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MARS - A Competitive PWR Based Completely on Passive Safety Systems

Key Features of the MARS Project

The MARS nuclear power plant is a cheap, simple, extremely safe PWR plant incorporating all main well-proven features of a "traditional" PWR plant.

A production plant is made of medium-power modules: each reactor module may produce up to some 1000 MWth; the design here presented refers to a size of 600 MWth. The whole design is strongly simplified, thanks to safety characteristics based on simple, almost completely static and exclusively passive-type emergency systems. The probability of core integrity failure is a vanishing event (according to rigorous PRA evaluations, core damage probability in the range of 1E-7 event/year).

The core cooling system includes one loop only, with recirculation-type steam generator; during normal operation, the primary coolant is pumped in the reactor vessel and in the primary loop, while, in emergency conditions, the coolant flow in the core is completely guaranteed through an independent cooling system, transferring heat to the external atmosphere with natural circulation and relying on only static components and one non-static, passive component (check valves, 400% redundant on two independent cooling trains).

In the design here presented, the MARS core (600 MWth) is of quite traditional design and includes 89 fuel assemblies, 17×17 pin array. 84 fuel assemblies have an irradiation period of 54 months, while 5 assemblies have a 72-months irradiation cycle. The refueling strategy is based on 1/3 core reloading, so the refueling operation will occur every (18+1) months. The safety margins are huge, also because of the total absence, in the MARS reactor, of fast thermal-hydraulic acci-

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dental transients. The nuclear design is based on traditional geometries and materials. At the CEA, Saclay, an innovative MARS nuclear design has been studied, incorporating the features of advanced French nuclear fuel and allowing quite higher irradiation cycles, together with simplification in primary coolant chemical control system and improved proliferation-resistance characteristics (BU achieved higher than 60.000 MWd/t).

The MARS reactor plant is equipped with a secondary pressurized containment filled with cold water, enveloping the full primary coolant pressure boundary. Deterministically, this does eliminate any loss of primary coolant. A containment building is foreseen to face external events in accordance with the strictest European regulations (aircraft impact). This containment building is able to withstand any internal pressurization, also in the incredible event of a complete loss of the core coolant pressure boundary.

LOCAs, ATWSs and LOFAs are eliminated in the MARS concept, making the plant an incredibly reliable, safe, easy-to-manage nuclear power plant.

The requirements of the radwaste system are quite limited, and the doses to operation and maintenance personnel are the lowest, thanks – among other – to the few components physically in contact with the primary coolant (only 7 components).

In spite of the exeptional safety and radiological protection features, the investment and operation costs for a MARS power production station are quite low.

This is a consequence of the huge plant simplification (drastic reduction of the number of systems and components, drastic reduction of components relevant to safety, reduction of concrete volumes) and of a design fully oriented to pre-fabrication and easy assembling/disassembling, that make the construction time incomparably short, limit the cost of all components, allow easy component substitution in case of failure instead of local repair.

The plant lifetime is 70-100 years, with the possibility of removal of components that could act as bottlenecks in the lifetime of the plant (as the reactor vessel itself).

Decommissioning of the MARS nuclear power plant is indeed quite an easy and fast job.

The total direct investment cost, including contingencies, has been evaluated with accuracy and is of 1650 US$/kWe (referred to a 3-reactor-module, 450 MWe station).

The cost of electric energy produced is equal to 0.035 US$/kWh (value that falls below 0.02 US$ after the first 20 years assumed for the debt service period). Further cost reduction may be achieved in the phase of final design.

The MARS NPP is a targeted solution for utilities requiring:

- Medium size NPP;
- Incomparably high safety and reliability, i.e., due to the closeness of the plant to highly inhabited towns (essential if the plant has to be used also for desalination purposes);
- Cheap electric production;

– Easy operation and maintenance;
– Fast plant construction;
– Possibility of multi - purpose production (electricity / heat / desalinated water).

