# THE NILUS REACTOR FOR THE CO-PRODUCTION OF ELECTRICITY AND POTABLE WATER


Carlo Lombardi
Politecnico di Milano
Dipartimento di Ingegneria Nucleare
via Ponzio, 34/3
20133 Milano, Italy
+39(02)2399-6397

Enrico Padovani
Politecnico di Milano
Dipartimento di Ingegneria Nucleare
Via Ponzio, 34/3
20133 Milano, Italy
+39(02)2399-6360

Marco E. Ricotti
Politecnico di Milano
Dipartimento di Ingegneria Nucleare
via Ponzio, 34/3
20133 Milano, Italy
+39(02)2399-6325



## ABSTRACT

The NILUS-1000 reactor, an innovative PWR with integrated layout and full natural circulation, is proposed for the adoption in those countries where the production of potable water is a major concern for the population, and where the nuclear culture could not be so developed and firmly established as the complexity of the nuclear industry would require.

The NILUS concept is briefly depicted, together with the reference desalination process and plant size selected for the coupling with the 1000MWth nuclear reactor.

The safety systems and the "hybrid" containment, a compact steel pressure containment with a pressure suppression pool, are described. The results of the preliminary safety analysis are reported for an ATWS accident, showing the validity of the concept design and the large grace period achieved in this beyond DBA transient.


## I. INTRODUCTION

In the last decade, many designs on small and medium power reactors[1] has been presented, all of them aiming at improving safety, reliability and economics. Their employment is devoted not only to electricity production but also to heat only and to combined heat and electricity generation, for different utilization purposes like district heating, seawater desalination, ship propulsion, steam production for industrial use.

The large part of these new designs[2-6] refers to integrated layout reactors, mainly of pressurized type and in forced circulation, with a wide use of passive safety systems, both for the primary system and the containment.

The paper deals with the preliminary description of the NILUS application to co-production of electricity and potable water. NILUS[7-8] belongs to the class of small-medium PWRs and is based on the key features of natural circulation, integrated layout, implementation of passive safety systems. Other characteristic features are: low core power density, long fuel cycle duration, no boron reactivity control, internally driven control rods, self-pressurizing system, easy accessibility to the internal steam generators and decay heat removal heat exchangers, pressure suppression containment.

The NILUS project aims at the definition of a set of requirements and design solutions that could assure not only enhanced safety and reliability to the plants, but also good economics and easy construction, management and maintenance, in particular if the adoption in developing countries is foreseen.

This is pursued by means of a general simplification of the systems and the adoption of a modular structure. As far as safety aspects are concerned, special effort are devoted to avoid core uncovering, in any circumstance.

The low core power density eases the possibility to substantially increase the fuel cycle duration; moreover, it is conceivable to adopt more innovative Thoria based fuel cycles, in order to reduce proliferation concerns and actinides production. All these aspects are of paramount importance for a wider use of the nuclear power in the world.

Here the main results of the project application to a 1000 MWth reactor, the NILUS-1000, and the reference process for water desalination are reported.

## II. NILUS CONCEPT AND DESALINATION PROCESS

The topical requirement of the NILUS (Natural circulation Integrated Layout Ultimate Safety) concept is to achieve a high degree of simplicity in order to obtain better cost economics, strongly reduced problems of repair and maintenance, easy control, operation and management of the plant, together with a high degree of safety. These features are of paramount importance, especially when the nuclear plant is to be placed near urban areas, as for the desalination and district heating purposes.

It must be recalled that the greatest number of problems encountered in the management of a nuclear plant is usually of non-nuclear origin and concern valves, pumps, pipes, controls, actuators, measuring instruments, etc. occurring not only during the full power plant operation but also during inspection, maintenance and other routine activities.

For such a reason, a major effort has been devoted to a strong simplification both of the whole plant systems and of the inspection, maintenance, testing, assembly and disassembly procedures. For example, the steam generator modules design has been carried out in order to get easy and independent accessibility to each collector, in the feedwater and in the superheated steam side. The shape of the collectors and the straight tubes of the bundle allow a very simple procedure for the in-service inspection and the plugging of the tubes, the assembly of the modules and the possible module substitution in the case of wide damage. Anyway, the presence of the high pressure primary fluid in the external side of the tubes, instead of in the internal side like in the usual configuration of the current steam generators, allows a dramatic reduction of the inter granular stress corrosion cracking (IGSCC) tube failure, one of the most important causes of tube rupture in the steam generators Moreover, the solution adopted for the NILUS reactor strongly reduces the radioactivity doses to the personnel during all the procedures related to the steam generator modules, having no direct contact with surfaces wetted by the primary fluid but only with the secondary side surfaces.

