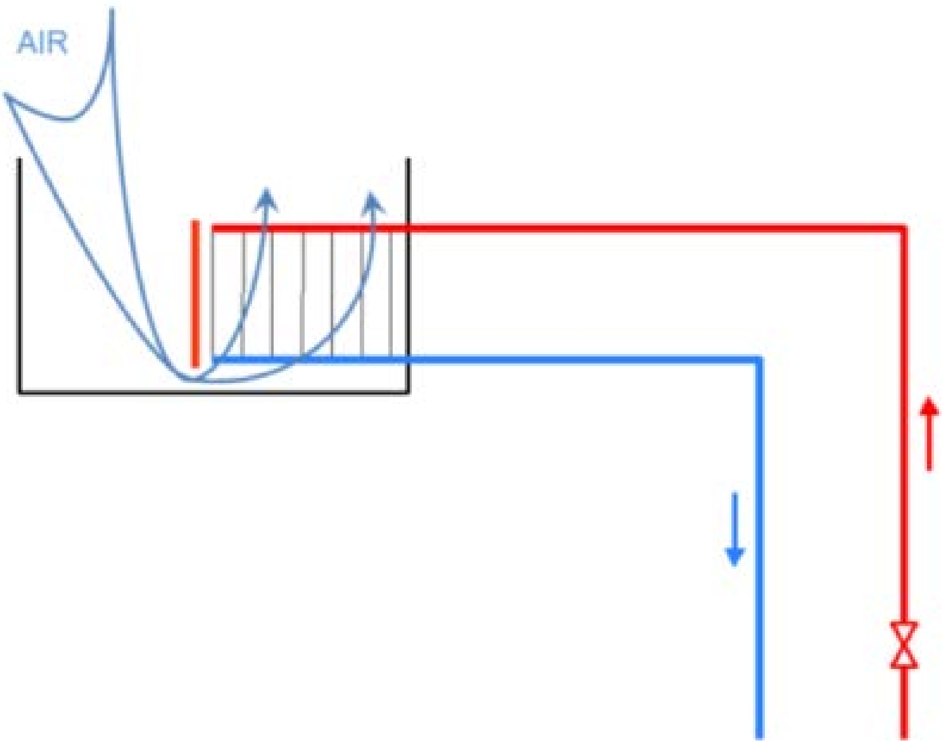


**Figure 15. PHX Air Only Heat Sink Condition**

The heat transfer coefficient is higher when the PHX operates with water, while air allows to remove a lower amount of power for an unlimited time in the last period of the transient; it should be noticed that the heat transfer coefficient roughly follows the trend of the decay heat generated in the reactor core after the shutdown.

Many degrees of freedom are left to the designer who should size the pool so that the shifting to the air only heat sink condition is sufficiently delayed in time, to be compatible with the reactor decay heat produced, preventing any unacceptable condition. At the same time the heat exchanger dimensions (i.e. tube number, length, diameter and pitch to diameter ratio) allow to increase the peak heat removal capability. A good design should take into account both these features to optimize the system to the specific application case.

The main features of the system are the high availability and reliability. Due to its static components, the system does not require intervention during stand-by conditions (only the valve has to be opened to guarantee the heat removal, but a special passive valve, such as the MARS one, can be adopted).

Many correlations have been found for natural convection on a vertical plane wall in open field; however, the geometry of the system creates many vertical channels, heated from the surface, in which the chimney effect takes place. In this condition, the air through the PHX will not behave according to a natural convection flow in an open field, but rather will behave in a forced flow. For this reason, the heat transfer coefficient has been evaluated, by means of an iterative process, with simpler correlation related to forced convection. The iterative process is necessary to evaluate the heat transfer coefficient in natural circulation condition since it depends on the mass flow rate that is a function of the density difference between hot and cold legs, which in turn depends on the heat transfer coefficient itself[13]. For the reference solution, applied to a nuclear reactor with a design thermal power of about 100 MW[14], the average heat transfer coefficient results to be about 12 W $m^{-2}$ $K^{-1}$.

[13] the Dittus-Boelter correlation has been used for the evaluation of the heat transfer coefficient in the air only heat sink condition:

$$Nu = 0.023 \cdot \mathrm{Re}^{0.8} \cdot \mathrm{Pr}^{0.4}$$

Where:

Nu is the Nusselt number [ ]: $Nu = \frac{h \cdot D}{k}$

Re is the Reynolds number [ ]: $\mathrm{Re} = \frac{\rho \cdot u \cdot D}{\mu}$

Pr is the Prandtl number [ ]: $\mathrm{Pr} = \frac{c_p \cdot \mu}{k}$

[14] The reference condition is equal to that described in the paper reported in the paragraph 4.4.

## 4.4 Paper on the innovative pool

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Title: An innovative pool with a passive heat removal system

Authors: D. Vitale Di Maio, A. Naviglio, F. Giannetti, F. Manni

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## Abstract:

Heat removal systems are of primary importance in several industrial processes. As heat sink, a water pool or atmospheric air may be selected. The first solution takes advantage of high heat transfer coefficient with water but it requires active systems to maintain a constant water level; the second solution takes benefit from the unlimited heat removal capacity by air, but it requires a larger heat exchanger to compensate the lower heat transfer coefficient. In nuclear power plants during a nuclear reactor shutdown, as well as in some chemical plants to control runaway reactions, it is possible to use an innovative heat sink that joins the advantages of the two previous solutions. This solution is based on a special heat exchanger submerged in a water pool designed so that when heat removal is requested, active systems are not required to maintain the water level; due to the special design, when the pool is empty, atmospheric air becomes the only heat sink. The special heat exchanger design allows to have a heat exchanger without being oversized and to have a system able to operate for unlimited period without external interventions. This innovative system provides an economic advantage as well as enhanced safety features.

**Keywords:**

Heat exchanger, Nuclear, Passive systems, Residual heat removal system, Emergency Core Cooling System.

## Introduction

In the core of nuclear reactors, when a shutdown occurs, fission chain reaction is stopped but, because of decay heat release by fuel, thermal power generation continues at considerable levels for a long time. In order to avoid the occurrence of a severe accident, safety systems (Decay Heat Removal Systems - DHRS) aimed at removing this thermal power, have safety functions of primary prominence. Quite demanding features are required to these systems, first of all regarding availability and reliability. The function of decay heat removal can be achieved through different systems, characterized by active or passive components. Both types of these safety systems must have all components in a safety grade class but the main difference, between active and passive systems, is due to their dependence on external mechanical or electrical power, signals or forces. Thanks to the reliance on natural laws only, passive systems require a lower number of components. Because of these features, DHRS are often based on passive systems in order to obtain higher safety performance with fewer components.

## The advantages of a completely static emergency decay heat removal system: impact on system reliability and plant safety

Because of the limited number of components involved in passive systems and the independence from human intervention, their use produces both economic advantages and safety enhancement.

Safety systems must have at least two different states: the first one corresponds to the normal operating condition and the second one corresponds to the safety operating condition. In order to change from normal to safety condition, and to be able to operate in safety conditions, it will be necessary to have: a signal to detect the need, a driving force to change the state and the necessary means to operate in the new state. If these three features rely on passive components, the system may be defined as passive.

Both systems, active and passive, have to be design according to the redundancy criterion. Moreover, an important difference between moving and static systems exists. The more similar the two operating states of a system are the less failure modes of the system will be. According to this, passive systems are classified, by the IAEA [1], in four different classes (from lower to higher passive characteristics) depending on their main features, i.e. mechanical moving parts, moving fluids, requirement of input signals.

The innovative system here proposed, is characterized by only moving fluids. Only a valve has a mechanical moving part, to allow the interception of the decay heat removal system during normal operating conditions. The simplified scheme Fig. 1 shows the valve present in the system, the only moving part of the whole safety system. This component is not analyzed in this paper, but different solutions have been studied and developed to design a passive component which may belong to the highest class for passive systems with mechanical moving parts.

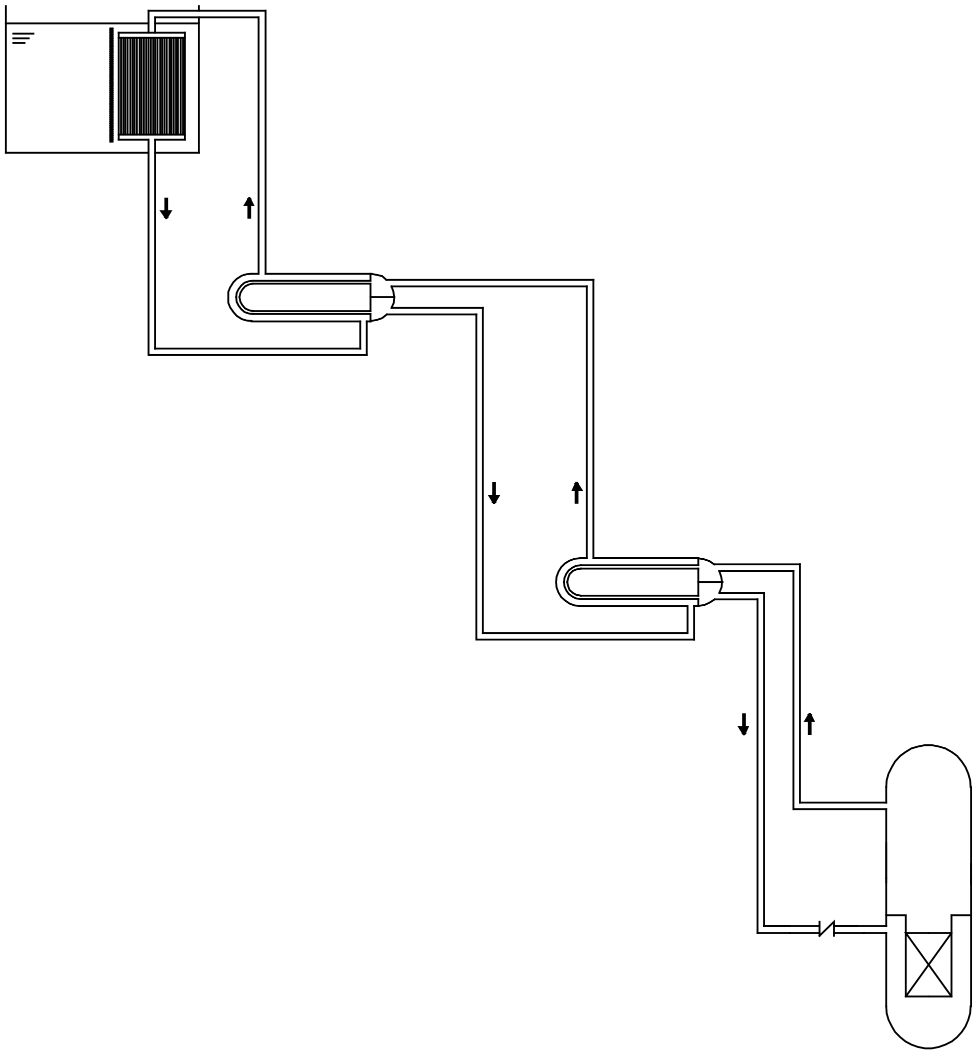

*Fig. 1. Simplified scheme of the upstream loops of the innovative DHRS.*

The rest of the system has, theoretically, no possibility to fail. The particular configuration of the system, and the only safety purpose that it has, allows easy ISI&R operations to further minimize the failure rate.

Since to develop a project, especially if it is innovative, it is necessary to identify all the failure modes and to define the failure rates of each system or component, a system equipped by static components is characterized by less branches in the event tree, that bring to core damage.

In a level 1 Probabilistic Risk Assessment (PSA), the overall CDF is the lowest if static components are used, if passive components are minimized and if active components are avoided.

## Possible solutions proposed: principles of operation

Usually decay heat removal systems use air or water as heat sink. The choice of the heat sink to adopt has to be based on the following criteria:

- The use of water allows to have higher heat transfer coefficient;
- The use of air allows to have unlimited operating time.

These different solutions are used in modern nuclear reactors [2]. For example, the ESBWR is equipped with a passive decay heat removal system releasing heat to a volume of subcooled water at the beginning of the transient and to boiling water later (the isolation condenser). Since no passive systems, aimed at maintaining the water pool inventory are present, a limited operating time of about 72 h is reachable without any external intervention. The VVER-1000, instead, has a passive decay heat removal system based on air heat exchangers releasing heat to the atmospheric air. This system uses the secondary side of steam generators to release heat to the atmosphere and valves to allow the circulation, it is characterized by an unlimited time of operating capability. The use of air as heat sink makes larger the air-cooled heat exchanger since it has to guarantee the residual heat removal from the system just after the reactor shutdown.

The choice of a water heat sink, thanks to the high heat transfer coefficients, is preferable in the presence of natural circulation systems because, for the same heat transfer surface, a lower temperature difference, between the hot fluid and the heat sink, is sufficient to start the system circulation. Lower temperature difference is as important as a cascade of loops in series, between the heat source (reactor core) and the heat sink. Since the heat sink temperature binds the upstream circuits' temperatures, the presence of water in the heat sink allows the system to keep temperatures lower up to the primary coolant.

In order to satisfy the utility requirements [3], the initial water inventory has to be large enough to allow 72 hours of safety operating conditions without any external intervention (neither systems nor operators). After 72 hours, it is foreseen the possibility to refill the water inventory thanks to active systems (pumps) or from outside (fire-fighter supply).

An air heat sink ensures an unlimited operating time without either operator or other system interventions. On the other hand, due to low air heat transfer coefficient, a higher LMTD is required with other conditions being the same. The starting of natural circulation will be smoother; if the safety system is equipped by multiple loops in series, the maximum coolant temperature at the exit of the core will not be limited as in the previous case. This problem can be compensated by an over-sized heat transfer surface that would result in a more expensive heat exchanger and larger space. The heat transfer surface has to be designed to remove the maximum decay heat, unless an additional heat capacity is provided. Since the decay heat trend is decreasing, the highest power is to be removed only during the first transient phases and the heat exchanger will have a heat transfer surface larger than necessary during the quasi-steady state of the transient.

## Description of the new-design, top-reliability heat sink for never ending decay heat removal system

Taking into account experiences gained in the MARS project [4] [5] and in particular concerning the passive DHR system, a new DHR, mainly suitable for small/medium sized nuclear reactors, has been developed and it is here described.

The DHRS of the MARS [6] is characterized by three loops in series where the most important one, due to its innovative features, is the last one. In the first and second circuits there is respectively, primary coolant and intermediate water. The hot water, from the second circuit, passes through a heat exchanger, characterized by horizontal tubes, releasing heat to the water presents within a pool.

The last loop is made up of a pool, a condenser, that allow to remove a large amount of heat thanks to the high latent vaporization heat of water at atmospheric pressure [7], and piping to connect both components. Therefore, the last loop is a closed circuit. During safety operating condition, after an initial transient period due to the reaching of the saturation condition in the pool, water vaporizes first and condenses later, releasing heat to the external atmosphere by means of the same atmospheric condenser which is installed inside a cooling tower (the atmospheric condenser uses air as heat sink; since the loop is closed it limits the loss of steam from the pool inventory to atmosphere and the water amount remains constant). The features of this system allow an almost unlimited operating time because it exploits air as heat sink and water, in closed loop, in all upstream circuits (only the atmospheric condenser has a small communication path between steam and air). Due to the system features, it is possible an accumulation of incondensable gases in the system. Since the presence of incondensable gases strongly affects the heat transfer characteristics of the system, a special system to flush out the gases was developed. For this reason, it is difficult to assess precisely the time length of capability of the system to operate without any external interventions (even if this may be expected as quite high).

Taking into account the above considerations, a new innovative system, characterized by a top-reliable heat sink, has been developed. Hereafter, the description will focus on the innovative heat sink features, while the upstream systems will be described only for the operating condition description. The heat sink is made up of a pool, directly connected to the atmosphere, a heat exchanger placed inside the pool and a special separating septum. Therefore, in the innovative solution, the water contained in the pool at the beginning will not be maintained during the transient phases (the heat sink will shift from subcooled water to air, passing through the pool boiling condition).

During normal operating condition of the plant, this system is in a stand-by condition characterized by: pool volume completely filled, subcooled water, and the fluid (i.e. water) within the heat exchanger tubes, stationary and at the same temperature as that of the water pool. The innovative system uses, as heat sink, water during the first transient phases and air later on, in order to join the advantages of high heat transfer coefficient of water, when heat to remove is greater, and air to have an unlimited period of system operating conditions (see Fig. 2). The operating conditions of the DHRS are characterized by three different stages established by the pool heat sink conditions; during the first stage of the transient, heat removal is guaranteed by the water present in the pool, later the heat sink is provided by air flow only. The change from the water heat sink to the air heat sink mode is not instantaneous: the shift is long enough to allow the complete vaporization of the water initially presents in the pool. The period characterized by a hybrid heat sink, water plus air, hereafter will be defined transition period. The target of the system is to guarantee the heat removal during all the three periods.

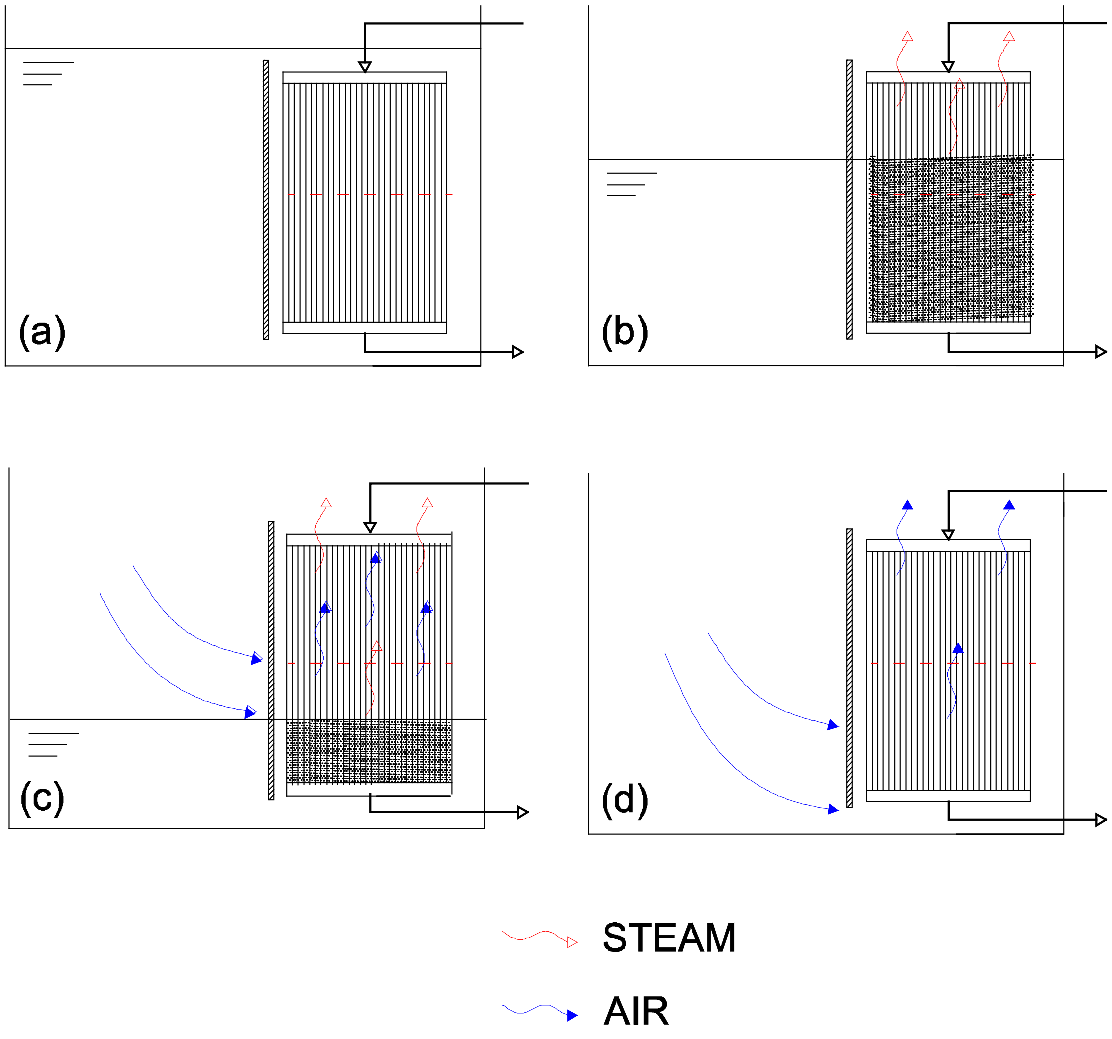


*Fig. 2. Scheme of the heat removal system's operating conditions: a) initial condition and during water heating up to saturation, b) boiling water condition with decreasing of its level, c) mixed heat sink with boiling water and air flow coexistence, d) air only heat sink: never ending operations. The horizontal dotted line is the upper edge of the openings.*

During these periods the heat transfer mechanisms are specific of each phase. In the first phase, when the pool is filled by subcooled water, the water level has to be high enough to completely overcome the heat exchanger tubes and the separating septum. In this condition, the warming water within the heat exchanger tubes will have a lower density compared with that of water presents inside the empty volume of the pool. The density gradient between the two pool volumes allows having a natural circulation through them and the whole heat capacity of the water pool will be exploited. In this way the start of the air only heat sink phase will be delayed. In this phase the heat transfer capability of the system is guaranteed by the natural convection condition on the pool side of the heat exchanger (see Fig. 2-a).

When boiling condition occurs the pool water level decreases. As soon as the free level reaches a height lower than the separating septum the pool boiling heat transfer condition will be reached. In this condition the cold water replacement within the pool volume is guaranteed only by communicating vessels principle rather than natural circulation due to density gradient (see Fig. 2-b). The main contribution to heat removal during the boiling period phase is due to boiling water, but a contribution which increase with decreasing level is due to the steam flow produced from the pool bulk. Once the openings are uncovered starts the transition period, in which the heat sink is made up of steam produced by the vaporization together with the air flowing through the septum openings (see Fig. 2-c). In this condition air flows in natural circulation and the steam flow within the heat exchanger enhanced the contribution of the air flow thanks to the chimney effect. During the transient period the lower zone of the heat transfer surface will be cooled by pool boiling condition while the upper zone by a mixed flow of air and steam. The contribution of the two different components changes during the transient from steam only to air only flow.

During the last transient phase, when air only heat sink phase occurs, the two volumes are filled by air. The empty volume will contain cold air while the heat exchanger volume will contain the hot air. The temperature difference that will establish between the two volumes is high enough to have natural circulation of air within the heat exchanger. In this condition the heat will be removed by convection on the pool side (see Fig. 2-d), as well as during the first transient phase with subcooled water within the pool.

On the tube side, within the heat exchanger, the flowing condition remains the same during the three different phases. The fluid can be flow in natural or forced convection, depending on the design features, but in any case the decay heat is removed by the convection heat transfer mechanism.

To achieve these purposes, the pool volume is divided into two interconnected regions, separated by a special septum. In the first region, the heat exchanger is placed, while the other region does not contain any mechanical component. The innovative components expressly designed for this system are the heat exchanger and the separating septum, both with particular features.

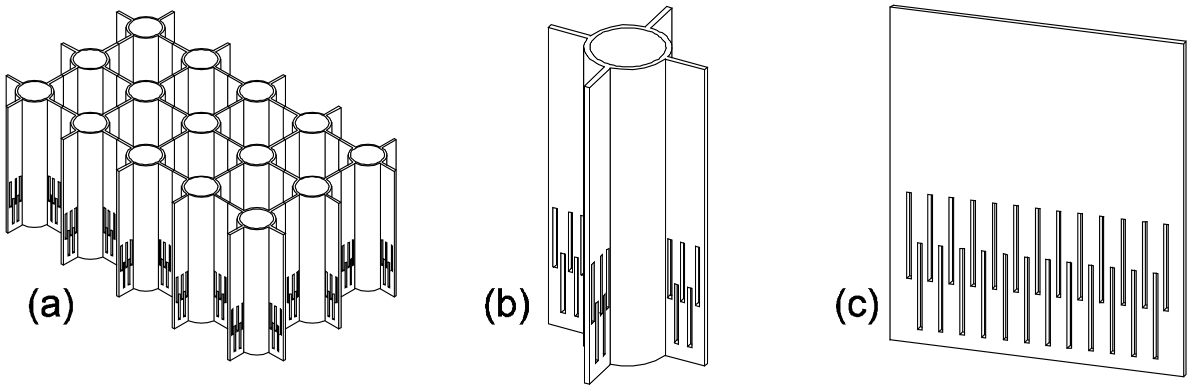


*Fig. 3. Views of main parts of the innovative heat exchanger: a) an isometric view of the heat exchanger, b) zoom of a single finned tube, c) the separating septum with openings.*

The heat exchanger is made up of vertical tubes, arranged in a squared matrix, equipped by four special fins. The fins of adjacent tubes are linked each other constituting about as many squared sub-channels as the tubes number (see Fig. 3-a). Each fin is provided, in the bottom part, by vertical openings, that allow mixing water of adjacent sub-channels first, and air flow during the transition heat removal period (see Fig. 3-b). In this way, the bottom volumes of the pool are in communication, while the upper volumes, upper heat exchanger tubes and upper empty volumes are independent. As for the fins, the separating septum is provided by vertical openings in the bottom region to allow air, from the empty volume, to reach the heat exchanger sub-channels (see Fig. 3-c).

The main features of the system are the high availability and the reliability. Due to its static components, the system does not require intervention during stand-by conditions. The only thing that has to be ensured is a level high enough to allow the pool to contain a water inventory sufficient to delay the passage to the air-only heat sink when the decay heat to remove is reduced to a selected, proper value. During normal plant operation, the level of the pool was not monitored, since the pool is communicating with the external environment, the desired water inventory would not be assured due to evaporation.

## Advantages of the innovative system

Many advantages are achievable with the use of this system, that make it interesting especially for nuclear power plants. Concerning the use of the system in NPPs, it could be easier to foresee this system if the size of the plant has a small/medium size. In this way, when the air-only heat sink operates, a quite limited thermal power (few MW) has to be removed, allowing to foresee a small heat exchanger. Concerning chemical plants, a system of this type can be applied to the plant foreseeing the possibility of a runaway reaction. The main advantages achievable are:

- High capability to decay heat removal during the first period after the reactor shutdown, exploiting the high heat transfer coefficient of water;
- Long term operability without any external intervention, thanks to the automatic shift from the water heat sink to the air heat sink.
- Compact heat exchanger, since the air-only heat sink takes the place of the water heat sink when a quite smaller power has to be removed.
- Possibility to delay the occurrence of the air-only heat sink simply increasing the water inventory initially present in the pool (the empty volume side can be enlarged as desired).

## The RELAP5 model used for the system

The reference system is an evolution of the MARS design; it is called mini-MARS, since it is applied to a plant characterized by a lower power, in addition to an integrated structure, a higher pressure and several innovative concepts; nevertheless, the applicability to plants with quite higher power level is absolutely possible. The DHRS is characterized by three loops; in the first one the primary coolant flows, in the secondary and in the tertiary ones, water flows, allowing the system to release the heat power to the external atmosphere. The presence of the second loop is required according to multi barriers criterion between primary system and external environment.

The simulation of the operating condition was carried out through a simplified model developed ad hoc and using the RELAP5/mod3.3 software [8], for an independent analysis. A simplified scheme of the system and the correspondent RELAP5 representation are shown in Fig. 4.

The RELAP5 simulation was carried out in two different steps. The first period is characterized by the heating of the pool water from initial temperature to saturation condition and by the period required to have a vaporization of water large enough to have pool level lower than the openings. The second period is characterized by the air-only heat sink. In this way, a conservative assumption was applied: in fact, the transition period in which both boiling water and air constitute the heat sink, are represented by an air-only heat sink, in which performance results strongly lower than in reality.

Due to this assumption, when the change of model occurs, trends of temperatures and mass flow, as well as the thermal heat removed by the pool, are characterized by visible discontinuities. Thanks to the special geometry of the system, due to the vertical openings in the structure, when the air contribution starts to be present, from the time when the water level drops below the top level of the openings, a high contribution to heat removal from the boiling water is still present. The geometry prevents the occurrence of flooding conditions, in which boiling water prevents air to enter into the heat exchanger volume. In this way, the assumption made is very conservative but totally coherent to show the effectiveness of the heat sink system. In Fig. 4, the models used for the water and air heat sinks are shown; the model of the water-only heat sink is represented in a conventional way, while the model in which an air flow is present required some other components represented by dotted lines. Concerning the RELAP5 model of the innovative heat exchanger, components of 300 and 400 series are the most important since they are respectively the tube and the pool sides of the heat exchanger. During the first transient phase, only the pool and the outlet atmosphere are represented while the heat sink is composed by subcooled water. The outlet atmosphere is necessary to allows steam flowing outside the pool volume after the saturation condition is reached. Conservative assumption was made during the first transient period, since openings and connections, between the two pool volumes, were neglected during the subcooled period. This assumption implies a pool condition and natural circulation between the two different volumes of the pool is neglected; in this condition the total heat capacity of the pool and the heat transfer coefficient on the pool side are underestimate.

Once the saturation condition is reached the produced steam is released in the outlet atmosphere, represented by a volume where pressure and temperature as boundary conditions are imposed. Change of the nodalization occurs when the pool level is lower enough to uncover the openings; in this condition the residual boiling water will be neglected and an air only heat sink will be simulated. In order to guarantee the natural circulation condition for the air flow, since the pool condition heat transfer with air would result excessively conservative, some other junctions and volumes are added. In particularly, beyond the outlet atmosphere also the inlet atmosphere is represented, by the 440 volume, with the same boundary conditions of the outlet one. Since cold air and hot air have to be represented in two different volumes to allow natural circulation simulation, the downcomer (volume 430) is added for the pool air flow. In the air only heat sink condition, the addition of the volume 430 causes an increasing in the pool volume that is not strictly connected with the real geometry. This difference between the reality and the model does not cause errors since, when the total heat capacity of the system is one of the main parameters (pool in subcooled condition), the pool volume is equal to the real geometry and when natural circulation occurs (during the air only heat sink condition), highs and pressure drop are equal to the real geometry since they are the main parameters.

During the air only heat sink, the mean heat transfer coefficient on the pool side was evaluated before and imposed in the RELAP5 input. The maximum uncertainty of the heat transfer coefficient could be up to 30%. Anyway, the instantaneously neglecting of the residual boiling water within the pool volume allows having large margins of conservatism in the system heat transfer capability.