Utilization of Passive Systems in the MARS NPP

The original idea behind the extensive use of passive safety systems in the MARS NPP was based on:
– simplification of the design, also to make it cheaper to build, operate and maintain;
– increase in the real safety of the plant through systems which were intrinsically simple and reliable, based on the action of 'natural' laws;
– improvement in the perception of safety of the plant for the same reasons.

A passive design strives to ensure that the three major safety functions can be carried out in a passive or quasi-passive manner. These functions are:
– reactor shut down in any plant condition;
– decay heat removal;
– fission product containment.

Passive safety emphasizes the use of natural forces (gravity, self-correcting neutronic feedback, thermal dilatation) and de-emphasizes systems which require large amounts of electricity (pumps) or energy, rapid automatic response, complex logic.

The design of the MARS (Multipurpose Advanced "inherently" Safe Reactor) plant started in 1983 at the Department of Nuclear Engineering and Energy Conversion of the University of Rome "La Sapienza", with the aim at proposing a new concept of fission-type nuclear plant to be used for a wide range of applications, including desalination and district heating.

The possibility of utilization of the plant in high-density population areas or in developing areas was the main reason for the search for improved safety requirements: the plant had to be simple, with an easy-to-understand and incontrovertible capacity to avoid any radiological hazard to the population. Typically, the safety approach had to be clearly understandable, on deterministic bases. The design was focused on a nuclear power generation capacity of 600 MWth, corresponding to about 170 MWe in the case of only electrical production, with a modular solution to satisfy progressively increasing power requirements (then power may vary, but very high power levels are not compatible with performance characteristics of some main passive safety systems).

The design was carried out using a step-by-step approach, which addressed all major issues regarding plant safety, performance and cost.

In particular, in order to make the MARS plant economically competitive with other nuclear plant solutions and with fossil-fueled thermal power plants, it was necessary to adopt a somewhat extreme simplification criterion in the design

approach. This simplified design involves mainly in-shop construction with only a "few" operations of easy assembling and simple substitution of all mechanical components for maintenance purposes, and easy removal for the final, complete decommissioning.

In May 1994 the preliminary design was completed, the preliminary safety report was submitted by ENEA, (the Italian State Agency for Energy, Environment and Innovation) to the Italian Nuclear Safety Authority, ANPA, for comments. Quite a lot of experimental tests on innovative components were performed.

The MARS nuclear power plant (NPP) design widely uses the well-proven technology and the operation experience of Westinghouse PWRs, but also incorporates several innovative features that hugely improve the safety performance, while keeping the cost of KWh competitive with traditional large power plants. Extensive use of passive safety, in depth plant simplification and decommissioning oriented design were the main guidelines for the design development.

The MARS NPP is designed to produce electric energy and/or industrial heat. The most efficient utilization of such kind of plant is definitively co-generation. Typical destinations of heat produced in a MARS NPP are low temperature utilizations of hot water or low pressure steam; among these, the following utilizations were analysed:

– water desalination using low temperature processes (as thermo-compression or multiple effects);
– district heating;
– food industry (conservation industry).

System Configuration

The MARS reactor is moderated and cooled by pressurized light water (PWR). The reference rated core thermal power, here referred to, is 600 MW. In case of only electric energy production, 150 MWe of gross power (146 MWe net power) are produced in a 600 MWth plant, with 25% gross efficiency (24.5% net efficiency).

In case of co-generation, the electric energy production strongly depends on the thermodynamic requirements of the produced hot water or steam. Co-generation cycles have been designed for the production of electric power ranging between 80 and 100 MWe and of hot water or steam at a temperature around 100°C.

The original MARS design incorporated only well proven nuclear reactor technologies, in order to make it easier plant licensing: major innovations concerned ECCS and some auxiliary systems. So, nuclear reactor characteristics here described are quite similar to well-known PWRs (primary loop type, core geometry and materials, reactor control type, etc.).

The MARS design here described is characterized by the following innovative solutions, representing completely passive safety features:

– an innovative, passive-type, quasi-static emergency core cooling system, based

only on natural circulation of cooling fluids and using external air as ultimate heat sink;
- an innovative, additional, passive-type scram system based on a two-metals core temperature sensor and operated by gravity (proposed as optional);
- full enclosure of the primary-coolant boundary in a pressurized containment filled with low enthalpy water (primary loop jacket). The absence of common mode failures, thanks to a special design, limits to irrelevant values the probability of loss of primary coolant.