A. Desalination process

A reference seawater desalination process[9-10] has been selected among the three more diffused ones, namely the Reverse Osmosis (RO), the Multiple Effect Distillation (MED) and the Multi Stage Flashing (MSF). Currently, the more attractive process seems to be the RO, both for technical and economic convenience, even if it is not as widely diffused as the MSF one, adopted in the 80% of the world desalination plant in the world.

The Reverse Osmosis plant to be coupled to the NILUS-1000 nuclear plant is of standard configuration, as already selected by an IAEA economic optimisation[11]: the main data are reported in Table I.

Two desalination plants could be coupled to the medium size nuclear reactor, in order to obtain a share of use for the electricity produced roughly of 50 % between the desalination plants and the grid: 165 $MW_e$ (55% of the total) to the grid for urban and industrial use, 135 $MW_e$ (45%) for a total potable water production of 576000 $m^3/d$, i.e. a large size desalination station.

A further option, conceived also by the Indian designers[10], foresees both electricity and heat production by a nuclear plant, with this ratio sizeable according to the local needs: this configuration allows the exploitation of electricity and heat for the water desalination processes, by adopting a double scheme including the RO and the MSF processes.

Table I

| Reverse Osmosis Plant | Config.: Hollow fibers |
|---|---|
| Number of Trains | 12 |
| Desalted Water Capacity:<br>Train<br>Total | <br>24 000 $m^3/d$<br>288 000 $m^3/d$ |
| Electricity consumption:<br>Specific Energy<br>Total Power | <br>5.64 $kWh_e/m^3$<br>67.7 $MW_e$ |
| Seawater inlet conditions:<br>Temperature<br>Total Dissolved Solids concentration (TDS) | <br>30 °C<br>45 000 ppm |
| Osmosis Membrane inlet Pressure | 72 bar |
| Desalted Water / Pumped Seawater | 0.35 |

Part of the electricity produced could be devoted to the RO plant, and the large amount of heat coming from the vapour spilled from the high and low pressure turbines is employed by the brine heaters, to raise the seawater temperature to the 125°C needed by the MSF process. An approximated evaluation of this mixed heat and electricity production, similar to that already done for district heating pourposes[12], with a double desalting plant configuration showed the possibility to reach roughly the same potable water capacity of the previous plant option, by assigning 500 $MW_{th}$ to the MSF plant for a capacity of 280 000 $m^3/d$, and 68 $MW_e$ to the RO plant for 300 000 $m^3/d$ of desalted water. The same amount of electric power (167 $MW_e$) is sent to the grid. This equivalence is obtained at the expense of the electric efficiency of the nuclear plant that drops down to 23.5%, and of the whole plant complexity, even if some advantage could be gained in terms of process flexibility. Anyway, the economic convenience must be thoroughly evaluated.

B. Integrated Layout description

The reactor pressure vessel (RPV) layout of the NILUS-1000 reactor is shown in Fig.1, while the main plant data are reported in Table II. The fully natural circulation of the primary fluid avoids the use of the primary pumps, leading to a significant simplification of the plant and an increased degree of safety, being practically excluded all the LOFA scenarios.

The 12 steam generator modules are located inside the RPV in the upper peripheral zone and are of once-through type in place of the recirculation type: this has been one of the major modification of the NILUS design, in order to achieve the better layout simplification and compactness of the containment, by eliminating the external steam drums. They are the more crucial and innovative component of the project, together with the internally driven control rods. A preliminary thermo-mechanical evaluation of the steam generator module has been performed, mainly for the conical collectors and the tube bundle, confirming the validity of the solution.