The RELAP5 model description is limited to the atmospheric heat sink and to the corresponding loop since the study is aimed at focusing the heat removal capability of the DHRS. Even though the detailed RELAP5 model of the whole circuits is not described, it is shown in Fig. 4 where the upstream loop, with its own heat exchanger, which ensures a physical separation between the outer loop and the primary coolant, is visible.

The heat removal, both with water and with air heat sinks, occurs thanks to natural laws since water and air employ the density difference between the cold column of fluid (inlet condition) and the hot column of fluid (outlet condition) as driving force.

The operating pressure of the loops will be defined, in each application, so that the maximum temperature reached is, in any case, lower than the corresponding saturation temperature.

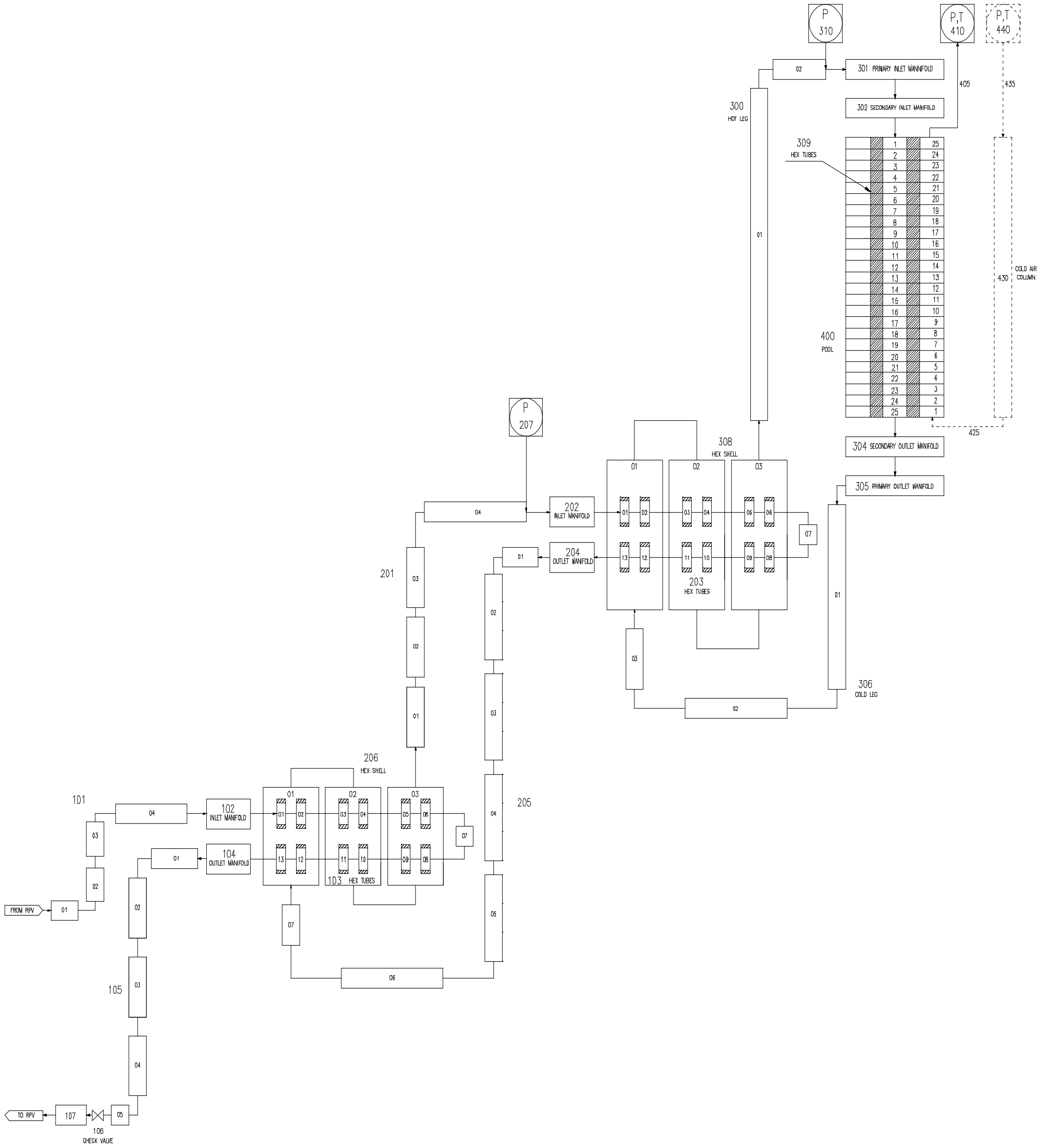


*Fig. 4. RELAP5 model of the innovative DHRS.*

The main features of interest in the operating simulation are listed in Tab. 1.

*Tab. 1. Main features of the system used for the simulation*

| | Value | Unit of measure |
|---|---|---|
| $l_t$ | 5 | m |
| $n_t$ | 1800 | - |
| $p_{at}$ | 0.101 | MPa |
| $\dot{Q}_0$ | 100 | MW |
| $t_{0a}$ | 20 | °C |
| $t_{0w}$ | 30 | °C |
| $V_{0w}$ | 40 | $m^3$ |

## Example of operation of the innovative system, with reference to a SMR

In this section, an example of the operating conditions of the innovative system is described. The reference conditions are Loss of Offsite and Onsite Power where, due to the characteristics of the system, the intervention of safety systems has been assumed in accordance with the following time scale.

*Tab. 2. Scheduled actions during simulation*

| Time, s | Action |
|---|---|
| 0 | ▪ Steady state. |
| 10 | ▪ Primary pumps disconnection;<br>▪ Start closuring of insulating valves of feed water and steam line of secondary circuit. |
| 11 | ▪ SCRAM;<br>▪ End closure of insulating valves, of feed water and steam line of secondary circuit. |
| 11.85 | ▪ Start opening of DHRS passive check valve. |
| ~ 56000 | ▪ Resolution model is changed since pool level reaches openings upper edge (from boiling water heat sink to air-only heat sink). |

Simulation of the safety operating condition is carried out for 72 hours after the reactor trip in order to satisfy the utility requirements. The time value when the change of the heat sink model is applied was identified through an appropriate control variable defined according to the geometric characteristics of the system (level from which fins and septum are provided with openings).

Following pictures (from Fig. 5 to Fig. 11) show the behavior of the DHRS, regarding parameters of interest.

The mass flow and temperatures of the ultimate loop of the DHRS (Fig. 5 and Fig. 6) show two different behaviors before and after the time value of 56000 s, at which the model change occur. During the first period mass flow rate presents a decreasing trend due to the reduction of the decay heat produced from the core. After the change of model due to the lower air heat transfer capability temperatures in the loop start to rise in order to have a larger temperature difference between external air and intermediate water, this allows system to continue the heat removal process.

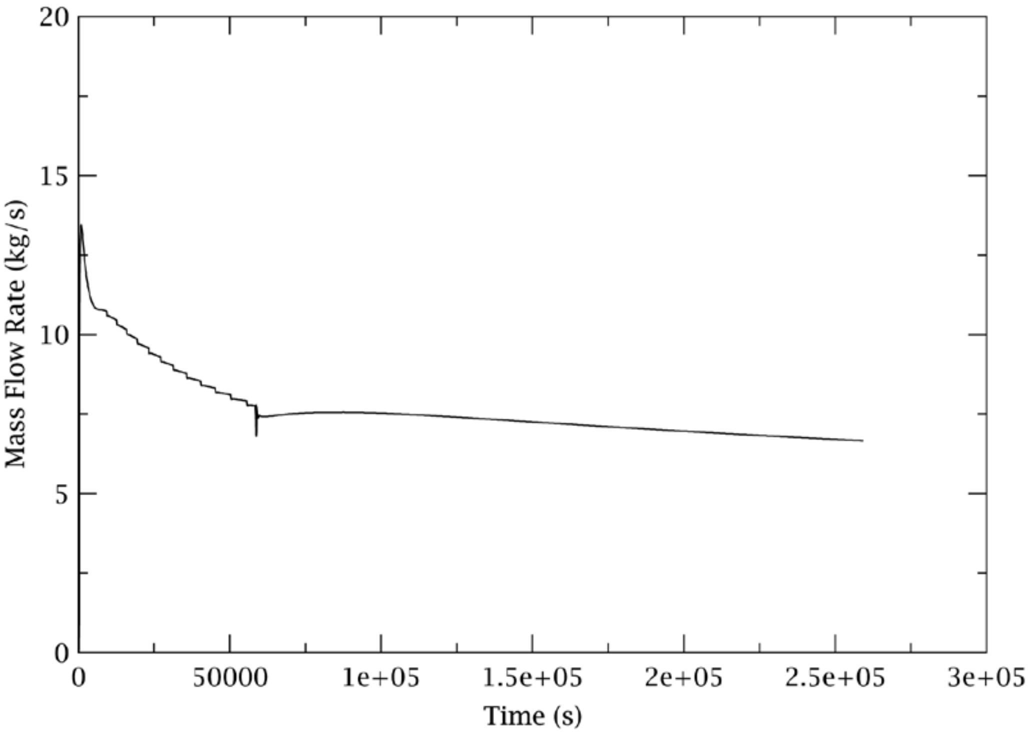


*Fig. 5. Mass flow trend in the last loop.*

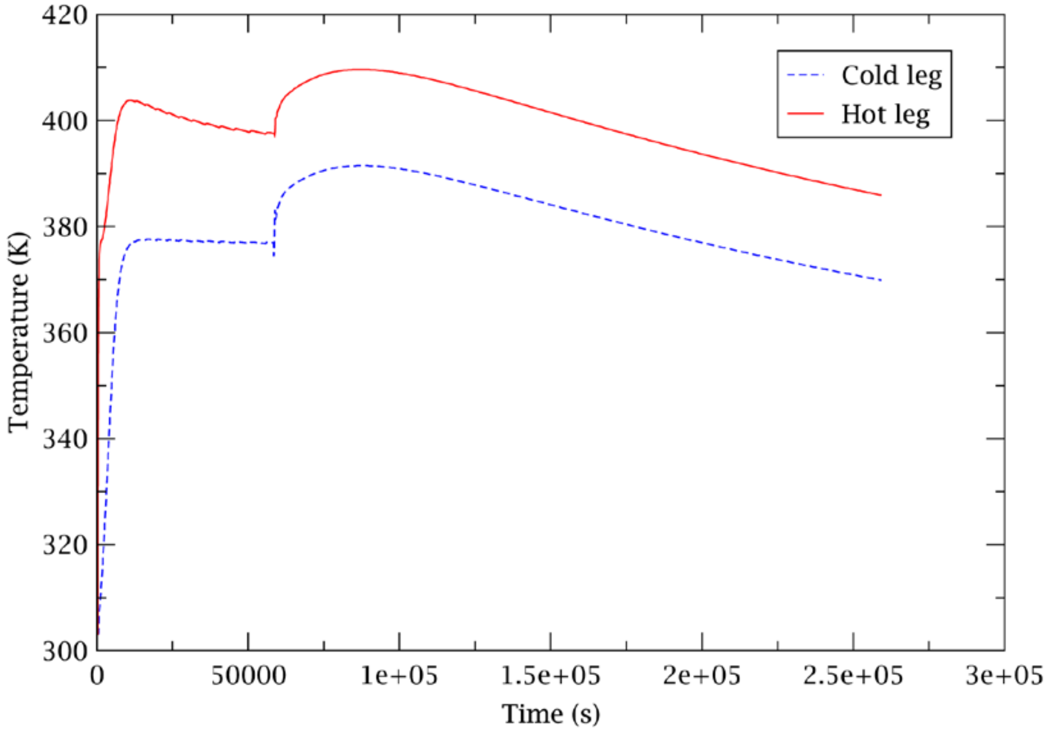


*Fig. 6. Cold (dotted line) and hot (solid line) legs temperatures in the last loop.*

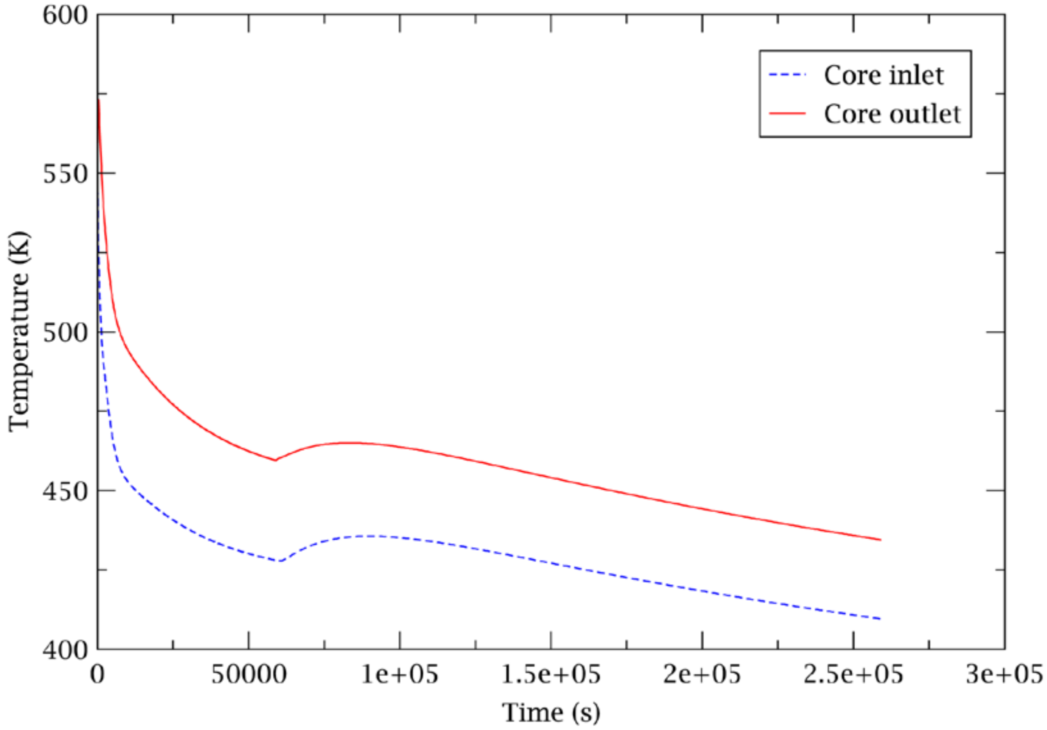


*Fig. 7. Primary coolant temperatures, at inlet (dotted line) and at outlet (solid line) of the core*

What happen inside the pressure vessel is very important, since the maximum coolant temperatures are reached at the core outlet. As visible (see Fig. 7), except for the first phase of the transient, primary coolant temperatures have been decreased; the time value, at which an increasing is reported, corresponds with the change of the heat sink.

Performance of the DHRS is evaluable by the comparison between decay heat produced in the core and removed heat from the heat sink. In order to evaluate the decay heat produced in the core, in the RELAP5 model the standard correlation ANS 5.1-1973 is used. Due to a delay requested for the starting of the DHRS, during the first period the decay heat is higher than the removed heat. This conduct to an increasing of the primary coolant temperatures, afterwards the released heat becomes higher and reactor has led to the cold shutdown condition.

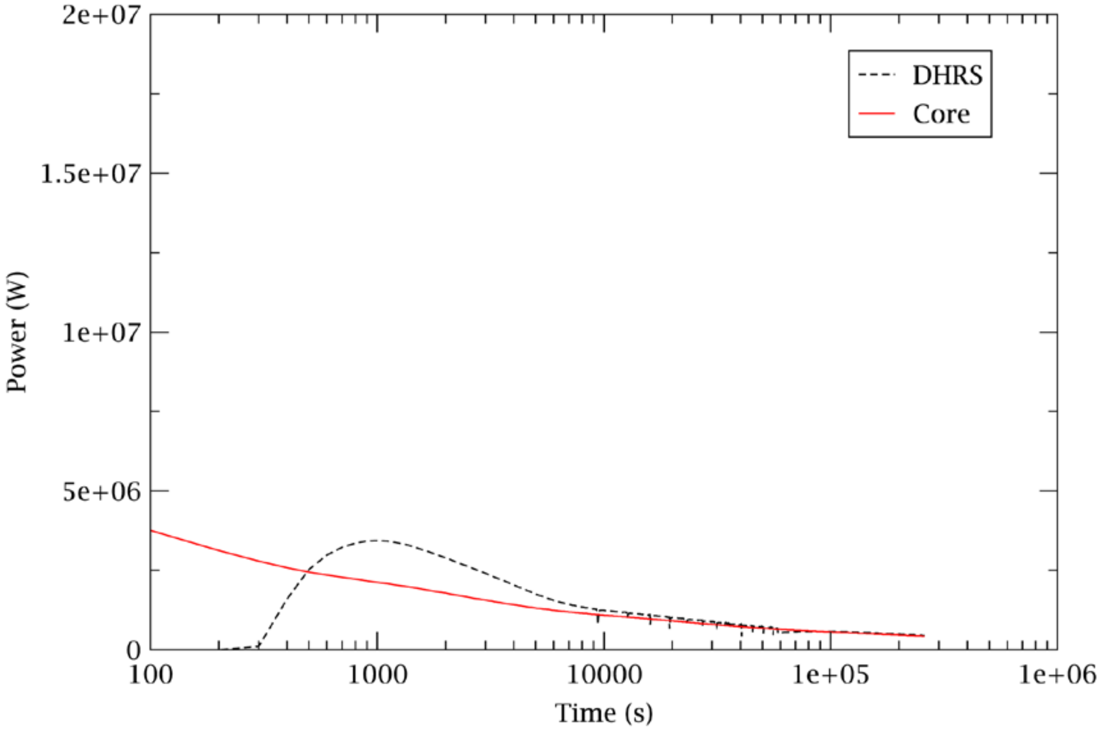


*Fig. 8. Decay heat power: Thermal power produced into the core (solid line) and removed by the innovative heat sink (dotted line).*

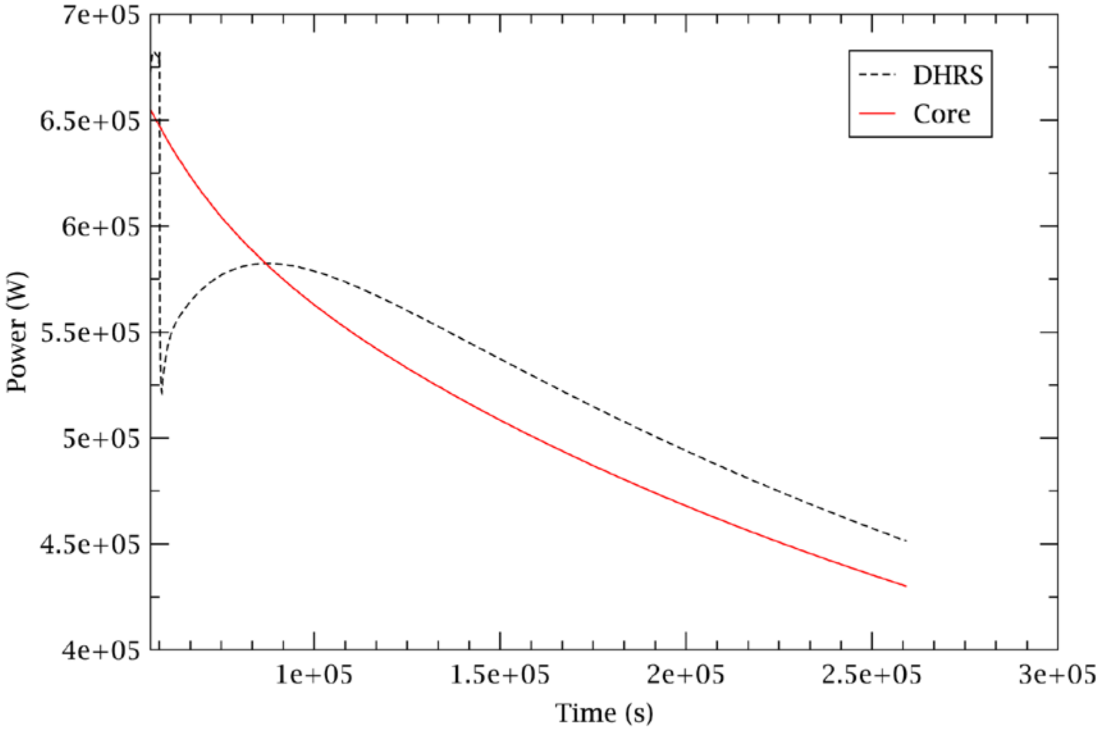


*Fig. 9. Decay heat power: Thermal power produced into the core (solid line) and removed by the innovative pool (dotted line) during the air only heat sink period.*

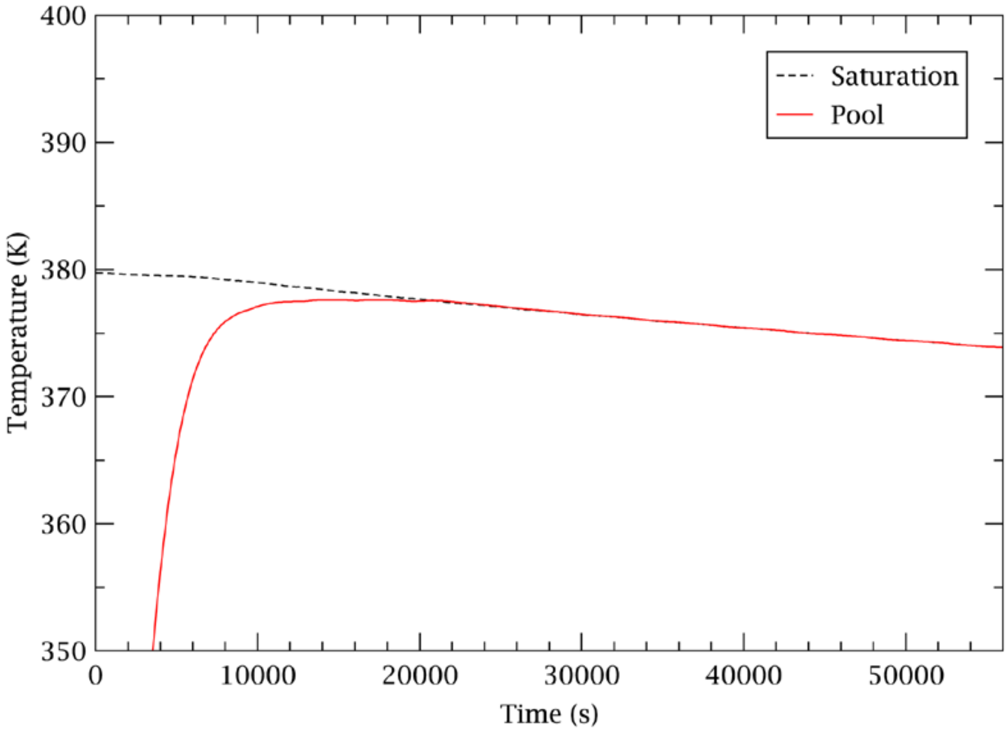


*Fig. 10. Innovative pool behavior: Mean temperature trend (solid line) within a middle pool volume compared with the saturation temperature (dotted line).*

Concerning the pool, the behavior of a middle height volume is considered for the temperature (Fig. 10). During operating condition pool water starts to rise its temperature reducing its density. The reduction in density causes a decrease in the saturation temperature since the static pressure due to the fluid column above is reduced. The water in the pool increases its temperature from starting condition to saturation. From this moment, the vaporization of the boiling water from the upper volumes, saturation temperatures and water level start to decrease (Fig. 11). Simulation of this first period was stopped when the openings started to be uncovered to allow changing the heat sink model.

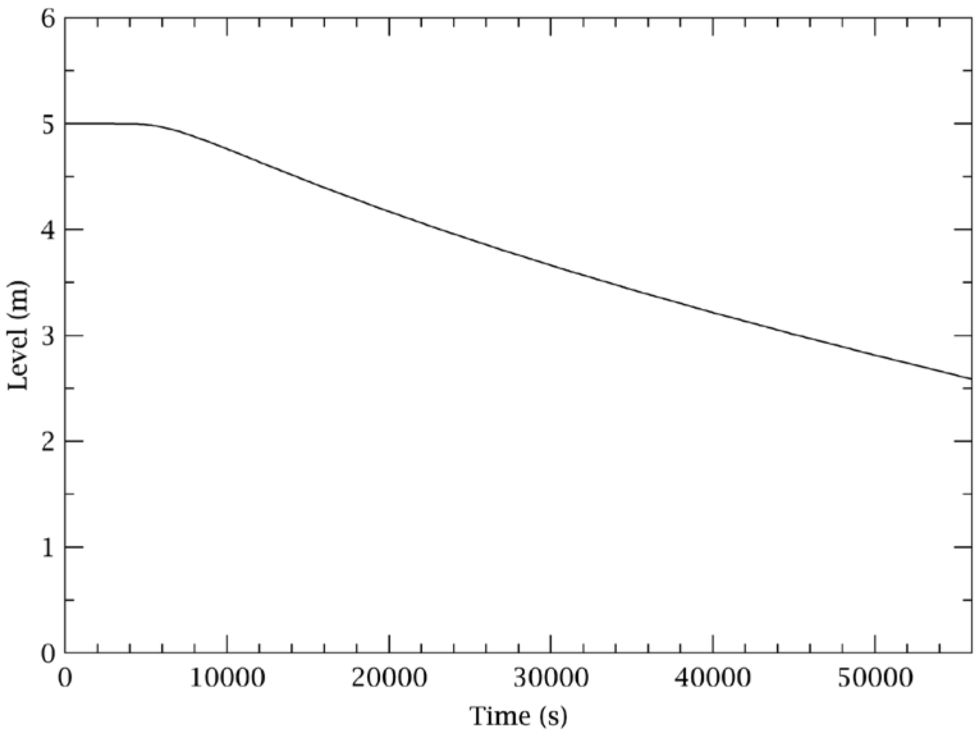


*Fig. 11. Innovative pool behavior: trend of the pool water level.*

## Conclusions

Due to safety requirements for NPPs, the availability of safety systems, as well as their reliability, is of primary importance. The innovative DHRS presented in this paper, shows some interesting aspects that allow minimizing the heat sink system dimensions, guaranteeing a high capability in heat removal when the core decay heat power is still high, thanks to the presence of water, and at the same time, a never ending heat sink thanks to the air.

Because of the conservative assumptions applied, the real trend of main parameters during the transition phase would be smoother than that with the change of air-only heat sink model. The evaluated values of process parameters are representative of the real phenomenon only before the transition phase and after the complete vaporization of the pool water.

The requirement of 72 hours without any external interventions can be easily achieved with this system. The features of the system allow designers to select the intervention delay, of the air only heat sink condition, to any value, simply changing the pool dimensions and the height of opening's upper edge. These features make the innovative system particularly attractive for small/medium size nuclear reactors. Since, after a limited period of time, it is sufficient to remove a thermal power of about 1% of the rated power, in a small/medium reactor, this value results low enough to be completely removed through an air-only heat sink, keeping the heat exchanger with very compact dimensions.

In standard DHRSs used in operating power plants, after a certain period of time it is necessary to refill the pool, used as heat sink, with water from outside in order to maintain the inventory. In special conditions, in which the power plant zone could be affected by flood or other types of events, it can be very difficult, or at worst not possible, to recover the inventory of water used as heat sink before it finishes. In these conditions benefits ensured by the innovative system here proposed, in term of safety, result very clear since it does not require any refill operation of the water inventory. Unlike other passive systems used in operating nuclear power plants, the DHR here proposed allows to simultaneously obtain an unlimited period of operation without any external intervention of operators and a heat exchanger small enough to be considered cost effective.

# Nomenclature

## Acronyms

| | |
|---|---|
| CDF | Core Damage Frequency |
| DHRS | Decay Heat Removal System |
| ISI&R | In-Service Inspection & Repair |
| LMTD | Log Mean Temperature Difference |

NPP Nuclear Power Plant

PSA Probabilistic Risk Assessment

## Symbols

$\dot{Q}$ thermal power, *MW*

*n* number

*l* height, *m*

*V* volume, $m^3$

*t* temperature, *°C*

*p* pressure, *MPa*

## Subscripts and superscripts

0 initial condition

a air

at atmosphere (inlet and outlet)

t heat exchanger tubes

w water

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## 4.5 The innovative pool and the Fukushima accident

The new system designed can provide special safety performance to the nuclear reactor itself. Especially after the Fukushima accident, the importance of very reliable and highly available DHRSs has been shown to be fundamental for the system safety and the management of any accidental condition. In the Fukushima NPP the most important problem was to guarantee the decay heat removal both from the reactor cores and from the spent fuel pools. Since the Emergency Core Cooling Systems (ECCS) were completely based on active components, pumps and emergency diesel generators were required to remove the decay heat. The tsunami flooded the emergency diesels and moved away the fuel tanks preventing the functioning of any available source of power needed for the pumps' operation. Should the tsunami had not flooded the diesels and moved away the fuel tanks, the condition of the site inaccessibility would probably have caused issues related to provide new fuel on the site to feed the emergency diesels. Some pictures of the Fukushima site conditions after the seismic and the tsunami events are reported below (Figure 16, Figure 17, Figure 18 and Figure 19). It is clearly visible that the damages and debris caused by these events make it harder any possibility of external intervention on the emergency safety systems. In this context, the characteristic advantages of the innovative pool heat sink here proposed are even more interesting. The standard design solutions and the Utility Requirements [7] foresee a system independence of at least 72 hours without any external intervention. The possibility of having a completely passive system, without any pump and power sources, able to operate for an unlimited period of time, makes this system especially suitable for the decay heat removal of nuclear power plants in emergency conditions. The analysis described in the previous paragraph (see 4.4) is referred to a SMR; since the Fukushima units were larger, the pool should have had larger dimensions to guarantee a sufficient delay in the air-only heat sink occurrence. This also implies special design issues to place the pool heat sinks accordingly to the plant layout. The design issues include the optimization between the maximization of the difference of height between the heat source and the heat sink and the minimization of the loop piping lengths.