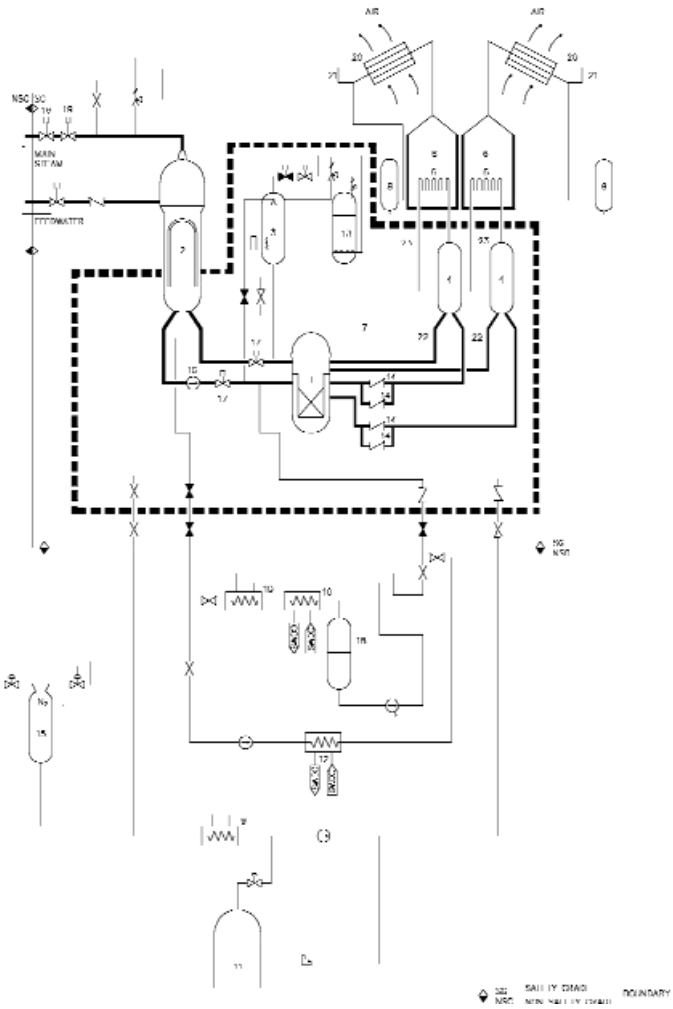
As said the fuel is low-enrichment uranium dioxide (in the fuel loading strategy here described, the loading U235 enrichment is 2.8%); the core includes 89 "standard" PWR fuel assemblies. The assemblies are zircalloy-cladded with rod array 17 × 17, including 264 fuel rods and 25 positions for zircalloy guide tubes for control rods (black, Ag-In-Cd; grey, stainless steel) or for burnable poisons (borosilicate). Fuel rod pitch is 1.26 cm. Fuel rod active length is 260 cm.

Light water flows in a single cooling loop. The average core coolant temperature is 234 °C. Reactor internals are AISI 304 made. Reactor vessel internal diameter is 3000 mm; the overall height of the assembled vessel is 11091 mm. The average fuel burn-up per cycle (3 irradiation cycles; 5 assemblies are irradiated for 4 cycles) is about 11,300 MWd/t. An alternative core design foresees a full core irradiation of 60,000 MWd/t.

The Innovative Safety Core Cooling System

The primary cooling system (Fig. 1) includes one loop only, with 25″ I.D. pipes, one vertical-axis U-tube steam generator and one canned rotor pump directly connected to the steam generator outlet nozzle. Connected to the reactor vessel is the Safety Core Cooling System (SCCS). A vapour-bubble pressurizer controls the pressure inside the primary cooling system. On/off valves (primary loop Main Isolation System, MIS) are installed in the primary cooling loop, in order to isolate, if necessary, the Steam Generator and the primary pump (i.e., in the event of a SG tube rupture).