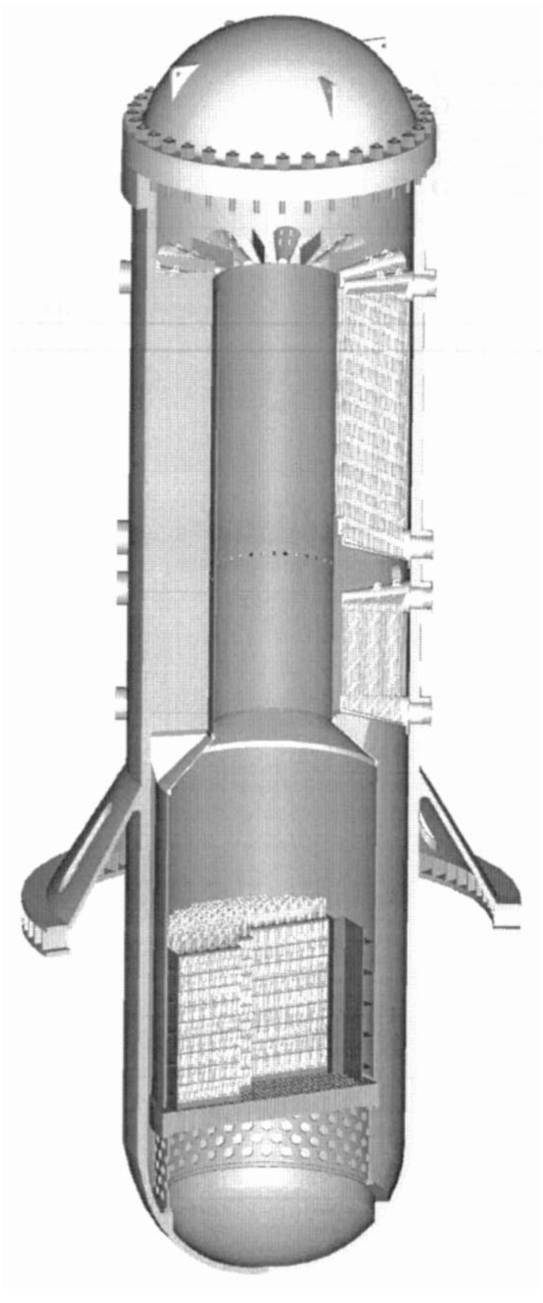

Fig.1 Layout of the NILUS-1000 reactor vessel.

In Fig.1 the decay heat removal (DHR) exchangers are visible, below the steam generators. This is a solution currently excluded in order to reach the maximum degree of simplicity for the RPV.

The cylindrical barrel is divided into two parts: the lower core barrel with larger diameter and the upper riser, an extractable section that allows the radial assembling and extraction of each steam generator module, being each module welded and flanged from the external side of the vessel. The primary water level into the RPV is well above the riser-barrel end, thus giving a suitable margin before the natural circulation is impaired during both operational transients and a large set of accident transients. Anyway, the natural circulation across the steam generators is assured in any circumstance by several holes at 1/3 of the riser-barrel height: this safety feature is acquired by paying the cost of a small bypass flow rate even at the operational reactor functioning.

The usual external pressuriser is eliminated and the primary pressure is controlled by means of a steam plus nitrogen volume on the primary water level in the vessel dome. The steam in this volume is at the partial pressure of 9900 kPa, corresponding to the saturation condition of the core outlet temperature, so that the nitrogen has to be inserted at a partial pressure of 2600 kPa. The pressure control is obtained by a feed of nitrogen and a bleed of the gas mixture, while the water level is controlled by the CVCS, an important system which is simplified by the elimination of the boron in the primary fluid. Due to the lack of the active controls of a conventional pressuriser (heaters and sprayers), the pressure is not as stable as in current PWRs, but the presence of the gas allows a significant dumping of the pressure oscillations and greatly simplify the system.

The cylindrical skirt under the core plate sustains the plate itself, the fuel, the control rods and the barrel. The holes diameter of the skirt and the skirt thickness are such that the pressure losses of the fluid are reduced and the mechanical strength is assured.

As far as the reactor core is concerned, the dramatic reduction in fuel power density, equal to 1/3 of current PWRs (45% of current fuel linear power), increases the safety margins but has also economical benefits, allowing the use of the 15×15 fuel assemblies with a slight reduction in fabrication costs, giving a lower Xenon poisoning which means higher reactivity and higher burnup, and leading to longer fuel cycles. Especially for the latter result, the reactors can be operated with the fuel loaded at periods of several years (>4) and this should be viewed as an advantage for the operation, for the economics due to higher load factors and for the acceptance in developing countries. However, the main innovations of the core are the presence of the hydraulic driven control rods and the elimination of the boron poisoning. As far as the former solution is concerned, it must be recalled that this type of control rods is already used in the Chinese NHR-5MWth demonstration reactor[2] and is designed to be used also in the NHR-200MWth. Experimental campaigns at reduced pressure have been already carried out by German designers.