**Figure 16. Near the Fukushima Power Plant**

**Figure 17. Street in the Fukushima Dai-ichi site**

**Figure 18. Street in the Fukushima Dai-ichi site**

**Figure 19. Debris after the tsunami in the Fukushima Dai-ichi site**

Due to safety requirements for NPPs, the availability of safety systems, as well as their reliability, is of primary importance. The innovative DHRS studied shows many interesting aspects that allow reducing the heat sink system dimensions, optimizing the pool dimensions to delay enough the pool boiling condition occurrence and hence the air-only heat sink. The adoption of this special system guarantees a high capability in heat removal when the decay heat to remove is still high, thanks to the sub-cooled pool water, and at the same time, a never ending heat sink exploiting the atmospheric air when the decay heat to be removed is lower. During normal operating conditions, some operative issues related to the pool water should be taken into account. The main parameters that can affect the system performance are related to the surrounding environmental conditions (e.g. temperature, humidity, etc.). Remote areas where SMRs are especially suitable to be installed can be characterized by extreme environmental conditions as very low or very high temperatures. Some solutions have been also hypothesized to comply with the sites' special requirements:

- High temperature environment:

  The loops that are foreseen upstream to the PHX are completely closed and the coolant circulation is prevented by dedicated check valves. In this solution the pool can be designed to have a volume large enough to guarantee a certain evaporation rate. This solution foresees a lay-out where the pool should be characterized by a water level higher than the required. A well-defined level variation, due to the evaporation rate, has to be allowed (Figure 20) and compensated each time the level decreases under a preselected value (minimum level that allows the system to guarantee the required safety function). In any case, also when the pool reaches its minimum level it has to guarantee a water inventory higher than the required. A second possible solution is to increase the water inventory, coupling the standard pool with an additional volume that has to be foreseen underground (Figure 21). This solution allows to reduce the surface-volume ratio and hence the evaporation rate.

- Low temperature environment:

The passive check valve, completely closed during the normal operating conditions, can be coupled with a small by-pass equipped with a control valve. In this solution, thanks to an active system, the by-pass flow rate can be easily controlled in order to avoid the freezing of the pool (Figure 22). ). In this condition, the solution leads to thermal losses that will reduce the overall efficiency, even if this reduction is quite limited and results to be much lower than one per cent.

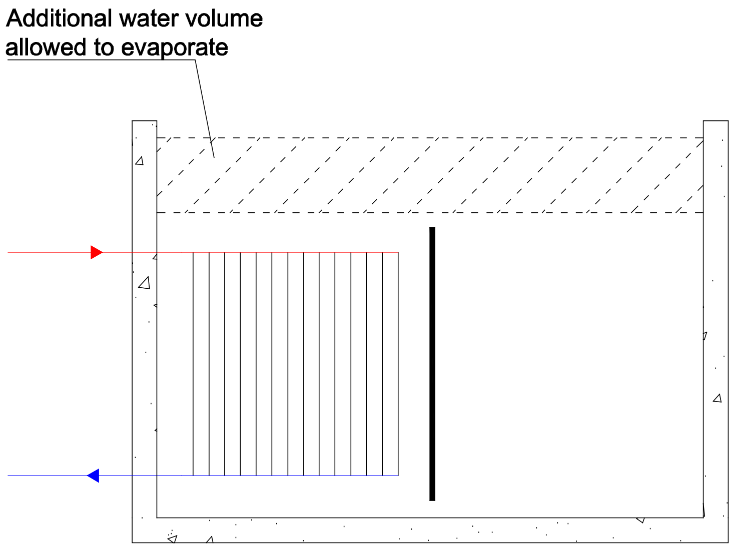


**Figure 20. Design solution allowing a certain evaporation rate**

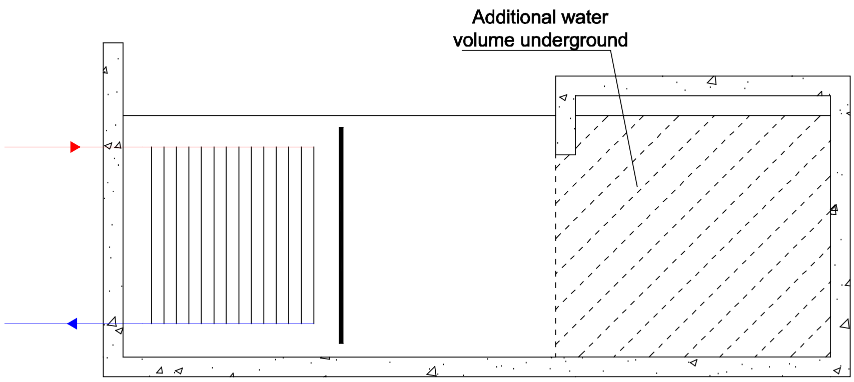


**Figure 21. Design solution with an underground additional water inventory**

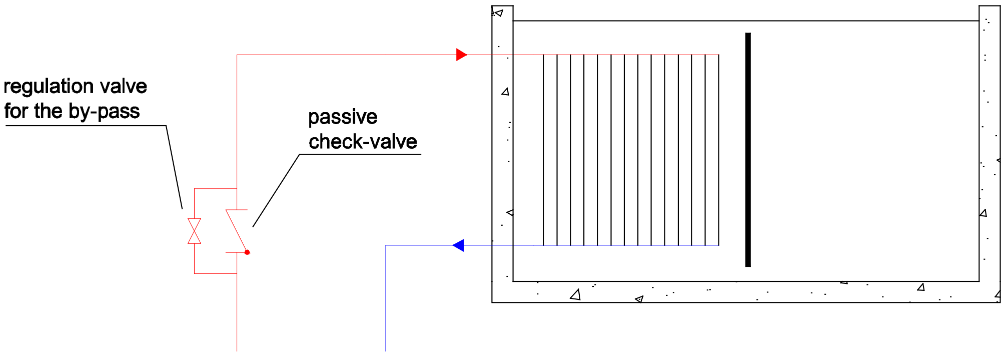


**Figure 22. By-pass solution to maintain the water inventory in liquid state**

The requirements of 72 hours without any external intervention can be easily achieved with this system; the 72 hours may be reduced to any value. The features and flexibility of the system allow designers to select the air-only heat sink delay occurrence at any desirable value. They make the innovative system especially attractive for applications on small/medium sized nuclear reactors. After a limited period of time it is sufficient to remove a decay heat of about 1% of the reactor rated power[15]. In a small/medium reactor, this value results low enough to be completely removed through an air-only heat sink, designing the heat exchanger with very limited dimensions. Such a system can be applied also on chemical power plants where, if a thermal runaway reaction occurs, it can be passively controlled removing the heat and the plant can be brought again in a safe condition.

The overall main advantages achievable with this system are:

- A high heat capacity due to the large water inventory and a good heat transfer coefficient when the pool is filled with subcooled water;
- Excellent decay heat removal capability after the first transient period following the reactor shutdown; the very high heat transfer coefficient of pool boiling water limits the upstream temperatures and allows a good heat removal;
- Long term operability without any external intervention, thanks to the automatic shift from the water heat sink to the air-only heat sink, passing through a mixed solution that includes air and boiling water at the same time;
- Compact heat exchanger, because the air-only heat sink takes the place of the water heat sink when a very limited power has to be removed (the power percentage to be removed by the air-only heat sink can be easily defined by the designer but some hundreds kW should be a good compromise with limited rated power of the NPP);
- Possibility to delay the occurrence of the air-only heat sink simply increasing the water inventory initially present in the pool (the empty volume side can be enlarged as desired).

[15] In a standard PWR a decay heat power equal to the 1% of the design thermal power is reached between 2 and 5 hours after the shutdown.

## 4.6 Patent application

The several advantages so far analyzed and the innovation characterizing this solution led to apply it for a patent as an “industrial invention”.

At the present time, systems aimed at passively removing the decay heat are generally based on water, but only a limited amount of it would be available, or on air, but very large sized heat exchangers would be required. Moreover, the passive check valve, responsible to passively activate the system, has not been studied and a solution as the MARS one was hypothesized, but it is beyond the present invention (see paragraph 4.1).

The possibility to extend the patent of  the system in some specific country is under consideration, and further development of the system can be considered in order to apply it to different specific cases.

Presently, the patent application has been submitted to the Italian patent office and then also to the World Intellectual Property Organization (WIPO) to have the possibility of publishing it also in other countries. The first page of the international patent application (n° WO 2012/063209 A2) extracted from the original document is reported in the Figure 23.

**(12) INTERNATIONAL APPLICATION PUBLISHED UNDER THE PATENT COOPERATION TREATY (PCT)**

**(19) World Intellectual Property Organization**
International Bureau

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**18 May 2012 (18.05.2012)**

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**Published:**

— *without international search report and to be republished upon receipt of that report (Rule 48.2(g))*

**(54) Title**: AIR AND WATER REFRIGERATION STATIC SYSTEM, INHERENTLY RELIABLE, WITH UNLIMITED FUNCTIONING, FOR HEAT REMOVAL FROM PROCESSES, PARTICULARLY FOR SAFETY PURPOSES

**(57) Abstract**: The instant invention relates to a refrigeration apparatus aimed at the removal of heat produced by a process, being based solely and exclusively on inescapable natural laws (water vaporization due to heat transfer, liquid circulation due to density gradient).

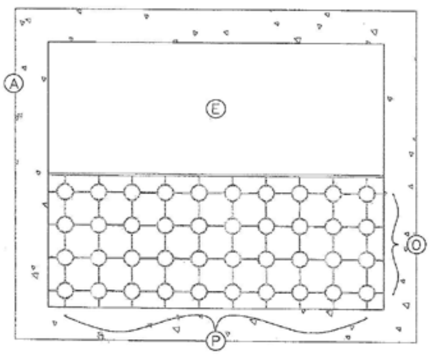


*FIG.1*

**WO 2012/063209 A2**

**Figure 23. First page of the PCT patent application**

# 5 SAFETY SYSTEMS APPLICABLE TO DIFFERENT NUCLEAR REACTORS TYPES (SFRS AND LFRS)

In the present chapter, two innovative components especially suitable for liquid metal cooled reactors are presented and their innovative features are also analyzed. Both systems are aimed at removing heat from the primary coolant, but the former system is aimed at the steam production for the thermal cycle, while the latter is especially aimed at removing the decay heat. Both of them are characterized by very improved safety features and they are especially aimed at avoiding any contact between the primary coolant (i.e. sodium, lead or lead-bismuth eutectic) and the heat transfer fluid used to connect the system to the final heat sink (i.e. water or diathermic oil). The system whose target is the steam production is called CANDLE (see paragraph 5.1), while the system aimed at the decay heat removal form Liquid Metal cooled Fast Reactors is called radiation-based DHX (see paragraph 5.3).

## 5.1 Paper on the Candle concept

Journal: International Journal of Risk Theory

Year: 2011

Volume: 1

Pages: 23 - 44

Title: A Proposal for Simplification and Cost Reduction of SFRs

Authors: A. Naviglio, D. Vitale Di Maio, F. Giannetti, G. Caruso, G. D’Amico

# Abstract:

The present work is aimed at describing an innovative plant solution able to greatly simplify and to make a fast neutron Sodium cooled nuclear reactor plant cheaper, without impairing safety. An innovative system has been developed, aimed at eliminating the intermediate sodium loop. It is based on a special heat exchanger that “joins” the intermediate heat exchanger, typical SFRs component, to the steam generator. The presence of an intermediate loop in Sodium Fast Reactors (SFR) is due to the high reactivity features of sodium with air or water. Thanks to this circuit a reaction between primary sodium (radioactive) and water/steam, within the energy conversion system, is prevented and, at the same time, the following presence of reaction product in the primary circuit, with the safety implication it would have, is avoided. Moreover, the plant solution proposed allows to maintain a physical double separation between primary sodium and water/steam, but avoiding any moving component within the fluid between primary sodium and water/steam.


The research leading to these results has received funding from the European Community’s Seventh R&TD Framework Programme (FP7/2007-2011) under grant agreement n° 232658.

# Introduction on Sodium Fast Reactors (SFRs)

## Key aspects of SFRs relevant for the new design

### Selection of sodium in Nuclear Power Plants

The use of Sodium in a Fast Neutron Nuclear Power Plant (NPP) is linked with several aspects that have to be considered. Sodium is characterized by special features that make it very attractive to be used for removal of thermal power produced within the core of a nuclear reactor:

- An excellent thermal conductivity of liquid sodium implies a very high heat transfer capability even at modest liquid velocities (high power density in fuel is possible);
- A high volumetric expansion coefficient of liquid sodium, that is useful to reach the condition of natural circulation of the coolant with modest temperature difference (coolability of fuel is easier to be achieved in accidental conditions);
- A high boiling temperature at low pressure allows a high energy conversion efficiency at quite low operation pressures.

At the same time, sodium physical-chemical characteristics have also negative effects:

- An excellent thermal conductivity has thermal-mechanical implications in fast temperature variations (thermal shock risks);
- A limited heat capacity implies higher temperature difference between inlet and outlet for given power and flow;

- A melting point at operating pressure (near atmospheric) higher than the room temperature requires a special heating system in order to avoid coolant freezing;
- A high chemical reactivity of sodium with air and water strongly affects safety and therefore has several implications on plant system and component design, as well as on plant operation and maintenance activities and costs.

## SFRs schemes

Solutions adopted in design of nuclear reactor systems using sodium as primary coolant, have always been aimed at preventing any possible contact between the primary coolant (active sodium) and the energy conversion fluid (water), as well as between active sodium and the environment. Usually, three independent circuits have been used to reach this purpose: the active primary coolant flows within the first loop including the nuclear reactor and an intermediate heat exchanger (IHX); the secondary coolant (sodium) flows within an intermediate loop including the secondary side of the IHX and the primary side of steam generators (SG) where the working fluid (water) vaporizes and is super-heated before reaching the steam turbine.

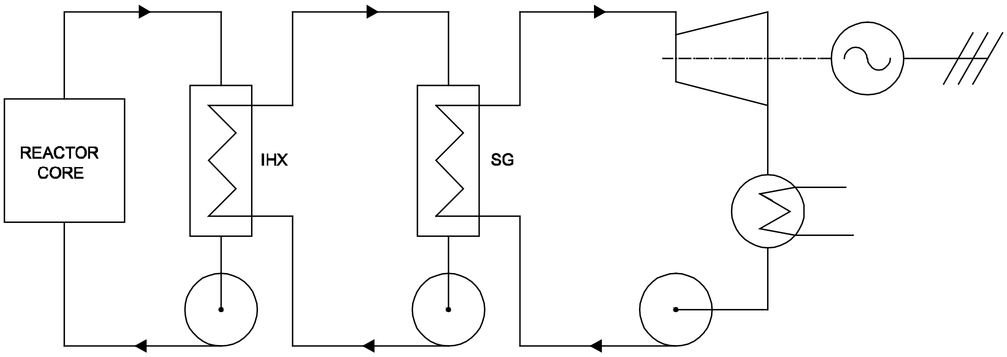


*Fig. 1. Scheme of a classical solution for SFRs*

The intermediate loop is necessary to prevent a possible interaction between the primary active sodium and high pressure water/steam in case of a separating barrier failure: sodium-water reaction products and water itself would enter into the primary coolant boundary, destructive phenomena could occur owing to the high energy generation due to water/sodium reaction, with possible release of radioactivity in the reactor building. Components of the primary circuit are generally arranged according to two different schemes:

- loop solution, characterized by:
    - limited primary vessel size and primary sodium amount in it;
    - pumps and intermediate heat exchanger easy to maintain;
    - long piping extension and large number of connections;
    - on the whole, it is not a compact solution.
- pool solution:
    - compactness of the whole system;
    - primary sodium presence only within the primary tank;
    - system easy to fill and drain;
    - higher thermal inertia due to the larger amount of primary sodium in the reactor tank;
    - particles within sodium may be deposited in low speed zones.

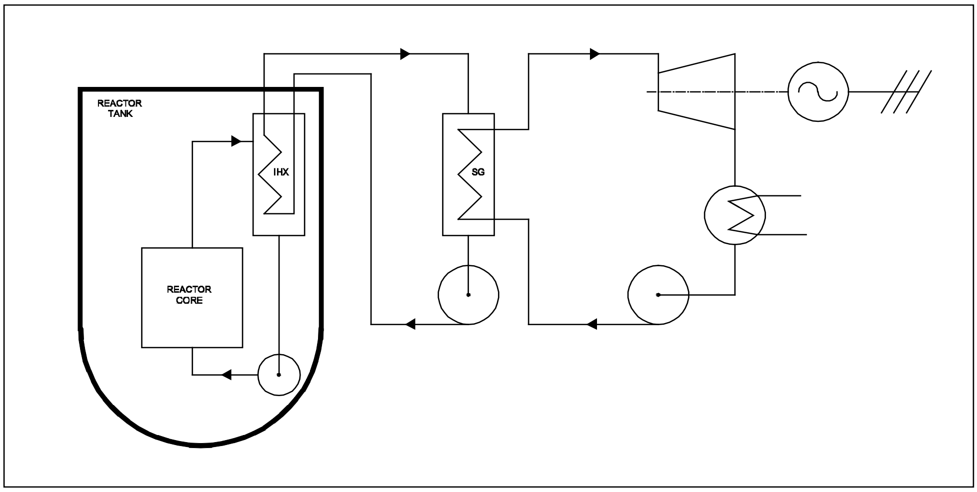


*Fig. 2. Scheme of a pool type Sodium Fast Reactor*

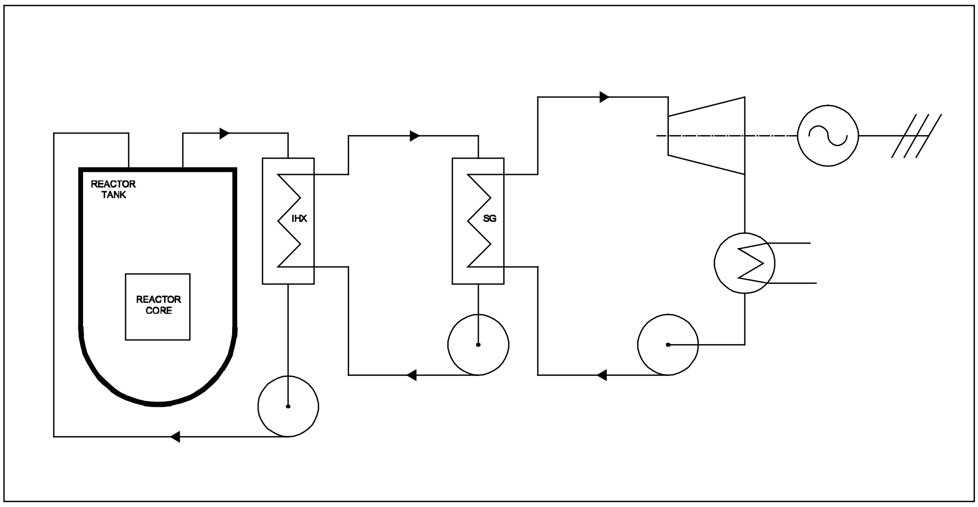


*Fig. 3. Scheme of a loop type Sodium Fast Reactor*

**Schemes of three typical SFRs projects (Superphénix, Monju, BN 600)**

Different countries have chosen different solutions and reactors of both types have been built. Three reactors are briefly analyzed below and their main characteristics are reported in the table.

*Tab. 1. Characteristics of three SFRs*

| Reactor (country) | Reactor power $MW_{th}$ (MWe) | First criticality | Final shut-down | Operational period (years) |
|---|---|---|---|---|
| Superphénix (France) | 3000 (1240) | 1985 | 1997 | 12 |
| BN-600 (Russia) | 1470 (600) | 1980 | (2020) | 31 |
| MONJU (Japan) | 714 (280) | 1994 | - | 17 |

**Superphénix**

The Superphénix NPP was based on a reactor that has been operating in France up to 1997. It was characterized by a pool configuration and each circuit (primary, secondary and tertiary) was made up of 4 parallel loops. The whole system included 4 SGs, 4 primary pumps and 8 IHXs. The primary sodium (about 2500 tons) flowed through the core between 395 °C and 545 °C. The primary sodium flowed within the shell side of the IHX where it released heat to the secondary sodium. Thanks to once-through SGs, steam at 487°C and 177 bar was produced.

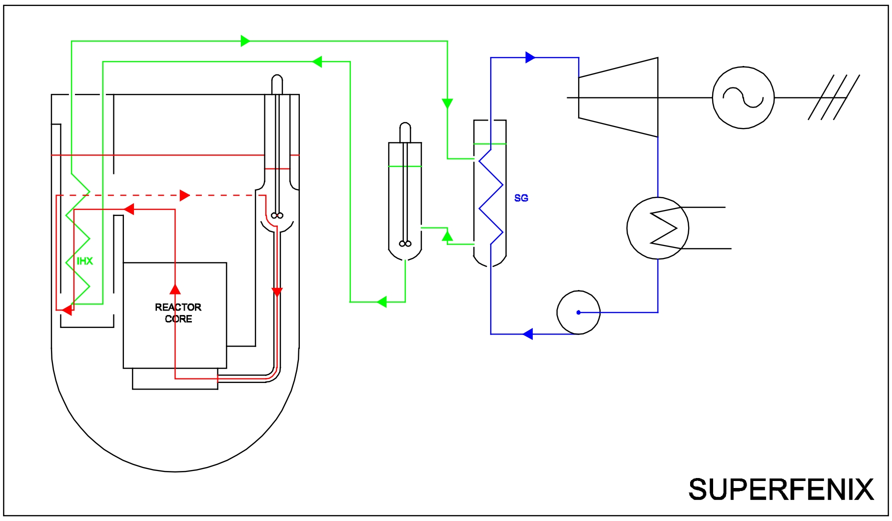


*Fig. 4. Scheme of the Superphénix reactor*

**BN-600**

The BN-600 reactor is a pool type, it was built in Russia, and it is still operative. Each circuit of the system (primary, secondary and tertiary) is characterized by 3 parallel loops. Primary sodium flows through the core, passing from 365 °C to 535 °C before reaching the IHX where the secondary sodium reaches 510 °C and produces steam at 500 °C and 132 bar.

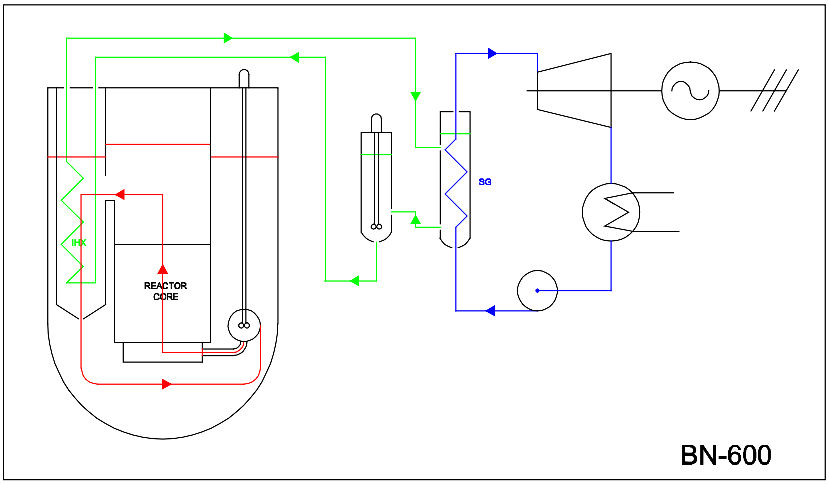


*Fig. 5. Scheme of the BN-600 reactor*

### Monju

Unlike the previously described solutions, the Monju reactor is a loop type, choice mainly due to seismic considerations, for easier accessibility of the components and to use the experience gained with the experimental reactor Joyo. The reactor was built in Japan and is still in operation.

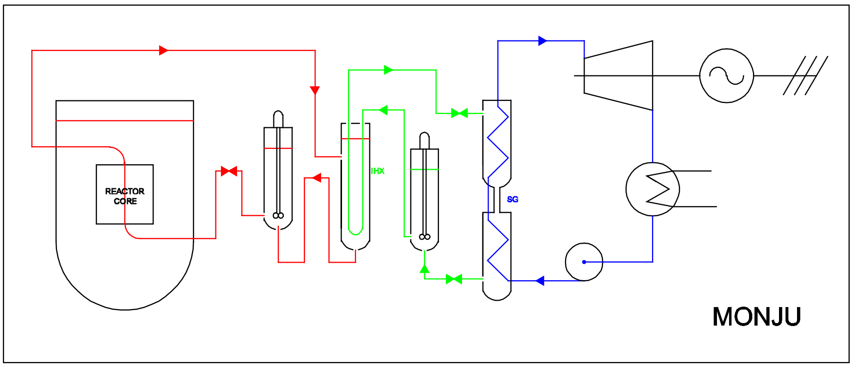


*Fig. 6. Scheme of the Monju reactor*

## Main aspects concerning Sodium as coolant

In order to design a NPP with sodium as primary coolant, the high chemical reactivity, the incompressibility of liquid sodium as well as the high thermal expansion coefficient have to be taken into account. The system foresees an inert argon volume to cover free sodium surfaces to prevent exothermic reactions between sodium and air/water. At the same time the free sodium level within the component, with argon above, is used for expansion purposes. The system components and pipes are traced with electrical heaters to always maintain sodium in liquid state. The main problems connected with the use of sodium as coolant are analyzed in the following sections.

### Sodium–water reactions

The main problem is due to the highly exothermic reaction between sodium and water (SWR)[16]. Sodium reacts with water/steam generating sodium hydroxide, hydrogen and heat accordingly with equation [1]:

[16] The SWR is a complex process where reactions take place in two different steps. Concerning safety analysis, the first step of reaction is the most important one because it generates heat, hydrogen and rapid pressure increase:

1. The first step is characterized by a high rate and release of gaseous hydrogen and heat:

$$Na + H_2O \rightarrow NaOH + \frac{1}{2}H_2 + 140\frac{kJ}{mol}$$

The reactions between sodium and water/steam have major implications in the design, material selection and protection system of sodium-heated SGs. These components are the most critical ones for successful operation of a sodium cooled fast reactor power plant. The implications in the SG design that have to be taken into account are high temperatures at reaction front, failure propagation to adjacent tubes, material erosion and high pressure within the sodium side of the SG. This reaction occurred several times during the operation of some reactors; some examples are reported below:

- Phénix: 30 kg of water injected within the sodium bulk.
- PFR: failure of a superheater tube due to bundle vibrations.
- BN 350 and BN 600: the worst event took place in January 1981, with 40 kg of water injected within the sodium.

The lessons learned from these accidents are:

1. In terms of protection:
    - need of high reliability and rapidity of the hydrogen detection system;
    - automatic shutdown accompanied by rapid depressurization on the steam side;
    - modularization on design of a casing around the SG, capable of confining even the most violent SWR;
    - existence of safety membranes to limit any pressure increase.
2. In terms of prevention:
    - better thermo-mechanical design;
    - more suitable materials;
    - extreme quality in manufacturing (100% inspection, non-extended welds, etc.);

---

$$Na + H_2O \rightarrow NaOH + \frac{1}{2}H_2 + 140\frac{kJ}{mol}$$

2. The second step is characterized by the reaction of products of the first step and additional sodium, according to these equations:

$$2Na + NaOH \rightarrow Na_2O + NaH$$

$$Na + \frac{1}{2}H_2 \rightarrow NaH$$

Then, the total reaction may be represented as:

$$4Na + H_2O \rightarrow 2NaH + Na_2O$$

- precautions in use (prevention of spurious thermal shocks, circuit well protected during drainages, etc.).

Superphénix did not have any SWR: nevertheless, the short reactor lifetime did not allow to validate the SG solution. Several minor problems were verified in plant operation and maintenance. Since 1991, the BN-600 and BN-350 reactors have had no further SWRs.

**Sodium–air reactions**

Liquid sodium reacts with air and oxidation reaction can occur in a runaway manner, leading to sodium fire [2]:

$$2Na + O_2 \rightarrow Na_2O_2 + 519\frac{kJ}{mol}$$

$$2Na + \frac{1}{2}O_2 \rightarrow Na_2O + 435\frac{kJ}{mol}$$

The ignition temperature of sodium is 220°C in damp air, 200°C in dry air and as low as 120°C in stirred liquid pool. Sodium burning is accompanied by production of dense sodium oxide fumes even if the produced heat is much less than conventional hydrocarbon fires. The system must be leak-tight to prevent sodium leaks. The piping/components must be equipped with leak-detection devices in order to limit the effects of fire.

**Sodium leaks**

Concerning the core cooling and integrity, primary coolant leaks are less hazardous in SFRs than in others reactors, because sodium is only slightly pressurized. Anyway, several sodium leaks took place on reactors, sometimes leading to sodium fires. The most serious sodium leak occurred at Monju (approximately 640 kg of Na) which resulted in reactor shutdown during more than 12 years. The leaks' amounts are very different, from less than one gram (detected during inspection) to massive events (BN-600, 2 leaks greater than 300 kg and one involving 1000 kg).

These sodium leaks can have very different origins:

- Construction defects or problems due to design;
- Material problems, such as 321 steel stress cracking;
- Thermal crazing leading to through cracks;

- Corrosion following air intake into the circuits;
- Operator error (for example, during thawing of the circuit and the corresponding expansion of the sodium).

Lessons have been learnt from these incidents in terms of design, circuit operating procedures, leak detection and protection from sodium fires.
The two latter points have led to the following:

- Diversified, redundant detection instrumentation ;
- Need for rapid drainage of sodium circuits;
- Sectoring around the sodium areas to limit the quantity of air available in the event of a possible fire;
- Insulation protection of the concrete floors and walls, covered with a metal plate;
- Possibility of inert gas blanketing of the areas involved.

## Cost aspects and Generation IV requirements

The Generation IV reactors are designed according to stringent specifications, in particular regarding sustainability, economics, safety and reliability, proliferation resistance and physical protection. Most of the designs under development are generally not expected to be available for commercial construction before 2030. The GIF (Generation IV International Forum) provided the criteria for identifying and selecting six nuclear energy systems to further develop. Some of these criteria, especially those related with economic and safety aspects, are summarized below [3]:

*"[…]*
*Economics*: Generation IV nuclear energy systems will have a clear life-cycle cost advantage over other energy sources.
*Economics*: Generation IV nuclear energy systems will have a level of financial risk comparable to other energy projects.
*Safety and Reliability*: Generation IV nuclear energy system operations will excel in safety and reliability.
*Safety and Reliability*: Generation IV nuclear energy systems will have a very low likelihood and degree of reactor core damage.
*Safety and Reliability*: Generation IV nuclear energy systems will eliminate the need for offsite emergency response.
*[…]"*

## The "Candle" concept: Intermediate Sodium loop not necessary

Core cooling in a nuclear reactor is the most important function of the whole power plant, together with a safe shutdown. Since a small interference in the core cooling could lead to a significant drifting in the reactor behavior, the primary coolant must operate under very well-known and controlled conditions (i.e. high purity, no coolant path obstructions, no cavitation phenomena, etc.).

In SFRs, a risk linked to the possibility that a SWR occurs and reaction products can circulate within the primary loop exists. Since a water/steam tube rupture can lead to a chemical exothermic reaction and the products are explosive ($H_2$), caustic and corrosive ($NaOH$), and can obstruct the fuel sub-channels within the core, prevention of any contamination of the primary coolant is of primary importance. For this purpose, as already anticipated in the paper, an intermediate circuit, between primary sodium and the tertiary circuit (water/steam as working fluid), is always adopted.

The use of an intermediate heat exchanger (IHX) in plant solutions already built implies the presence of another hydraulic loop (possibly, multiple), which complicates the overall design, increases the plant costs and adds other maintenance issues; thus its elimination, or drastic simplification, would be extremely advantageous from the standpoint of efficiency, costs and safety.

The innovative solution, here presented and called “candle”, is based on a double barrier separation between the water and the primary sodium, replacing the conventional intermediate sodium loop with a simple heat transfer system, working in absence of moving fluids. Thanks to this solution all the secondary sodium auxiliary systems (pumps, valves, dump vessels, etc.) are no longer necessary. The merging of the IHX with the SG allows achieving all the above mentioned objectives concerning reliability, economics and safety.

### Introduction on the "Candle" concept

The candle is the key-component of the proposed innovation in SFR design; it has been foreseen with a very simple configuration, as shown below (Fig. 7).

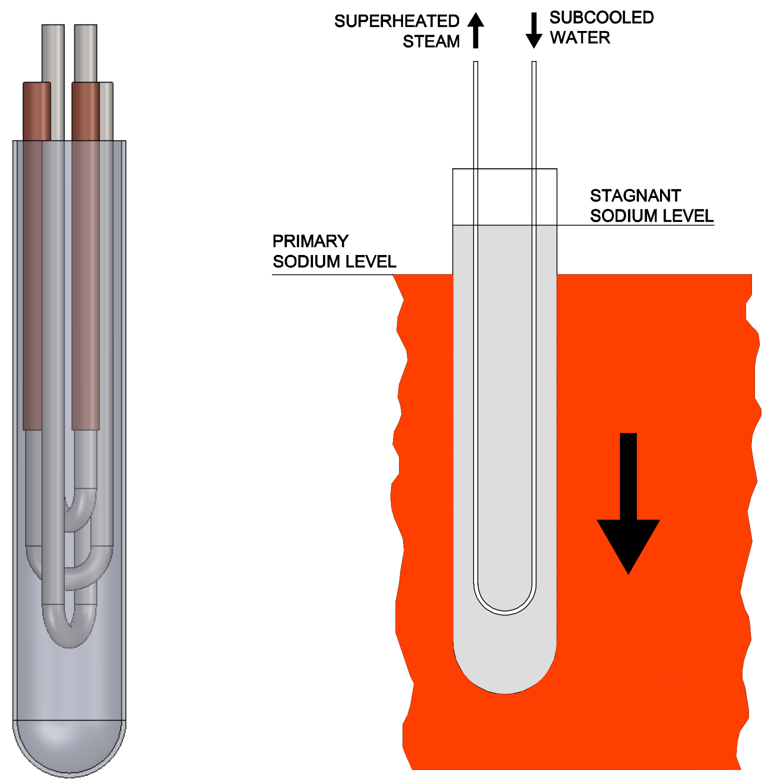


*Fig. 7. The “Candle” concept and its working scheme*

The candle consists of a small diameter vessel, containing three “U-bend” tubes (arranged in angles of 120°); the straight vessel has an easy–to-manufacture geometry and it presents no irregularity, while the inner tubes are foreseen with a relatively large U-bend, in order to achieve a relevant manufacturing easiness, and to have very low stresses. The structural material for the vessel and the tubes is assumed to be stainless steel (having a high corrosion resistance with sodium and mechanical strength). The candle is submerged in primary sodium, and it is filled with a coupling fluid, which thermally connects the primary sodium and the water flowing within the tubes.

The primary sodium, outside the candle vessel, flows downward transferring heat. The filling fluid has been chosen taking into account its thermal-physical properties since it has the purpose of transferring heat from the core coolant (primary sodium) to the working fluid (water) mainly by conduction. In the reference case, sodium has been chosen as filling fluid.

## Advantages of the concept and critical points to focus

Thanks to the innovative proposed concept, the overall NSSS size is greatly reduced. In Fig. 8 and Fig. 9, it can be seen that the presence of the candle inside the system greatly reduces the number of components and systems required, especially avoiding any moving part within the secondary circuit.

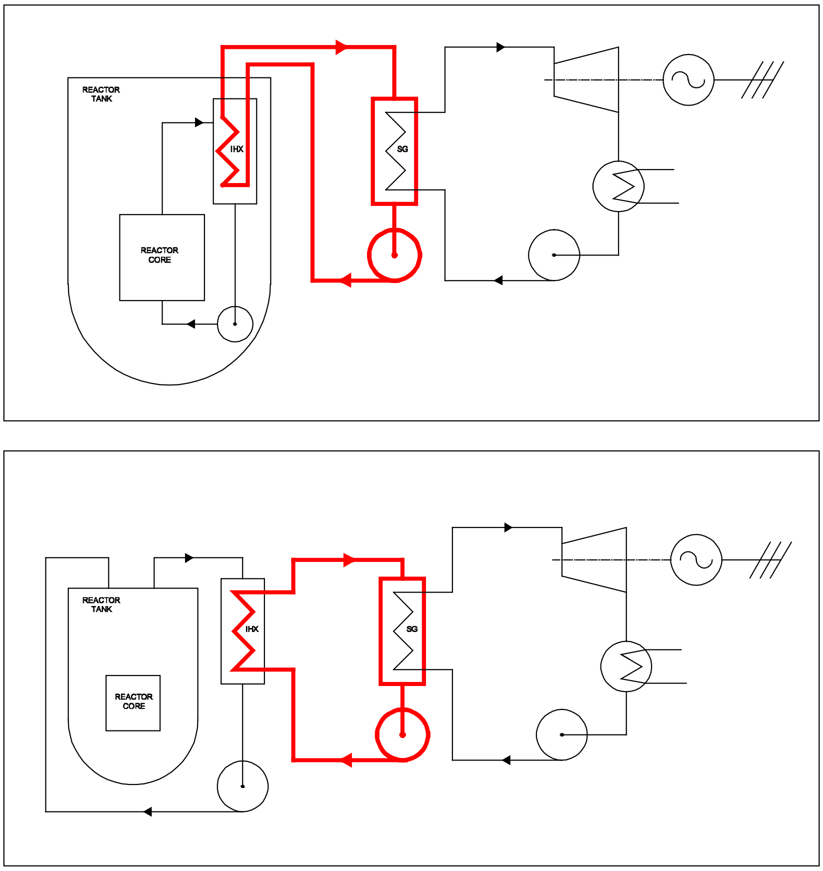


*Fig. 8. The pool and loop standard concept schemes (secondary circuits in red)*

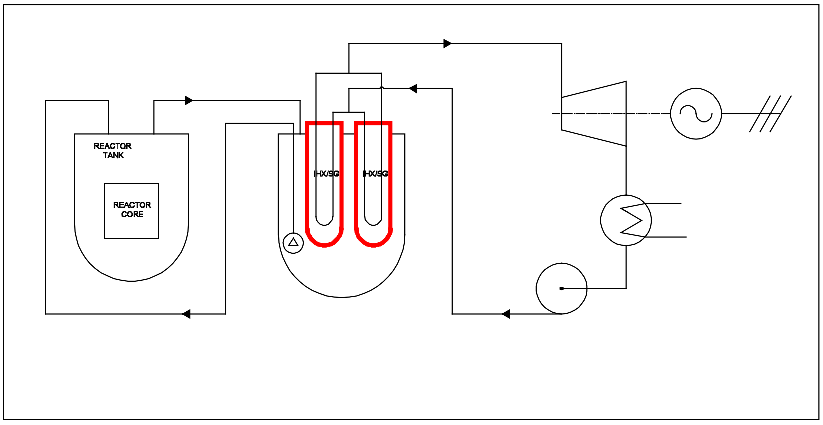


*Fig. 9. The “Candle” concept scheme (secondary circuit in red)*

The presence of the candles implies also simpler construction processes. Such an innovative solution requires a dedicated safety analysis to verify the feasibility of the system. In particular, the heat exchanger characteristics and the problems due to the presence of impurities (eventually, generated from a sodium water interaction) in the primary coolant should be verified in the reference accident conditions to assure that:

- No consequences derive from the interaction of primary sodium and secondary stagnant coolant;
- The failure of a water/steam tube and the consequent interaction between secondary stagnant coolant and the working fluid do not compromise the integrity of adjacent tubes within the same candle;
- The failure of a tube and the consequent interaction between secondary stagnant coolant and the working fluid do not compromise the integrity of candle vessel, in order to prevent any possible interaction between water and primary sodium and the possibility that reaction products reach the primary circuit and hence the core.

# Engineering aspects of the proposed design for the Candle

## Design criteria

### "U" tubes radius selection

The selection of the "U" tubes bend radius, within the candle vessel, is a key parameter since it affects the manufacturing costs of the steam generator and, indirectly, the minimal candle vessel diameter. The radius is a function of the tube thickness and diameter that were selected as follows:

- Tube thickness to withstand the pressure difference between the working fluid (about 185 bar) and the secondary stagnant fluid (about 3 bar). Conservative evaluation implied atmosphere pressure as reference pressure instead of the real operating pressure;
- Tube internal diameter to maintain the water/steam velocity lower than defined values having acceptable heat transfer coefficient. In any case the minimization of the total number of the candles within the system has to be considered as one of main targets.

The selection of the “U” tube radius affects the candle vessel diameter and generally an iterative process is required in order to define the optimum “U” tube diameter, the “U” tube bend radius and hence the candle vessel diameter.

**Candle diameter selection**

The choice of the candle vessel diameter has to be made taking into account the following issues:

Minimization of the overall Candle dimensions;

Costs minimization:

- Use of standard commercial pipe for vessel and tubes for “U” tubes;
- Candle vessel diameter large enough to allow a sufficient space to have a 180° “U” tube bend with a standard radius and easy manufacturing processes;
- Diameter large enough to have a sufficient heat transfer surface from primary sodium to the stagnant secondary coolant;
- Diameter large enough to allow an easy discharge of the stagnant secondary sodium (if any) in case of a sodium water reaction (failure of a tube), in order to minimize the severity of the pressure transient;
- Diameter small enough to minimize the stagnant sodium thermal resistance.

Taking into account these issues, a diameter of 0.1 m was selected in correspondence to a ¾” BWG 13 tube selected as reference for the “U” tubes.

**Finned/not finned**

In order to enhance the heat transfer performance of the steam generator, or to reduce the tube length with other parameters being equal, fins can be added in the riser tubes section. The presence of finned tubes could improve the heat transfer capability of the system but is strongly affected by the filling fluid choice in addition to the type and material of fins. In fact, if the thermal conductivity of the filling fluid, in which heat transfer is mainly based on conduction, is higher than that of the fins material, a solution based on not finned tubes is preferable.

**Heat transfer enhancement**

A common issue related with once-through steam generators is the strong reduction of the heat transfer capability of the system in the post dry-out region. The dry-out corresponds to the point in which the drying of the inner wall of the tube occurs and the heat transfer coefficient decreases because of the steam presence against the tube wall.

Once the post dry-out condition is reached, the complete vaporization of the droplets entrained into the steam flow becomes more difficult because specific power decreases, due to heat transfer occurring basically with a vapor phase. Since this issue strongly penalizes the heat transfer performance of the steam generator, and the dimensions of the candles, possible solutions have been investigated in order to minimize this effect. Each solution analyzed has been evaluated to verify: first, the feasibility and secondly, possible contras that can affect the whole system. Among the analyzed systems, taking into account problems related to the manufacturing complication of the candle and to the increase of total pressure drop within the system, the choice has regarded a special use of a standard heat transfer enhancement device; in particular, the chosen device was a “twisted tape”.

Twisted tapes are usually used in single-phase zones, where the helical flow allows to locally increase the velocity and hence the heat transfer coefficient. The special use within the candle system concerns the use of the turbulence enhancement system from the dry-out point up to the outlet section, achieving two benefits:

1. Within the post dry-out zone (liquid deficient zone), thanks to the centrifugal force, the droplets entrained into the steam flow are forced outward against the tube wall and immediately vaporize guaranteeing a high heat transfer coefficient also within the liquid deficient zone. The heat transfer coefficient, since the liquid droplets are in saturation condition, results as high as in the nucleate boiling and the droplets can completely vaporize in a very short length.
2. Since the post dry-out region is reduced by the inclusion of the twisted-tape, the superheating zone is longer for a same tube length. Moreover, thanks to the presence of the twisted-tape, an average increase of about 1.5 times in the heat transfer coefficient is expected.

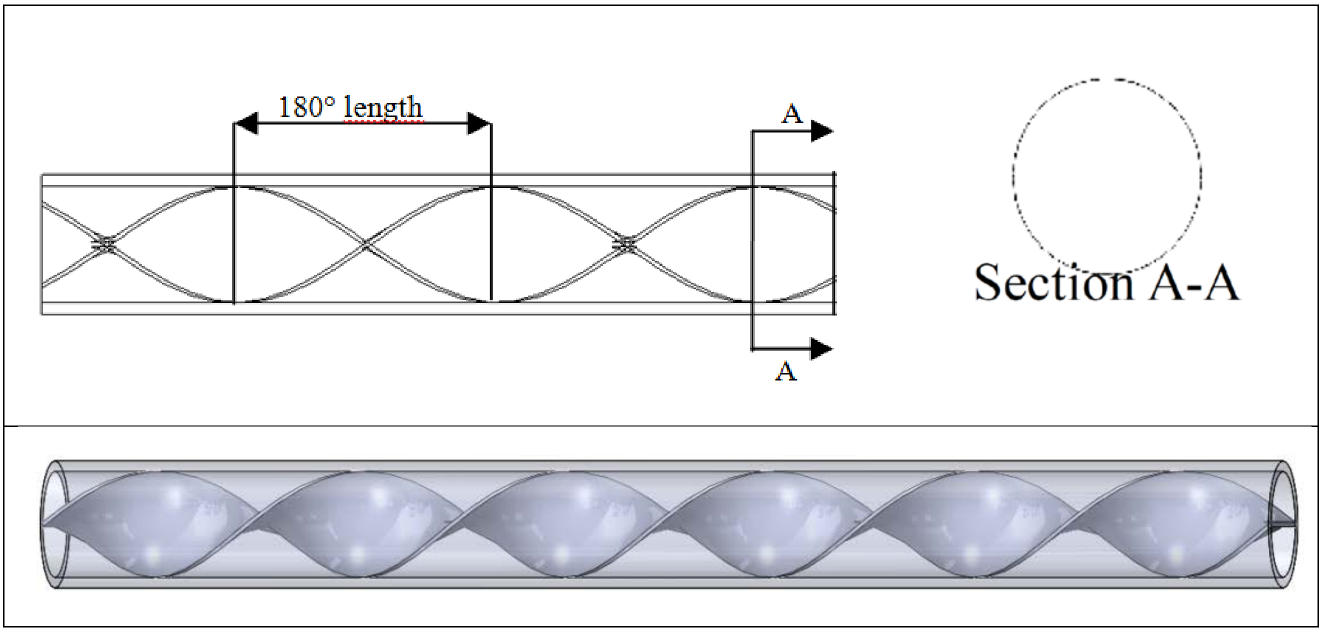


*Fig. 10. Scheme and main parameters of a twisted tape insert*

Obviously, it is of primary importance the definition of the dry-out point from where the twisted tape has to be foreseen. In fact, the presence of twisted tape within the tube causes an increase in pressure drops especially concerning the two-phase region.

The optimum, from an engineering point of view, is that the twisted tape starts from the dry-out point and arrives up to the outlet section (see Fig. 11).

Since the heat transfer coefficient, in the post dry-out region, falls down, an evaluation of the improvement in the heat transfer coefficient due to the twisted tape insertion has been carried out (see Fig. 12).

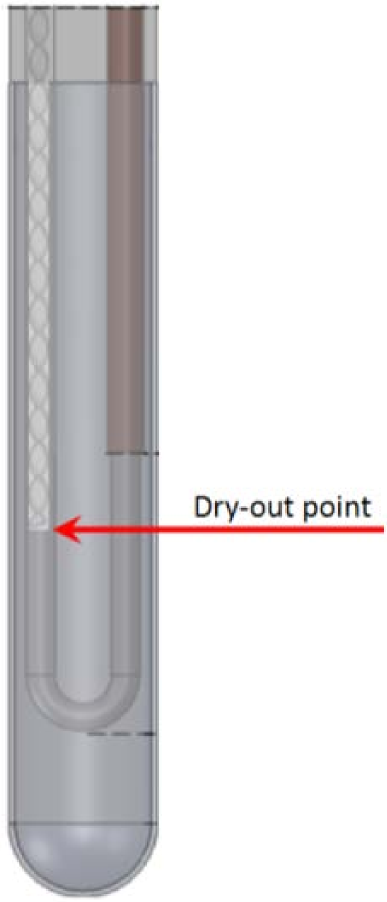


*Fig. 11. Scheme of the candle concept with the insertion of a twisted tape above the dry-out point.*

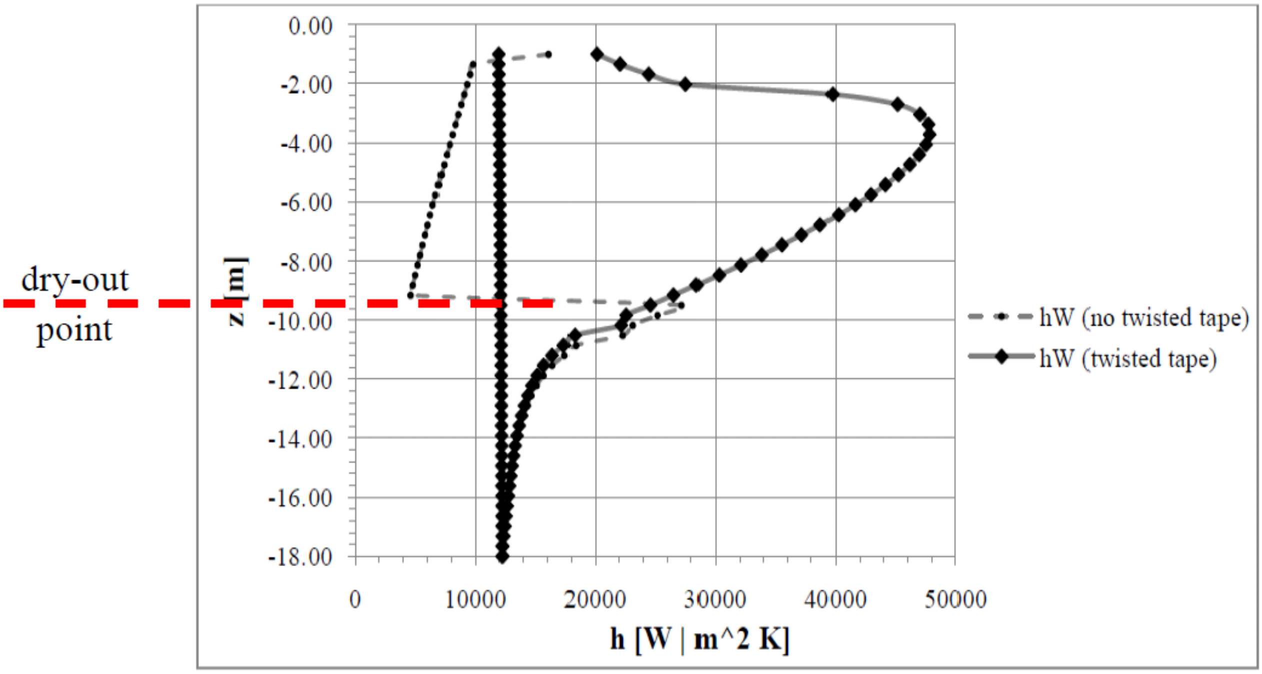


*Fig. 12. Evaluation of the heat transfer coefficient trend as a function of the "U" tube length*

**Insulator length**

The candle heat exchanger consists of "U" tubes and, within the preheating section, the working fluid flows downward against the natural circulation direction. To maintain at acceptable values the density difference between the inlet section and the lower section of the “U” tube, the saturation condition of the working fluid has to be avoided. In order to prevent reaching the saturation point within the downcomer, an insulating layer aimed at limiting the heat transfer from primary sodium (through the stagnant fluid) to the working fluid has been foreseen around the downcomer tube. In order to optimize the performance of the heat exchanger, the length of the insulator has to be long enough to cause saturation condition of the working fluid in the bottom part of the riser section in order to support natural circulation.

**Choice of filling fluid**

In order to choose the best filling fluid for the secondary stagnant coolant, a comparison among different liquid metals has been performed. For each material, pros and contras have been compared in order to evaluate the best solution for the filling fluid. In particular, the comparison has been based on costs, chemical and physical risks, thermal-physical properties, etc. This analysis considered Sodium, Lead, Gallium and Indium.

Trying to find a compromise between the overall system performance and its reliability, taking into account safety aspects and issues concerning the corrosiveness of alternative liquid metals selected, the most appropriate filling fluid has been considered sodium.

For the candle, a filling fluid characterized by a high thermal conductivity is necessary, in order to reduce thermal losses within the candle and to minimize the number of required candles. From the safety point of view, an intermediate fluid equal to the primary coolant is a huge advantage: in case of an integrity failure of the candle vessel two identical fluids will be put in contact without any consequence for the plant safety. Moreover, sodium is a relatively light metal and this is an advantage both concerning structural solutions (also taking into account abnormal design loads as seismic ones) and for candle draining process.

The choice of the filling fluid has fallen on sodium, mainly for three reasons:

- The high thermal conductivity allows to have high heat transfer coefficient even in stagnation conditions, where the heat transfer capability of the system is mainly based on conduction.

- The low density value of sodium implies smaller structure compared with alternative filling fluids analyzed and hence lower costs.
- Chemical behavior of fluids and criticality in material selection.

## In-sight on safety of a Candle-based integrated IHX/SG

### The Candle resistance

Preliminary studies have been carried out in order to identify the minimum candle vessel thickness to withstand loads in accident conditions, when a sodium water reaction occurs. The rough evaluation showed that the pressure transient following the SWR occurrence is strongly influenced by the time required for sodium removal from the candle volume. At present, detailed CFD analyses are being carried out in order to obtain a detailed pressure and temperature transient during the accident evolution. The pressure difference existing between the filling fluid and the working fluid implies the critical flow condition that limits the water mass flow rate and hence the quantity of water that can react with sodium per unit time. The safety margins related with maximum values of pressure and temperature can lead to modify the candle vessel diameter in order to reduce the required time to eliminate sodium form the candle volume.

## The need not to involve surrounding Candles

If a water tube rupture occurs within a candle, or if a water leakage from the distribution system occurs, the water could flow, from the SG upper volume, inside the candle vessel. With the purpose of avoiding, or at least minimizing, any water intake into the stagnant sodium, each candle is provided with a plug characterized by the following features:

- it must eliminate or limit the water/steam flow from the volume above the candle itself (dome);
- it must prevent the candle pressurization, in case of internal SWR, so it has to operate as a rupture disc;
- it must allow the differential thermal expansion of the hot and cold legs of the “U” tubes within the candle;
- it must provide a way to fill and drain the candle with sodium.

For this purpose, an idea has been developed (see Fig. 13) where:

- the axial labyrinth seal provides a barrier against external steam/water inlet, but allows the differential thermal expansion of the U-tube legs;
- the central rupture disc will break in case of a limited candle pressurization, but it can be removed in order to allow access to the filling sodium (for purging/filling operations, as well as for ISI&R purposes)

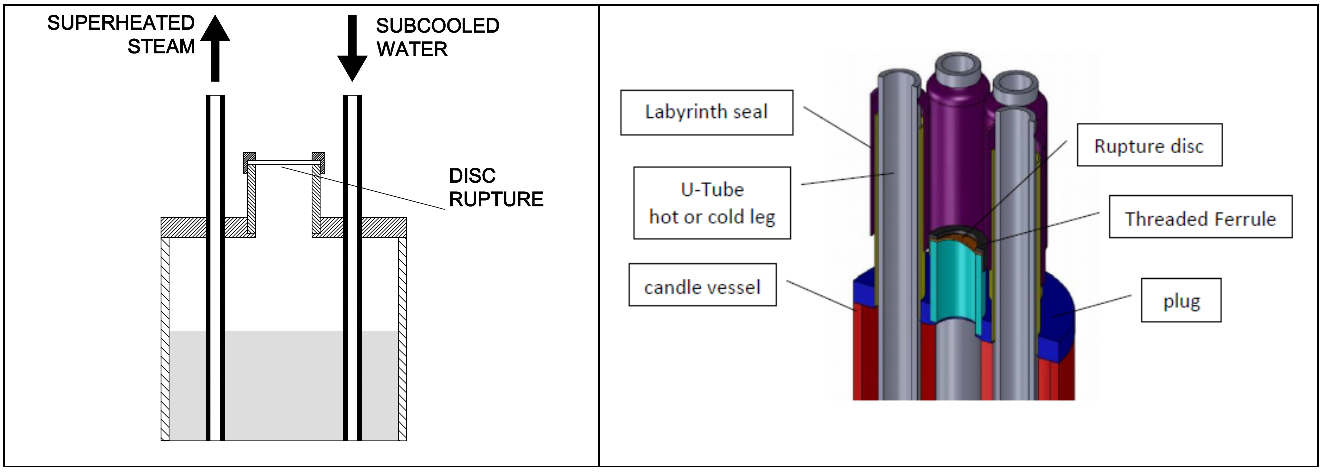


*Fig. 13. The candle vessel plug and scheme of the rupture disc*

# Engineering of the tanks hosting the Candles

## The support of piping in the Candles

Important issues linked with thermal expansion phenomena must be taken into account. The cold leg of the “U” tube, the downcomer, is totally filled with subcooled water, while the hot leg, riser, is filled with saturated water and superheated steam at the same pressure. For this reason, the mean temperatures between the two sides of the “U” tube are very different. Due to the height of the water tube (more than 15 m), a significant length increase, due to thermal expansion of the steel, is expected. A rough calculation has been carried out, leading to the following results:

$$\Delta l = \alpha \cdot \left(T_{riser} - T_{downcomer}\right) \cdot h_{"U"\,tube} = 0.037\,m$$

*where*:

$$\alpha = steel\ thermal\ \exp ansion = 17.5 \cdot 10^{-6}\ \frac{m}{m}$$

$$T_{riser} = 420\,°C$$

$$T_{downcomer} = 280\,°C$$

$$h_{"U"\,tube} = 15\,m$$

The result shows a considerable differential thermal length increase, which should be taken into account in a further stage of design, for example, implementing a device based on a flexible support as that shown in Fig. 14.

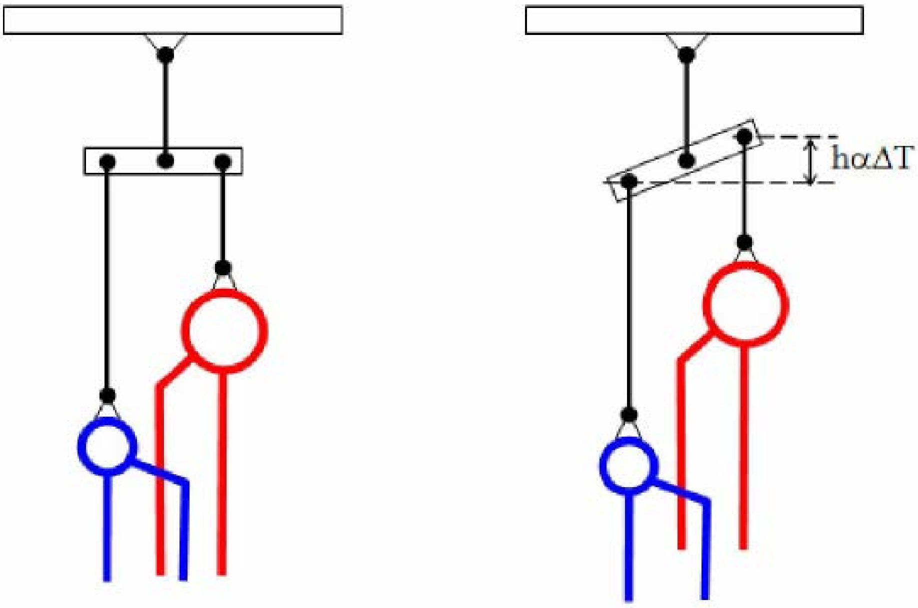


*Fig. 14. Schematic representation of the flexible support that allows free thermal expansion of the "U" tubes*

## The grouping of candles into modules

The whole steam supply system is made up of several candles and it is impossible to move each candle separately. Moreover, the very high aspect ratio (H/D) of the candle implies low structural strength. In order to take into account all these aspects and to make the construction easier, 36 candles are grouped within a single module (see Fig. 15). This solution thus reduces time and costs required for manufacturing of the system, responding at the economic requirements for generation IV reactors.

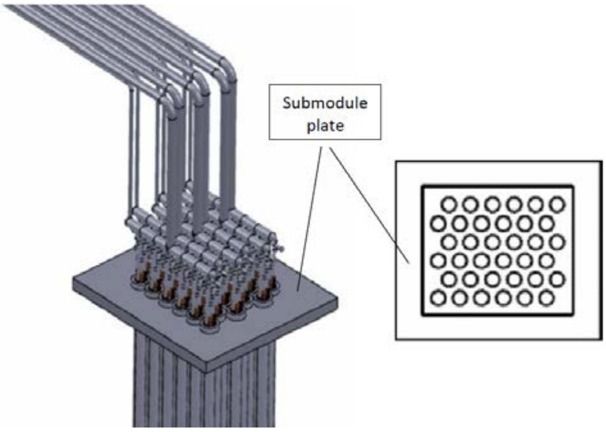


*Fig. 15. Single module made up of 36 candles*

## The dome

Since the required candles are a lot, they are arranged within special tanks outside the primary tank (the NPP configuration becomes a mixing between the “pool” and the “loop” concepts). Each SG tank will be enclosed, in the upper part, in a metallic dome, which has a double goal:

- it must contain sodium/water leakage from the circuits (avoiding any environment contamination), and it must be able to withstand the reference accidental conditions
- the pump supporting structure can be placed within such volume.