Table II
NILUS-1000 Main Plant Data

| | | | |
|---|---|---|---|
| **CORE** | | | |
| **Thermal power** | 1000 MWth | **Cladding thickness** | 0.62 mm |
| **Gross electric power** | 300 MWe | **Pitch** | 14.27 mm |
| **Active height** | 2.44 m | **Assembly equivalent diam.** | 13.57 mm |
| **Active height/core diam. ratio** | 0.65 | **Overall assembly width** | 214 mm |
| **Fuel assemblies (15x15)** | 241 | **Average fuel linear power** | 8.32 kW/m |
| **Rod diameter (outer / pellet)** | 10.72 / 9.29 mm | **Average fuel specific power** | 32.42 kW/l |
| **VESSEL** | | | |
| **Inner diameter** | 4.97 m | **Free weight** | 419 tons |
| **Thickness** | 0.20 m | **Pressure** | 12.5 MPa |
| **Vessel-barrel gap** | 0.40 m | **Core inlet temp.** | 264.6 °C |
| **Total height** | 21.35 m | **Core outlet temp.** | 310 °C |
| **Free volume** | 335 $m^3$ | **Primary flow rate** | 4138 kg/s |
| **STEAM GENERATORS** | | | |
| **Type** | Once-through | **Total external heat transfer surface** | 10130 $m^2$ |
| **Material** | Stainless-Steel | | |
| **Number of modules** | 12 | **Secondary pressure** | 4.4 MPa |
| **Tube diam. (outer / inner)** | 11.2 / 9.2 mm | **Secondary total flow rate** | 495.7 kg/s |
| **Pitch/diam ratio** | 1.4 | **Inlet feedwater temp.** | 203.8 °C |
| **Tube length** | 5.90 m | **Outlet steam temp.** | 281 °C |
| **Total number of tubes** | 48800 | **Outlet steam superheating** | 25 °C |
| **INTERNAL PRESSURISER** | | | |
| **Type** | Steam+$N_2$ | **Total volume** | 42.5 $m^3$ |
| **Nitrogen partial pressure (nominal)** | 2.5 MPa | **(liquid volume)** | (22.5 $m^3$) |
| | | **(vapour volume)** | (20.0 $m^3$) |
| **ACTIVE & PASSIVE HEAT REMOVAL EXCHANGERS** | | | |
| **Number of trains** | **4 (2 active + 2 passive)** | **Number of modules per train** | **2** |