The volume within the dome must be filled with an inert gas, in order to prevent any air-sodium interactions (sodium fires). In this study, argon has been assumed as cover gas. The coupling of the dome with the cylindrical vessel can be realized, for example, with a flange along the full perimeter. Since the upper volume of the tank holds the sodium distribution piping, the inner diameter (15 m) is sized taking into account the dimensions of the tubes (in particular the hot ones).

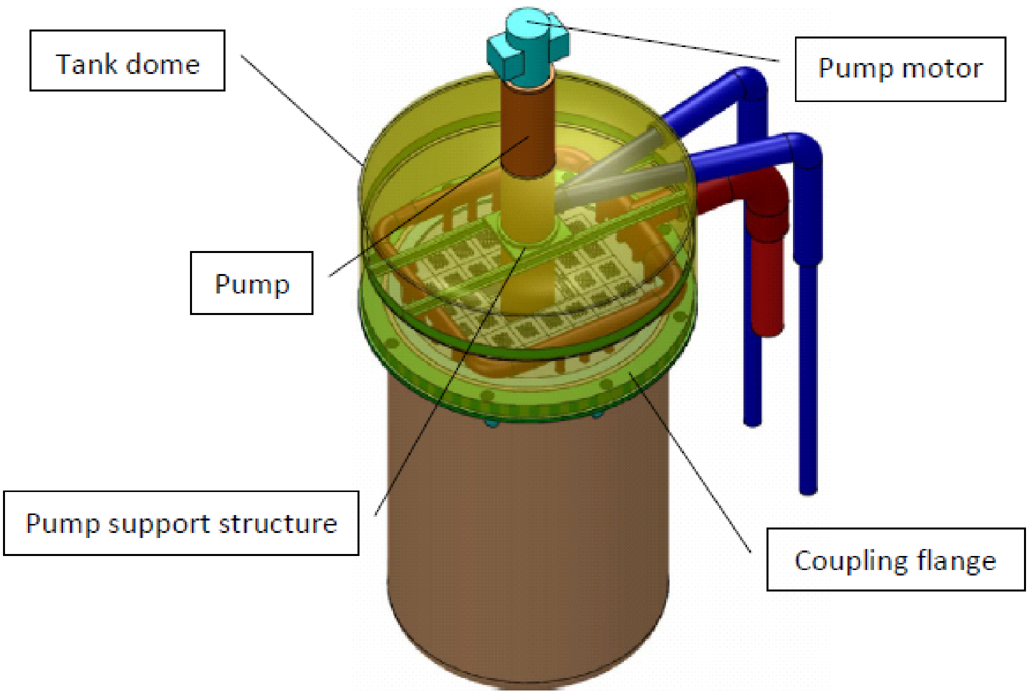


*Fig. 16. Representation of the dome above one of the candle tanks*

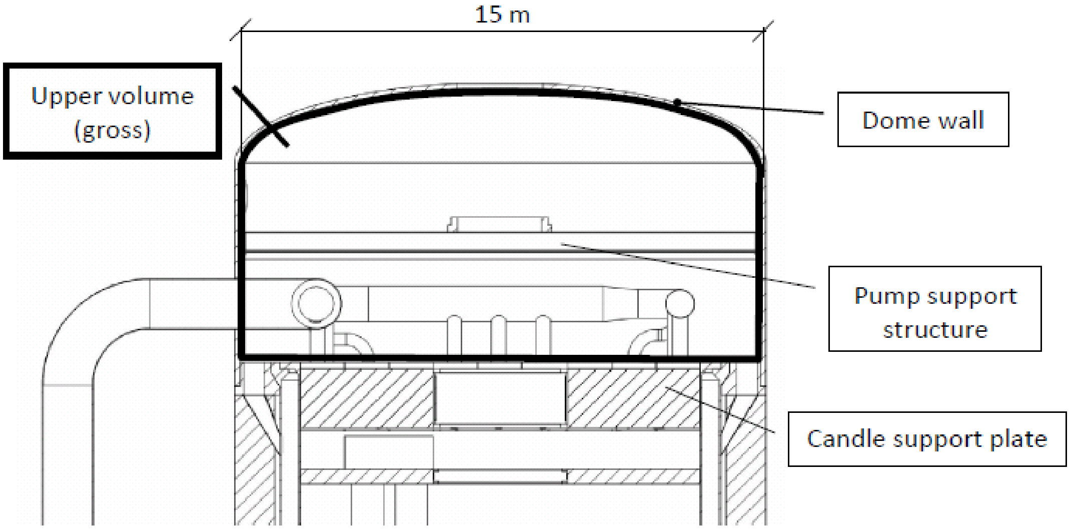


*Fig. 17. Structural details of the dome and of sodium distribution piping within it*

A preliminary evaluation of the necessary free volume within the dome has been carried out and later verified. Two independent evaluations of the dome pressurization due to the release of a high pressure water flow within a candle have been carried out (1. RELAP5/mod3.3; 2. Computer code developed ad hoc). This preliminary evaluation did not take into account the thermal heat produced within the candle due to the sodium-water reaction. The simulation does not result conservative but it can give a rough indication of the pressure equilibrium value at the end of the accident transient phase. Since the pressure transient has been evaluated to be very short (less than 1 second), the pressure value that affects the structural design of the dome is mainly due to the high pressure of the water/steam circuit; while concerning the structural design of the candle, it could be mainly influenced by the temperature and pressure values following the SWR.

The results of the simplified simulation concerning the failure a “U” tube within the candle, in terms of final pressure and temperature in the dome volume, are reported below:

*Tab. 2. RELAP 5 DBA simulation results*

| Dome final conditions | | |
|---|---|---|
| Vapor pressure | Total pressure | Temperature |
| 0.535 bar | 1.857 bar | 280.1 °C |

The main consequences of the reference accident condition on the structures are reported below:

- The temperature increase (+18 °C), within the dome, involves steel yield stress decreasing of about 2.4%, while the ultimate stress will remain almost constant (–0.15%);
- The dome volume pressure increment (+48.8%) causes an increase of the mechanical stress of the dome wall;
- The pressure rising within the dome volume causes a linear increase in the load carried by the plate itself.

Results show that the effects of this accident are acceptable. Nevertheless, should more than one candle be involved in the accident, the amount of released energy could be too high. Thinking to this case, it is advisable to provide a relief tank in order to increase the total volume available for gas expansion.

### The mechanical support of candles and thermal protection

Due to the severe conditions in which the candle steam generators have to work, special solutions have to be provided in order to guarantee the integrity of all structures and not to reach extreme conditions (temperature and pressure values). In particular, in order to avoid any reduction of the mechanical resistance of the support plate, where the weight of the candle modules is borne, an intermediate plate, without any structural function, between the free sodium level and the support plate has been foreseen. In this way, the support plate operates at a lower temperature and higher mechanical performance is guaranteed.

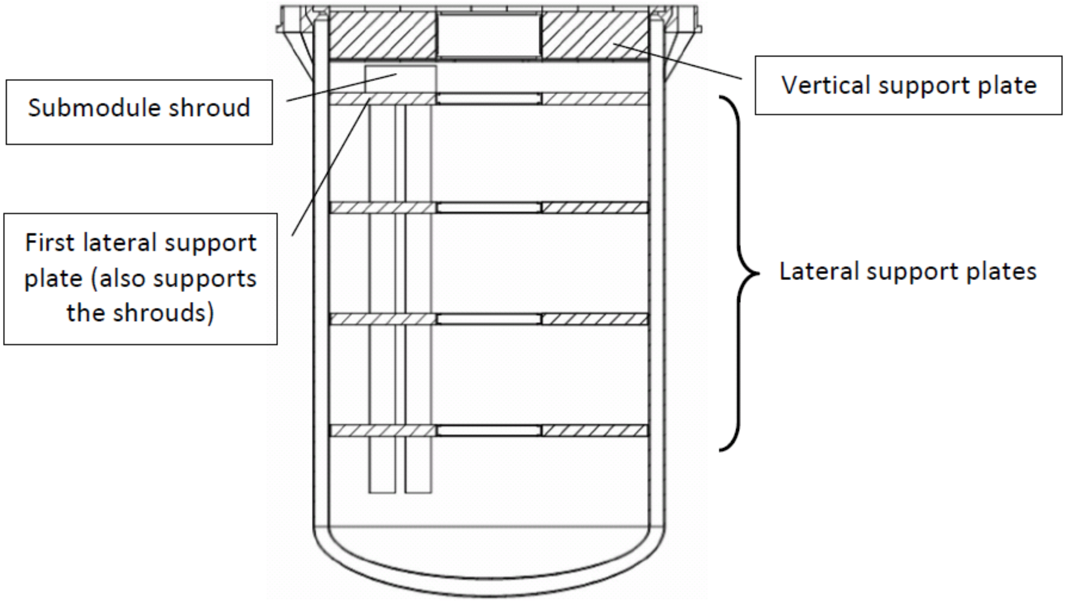


*Fig. 18. Scheme of the internal support structure within a candle tank*

## Seismic candle protection

Due to the considerable length of the candles and to the presence of only upper mechanical constrains, in case of a seismic event candles could move horizontally causing possible candle vessel failure. In order to minimize the free horizontal movement of each candle, a series of lateral support plates has been foreseen, within the height of the candles, of the types shown below (see Fig. 19).

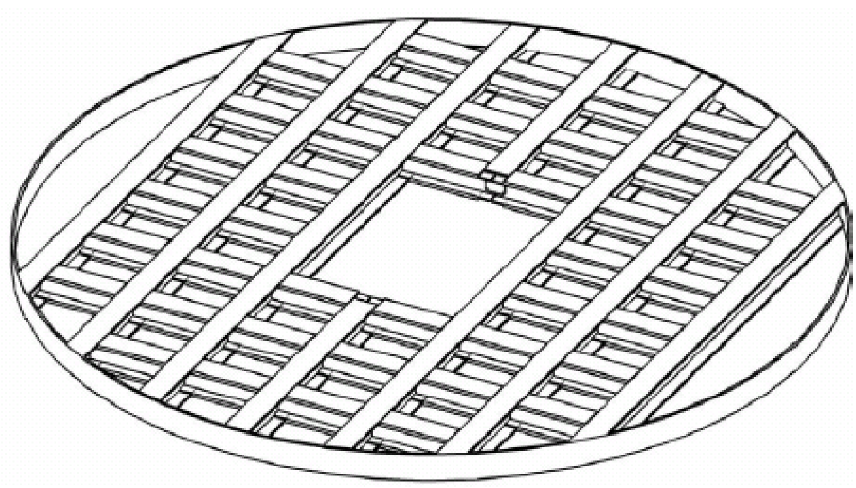

*Fig. 19. Detail of a lateral support plate for the candle modules*

# The Balance of Plant

## Piping lay-out

In order to produce electricity, the steam has to be sent to a turbine and a suitable piping system is required. Due to the high number of candles and tubes (reminding that each candle is characterized by three feedwater inlet and three steam line outlet), a complex system of manifolds and pipes has been studied.  During the development of the piping system of water and steam, a preliminary solution was selected; later, with the target of cost minimization, an alternative solution, based on standard components only, has been developed.

Since the first solution analyzed would require non-standard components, in spite of the possibility of lower diameter for the candle tank, an increase in costs would be experienced. Taking into account a solution described before (grouping several candles in a single module) and the use of standard components only, an alternative manifold solution has been identified. Each candle module is includes 36 candles arranged on 6 parallel rows; each row can have an independent manifold concerning both the feedwater and the steam. This solution strongly simplifies the piping management since pipes with lower diameter can be used and also within a single module there are independent sub-systems made up of 6 candles that contribute to the complete modularization of the steam supply system.

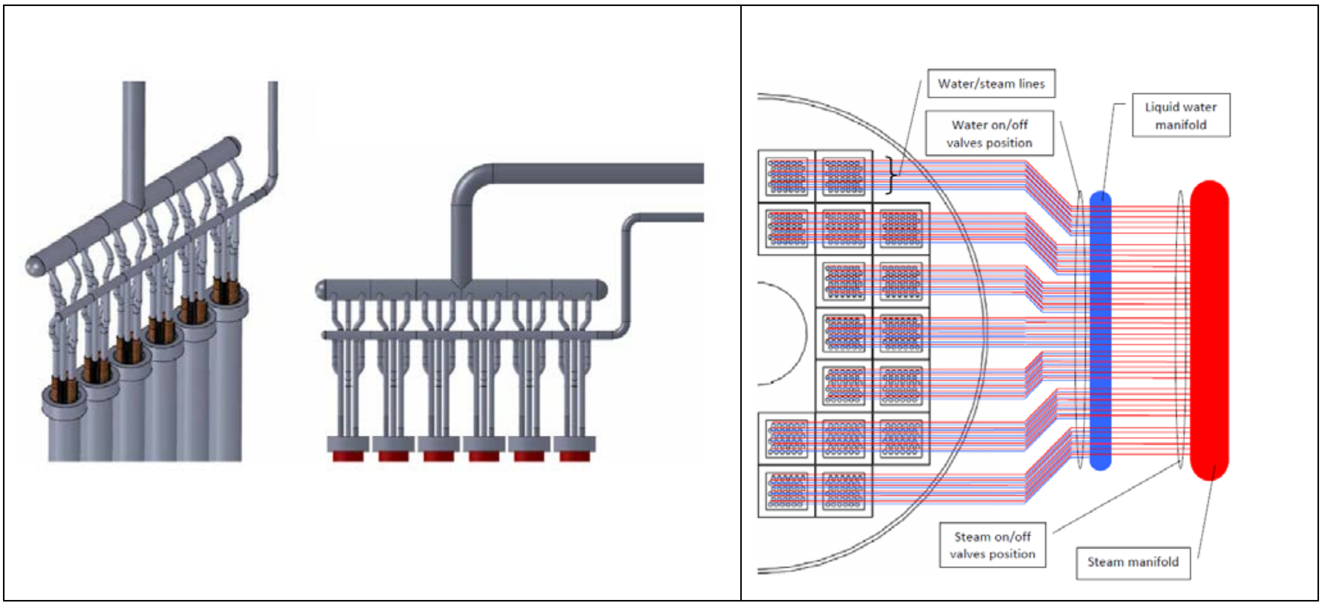


*Fig. 20. Schemes of a sub-system made up of 6 candles within a module of 36 candles and the schematic representation of piping connections to the main manifolds.*

## Cost considerations

### Rough evaluation, based on weights of machined steel in SPX2 and on the Candle-based-SFR weights

The investment cost, due to the realization of the proposed Candle SG, is an essential parameter for its feasibility evaluation. At the present phase, detailed characteristics are not yet available, and no provision has been made yet on the realization phases of all the components necessary for a full integration in a nuclear power plant (i.e. concrete vault, instrumentation, safety cooling systems, dedicated radiological shielding for devices operating in radioactive environment, etc.). Therefore, at present only a very rough evaluation is possible for cost assessment, taking into account only the material cost. A preliminary whole evaluation of the steam generator system cost has been performed. As result of this evaluation, the comprehensive cost of a single tank including its internal components (candles, “U” tubes, piping, manifolds, valves, etc.), in a solution made up of three heat exchanger tanks, should be in a range between 40 and 50 M€. Concerning the manufacturing costs, a percentage close to 50% of the whole cost could be assumed.

## Conclusions

Studies carried out during the last years showed some very attractive features concerning the concept of the “Candle”-type steam generator for application in SFRs, especially in view at the elimination of the intermediate sodium circuit, through the “fusion” of the traditional IHX with the SG. The elimination of the intermediate sodium circuit, together with its components, results in a great saving in capital cost as well as in a reduction of component number and in ISI&R strategies simplification.

In spite of the great simplification due to the introduction of this system, the safety performance of the NPP are not compromised, but even improved.

In case of a possible double wall failure, of “U” tubes and of the candle vessel, since the SWR occurs outside the primary tank, only the reaction products can affect the operation of the reactor rather than the pressure transient too. Isolation valves on the sodium piping could, in any case, avoid also this remote problem.

Concerning the development of the innovative system to a more advanced stage, further studies should be performed in order to:

- Optimize the candle geometry according to results obtained in the safety analyses (possibility to increase the candle diameter to reduce the required draining time, evaluation of exact volume and other dimensions of the tank dome, etc.);
- Carefully evaluate the costs of the whole system manufacturing, in order to compare this solution with the alternative ones within the scope of GEN IV economics requirements.

## Nomenclature

| | |
|---|---|
| GIF | Generation IV International Forum |
| IHX | Intermediate Heat eXchanger |
| ISI&R | In Service Inspection and Repair |
| NPP | Nuclear Power Plant |
| SFR | Sodium Fast Reactor |
| SWR | Sodium Water Reaction |

## References

[1]. *“Sodium-Water Interactions in steam generators, sodium fire: tests and analyses”* – IAEA

[2]. *“Impianti Nucleari”* - Maurizio Cumo. Ed. “La Sapienza”

[3]. *“Generation IV Roadmap”* – Generation IV International Forum

## 5.2 Further consideration on the Candle solution

### 5.2.1 Filling fluid selection

Different filling fluids have been investigated but, for different reasons, sodium has been selected also as filling fluid. This choice has been mainly due its thermal properties, despite its high chemical reactivity causes many drawbacks. In the Tab. 3 some characteristics of filling fluids analyzed and compared for the same purpose are reported.

*Tab. 3. Some relevant parameters of three filling fluids analyzed*

| | | **Sodium** | **Lead** | **Gallium** |
|---|---|---|---|---|
| **Thermal conductivity (@m.p.)** | [ W \| m K ] | 89 | 15.8 | 28.22 |
| **Density (@m.p.)** | [ kg \| $m^3$ ] | 927 | 10650 | 6116 |
| **Melting point** | [ °C ] | 97.72 | 327.46 | 29.76 |

The data are obtained from correlations available in literature ([8], [9] and [10]).

Since the thermal coupling between the primary sodium and the working fluid is obtained by means of a stagnant filling fluid, the thermal conductivity is one of the most important parameters affecting the heat transfer performance of the whole system.

The choice to use sodium as filling fluid has been mainly due to the following considerations:

- The liquid Sodium has a higher thermal conductivity compared with Lead and Gallium;
- Since each candle vessel is submerged in the primary sodium, using sodium as filling fluid allows to reduce the mechanical stresses, which would be due to static pressure differences;
- The choice of sodium as filling fluid allows to use systems of the same type to prevent and/or mitigate incident due to the sodium chemical reactivity (the use of a different filling fluid can add further precepts to the plant management).

### 5.2.2 CANDLE design criteria

**Candle diameter selection**

The choice of the candle vessel diameter has to be made taking into account the following issues:

- Minimization of the overall Candle dimensions;
- Costs minimization:
    - Use of standard commercial pipe for vessel and tubes for “U” tubes;
    - Candle vessel diameter large enough to house the three “U” bend tubes guaranteeing a gap to be filled with the stagnant sodium;
- Diameter large enough to have a sufficient heat transfer surface from primary sodium to the stagnant secondary coolant;
- Diameter large enough to allow an easy discharge of the stagnant secondary sodium (if any) in case of a SWR (failure of a tube), limiting the maximum pressure in order to minimize the severity of the accidental transient;
- Diameter small enough to minimize the stagnant sodium thermal resistance.

Taking into account these issues, a diameter of 0.1 m was selected in correspondence to a ¾” BWG 13 tube selected as reference for the “U” tubes.

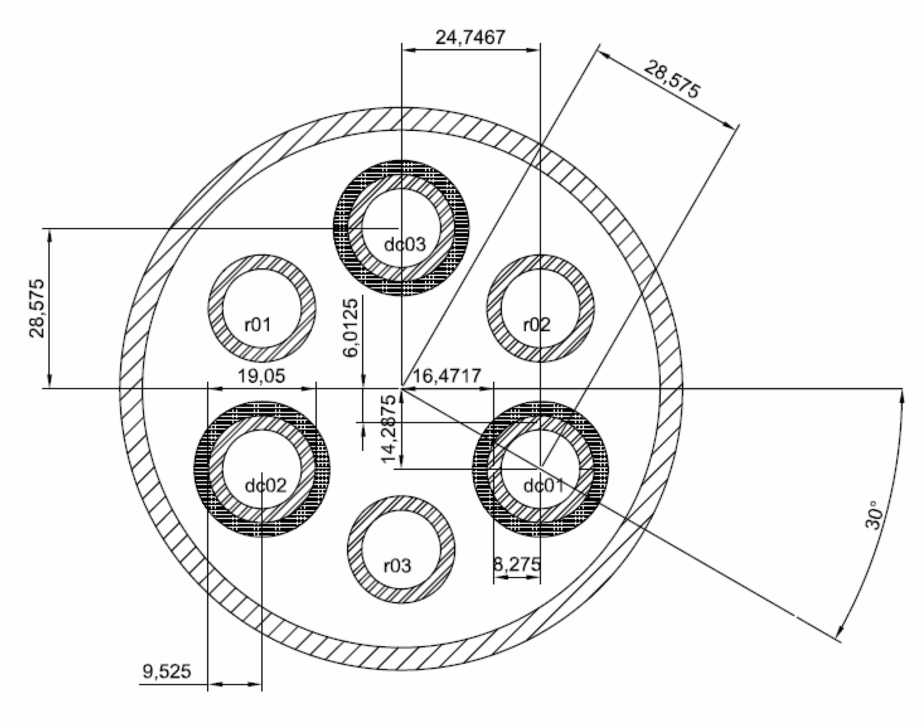


**Figure 24. Main dimensions of the "Candle" concept**

## Finned/not finned

In order to enhance the heat transfer performance of the steam generator, or to reduce the tube length with other parameters being equal, fins can be added in the riser tubes section. The presence of finned tubes could improve the heat transfer capability of the system but is strongly affected by the filling fluid choice in addition to the type and material of fins. In fact, if the thermal conductivity of the filling fluid, in which heat transfer is mainly based on conduction, is higher than that of the fins material, a solution based on not finned tubes results, obviously, preferable.

In the reference case, where sodium has been selected as filling fluid, the not finned solution results to be the preferable since at 500 °C, AISI 316L has a thermal conductivity of about 20 W $m^{-1}$ $K^{-1}$ while the molten sodium has a thermal conductivity of about 65 W $m^{-1}$ $K^{-1}$. The use of the not finned design solution means a reduced thermal resistance to the conductive heat transfer mechanism.

## 5.3 Paper on the radiation based DHX

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Title: Innovative radiation-based Decay Heat Removal System

Authors: Andrea De Santis, Matteo Nobili, Damiano Vitale Di Maio, Antonio Naviglio, Fabio Giannetti

## Abstract

One of the main issues, regarding the development of Generation IV nuclear power plants, is the safety. In order to design nuclear systems characterized by enhanced safety features, an innovative and passive system, aimed at decay heat removal, especially suitable for sodium cooled fast reactors, has been developed. The attention is here mainly focused on an innovative, passive, decay heat removal system, proposed for liquid metal cooled reactors and based on a radiation-based bayonet tube heat exchanger that allows to remove decay heat from the primary coolant. Heat transfer is mainly guaranteed by the radiation mechanism, since each bayonet tube is equipped with a vacuum gap required to better decouple primary and secondary coolants and to detect possible leakage. The secondary fluid flows within the bayonet tubes in natural circulation. The proposed heat exchanger presents important advantages from the safety point of view, including the good compatibility between primary and secondary coolants, the good heat transfer properties of secondary fluids and, above all, the possibility to detect immediately any leak from both primary and secondary side, thanks to a continue monitoring of the vacuum gap pressure. In this paper, several CFD analyses, aimed at analyzing different parameters' influence on the heat transfer capability of the system, have been discussed.

## Introduction

The developing of the next generation of nuclear reactors (GEN IV) requires many efforts in terms of R&D and in the safety analysis field. Within the selected type of GEN IV nuclear reactors, Sodium Fast Reactors and Lead Fast Reactors represent, in Europe, first steps towards the future nuclear energy system. According to Generation IV International Forum, future-generation nuclear energy system must be designed with revolutionary improvements, in particular regarding sustainability, economics, safety and reliability, proliferation resistance and physical protection. Focusing on Safety and Reliability three points have been defined:

1. Generation IV nuclear energy systems operation will excel in safety and reliability
2. Generation IV nuclear energy systems will have a very low likelihood and degree of reactor core damage

3. Generation IV nuclear energy systems will eliminate the need for offsite emergency response

To fulfill these requirements new projects and concepts have been developed, often foreseeing the introduction of passive components with a full capacity to sustain every kind of transient generated by an accident. In particular, a great interest is placed on Decay Heat Removal Systems (DHRS) which have to guarantee long term refrigeration of the core. In this paper the attention will be focused on the primary heat exchanger of a radiation-based DHRS and the heat transfer performance of the component will be analyzed.

## Decay Heat Removal System (DHRS)

The purpose of the system is to maintain the primary coolant within a limited temperature range, in order to ensure an efficient cooling of the core and prevent any possibility to have a phase change that adversely affect the proper operation of the reactor. In a nuclear reactor the power produced into the fuel elements decreases with time, after the shutdown, with an exponential law $P = P_0\ \beta t^{-\alpha}$, where $\alpha$ and $\beta$ are numerical coefficients. According with this condition, the DHRS has to remove the largest amount of decay heat in the first time of the transient and after a certain period of time it has to guarantee a lesser effective heat removal based on the lower limits for the temperature. This is much more true in case of Lead Fast Reactor, for which the freezing temperature of the primary coolant is closer to the operative one (about 327°C). [1]

### DHR Heat Exchanger

Within the whole system, a fundamental role is played by the primary decay heat exchanger (DHX). The standard proposal for such a type of DHX foresees to use a shell and tubes heat exchanger with sodium as secondary coolant. The secondary coolant allows releasing heat to the external atmosphere and, in order to do that, it has to reach the atmosphere outside the reactor building. The innovative DHRS utilizes a radiation-based bayonet tubes DHX. In this solution the primary coolant, driven by perforated shells, flows downward in contact with the external surface of the submerged DHX. The heat transfer between primary sodium and secondary coolant is mainly based on radiation between the hot internal surface of the outer tube and the cold external surface of the inner tube where the primary coolant is in contact with the outer tube while the secondary coolant flows within the inner tube. Each bayonet tube is composed of three coaxial tubes (inner tube, intermediate tube and outer tube). The outer and the intermediate tubes are separated by a gap in which a high vacuum is obtained, in order to maximize the surface to surface radiative heat transfer mechanism (Fig. 1).

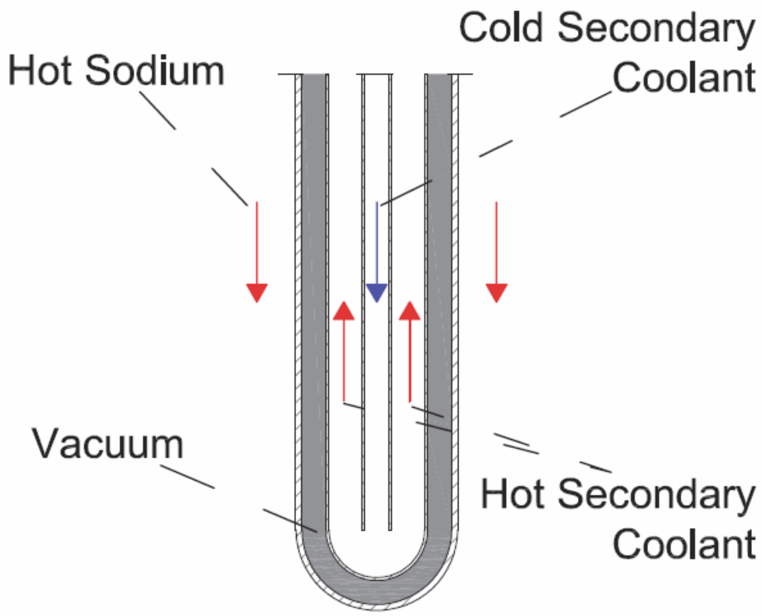


*Fig. 1. Bayonet tube concept*

The secondary fluid enters on the top by a cold header and circulates downward in the inner tube. The bottom part of the intermediate tube is closed by means of an hemispheric head while the inner tube is open at the bottom. In the lower head, the fluid turns back and circulates upward in the annular gap between the inner and the intermediate tubes. The vacuum gap allows an efficient radiative heat exchange between primary and secondary coolants. To improve thermal performance, the two surfaces separated by the vacuum gap are provided of fins (Fig. 2). The pressure in the annulus between the intermediate and the outer tubes should be continuously monitored to promptly detect any leakage from either side. The preliminary dimensions chosen for DHX tubes are reported in Tab. 1. A small inner tube diameter has been selected in order to analyze the system corresponding to the minimum DHX whole diameter. An optimization phase is foreseen in further design phase to meet fins manufacturability, pressure drop, heat transfer and overall DHX dimension requirements. A scheme of the bayonet-tubes distribution within the whole DHX is reported in Fig. 3. The design solution has been developed to be used in both Sodium Fast

Reactors and Lead Fast Reactors. In sodium application the secondary coolant could be diathermic oil, which guarantees higher safety grade with respect to water based systems, characterized by high risk of sodium water reactions occurrences. In lead applications, instead, water can easily be used as secondary coolant thanks to their compatibility. In this case the secondary circuit pressure has to be maintained high enough to ensure water to be always in liquid phase.

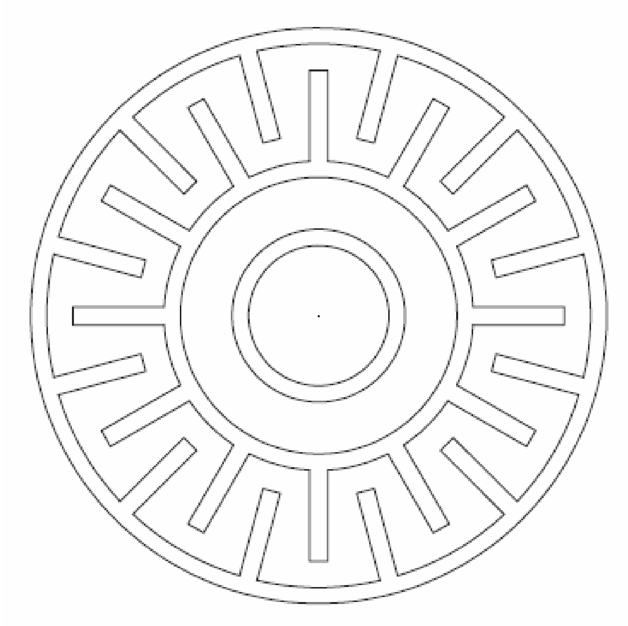

*Fig. 2. A scheme of the bayonet-tubes*

*Tab. 1. Preliminary geometrical parameters*

| | | |
|---|---|---|
| Total DHX diameter | 1.6 | [ m ] |
| DHX tube length | 5 | [ m ] |
| External tubes OD | 0.0318 | [ m ] |
| Intermediate tubes OD | 0.0171 | [ m ] |
| Inner tube OD | 0.0095 | [ m ] |
| Tubes thickness | 0.0009 | [ m ] |
| Annular thickness | 0.0065 | [ m ] |
| Fin height | 0.0050 | [ m ] |
| Fine thickness | 0.0010 | [ m ] |
| Fins for each surface | 12 | |
| Tube height | 5 | [ m ] |
| Pitch/diameter ratio | 1.3 | |

As said, oil is an attractive fluid since it does not react violently with sodium and for its heat transfer capability. In order to avoid oil cracking and system pressurization, the oil has to operate below 385°C. For the simulations a synthetic and aromatic oil has been used, namely Therminol®75. It has a liquid operating temperature range between 80°C and 385°C. Thermal-physical properties used for CFD analyses have been taken from a technical bulletin for the oil [2] and from a technical report by Argonne National Lab for Sodium [3].