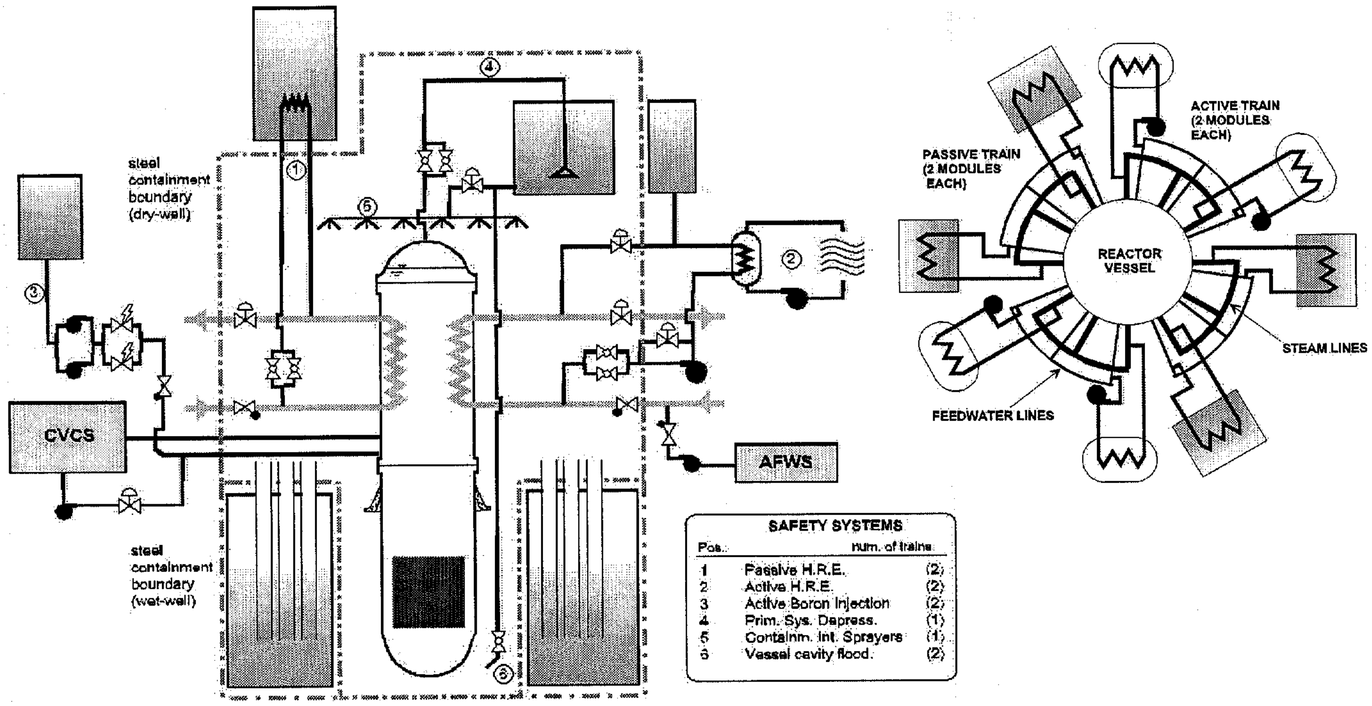


Fig.2 Scheme of the main safety systems of the NILUS-1000 reactor.

The study of a different mechanical solution together with CFD analyses on the device are currently performed by the Authors: instead of the cruciform control rods, used by the Chinese and German designers and very similar to those adopted in the BWRs, the "spider structure" of the current PWRs control rods should be maintained, with the hydraulic piston located in the centre of the assembly in place of the four central rods.

## C. Safety Systems description

A diagram showing the possible location of different active and passive systems is reported in Fig.2. Particular attention is devoted to the selection of the number and type of the safety systems, with a specific aim to the simplification of the plant, once the selected degree of safety has been achieved: the deterministic accident analysis and a preliminary PRA will define the final configuration of the plant.

With reference to the diagram, the main characteristics of each system is briefly depicted.

1.Passive Heat Removal System. Is a simple natural circulation loop connected to the steam generator feedwater lines and to the steam lines, activated by a fail-to-safe valve, energised by a dedicated supply; the loop assures the condensing flow rate to feed the steam generators and extracts the decay heat via a heat exchanger submerged in the external pool. There are two trains, with two loops each. A single train is designed to extract the decay heat in the worse condition, i.e. in presence of stagnant steam into the RPV at the pressure of 2 bar and with a minimum content of non-condensable gas equal to 10%.

2.Active Heat Removal System. Is made up by two trains, four loops total, with secondary heat exchangers rejecting the decay heat to the heat sink (e.g. cooling towers, sea or river flow). The heat exchangers are connected to the feedwater and steam lines of the modular steam generators, as in the passive system. The forced circulation is controlled by a pump and a control valve, the system is employed also during start-up operation or other operational transients.

3.Active Boron Injection System. Is activated only if the double control rods scram system does not work. It is connected to the 4" feed line of the Chemical and Volume Control System (CVCS).

4.Primary System Depressurisation Lines. Two pipe lines starting from the top RPV connect the steam dome with a condensing pool located inside the containment. The fail-to-safe valves, energised by a dedicated supply, open on low-pressure, high-pressure, low-level signals; they have the only purpose to slowly depressurise the RPV to avoid any scenario involving LOCA at high pressure, in order to keep covered the core even during large LOCAs.

5.Internal Containment Sprayers. It is a gravity driven spray system; it uses the water located inside the depressurisation pool and contributes to condense the steam in the dry-well.

6.Cavity Flooding Line. A pipe line is directly connected to the depressurisation pool and a check valve activates when the temperature of the dry-well environment rises, due to the occurrence of a LOCA. The water of the pool, together with that condensed on the containment inner surfaces, is collected into the vessel cavity of the dry-well. This procedure assures the bottom RPV external cooling.

## D. Containment

Currently, two options are under evaluation for the NILUS concept: a concrete pressure suppression containment, widely used in the BWRs, and an innovative hybrid steel containment, designed to accommodate the internal pressure as the usual pressure containment but with a wet volume to suppress the peak pressure. The two schemes are reported in Fig.3. Here only the hybrid containment is presented.

The main design goal is to achieve a compact containment with a high degree of passive safety, mostly relying on natural circulation flow for the decay heat removal. The dry-well plus the wet-well volumes configure a steel bottle-like containment, with the external surfaces partially submerged in water walls.

The dry-well is shaped in three volumes: a large upper cylinder of 6500 $m^3$ about (5000 $m^3$ free volume), accommodating the upper part of the RPV including the steam generators lines and the depressurisation pool, a smaller lower cylinder below the support RPV skirt, that is the RPV cavity and a truncated cone connecting the previous two volumes, which allows the water condensed on the containment walls to flow into the RPV cavity.