## CFD analysis

To obtain a numerical characterization of the system, a CFD analysis for Sodium/Oil fluids has been performed. The analysis has been carried out with the aim of evaluating the influence of the temperature of the hot fluid on the thermal power exchanged by DHX.

### Simulation domain

Simulations have been performed using a commercial CFD code based on the finite volume method. The computational domain reproduces one twelfth of the actual geometry of a single element, exploiting the symmetry which characterizes the hexagonal sub-channel. A schematic representation of this domain is shown in Fig. 4. For turbulence modeling the RNG $\kappa-\varepsilon$ model with wall functions has been used. The surface-to-surface radiation (S2S) model has been used in this simulation. This model can be used to take into account the radiation exchange in an enclosure of gray-diffuse surfaces. According to this model the energy exchange between two surfaces depends in part on their size, separation distance and orientation. All these parameters are accounted for by a geometric function called *View Factor*. The main assumption of this model is that any absorption, emission or scattering of radiation can be ignored, therefore only surface-to-surface radiation need be considered for analysis. The mesh is composed of $1.1\times10^6$ prisms with a triangular base, characterized by minimum dimensions variable according to the subdomain considered. A streamwise section of the mesh is shown in Fig. 5. shown in Fig. 5. A simulation flow time of 150 s has been used for the Sodium/Oil system in order to reach a steady state condition; for this purpose a time step of 0.03 s has been selected.

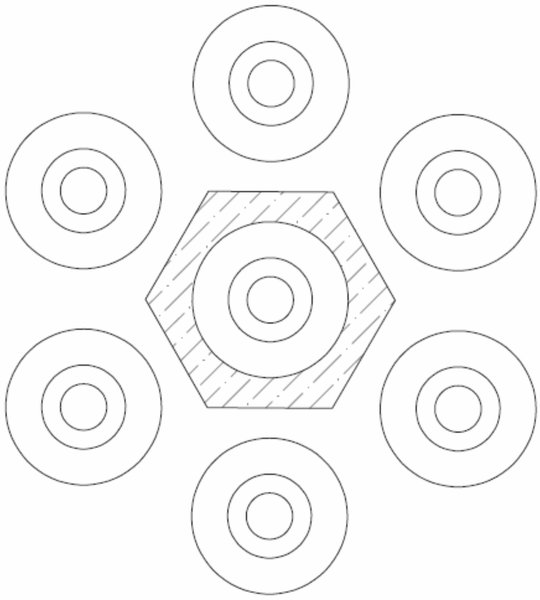

*Fig. 3. Bayonet tubes distributions*

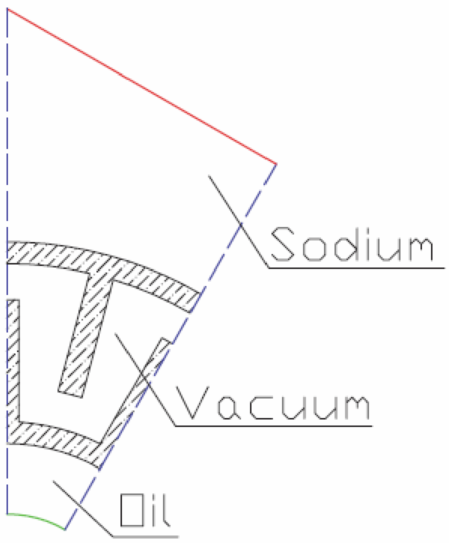


*Fig. 4. Computational domain section*

## Boundary conditions

The boundary conditions, referred to Sodium/Oil system, are reported in Tab. 2.

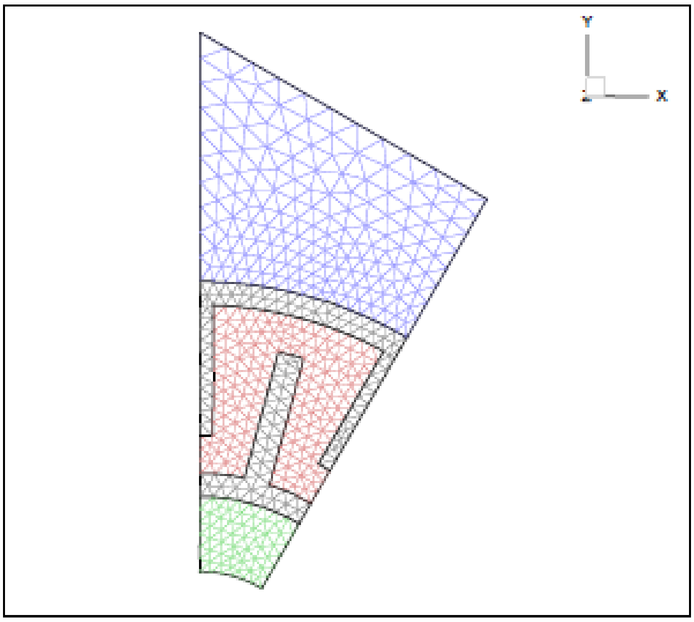


*Fig. 5. Mesh section*

*Tab. 2. Sodium/Oil system boundary conditions*

| | | Thermal Oil | Sodium |
|---|---|---|---|
| Inlet | | | |
| | Velocity [m/s] | 0.167 | 0.179 |
| | Temperature [K] | 423 | 623 to 923 |
| | $D_H$ [m] | 0.00574 | 0.014 |
| | T.I.[a] | -[b] | 4% |
| Outlet | | | |
| | Rel. pressure [Pa] | 0 | 0 |

[a] Turbulence intensity

[b] Laminar Flow

Concerning the computational domain, the following boundary conditions, referred to Fig. 4, have been applied:

- blue surfaces: *Symmetry*
- green surface: Wall - No Slip
- red surfaces: Wall - Free Slip

The finned surfaces' emissivity has been conservatively chosen equal to *0.8*.

The oil flows in laminar condition since the inlet Reynolds number is about 600 while for the Sodium side the flow is turbulent ($Re$ = 8300).

Further simulations involving Lead/Water system will have to be performed, imposing appropriate condition of similarity with respect to the Sodium/Oil System, with the aim of comparing different solutions.

# Results

In order to evaluate the system performance several simulations have been performed. Since the heat transfer mechanism is mainly based on the radiation phenomenon, the most important parameter is the inlet temperature of the hot fluid. For this reason different values of the hot fluid inlet temperature have been analyzed. The heat transfer performance at different inlet temperature has been compared with a typical DHX suitable for SFR ($T_{Na-inlet}$ =550°C).

## Sodium/Oil System

The Tab. 3 and the Fig. 6 show, for each case, the inlet temperature of the sodium and the exchanged thermal power in a single element.

As shown in Fig. 6, the thermal power exchanged as function of Sodium inlet temperature can be well fitted by $\dot{Q} = 1.909\times10^{-8}T^{4.015}$. The fourth degree function is coherent with the main heat transfer mechanism (radiation) that characterizes the system.

*Tab. 3. Sodium/Oil system*

| $T_{inNa}$ [K] | Q[W] | $T_{outOil}$ [K] |
|---|---|---|
| 623 | 3216 | 480.32 |
| 723 | 5670 | 520.49 |
| 823 | 9548 | 579.98 |
| 923 | 15624 | 665.03 |

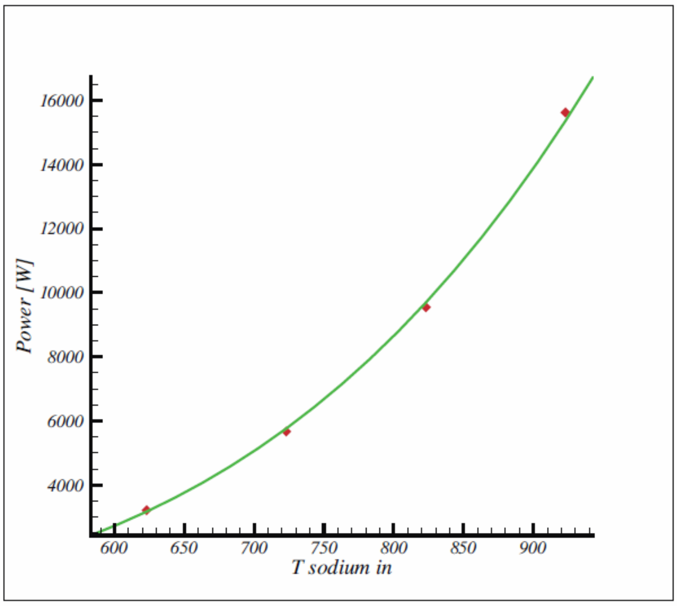


*Fig. 6. Sodium/Oil System - Power vs. Temperature*

Fig. 7 shows the evolution of mean temperatures of the two fluids as function of the heat exchanger height for the reference case. It can be noticed that the trend seems like that usual for a classical counter-flow heat exchanger. Oil temperature at the outlet section changes as reported in Tab. 3. As shown in Tab. 3, the maximum oil temperature remains within the operating temperature range in all cases except the last one. In this condition the oil cracking rate could be too high and a purification system would result necessary. Concerning the vapor pressure, a slight pressurization can be foreseen in order to maintain the oil in a single phase condition. In any case this condition corresponds to 650°C that is not a realistic condition for SFRs. To provide a visual representation of axial and radial temperature fields, the temperature contours for the reference case are reported in Fig. 8. Pitch between each slice is equal to 0.5 meters.

## Lead/Water System

Further studies will involve using of radiation-based bayonet-tubes DHX for Lead Fast Reactors. A LFR usually operates in a different and lower range of temperature respect to the Sodium Fast Reactor. In particular the ELFR project operates in the range 400/480°C inlet/outlet core during normal operations [4]. As known the radiative heat exchange is strongly influenced by the highest temperature of the system. It is reasonable to expect a less efficient radiative heat exchange with respect to the Sodium system. On the other hand the absence of high reactivity features with water allows using this coolant in place of diathermic oil. The use of water imposes the pressurization of secondary side to avoid vaporization. Considering an inlet temperature for Lead side equal to 480°C and an inlet temperature for Water side equal to 150°C, the pressure of secondary side should be about 70 bar ($T_{sat}$ = 285.8°C) to allow temperature variation of the water greater than 100°C.

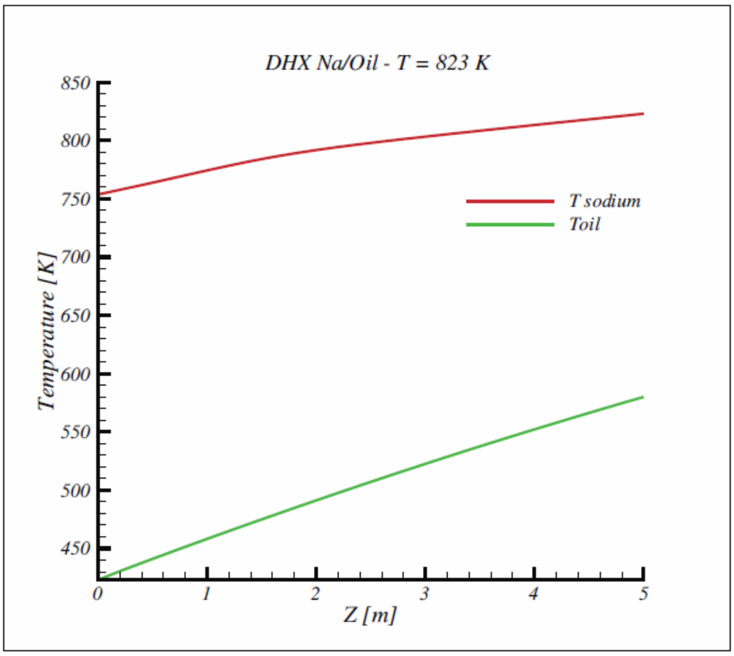


*Fig. 7. Sodium/Oil System - Power vs. Height*

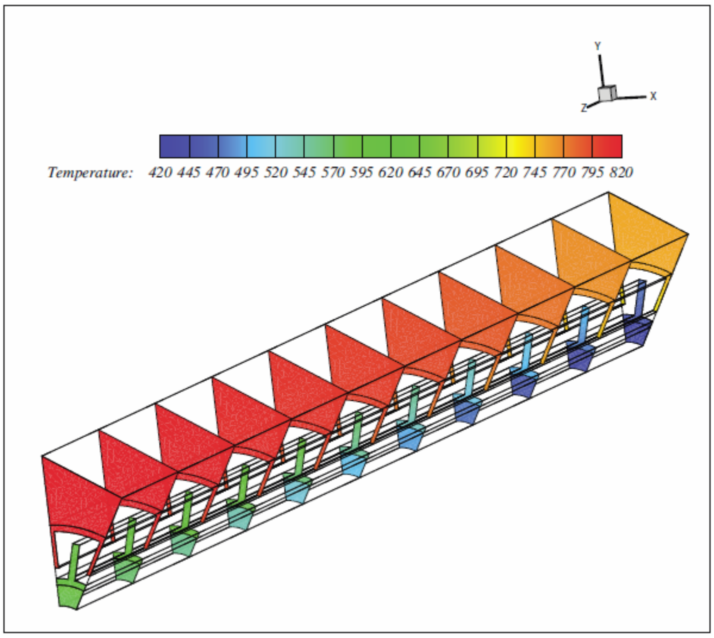


*Fig. 8. Sodium/Oil System - Temperature Fields*

## Conclusions

In order to evaluate the performance of a radiation-based DHX, several CFD simulations have been performed. In particular, with reference to the SFRs, the response of the heat exchanger to temperature variations of the hot fluid has been evaluated. The data obtained for the exchanged thermal power as a function of the temperature of sodium can be fitted with a fourth degree curve, confirming the dominant character of irradiation in the operation of the device. Hot fluid temperature influence makes the heat exchanger particularly suitable for application in the nuclear field as DHX especially in next generation of nuclear power plants. This is because the features of the device allow to easily adapt the system to the power decay of the core. The proposed DHX coupled with an appropriate heat sink for the secondary fluid, as described in [5], allows to remove the decay heat indefinitely in time without any external intervention. Further developments of the project will include the geometry optimization and the engineering of the heat exchanger. Additional CFD analyses will be performed in order to verify the behavior of the DHX with different primary and secondary fluids such as lead or lead-bismuth eutectic and water.

## Nomenclature

| | |
|---|---|
| $D_H$ | Hydraulic Diameter [m] |
| *DHRS* | *Decay Heat Removal System* |
| *DHX* | *Decay Heat Exchanger* |
| *ELFR* | *European Lead Fast Reactor* |
| $\dot{Q}$ | *Total thermal power for a single DHX element [W]* |
| *SFR* | *Sodium Fast Reactor* |
| $T_{inNa}$ | *Sodium Inlet Temperature [K]* |
| $T_{outOil}$ | *Oil Outlet Temperature [K]* |
| $T_{sat}$ | *Saturation Temperature [°C]* |

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# 6 JOINED APPLICATION OF DEVELOPED SAFETY SYSTEMS

## 6.1 Paper on the application of the radiation-based DHX to the innovative pool

Journal: International Journal of Risk Theory

Year: 2012

Volume: 2 (no.1)

Pages: 11 - 31

Title: Proposal of a high reliability DHR system for NPPs with never ending capacity

Authors: G. Caruso, L. Ferroni, F. Giannetti, A. Naviglio, D. Vitale Di Maio

## Abstract

The present paper aims at describing a totally innovative top-reliability Decay Heat Removal (DHR) system suitable for application in Nuclear Power Plants (NPPs). It is completely passive and with a never ending capacity. The configuration here described is mainly addressed to liquid metal cooled nuclear reactors, but is based on a sub-system also applicable to any type of nuclear power plant and even to some chemical plants. The main subsystems of the innovative DHR System are patent pending.

The paper is mainly based on the work carried out in the framework of the CP ESFR (European Sodium Fast Reactor) project (7th R&TD FP of the EU); the research leading to these results has received funding from the European Community's Seventh R&TD Framework Programme (FP7/2007-2011) under grant agreement n° 232658.

The innovative DHR system is based on a Direct Reactor Cooling System (DRC): the DRC is basically made of a loop (in this case, with oil as coolant) receiving nuclear decay heat through a Radiation-based bayonet Decay Heat Exchanger (DHX) and transferring the heat through natural circulation to an innovative external Air cooled Heat Exchanger (AHX), using water and air as heat sink during the cooling phase. The patent pending DHX is a sodium / vacuum / oil Bayonet-tubes type. The thermal power is transferred from primary nuclear coolant (in the paper, sodium) to oil of the DRC loop through radiation in the vacuum gap of the bayonet.

The patent pending AHX is an oil/ water-air pool heat exchanger, which transfers thermal power from oil to different fluids: water at the beginning of the transient, when the decay thermal power to remove is the highest; then, following the decrease of the decay heat, and after the vaporization of the boiling water, thermal power is transferred from oil to air (that replaces water in the pool without any external intervention). This AHX was conceived and developed for top-reliability cooling in PWRs, but has a generalized applicability. Preliminary analyses have demonstrated the reliable operation of the two innovative sub-systems (DHX and AHX) coupled in the SFR project, in an absolutely reliable natural circulation DRC.
The absence of risks due to anticipated failure of the barriers either/both on the primary or/and on the secondary side of the DHX, due to the vacuum gap presence as well as the perfect compatibility of the secondary oil with the primary coolant (sodium), make this concept a very attractive solution for the diversified ESFR DRC passive system. The same concept is applicable to Lead Fast Reactors as well. It is interesting to underline that the innovative DHR concept is easily implemented in every industrial plant where, should an hazardous accident occur, a heat removal system can avoid more dangerous consequences (i.e.: in Nuclear Power Plants, in severe accident management, eventually substituting oil with other coolant; in chemical plants to manage run-away reactions).

## Decay Heat Removal relevance in nuclear power plants

It is well known that during operation of a nuclear power plant, heat is generated in the nuclear fuel (heat generation in other components is irrelevant in the scope of this paper). The main contribution to heat generation is due to nuclear fission reactions, originated by interaction between neutrons and fissile isotopes ($U^{235}$ and $Pu^{239}$ in thermal reactors, or several isotopes of Uranium and other transuranic elements in fast reactors); a typical fission reaction is shown in Fig. 1.

$${}_{0}n^{1} + {}_{92}U^{235} \rightarrow {}_{92}U^{236} \rightarrow {}_{56}Ba^{137} + {}_{36}Kr^{97} + 2{}_{0}n^{1}$$

*Fig. 1. Example of a typical fission reaction*

In the fission reaction shown (a high probability fission reaction, applicable to thermal reactors but representative of any fission reaction), a neutron interacts with a heavy isotope (in this case, U235), an instable nucleus of U236 is generated, which breaks into two “fragments”, namely two medium weight isotopes (in this case, Ba137 and Kr97) and two fast neutrons.
Since in the reaction a loss of mass occurs (of the order of 0,2 atomic mass unit, that is the ratio between 1 g and the Avogadro number, the number of atoms in a g-mole), this is translated into energy release (about 200 MeV per fission).
Most of this energy is immediately released as kinetic energy of the reaction fragments (including fast neutrons; about 85%) and this causes heat generation in the fuel assemblies (and moderator, if present); a part of the energy (about 4%) is released through immediate emission of fission gamma rays and a part of energy is “accumulated”

in the form of a potential energy in the nuclei of main fission fragments, which are in a unstable configuration. This instability is the cause of modifications in these nuclei, with new nuclear reactions occurring in time, leading to nucleus modification and to the delayed emission of energy (this process may occur in several steps, involving several “fission products”). As shown in Tab. 1, delayed energy emission is mainly associated to electrons (beta radiation) and additional gamma rays; this process is named isotope decay and takes time to occur (each isotope decay is characterized by a given type and energy emission, and by a given “average” time, identified through the “half-life” of that decay). This means that, after fission, a “decay heat” continues to be generated in the fuel.

*Tab. 1. $U^{235}$ fission energy distribution*

| | MeV/fission | % |
|---|---|---|
| Instantaneous energy release | | |
| • Kinetic energy of fission fragments | | |
| • $\gamma$ rays energy | 169 | 83 |
| • Kinetic energy of neutrons | | |
| | 8 | 4 |
| Delayed energy | 5 | 2.5 |
| • $\beta$ from fission products | | |
| • $\gamma$ from fission products | | |
| • $\gamma$ from neutronic captures | 8 | |
| | 7 | 3,5 |
| Tot | 7 | 3,5 |
| | | 3,5 |
| | 204 | |
| | | 100 |

An additional contribution to the “decay heat” in nuclear fuel involves the absorption of neutrons mainly by high weight isotopes (namely actinides or other isotopes of heavy elements): also in this case, the absorption leads to the generation of isotopes not stable, subject to modifications in their nucleus, with the delayed release of radiations (gamma rays, electrons, etc.), so of thermal power. A typical neutron absorption reaction is show in Fig. 2.

$$_{92}U^{238} + {}_{0}n^{1} \longrightarrow {}_{92}U^{239} + \gamma$$

$$_{92}U^{239} \xrightarrow[23\ m]{\beta} {}_{93}Np^{239} \xrightarrow[2,3\ g]{\beta} {}_{94}Pu^{239}$$

*Fig. 2. Example of a typical neutron capture reaction and following decays*

For these two causes, one directly connected to fission, the other one to neutron capture, in nuclear fuel heat continues being generated also after the reactor shutdown, i.e., after the end of the fission process.

In Fig. 3, a typical decay heat generation curve is shown: at the shutdown time from a rated power generation, the decay heat drops instantaneously to some 7% of the rated power; after one hour it is reduced to some 1,5%, after one day to some 0,5%. In case of a nuclear reactor generating 3000 MWth in the core at rated power (corresponding to some 1000 MWe), after one day the generation is 15 MWth,

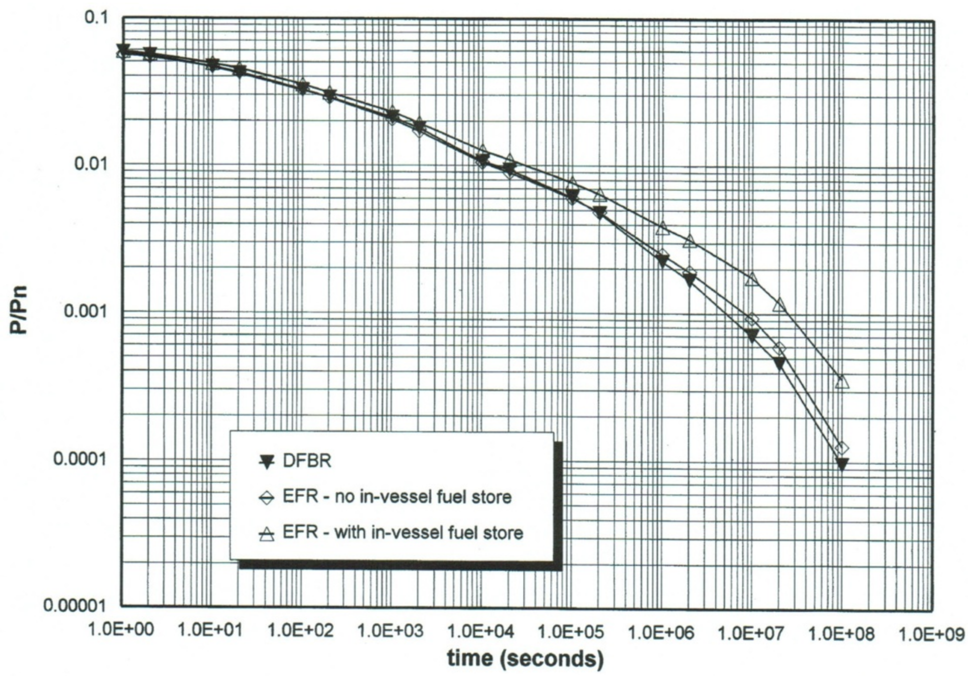



*Fig. 3. Example of typical decay heat generation curve for SFR*

Decay heat has to be removed, even in nuclear plant accidental conditions, to guarantee fuel integrity, so avoiding quite worse hazards.

In fact, it has to be remarked that the isotopes generating the decay heat are the same isotopes responsible of the release of radioactivity into the environment in case of an accidental scenario involving their release into the environment (not all isotopes have the same behaviour in the release phase and their effect on biosphere is characteristic for each isotope).

In our scope, what is essential is that in the reactor core of nuclear reactors, when a shutdown occurs, the fission chain reactions are stopped but, because of decay heat release by fuel, thermal power generation continues in the fuel at considerable levels for a long time. This power must be absolutely removed in order to guarantee fuel cooling and, consequently, its integrity, without release from fuel cladding of radioactive isotopes and, even worse, without cladding and fuel melting (severe accident). Therefore, in order to avoid unacceptable fuel heating, in all NPPs suitable systems aimed at removal of Decay Heat are always foreseen.

Generally, in non-accident conditions, after a reactor shutdown Decay Heat removal is initially guaranteed through the nuclear heat utilization system (steam generators feeding plant turbine), while in a second phase a dedicated heat removal system enters into operation, bringing nuclear coolant temperature to levels compatible with the refuelling operations. Nevertheless, due to the extreme importance of Decay Heat Removal even in accident conditions, dedicated safety systems (safety-grade Decay Heat Removal Systems - DHRS) are foreseen and their design specifications are coherent with the highest safety standards.

Most existing safety DHR systems rely on the operation of cooling circuits including pumps, heat exchangers, valves; part of the primary coolant is spilled and it releases heat to a second fluid in a closed circuit. This second fluid, in turn, releases heat to another fluid, that can be the final heat sink (river or sea water; air through cooling towers, etc.). Many active components allow the operation of the safety DHR system, and electric power is needed both for the operation of pumps and for control devices and instrumentation. A safety electric supply system is needed; the application of the highest design codes and the application of criteria of redundancy and diversification contribute to the reliability of the safety Decay Heat Removal system.

Nevertheless, in a new perspective of plant safety, very high reliability levels may be achieved avoiding the bottleneck of availability of electric supply systems: the safety is based, in this case, not bringing to the extreme consequences the active system duplication concept, but identifying a robust design based on the use of simple physical laws, simple components, natural circulation of fluids, with a never ending cooling capability.

This approach has been applied in the proposed innovative DHR design, with a specific application in SFRs.

## Design criteria for safety DHR systems in IV Generation

General safety design criteria in ESFR project include design margins, diversity, redundancy, structural protection and physical separation of the safety relevant systems, structures and components and the effectiveness of the defence-in-depth concept.

As already stated, the main goal of the safety DHR system is to provide a safe heat removal from the primary system to assure, in case of reactor shutdown or trip, in any plant condition, the integrity of the reactor core, remaining at an acceptably low temperature.

The function of decay heat removal can be achieved through different systems, characterized by active or passive components. Both types of these safety systems must have all components in a safety grade class but the main difference, between active and passive systems, is due to their dependence on external mechanical or electrical power, signals or forces. Thanks to the reliance on natural laws only, passive systems require a lower number of components for a same system reliability target.

This is why new DHRs are often based on passive systems, in order to obtain higher safety performance with fewer components, according with one of the main goals for Generation IV plants that is “to eliminate the technical need for off-site emergency response”.

In particular, to guarantee an adequate and reliable core decay heat removal function when the reactor is in shutdown conditions, the following safety features are required to the dedicated system:

- to guarantee a long term core cooling;
- to maintain structural integrity and operability of primary circuit components.

To fit with such features, the design criteria for the Decay Heat Removal system must include:

- Reliability, system diversity and independence, redundancy

  The passive safety DHR system must include redundant systems, preferably based on different concepts and components, designed so as to avoid common mode failures (in this paper, one possible DHR system is described)

- Maintainability

Since safety DHR systems may be required to guarantee the highest design reliability for a long period, the components must be easily maintainable: easy accessibility and a limited number of components to be maintained are, obviously, important.

Referring to the design criteria for safety DHR systems in IV Generation NPPs, ESFR DHR systems are to achieve the DHR function with a Direct Reactor Cooling (DRC) system, composed of passive and active loops, by means of dip coolers (DHX), that extract heat from the primary sodium vessel, as shown in Fig. 4 for a pool-type EFSR NPP reactor tank.

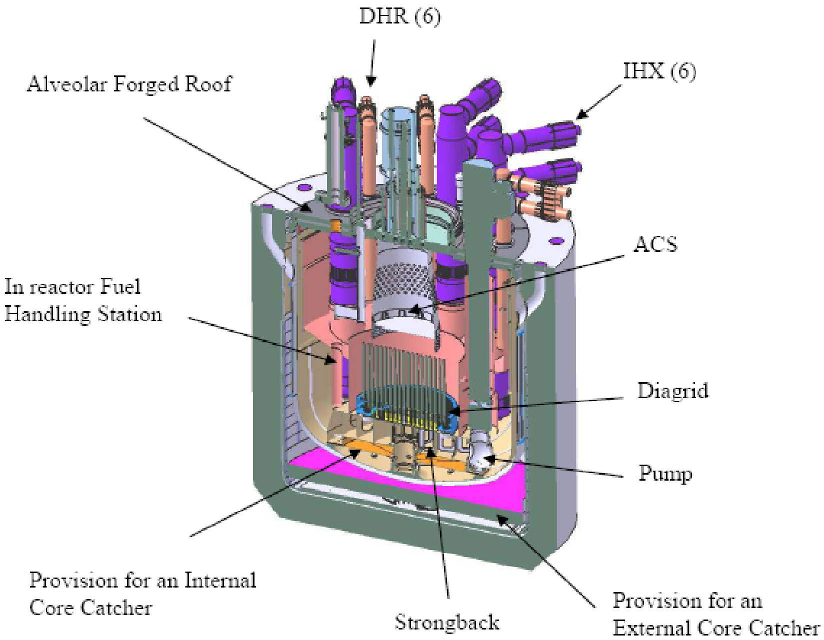



*Fig. 4. Pool EFSR NPP tank scheme*

All loops extract heat from the primary sodium of the hot pool (T Na = 530°C) by means of immersed sodium/sodium dip coolers (DHX), and reject the heat to the environment using sodium/air heat exchangers (AHX) located, at a certain elevation, near the reactor building; a simplified scheme is shown in Fig. 5.