During a LOCA, the steam condensed on the dry-well walls in the upper volume plus the liquid water discharged through the rupture are collected into the RPV cavity, keeping the external side of the RPV submerged: from a parametric study carried out with different break locations and rupture sizes, the lower cylinder of the dry-well is completely filled by water in almost all the cases. In the remaining cases the water level is largely above the top of core elevation. The shape of the cavity, the large amount of water both in the vessel and in the dry-well reasonably lead to the exclusion of the ex-vessel severe accident scenario and to the management of the in-vessel one. A deterministic evaluation of these scenarios is under way.

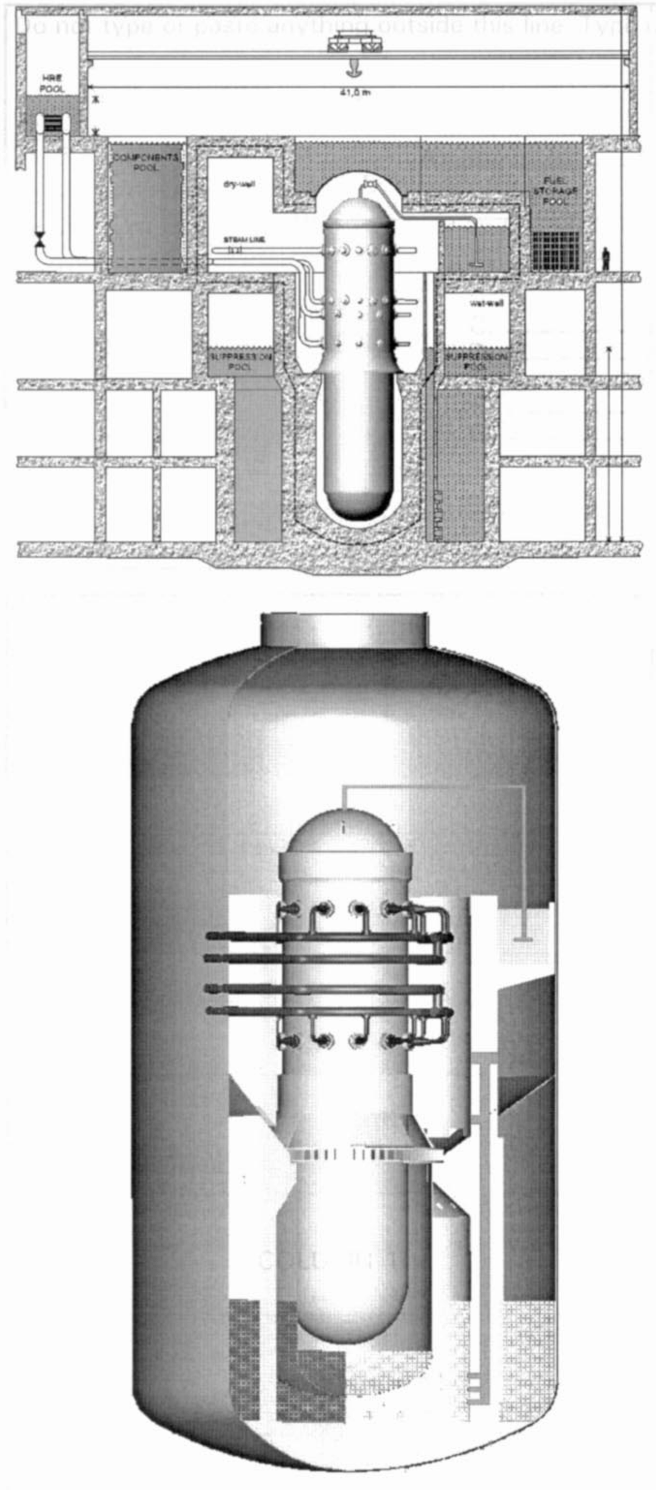

Fig.3 Concrete Containment (upper scheme) and Steel Containment (lower scheme) options for the NILUS-1000 reactor.

The wet-well volume (700 $m^3$ of water plus 1500 $m^3$ of free volume) has steel boundaries and an annular shape: the inner boundary allows the heat transfer between the water inside the RPV cavity and the water in the wet-well pool, while the outer boundary is faced to the water filling the external space between the steel containment and the concrete containment, in order to strengthen the thermal power dissipation to the external environment. The concrete containment is necessary for the plant to resist to external missiles and environmental hazards.

## III. SAFETY ASSESSMENT

The preliminary safety assessment of the NILUS-1000 reactor is in progress, with the aim not only to assess the safety level of the concept but also as fundamental feedback for the safety systems selection.