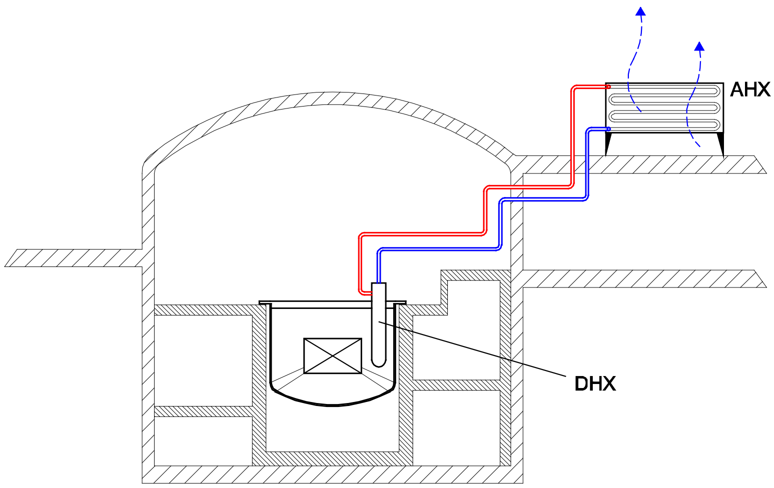



*Fig. 5. Pool EFSR NPP DHR (DHX + AHX) scheme*

With reference to safety-grade Direct Reactor Cooling, a new concept for an ESFR passive, inherently safe Decay Heat Removal (DHR) system is described in the following: both the DHX and the AHX are based on innovative concepts; in the following, the innovative AHX system is called PHX.

The DRC loop includes a Radiation-based bayonet DHX and an innovative AHX system, using water and air as heat sink.

The DHX is a sodium / vacuum / oil Bayonet-tubes type. Thermal power is transferred from primary sodium to secondary oil through radiation in the vacuum gap of the bayonet.

The PHX is an oil/ water-air pool heat exchanger, which transfers thermal power from oil to water at the beginning of the transient, when the decay thermal power to remove is the highest. Then, following the decrease of the decay heat, it starts transferring heat from oil to air that replaces water in the pool without any external intervention.

# Innovative DHR description

## Introduction

The innovative safety DHR here proposed refers, as for the geometrical and functional data, to conditions indicative of the ESFR design. “Easy” modifications may apply in case of different operational specifications.

The proposal of the innovative safety DHR system is based on the consideration that because of the limited number of components involved in passive systems and of the independence from availability of power at the site and from human intervention, they may, in principle, lead both to economic advantages and to safety enhancement.

Safety systems must have at least two different states: the first one corresponds to the normal plant operating condition and the second one corresponds to the safety operating condition. In order to change from normal to safety condition, and to be able to operate in safety conditions, it will be necessary to have: a signal to detect the need, as well as a driving force to change the state and the necessary means to operate in the new state. If these three features rely on passive components, the system may be defined as passive.

Moreover, referring to passive systems, an important difference between moving and static systems exists. The more similar the two operating states of a system are, the less failure modes of the system will be. According to this, passive systems are classified, by IAEA, in four different classes (from lower to higher passive characteristics) depending on their main features, i.e. mechanical moving parts, moving fluids, requirement of input signals.

The innovative system here proposed, is characterized by only moving fluids.

The DHX is made of bayonet tubes with oil as secondary fluid and vacuum in the gap (patent pending). The thermal power is transferred from primary sodium to secondary oil through radiation in the vacuum gap of the bayonet (in this paper, oil is the reference fluid for the secondary circuit but also other fluids and geometries may be considered, in order to optimize the solution). Thanks to the special configuration of the DHX, the contact between primary sodium and secondary oil can be considered impossible, because the vacuum gap allows detecting any possible leak both from the primary and the secondary sides. In any case, oil is an attractive alternative fluid instead of water or air, because it does not react violently with sodium and it presents interesting heat transfer capability (better than air, but worse than water). Nevertheless, oil has to flow at temperatures lower than 350-400°C (and also beyond 400°C, depending on the specific oil used) because, beyond the maximum allowable temperature, there is a risk of oil cracking.

The PHX is a special oil / water – air pool heat transfer system (patent pending as well), already proposed in an innovative LWR design. It allows transferring thermal power from oil to water in the pool, when the decay power has its highest value (first phase of the transient). Then, when the water content in the pool has partially/totally vaporized, air replaces water inside the pool allowing a never ending heat removal. High heat transfer efficiency is guaranteed also with air by a special design of the pool heat transfer system.

The whole system operates in natural circulation.

A simplified scheme of the proposal innovative safety DHR system is shown in Fig. 6.

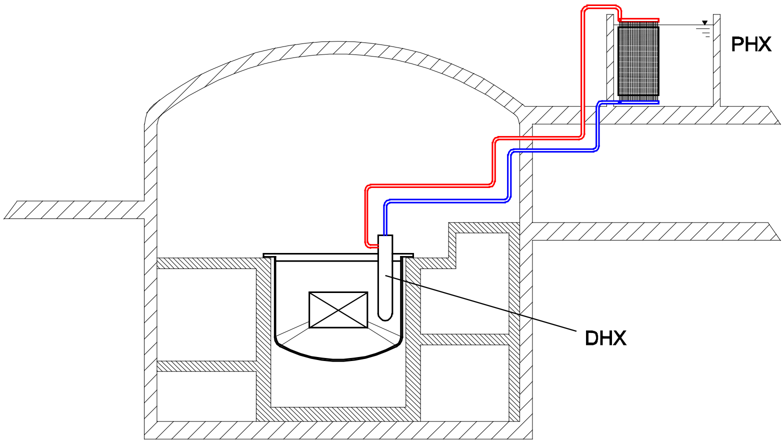



*Fig. 6. Simplified scheme of the proposal innovative safety DHR system*

The preliminary innovative DHR design has been carried out referring to thermal power initially to remove equal to 15 MW through each loop.

## DRC loop description

The proposed DRC loop is schematically shown in Fig. 7: the primary sodium flows downward in contact with the external surfaces of the Radiation-based Bayonet DHX immersed in the sodium tank. The secondary fluid is oil (alternative fluids may be considered).

Heat transfer between primary sodium and secondary oil is mainly based on radiation between the hot external tube surfaces, in contact with sodium, and the cold intermediate tube surfaces, in contact with oil. To allow better thermal performance, high vacuum has been foreseen in the gap between the outer and the intermediate tubes. Furthermore, the radiation surfaces have been designed finned to increase the heat transfer area.

The secondary / tertiary fluid heat exchanger is an innovative oil/water-air pool HX (PHX; again, oil is foreseen in this paper, but other fluids may be considered as well). The heated oil, which comes from the bayonet DHX, flows inside a tube bundle immersed in the PHX pool, which, at the beginning, is full of water. The PHX allows transferring thermal power from oil to water, when the decay power has its highest value (starting of the transient). Water at first is heated, than boils and finally air replaces water within the pool, allowing a never ending heat removal.

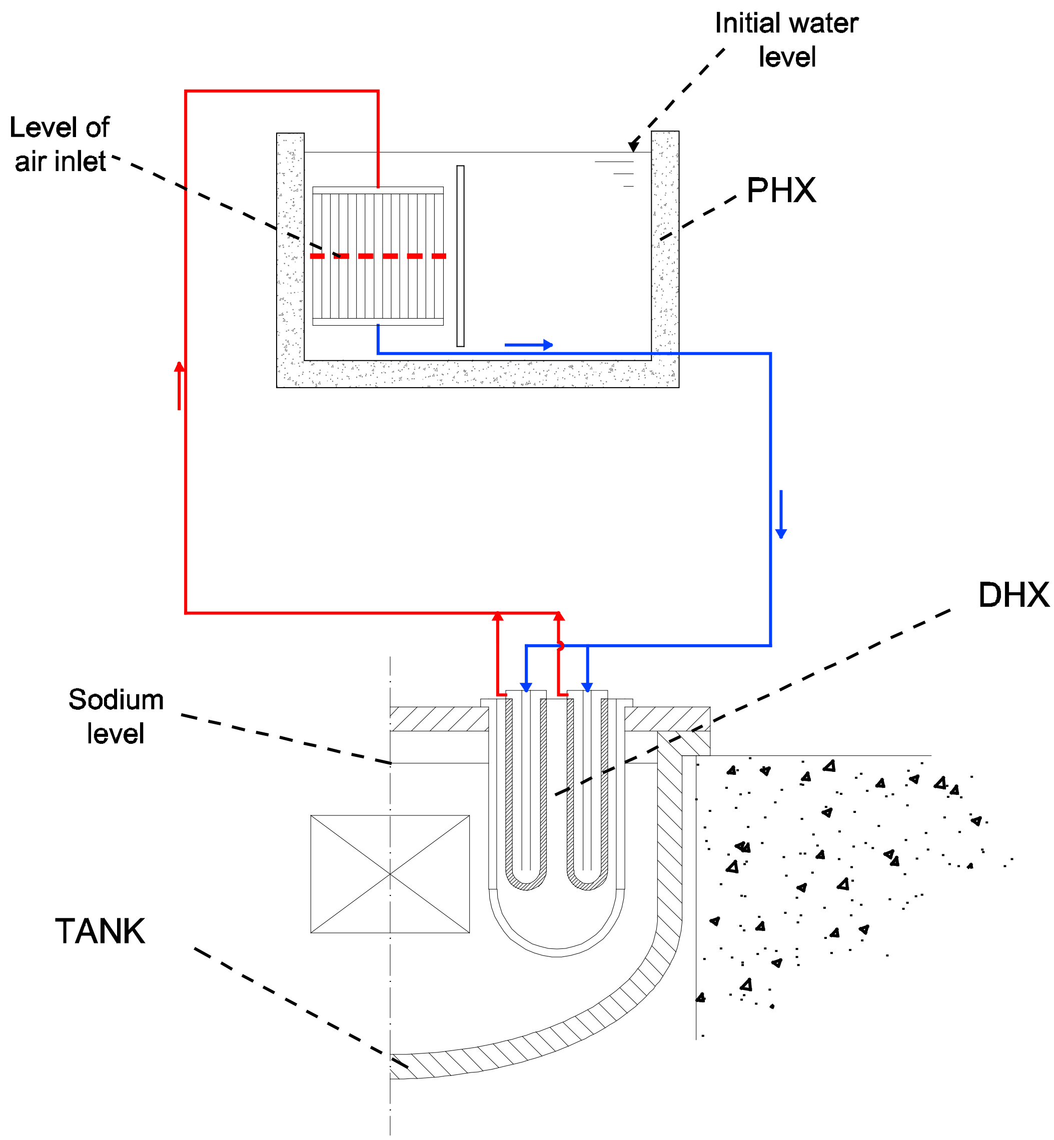



*Fig. 7. Simplified operational loop of the innovative DRC*

## The new DHX Concept description

The proposed concept is a bayonet tube exchanger which uses oil as secondary fluid.

Each bayonet tube is composed of three co-axial tubes (inner tube, intermediate tube and outer tube). The outer and the intermediate tubes are separated by a gap in which high vacuum is obtained as shown in Fig. 8.

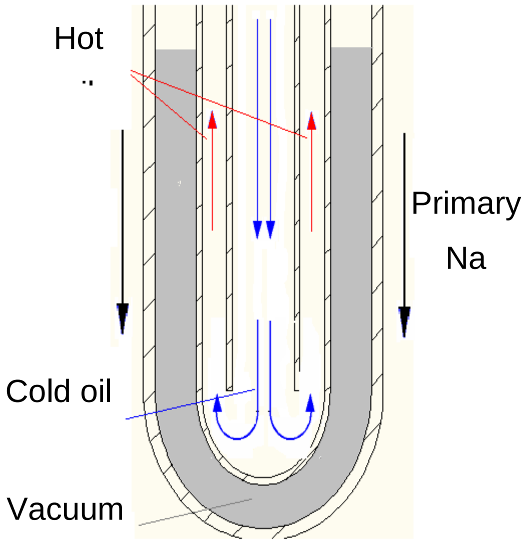



*Fig. 8. Na / vacuum / oil bayonet tube concept*

The secondary fluid (thermal oil) enters on the top from a cold header and circulates downward in the inner tubes. In the tube lower part, oil turns back and circulates upwards in the annular gap between the inner tube and the intermediate tube. Oil is separated from the primary sodium by two walls. The vacuum gap, within these two walls, allows an efficient radiation heat transfer between the internal surface of the outer tube and the external surface of the intermediate tube. Fig. 9 shows a possible type of fins to improve radiant heat transmission.

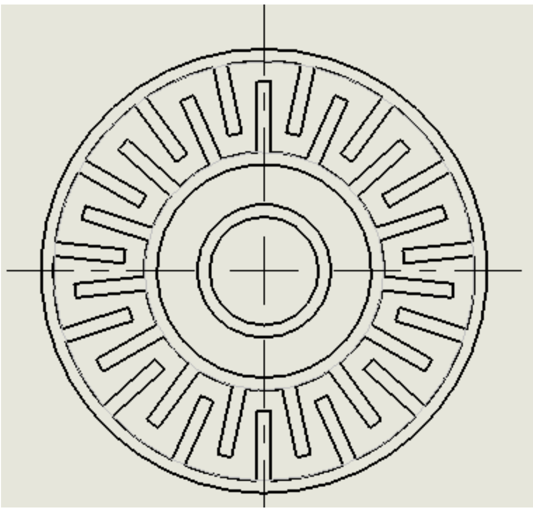


*Fig. 9. Na / vacuum / oil bayonet tube section*

Pressure in the vacuum annulus between the outer and the intermediate tubes may be continuously monitored to promptly detect a leakage from either wall.

Primary sodium circulates downwards outside the tubes bundle, guided by an outer perforated shell, as shown in Fig. 10.

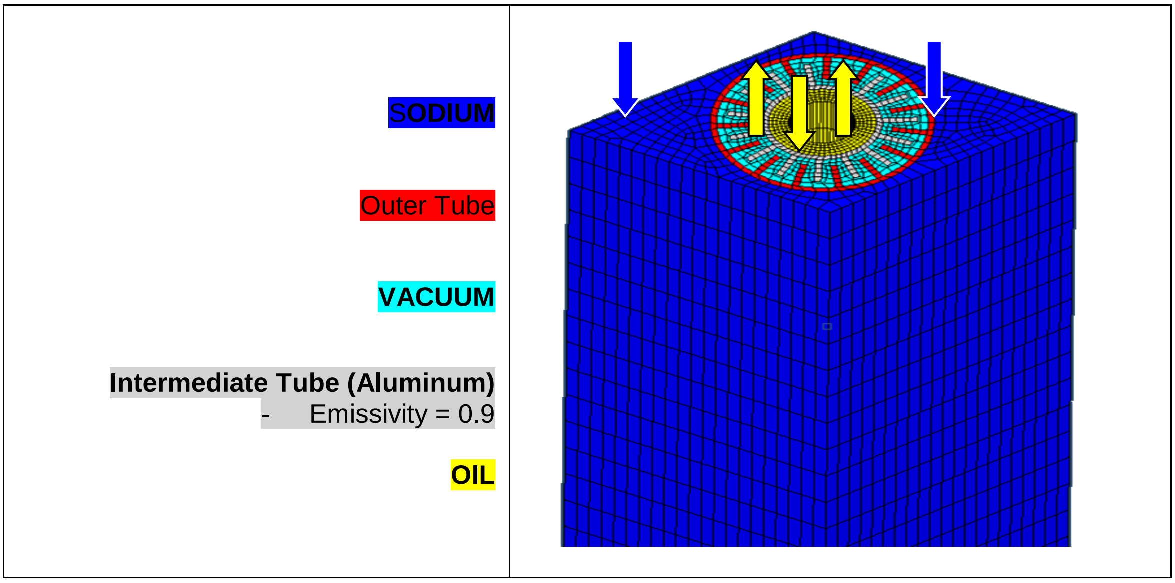



*Fig. 10. Simplified operational mode of the innovative DHX dipped in the tank*

The heat transfer, allowed by this kind of bayonet tubes, can be divided as follows:

- Convection in the primary fluid (sodium);
- Conduction in both walls between oil and sodium;
- Radiation in the vacuum gap;
- Convection in the secondary fluid (oil).

In Fig. 11, a tubes bundle section is shown, where oil headers (hot and cold headers) and external perforated shell are shown.

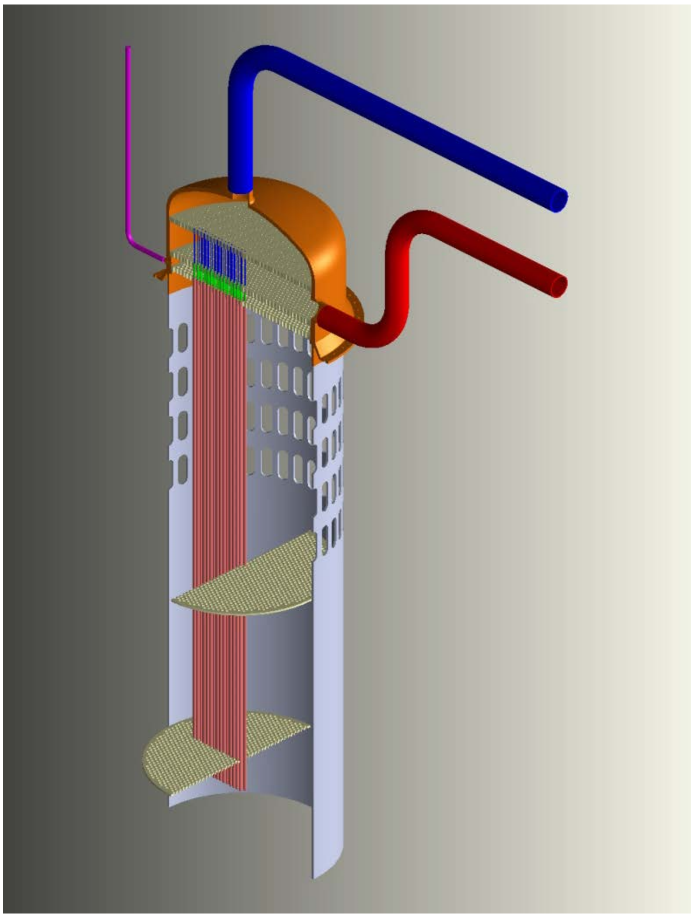
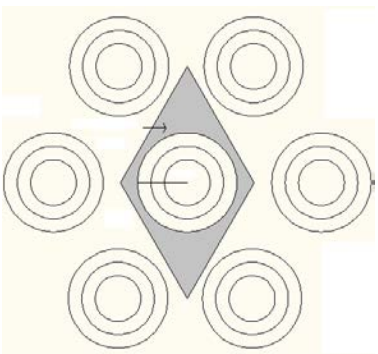


*Fig. 11. Innovative DHX tube bundle section*

### Geometry

The dimensions preliminarily chosen for the Radiation-based Bayonet DHX tubes are reported in Tab. 2 and in Fig. 12.

*Tab. 2. Radiation-based bayonet DHX main geometrical parameters*



| | | |
|---|---|---|
| DHX Tubes number | 1400 | [ ] |
| Total DHX diameter | 1.9 | [ m ] |
| DHX tube length | 5 | [ m ] |
| External Tubes OD | 0.0377 | [ m ] |
| Intermediate tube OD | 0.0200 | [ m ] |
| Inner tube OD | 0.0138 | [ m ] |
| Tubes thickness | 0.0009 | [ m ] |
| Annular thickness | 0.0065 | [ m ] |
| Fin height | 0.0050 | [ m ] |
| Fin thickness | 0.0010 | [ m ] |
| N° of fins (for each surface) | 15 | [ ] |
| Tube height | 5 | [m] |
| Pitch/diameter ratio | 1.3 | [ ] |

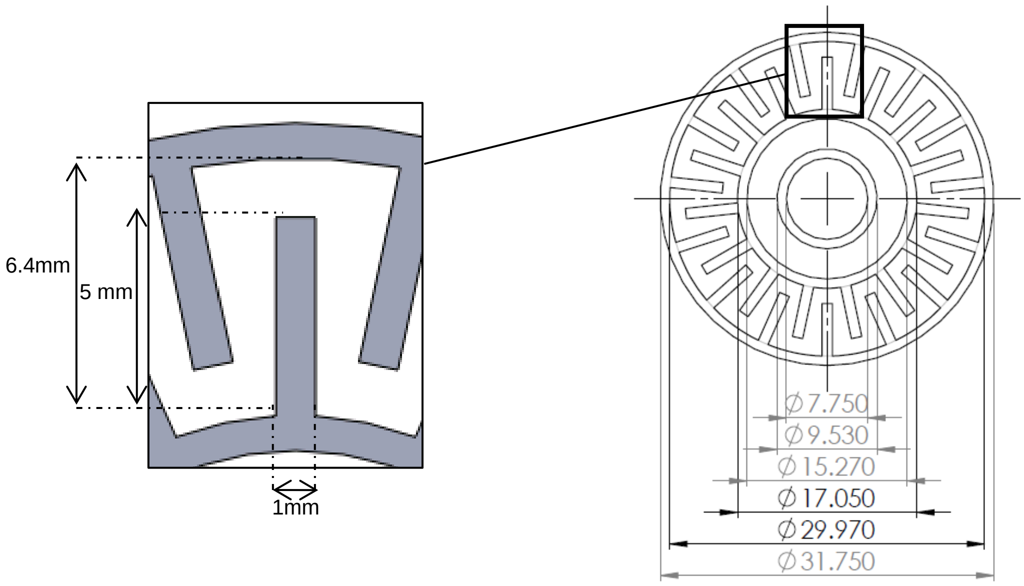



*Fig. 12. Radiation-based bayonet DHX fin dimensions*

The external and the internal tubes of each bayonet are made of Austenitic Stainless Steel. The intermediate tube of the bayonet is made of aluminum. The selected materials are compatible with fluids and simplify the feasibility and the manufacture of the bayonet tube. The emissivity of both the finned radiation surfaces (the inner surface of the outer tube made of stainless steel and the outer surface of the intermediate tube made of aluminum) has been hypothesized equal to 0.9. It is a realistic value due the possibility of surface treatments industrially achievable. Other geometries with different DHX tube dimensions have been studied. The small diameter allows to minimize the whole DHX diameter but in presence of a limited hydraulic head higher diameters may be required to guarantee acceptable performance in natural circulation.

**Secondary fluid**

Oil is an attractive fluid since it does not react violently with sodium (the possibility of interaction between the two fluids is associated to a double barrier rupture) and it presents interesting heat transfer capability.

As a possible oil candidate we refer to a synthetic heat transfer fluid used in non-pressurized / low-pressure, indirect heating systems, which has to operate at temperatures lower than 350°C to avoid cracking. The high boiling point of this thermal oil helps reducing the volatility and fluid leakage; moreover, it is noncorrosive to metals commonly used in heat transfer systems. The main thermal-hydraulic parameters of the chosen oil are reported in Tab. 3.

*Tab. 3. Reference oil thermodynamic characteristics*

| Temperature | Density | Specific heat ($c_p$) | Cinematic Viscosity | | Thermal conductivity |
|---|---|---|---|---|---|
| *[°C]* | *[kg \| m^3]* | *[J \| kg °K]* | *[cStoke]* | *[m^2 \| s]* | *[W \| m K]* |
| 25 | 851 | 1910 | 60 | 6.00E-06 | 0.1313 |
| 50 | 835 | 2001 | 21 | 2.10E-06 | 0.1293 |
| 100 | 802 | 2185 | 5 | 5.00E-07 | 0.1253 |
| 200 | 738 | 2551 | 2.4 | 2.40E-07 | 0.1173 |
| 250 | 705 | 2734 | 2.2 | 2.20E-07 | 0.1133 |
| 275 | 689 | 2826 | 2.19 | 2.19E-07 | 0.1113 |

## The innovative PHX concept

The PHX is a heat transfer sub-system characterized by a simple configuration but including special devices (patent pending) allowing an optimal use for heat removal whenever the power to remove decreases with time. It was developed for new LWRs and for applications in the chemical field.

The heat exchanger is made of several vertical finned tubes, arranged in a square matrix. Each tube has four fins with a height equal to half the space between two adjacent tubes; in this way tubes and fins give place to a number of sub channels almost equal to the number of tubes. The fins of adjacent tubes are welded in order to create closed channels; all fins of the HX tubes have loopholes in their lower zone creating communications among channels and to let the cooling fluid distribution inside channels. Views of main parts of the innovative heat exchanger are shown in Fig. 13.

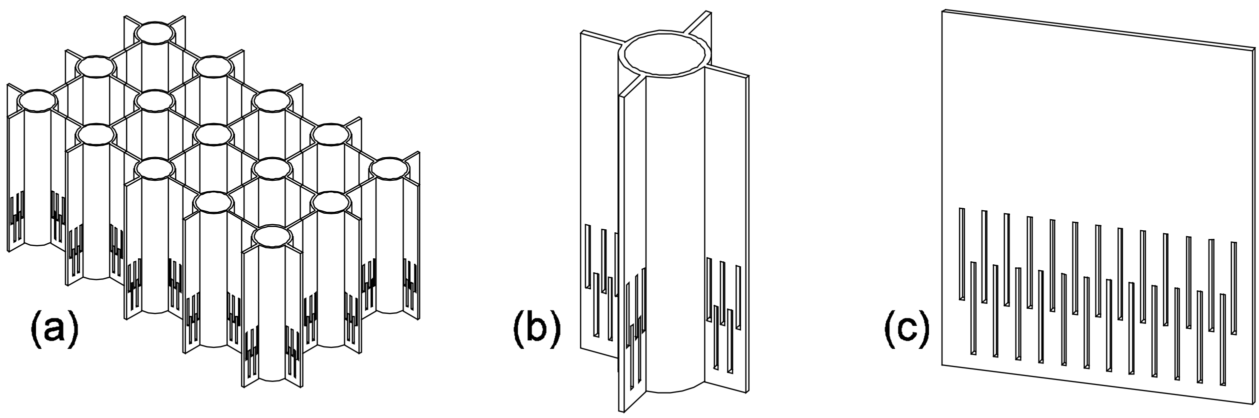



*Fig. 13. Views of main parts of the innovative heat exchanger (not designed according to real dimensions): a) an isometric view of a part of the heat exchanger, b) a single finned tube, c) the separating septum with loopholes.*

The PHX is foreseen to be dipped in a pool, initially filled with sub-cooled water, made of two different volumes; the first one hosts the heat exchanger while the second one has no mechanical components inside. The two volumes are divided by a separating septum which is equipped in the lower half part of its height by loopholes as in the case of the fins, as shown in Fig. 14.

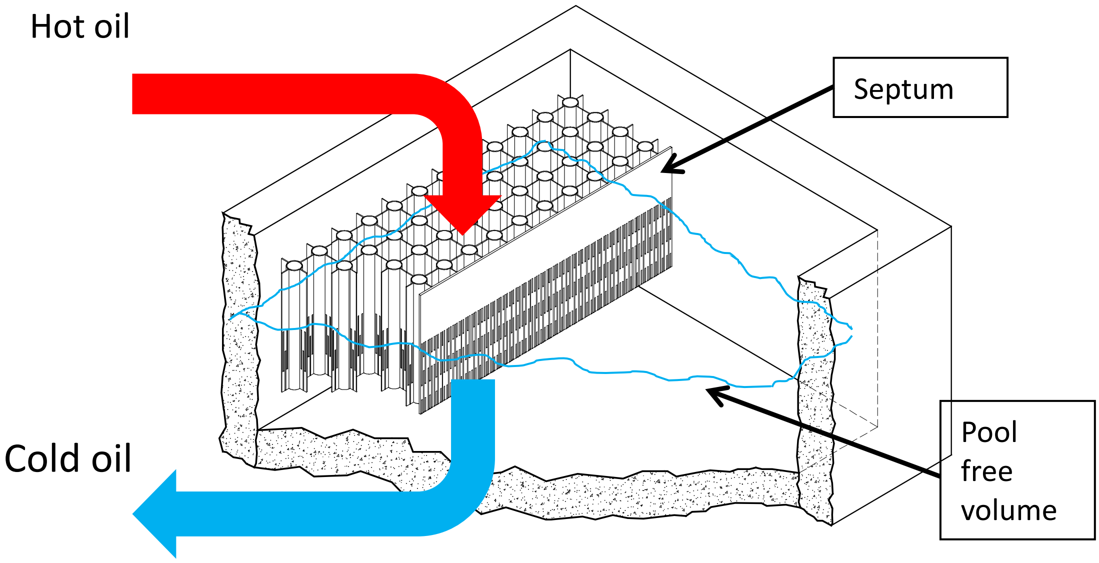



*Fig. 14. PHX pool section*

The loopholes foreseen on the septum and on the fins allow both water flow towards the sub-channels where water vaporizes, and automatically, progressive changing of the tertiary fluid, with substitution of water by air with the progressive lowering of water level, without any external intervention; this system configuration allows a never ending heat removal process.

**Operating condition description**

During normal operation of the plant, the PHX is in a stand-by condition thanks to a special component (i.e. a special passive valve, not analyzed in this paper) that limits the oil circulation in the secondary circuit avoiding a continuous release of large amount of thermal power to the heat sink. The valve characteristics must allow a suitable maximum oil temperature.

In this condition, pool water evaporation is strongly limited but, in any case, a feed water circuit with a quite limited flow is foreseen, to maintain the desired water level in stand-by conditions (no requirement of intervention for the safety DHR system).

When the system has to intervene, the valve will open, with an increased oil flow. The decay heat will be released to sub-cooled water (35°C) flowing in natural circulation through the sub-channels inside the pool, during the first transient phase; afterwards, when the water level begins to uncover the loopholes because of vaporization, the heat sink becomes a mixture of air, vapor and water; the selection of component dimensions (and of the amount of water) will allow the heat removal by air only, when the decay heat will be low enough for a completely safe operation.

The system operating condition can be divided into three different periods. Referring to Fig. 15, during the first transient period pool water temperature rises until the saturation point is reached (Fig. 15 - a). At that time, due to vaporization, the water level starts to decrease (Fig. 15 - b); during preliminary system design, the possible low feedwater refill is not taken into account for safety reasons. The second period is characterized by a "mixed" heat sink made of air together with boiling water (Fig. 15 - c); this period starts when the water level is low enough to reach and uncover the loopholes in the septum and in the fins. The third and last period starts when water is no more present and external air is the only heat sink, entering and flowing upwards into the sub-channels through the holes (Fig. 15 - d).

Thanks to the decreasing trend of the decay heat, the air heat sink capability, even if lower than the water heat sink capability, will result high enough to remove the decay heat, keeping the oil temperature (secondary fluid) under the upper limit.

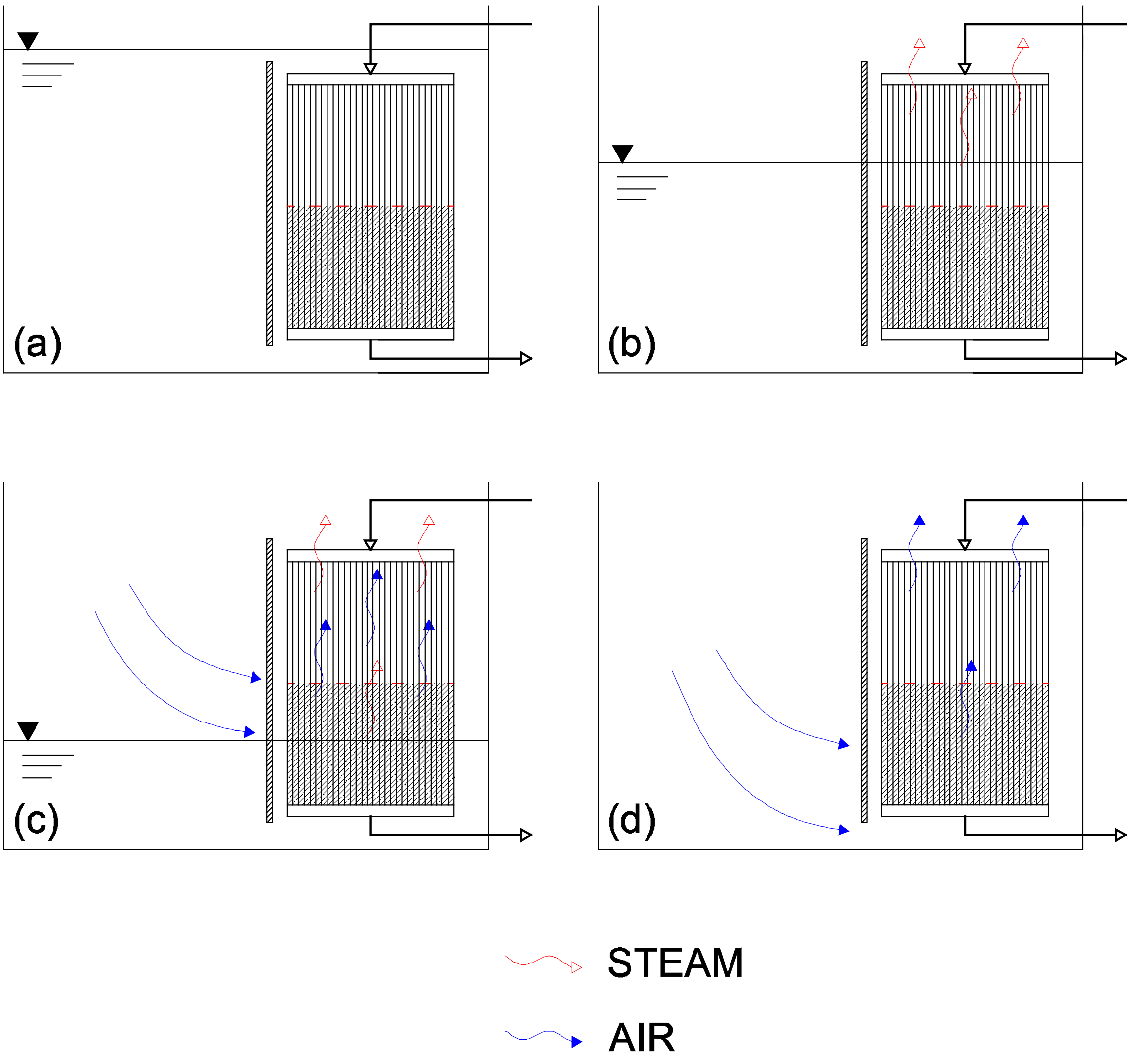



*Fig. 15. Simplified scheme of the operating condition of the PHX system*

The secondary fluid (thermal oil) enters on the top of the heat exchanger and flows downward, within the tubes, releasing heat to the tertiary fluid (water / air). On the pool side, water at first, and air later on, flow upward, through the sub channels, in natural circulation.

In order to improve the heat transfer capability when air is the only heat sink, each tube may be equipped with a higher number of fins (greater than four).

**Geometry**

The main geometrical parameters of the system are reported in Tab. 4 and in Fig. 16. Some details of the heat exchanger are shown in Fig. 16. PHX HX tube main geometrical parameters [mm]

*Tab. 4. PHX pool main geometrical parameters*

| | | |
|---|---|---|
| Tubes OD | 0.0127 | [ m ] |
| Tubes thickness | 0.00125 | [ m ] |
| Fin height | 0.0064 | [ m ] |
| Fin thickness | 0.00125 | [ m ] |
| Number of fins | 4 | [ ] |
| Tube height | 8.00 | [m] |
| Pitch/diameter ratio | 2 | [ ] |
| Fin loopholes maximum quote | 4.00 | [m] |

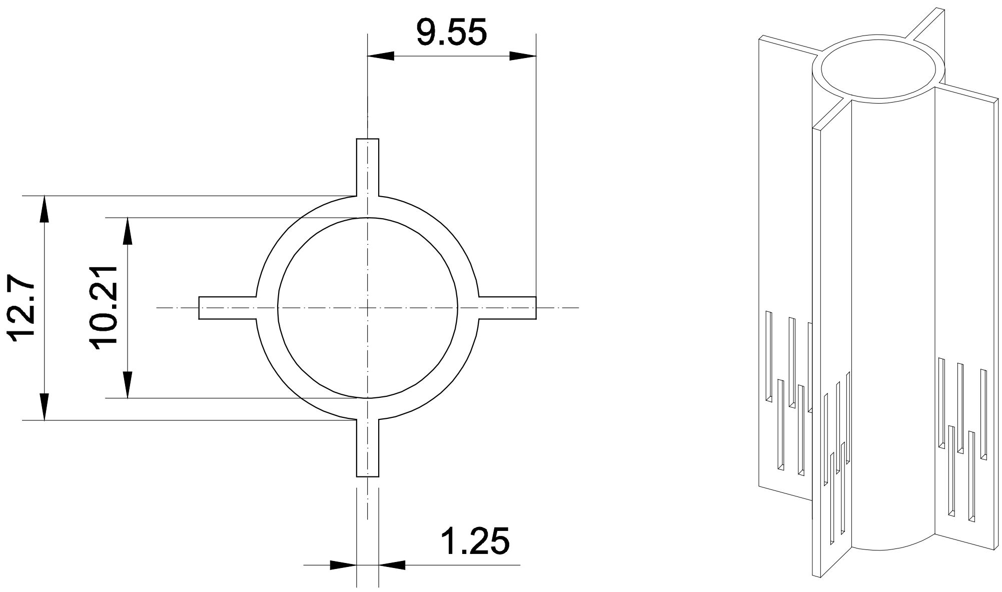



*Fig. 16. PHX HX tube main geometrical parameters [mm]*

*Fig. 17. Details of the Heat Exchanger of the PHX system (immersed tubes and headers)*

**Tertiary fluid**

Thanks to the special features of the heat sink, the tertiary fluid change during the transient operating period. The tertiary fluid, whose behavior is analyzed in the following, will be:

- Sub-cooled water during the first transient period;
- Boiling water plus air during the second transient period;
- Air during the last transient period.

The reference boundary conditions of the pool here considered are:

- Sub-cooled water starting from 30°C at the beginning of the transient period;
- Saturation temperature (at atmospheric pressure) during the boiling period;
- Air at 35°C at the pool inlet.

## First heat removal calculations

Assuming a nominal power of 3600 MWth for a large SFR, it is hypothesized that two safety DHR systems intervene when some 0.8% of nominal power is reached, which is about 30 MWth of thermal decay power to remove, as shown in Fig. 18.

It may be assumed that two independent safe DHR systems are available, one active and one passive, or two passive but different, each one equipped with three independent DHR loops; it may be also assumed that each loop extracts 15 MWth, that is to say a total of 90 MWth can to be removed (heat removal capacity by both systems: 300% nominal heat decay to remove).

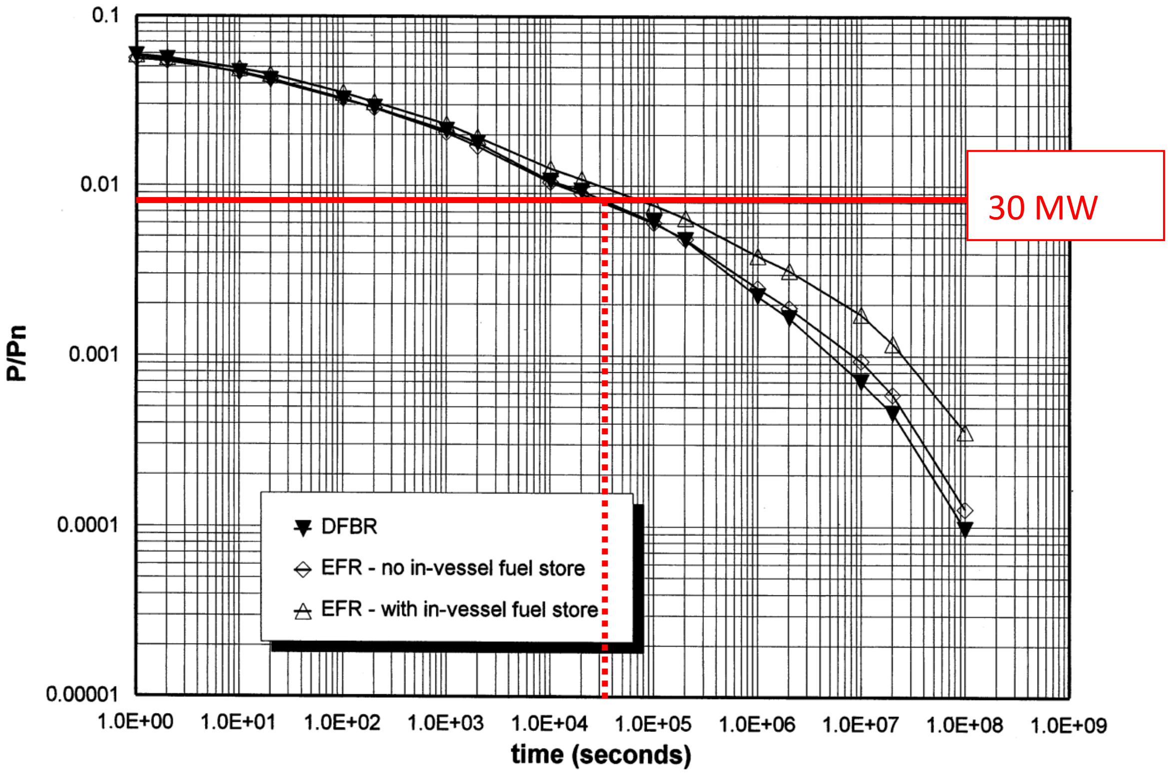



*Fig. 18. Decay heat to remove in the reference configuration*

The reference configuration (reference thermal power equal to some 30 MW), extracted from the previous figure, occurs at 34525 s after the reactor shutdown.

Each innovative DHR system loop is verified for removal of 50% of that power.

The number of tubes in the PHX is assumed as 50.000. Optimization could lead to minor number of tubes (depending also on oil cracking characteristics).

The primary flow rate and the inlet temperature of sodium have been defined according to possible design values of SFRs.

The secondary flow rate has been identified through the equilibrium between the pressure losses and the natural circulation head on the secondary / tertiary circuit.

The thermal-hydraulic characteristics of the DRC loop in the reference configuration are reported in Tab. 5 (transient starting time).

*Tab. 5. DRC thermal-hydraulic parameters*

| | |
|---|---|
| Sodium initial temperature | 530°C |
| Primary sodium mass flow rate | 181 kg/s |
| Water temperature | 30 °C |
| Power removed | 15 MW |

Several pool configurations have been analyzed: a series of different reference thermal power have been chosen, simulating the different pool conditions to evaluate the thermal performance of the DRC loop. Five different test cases have been taken into account:

- PHX pool full of water (100%) at 30°C
- PHX pool full of water (100%) at 80°C
- PHX pool full of water (100%) at saturation conditions
- PHX pool with 75% of air entering at 35°C; to be conservative the presence of water, present in 25% volume and removing heat through boiling, has been neglected in the evaluation of thermal performance
- PHX pool full of air (100%) entering at 35°C.

The thermal - hydraulic characteristics of the DRC loop for the chosen test cases are reported in Tab. 6.

*Tab. 6. DRC thermal-hydraulic parameters for different thermal cases*

| **Power** | **Time** | **Point - State** | **Pool temperature** | **Pool temperature (exit of sub-channel)** | **Hot Oil temperature** | **Cold Oil temperature** | **Oil Flow rate** |
|---|---|---|---|---|---|---|---|
| ***[W]*** | ***[ s ]*** | | ***[ °C ]*** | ***[ °C ]*** | ***[ °C ]*** | ***[ °C ]*** | ***[ kg \| s ]*** |
| 1.50E+07 | 34525 | A - Sub-cooled Water (100%) | 30 | 32 | 275 | 33 | 26.0 |
| 1.12E+07 | 84501 | B - Sub-cooled Water (100%) | 80 | 81 | 226 | 83 | 32.7 |
| 1.03E+07 | 109036 | C - Saturation water (100%) | 100 | - | 232 | 103 | 33.0 |
| 5.80E+06 | 685852 | D - Air (75%) | 35 | 181 | 223 | 138 | 27.5 |
| 3.91E+06 | 1058299 | E - Air (100%) | 35 | 138 | 164 | 84 | 21.5 |

As can be noted, there is a margin between the maximum oil temperature shown in Tab. 6 and the maximum allowable oil temperature (350°C); a second PHX pool configuration could be analyzed where the number of the PHX tubes decreases from 50.000 to allow a further reduction of the PHX dimensions and costs.

Referring to thermal-hydraulic parameters in Tab. 6, the oil hot leg temperature trend, related to the reference configuration, is shown in the Fig. 19, where - in different colors -

the three different states of the pool heat sink are represented. The temperature trend, during the shifting phase from boiling water heat sink to air-only heat sink, is estimated with conservative assumptions: during the period of coexistence of boiling water and air flow in the mixed heat sink, only the air contribution is taken into account. The real trend of the oil hot leg temperature, that can be obtained in further stages of the study, is expected to be lower than (or equal to) the trend shown in the figure.

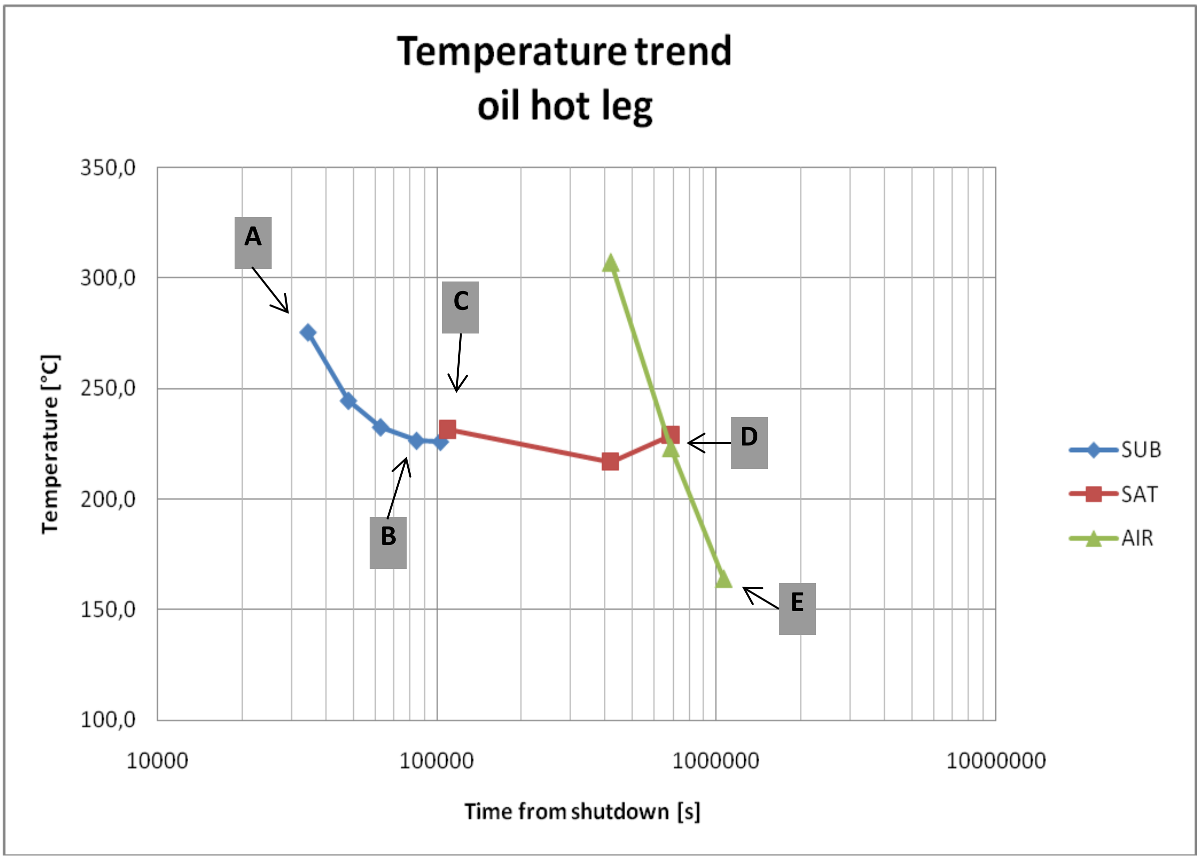



*Fig. 19. Oil hot leg temperature trend after shutdown*

Tab. 7 summarizes the geometrical and thermal–hydraulic parameters, only concerning the initial period of the transient phases (starting time), of the new proposed safety DHR system, made of a Radiation-based bayonet DHX and an innovative PHX, pool type special heat transfer system.

*Tab. 7. New proposed safety DHR system - main geometrical and thermal–hydraulic parameters*

| **New DRC concept** | | |
|---|---|---|
| Thermal – hydraulics parameters concerning initial time | Power removed | 15 MW |
| | Sodium temperatures | 530°C/464°C |
| | Primary mass flow rate | 181 kg/s |
| | Oil temperatures | 275°C/ 33°C |
| | Secondary mass flow rate | 26.0 kg/s |
| | Pool temperature | 30 °C |
| | | |
| Geometrical features of the: Radiation-based bayonet Na / vacuum / oil DHX | Outer tube (O.D. and t) | 0.0377 m (0.00089 m) |
| | Intermediate tube (O.D. and t) | 0.0200 m (0.00089 m) |
| | Inner tube (O.D. and t) | 0.0138 m (0.00089 m) |
| | Number of tubes | 1400 |
| | Bundle diameter | 1.9 m |
| | Increase in the tube bundle diameter compared to DHX with “U” tubes | ~110% |
| | Exchange area (considering outer diameter of the outer tube) | ~ 800 $m^2$ |
| | | |
| Geometrical features of the: AHX Oil / water -> air | tube (O.D. and t) | 0.0127 m (0.00125 m) |
| | Number of tubes | 50000 |
| | Pool dimensions | 20 m x 20 m x 8 m |

## Conclusions

The proposed innovative passive safety DHR system, DRC type, shows quite a number of interesting aspects, mainly in terms of safety.

Regarding the Radiation-based bayonet DHX, the use of oil as secondary fluid (which is perfectly compatible with sodium) together with the vacuum gap (which allows to promptly detect any leakage) allow to exclude any risk both on primary and on secondary side of the heat exchanger.

In further stages of design, fluids alternative to oil may be considered; the reference to the final layout of the various components could lead to a different DHX diameter.

Also the innovative concept of the PHX (which is, among others, extremely simple and cheap) ensures a high heat transfer capability in the first part of the transient (phase of cooling by water) and a never ending heat removal capability in natural circulation (phase of cooling by air). The oil temperatures remain lower than the maximum allowable oil temperature needed to avoid oil cracking (300 –400 °C).

The technological feasibility is possible, also thanks to the chosen material for the intermediate tube (which is aluminum). It must be remarked that the selected solution avoids any amount of sodium (not only primary sodium) flowing outside the reactor tank: this is considered a quite innovative feature, extremely relevant for safety and for simplicity and reliability of nuclear plant operation and maintenance.

## Nomenclature / Acronyms

| | |
|---|---|
| AHX | Air cooled Heat eXchanger |
| DHX | Decay Heat eXchanger |
| DRC | Direct Reactor Cooling system |
| NPP | Nuclear Power Plant |
| PHX | Pool water/air cooled Heat eXchanger |
| SFR | Sodium Fast Reactor |

## References


[1]. *"Generation IV Roadmap"*, Generation IV International Forum.

[2]. B. Farrar, J.C. Lefevre, S. Kubo, C.H. Mitchell, Y. Yoshinari, S. Itooka, *"Fast reactor decay heat removal; approach to the safety system design in Japan and Europe"*, Nuclear Engineering and Design 193 (1999).

# 7 CONCLUSIONS

In the present work some innovative concepts, conceived and studied in these years, have been analyzed. In order to make possible a widespread diffusion of nuclear reactors, increased, easy-to-demonstrate and unaffectable safety features have to be guaranteed.

The main purpose of the present work is to show the unique features that innovative passive safety systems can guarantee. The development of not industrialized regions will lead to nuclear reactors construction also in areas where electricity is not, at present, produced through NPPs. In these regions it is particularly important that the plant are easy to operate and have special safety features: hence the NPP has to be characterized by very enhanced safety standards. The best way to reach highest safety standards is to avoid, as much as possible, initiating events that could lead to accidental conditions, preventing them by design. This philosophy, called “safety by design”, allows reducing accidental conditions occurrence.

The systems here analyzed are mostly based on this criterion, since the pool allows reducing the DHRS failure probability exploiting a water heat sink and later an air heat sink only by means of natural laws. Natural laws are exploited for activation, for operation and for shifting from water to air heat sink. The pool system, always available and extremely reliable, minimizes the decay heat removal function failure probability.

The radiation-based DHX foresees a further barrier between the primary sodium and the diathermic oil. This further barrier, characterized by a vacuum volume, allows at the same time to obtain a promptly detection of any leakage, the minimization of primary / secondary coolants interaction even if the use of the diathermic oil, instead of water, makes the interaction less hazardous. The elimination by design of the SWR occurrence means that the system design requirements are less stringent and hence cheaper. Moreover, the use of a complete decoupling between primary sodium and diathermic oil makes it possible to avoid also an intermediate loop to uncouple the primary sodium and the heat transfer fluid. The use of diathermic oil instead of secondary sodium makes a loop leakage, outside the containment building, not hazardous since the oil does not react with air as sodium. In any case, depending on the oil operating temperature, some issues related to the auto-ignition of the oil has to be considered.

Concerning the candle concept, considerations similar to that of the radiation-based DHX can be done.

The examples here described are clearly related to innovative designs that are, at the present, under development. The increasing in the safety features will be a key parameter for selecting one specific design instead of another one. Considering the possible consequences related to an accidental event, especially after the Fukushima accident, including the releasing of radioactive materials, the strategic policy related to a specific remote area (e.g. mineral ore, oil field, military base, etc.) can make an inherently safe nuclear reactor more interesting than a cheaper standard nuclear reactor (it is possible that a nuclear reactor with a standard design could reach higher performance).

Future nuclear reactors, for which a minimization of the environmental impact should be guaranteed also in an accidental event occurrence, will be mainly based on safety systems characterized by passive activation and operation features as well as an inherently safe systems (according to IAEA: higher the passive category of the system is and safer the plant will result). Probably, for some specific applications, the safety performance should be the reference parameter to select a specific design. At the present time, several designs are characterized by enhanced safety features, especially those designs focused on SMRs (e.g. NuScale, mPower, etc.). In any case, also large-size nuclear reactors can adopt passive safety systems to comply with the needed safety requirements (i.e. Westinghouse AP1000) while other designs are mainly based on diversification and physical separation criteria (i.e. Areva EPR). According to my personal opinion, once analyzed and deeply studied several proposed solutions, I am strongly convinced of the importance and of the unequaled characteristics that suitable passive safety systems can guarantee (these must be based on operation principles that exclude events able to negatively affect their performance). Specific design solutions that would increase the *grace time* and would make the reactor a walk away system will make it possible in the future to perceive safer the nuclear power utilization.

During the research work some problems affected the activities and the most important problems founded can be summarized by the following statements:

- Needs to simplify the system model, focusing the main system pros and cons, in order to understand if the system has features to be developed or to be discarded;
- Systems developed that are applicable to special context often face with the lack of experimental correlations to represent the selected phenomenon (e.g. heat transfer coefficient), in these cases a simplification to correctly model the phenomena has been required;
- Innovative features of the systems make them often un-comparable with existing designs, preventing any possibility to further develop the system, on the basis of open issues already identified, before experimental test campaigns;
- Concerning some existing designs, comparable with those developed, only very limited information are available (also limited to special features of the system itself).

The systems here described are the final results of the evolution and the optimization processes applied to some solutions identified at the beginning. The effort done has been the result of many brainstorming among designers, researchers and specialists on specific issues. Further studies, especially finalized to some experimental tests as well as to some more specific performance definition, could be desirable.

# 8 ACRONYMS

| | |
|---|---|
| AHX | Air cooled Heat eXchanger |
| ATSS | Additional Temperature-actuated Shutdown System |
| CDF | Core Damage Frequency |
| CPP | Containment for Primary loop Protection |
| DHRS | Decay Heat Removal System |
| DHX | Decay Heat eXchanger |
| DRC | Direct Reactor Cooling system |
| ECCS | Emergency Core Cooling System |
| FR | Fast Reactor |
| GIF | Generation IV International Forum |
| IHX | Intermediate Heat eXchanger |
| ISC | Intermediate Safety Circuit |
| ISI&R | In-Service Inspection & Repair |
| LFR | Lead Fast Reactor |
| LMTD | Log Mean Temperature Difference |
| LOOP | Loss of Offsite and of Onsite Power |
| LWR | Light Water Reactor |
| MARS | Multipurpose Advanced Reactor inherently Safe |
| NPP | Nuclear Power Plant |
| NSSS | Nuclear Steam Supply System |
| PHX | Pool water/air cooled Heat eXchanger |
| PSA | Probabilistic Risk Assessment |
| PSC | Primary Safety Circuit |

| | |
|---|---|
| PWR | Pressurized Water Reactor |
| RHRS | Residual Heat Removal System |
| SCCS | Safety Core Cooling System |
| SFR | Sodium Fast Reactor |
| SG | Steam Generator |
| SMR | Small and Medium Reactor |
| STP | Standard Temperature and Pressure |
| SWR | Sodium Water Reaction |
| TSC | Tertiary Safety Circuit |
| WIPO | World Intellectual Property Organization |

# 9 RESEARCH PRODUCTS

## INTERNATIONAL JOUNALS

1. A. Naviglio, D. Vitale Di Maio, F. Giannetti, G. Caruso, G. D'Amico, "A proposal for simplification and cost reduction of SFRs", International Journal of Risk Theory, Volume 1, 2011, pp. 23-44, ISSN 2248-1672.
2. G. Caruso, L. Ferroni, F. Giannetti, A. Naviglio, D. Vitale Di Maio, "Proposal of a high reliability DHR system for NPPs with never ending capacity", International Journal of Risk Theory, Volume 2, 2012, pp. 11-31, ISSN 2248-1672.
3. D. Vitale Di Maio, A. Naviglio, F. Giannetti, F. Manni, "An innovative pool with a passive heat removal system", Energy 45 (2012), pp. 296-303, ISSN 0360-5442, 10.1016/j.energy.2012.03.053.
4. G. Caruso, D. Vitale Di Maio, A. Naviglio, "Condensation heat transfer coefficient with noncondensable gases inside near horizontal tubes", Desalination 309 (2013), pp. 247-253, ISSN: 0011-9164, 10.1016/j.desal.2012.10.026.
5. G. Caruso, D. Vitale Di Maio, A. Naviglio, "Film Condensation in Inclined Tubes with Noncondensable Gases: An Experimental Study on the Local Heat Transfer Coefficient", International Communication in Heat and Mass Transfer. ***SUBMITTED PAPER (IN PRESS)***
6. A. de Santis, D. Vitale Di Maio, G. Caruso, F. Manni, "Innovative radiation-based Decay Heat Exchanger (DHX) for liquid metal cooled reactors", Nuclear Engineering and Design. ***SUBMITTED PAPER (UNDER REVIEW)***

## INTERNATIONAL CONFERENCES

7. D. Vitale Di Maio, A. Naviglio, F. Giannetti, F. Manni, "An Innovative Pool with a Passive Heat Removal System", 24th International Conference on Efficiency, Cost, Optimization, Simulation and Environmental Impact of Energy Systems, 2011, 4-7 July, Novi Sad, Serbia, pp. 2873-2884, ISBN 978-86-6055-016-5.

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10. G. Caruso, M. Nobili, D. Vitale Di Maio, "An innovative integrated IHX/SG for Sodium cooled Fast Reactors: a CFD analysis of the Sodium Water Reaction", The 15th International Topical Meeting on Nuclear Reactor Thermal-hydraulics, NURETH-15, 2013, 12-15 May, Pisa, Italy. ***SUBMITTED PAPER (ACCEPTED)***

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13. D. Vitale Di Maio, F. Giannetti, A. Naviglio, L. Gramiccia, L. Ferroni, "Analysis of passive safety systems in a small/medium size nuclear reactor", XXIX Congresso UIT sulla Trasmissione del Calore, 2011, 20-22 June, Torino, Italy, pp. 415-419, ISBN: 978- 88467-3072- 5
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