The analyses we carried out, refer to both operational and accident transients. For the sake of simplicity and brevity, the results of a single but significant transient are reported: the analysis refers to a large break LOCA (14" diameter), occurred directly in the reactor vessel at the elevation of 11 m from the vessel bottom. The accident scenario is a beyond design basis ATWS: neither the reactor scram systems, nor any other safety and decay heat removal systems works, both active and passive.

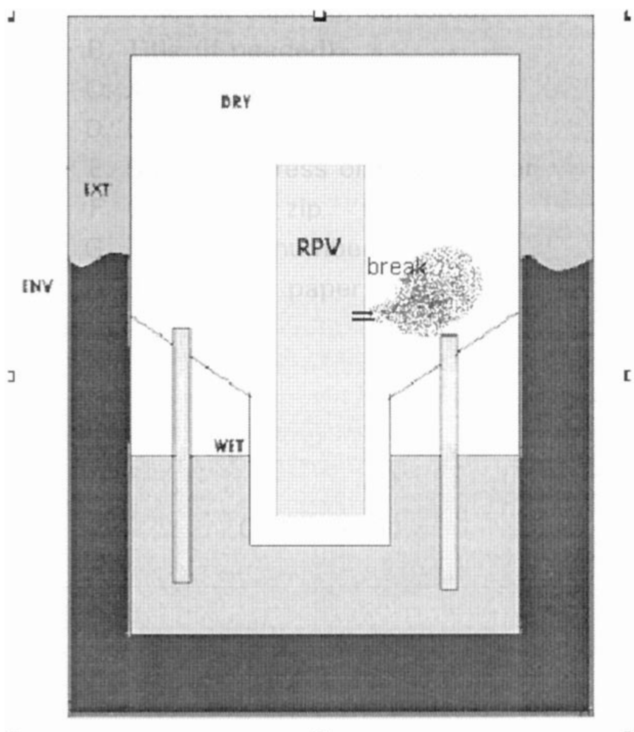


Fig. 4 Conceptual scheme of the Vessel plus the steel Containment, for the LOCA simulation with ECART code.

The plant scheme adopted for the nodalisation is reported in Fig.4. The code used to perform the transient analyses is the ECART[13] code, developed by ENEL and EDF for the severe accident scenarios.

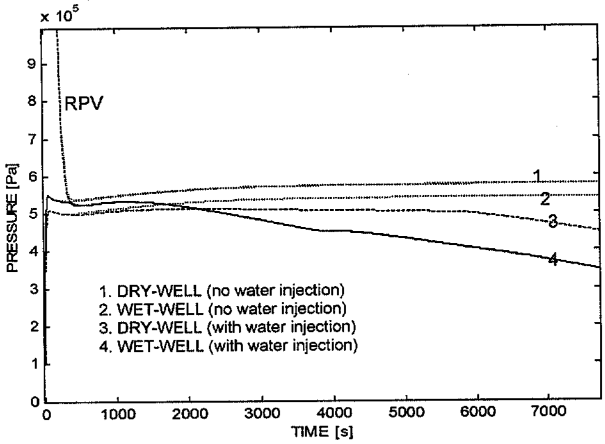


Fig. 5 Time profiles of the pressures during the ATWS, 14"-break LOCA. The figure shows the trends for the accident both with and w/o water injection from the CVCS.

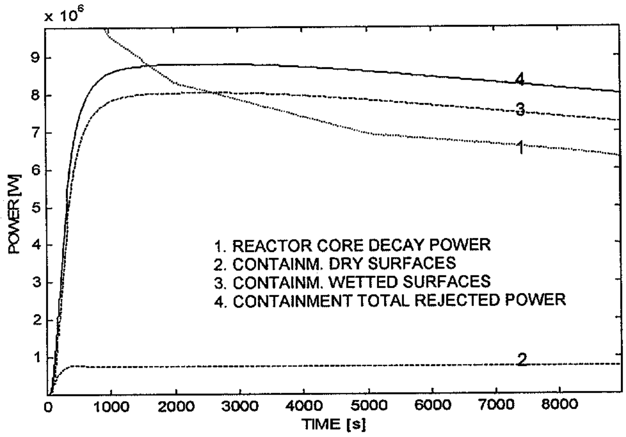


Fig. 6 Time profiles of the power exchanged by the containment surfaces during the ATWS, 14"-break LOCA.

The pressure behavior throughout the transient for the RPV, the dry-well and the wet-well is shown in Fig.5. We can outline three phases:

1. Initial pressure peak (at 5.5 bar absolute). At the onset of the accident a huge flow of air and steam through the vents occurs, as soon as vents have been depleted of water; this air-steam mixture is initially cooled down by contact with the wet-well steel walls. Consequently the pressure goes down.
2. Slow pressurisation. The cooling action ends as the inner wall starts to heat it up, and eventually a positive thermal flux from dry-well to wet-well through this wall would start, leading to a slow increase of wet-well atmosphere temperature and pressurization of the system. Dry-well pressure would reach 5.75 bar absolute, and the thermal flux through the steel wet walls would release in wet-well about 3 MWth. The difference between dry-well and wet-well pressure is roughly 0.3 bar, as water head in the pool is about 3 m.
3. Depressurization due to fuel uncovering. After 9500 s from the onset of the accident, the core would be completely uncovered: there is no bulk production of steam in RPV, so that no new steam would reach the dry-well to compensate vented and condensed steam: dry-well pressure would steeply begin to decrease; wet-well pressure, instead, is nearly unchanged, because it is due almost entirely by non condensable partial pressure. Differential pressure between dry-well and wet-well starts to decrease, hampering steam flow through the vents when it is not enough to withstand pool head. When the difference of pressure is negative enough to overwhelm vent gravitational head (now they are completely full of water) a reverse flow from wet-well to dry-well would occur, increasing the flooding of the RPV cavity bottom.

However, from the whole set of preliminary evaluation of LOCAs even with the extreme hypothesis of no intervention of any safety system, the water level remains well above the core for a very long period of time, thus indicating the high safety margins with respect to core uncovering.

In Fig. 6, the thermal power exchanged from the core, compared to the power released by the containment walls in the external environment is shown. It can be noticed that, after just 1000s from the onset, the structure would release all the power from the fuel. Cooling action is most effective on wet-well walls, despite their smaller area due to water-steel-water coupling; higher efficiency could be reached simply increasing the submersion level of the steel containment.

As a further step of analysis, the previous transient has been modified including the effect of a high pressure injection device, typically a charging pump of the CVCS, which provides a mass flow rate of 20 kg/s to the RPV, starting after 100 s from the onset of the accident. The simulation shows a slight decrease of peak pressure, since the low-enthalpy water injection decrease the steam production rate, and venting/condensation phenomena in the dry-well are not balanced by steam supply from RPV.

After 2000 s, the dry-well pressure falls below the wet-well one, eventually leading to the flooding of dry-well by an inverse flow arising from the pool, accelerating the rate of depressurization.

As far as RPV water level is concerned, the core remains covered throughout the event, the core head level being about 5 m from the bottom, while without CVCS pump operation its uncovering and eventual melting is not prevented.

The Fig.7 shows RPV water level with and without CVCS injection (full and dotted line respectively).

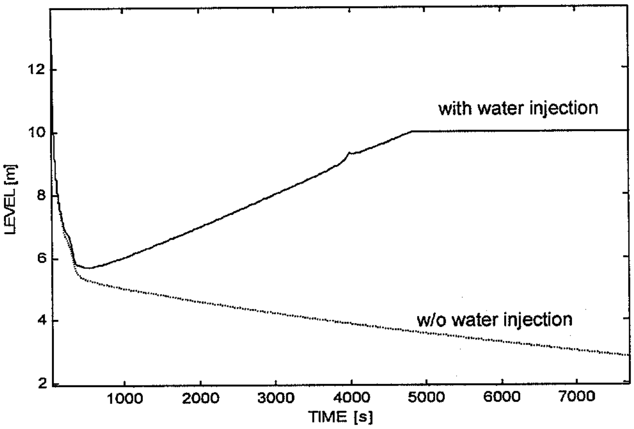


Fig. 7 Water level into the Reactor Pressure Vessel during the ATWS, 14"-break LOCA (with and w/o water injection).

In conclusion, the following remarks can be stated.

- The hybrid steel containment is a very effective device to keep the containment pressure within design value; furthermore, thermal exchange through the steel containment walls provides a long-term depressurization action, whose 'grace period' will be determined by external pool inventory.
- There is a great improvement in time interval between accident onset and core uncovering events if compared to the existing LWRs: these kinds of accidents (LBLOCA with total loss of active and passive safety systems) would lead to core uncovering within several minutes in a conventional reactor, while in the case of NILUS it is almost 3 hours, at least.
- A small injection is useful not only in keeping the core uncovered, but also as containment pressure suppression device, both directly (decreasing steam production rate), and indirectly (causing flooding of the dry-well by pool water).

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