The primary cooling system and the Safety Core Cooling System are inside a pressurized containment, filled with water at the same pressure as the primary coolant, but at a lower temperature (70 °C), called CPP (pressurized Containment for Primary loop Protection – Fig. 2) which allows the reduction (even the elimination) of primary stresses on the primary coolant boundary and provides an intrinsic defence to loss of coolant.The cooling of MARS core in emergency conditions is provided by the Safety Core Cooling System (SCCS – Fig. 3) is designed to transfer the core decay heat directly from the reactor pressure vessel to the external air, without the intervention of any energized system or component. The system operating principle relies on water density differences, due to temperature differences between two vertical fluid columns, causing the fluid circulation.



Legend

1. reactor
2. steam generator
3. pressurizer
4. heat-exchanger (reactor coolant/intermediate coolant)
5. heat-exchanger (intermediate coolant/final heat sink coolant)
6. water reservoir
7. pressurized containment for primary loop protection (CPP)
8. intermediate loop pressurizer
9. heat-exchanger (primary containment water cooling system)
10. chemical and volumetric control system heat-exchangers
11. water storage tank
12. residual heat removal system heat-exchanger
13. pressurizer relief tank
14. safety core cooling system check valve
15. primary containment pressure control system pressurizer
16. main coolant pump
17. primary loop on/off valve
18. VCS tank
19. steam line on/off valve
20. ultimate heat sink condenser
21. communication path with the atmosphere
22. safety core cooling system primary loop
23. safety core cooling system intermediate loop

Fig. 1. MARS primary cooling system.

The presence of multiple circuits (Primary Safety Cooling loop, PSC; Intermediate Safety Cooling loop, ISC; and pool and condenser loop (Third Safety Cooling loop, or TSC)) in a cascade functional operation chain provides redundant barriers between activated reactor coolant and external environment.The SCCS includes two trains; each train may remove 100% of the core decay power. In an accidental event causing the reduction of the core coolant flow (such as station black-out or primary pump trip), its activation is automatic (without any intervention either by the operator or by the control and supervision system, because the PSC interception valves are kept in closed position by the forces due the primary coolant flow

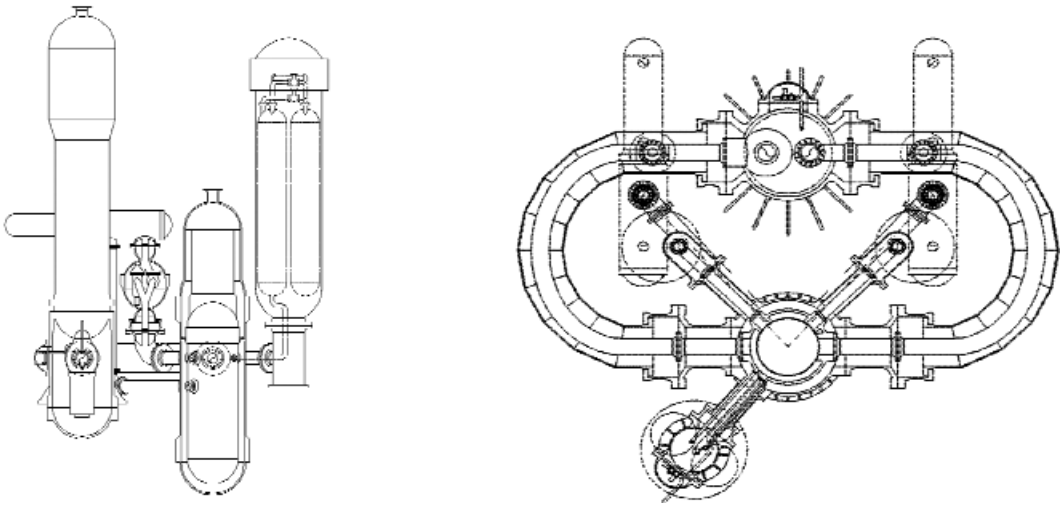


Fig. 2. Pressurized containment for primary loop protection (CPP).

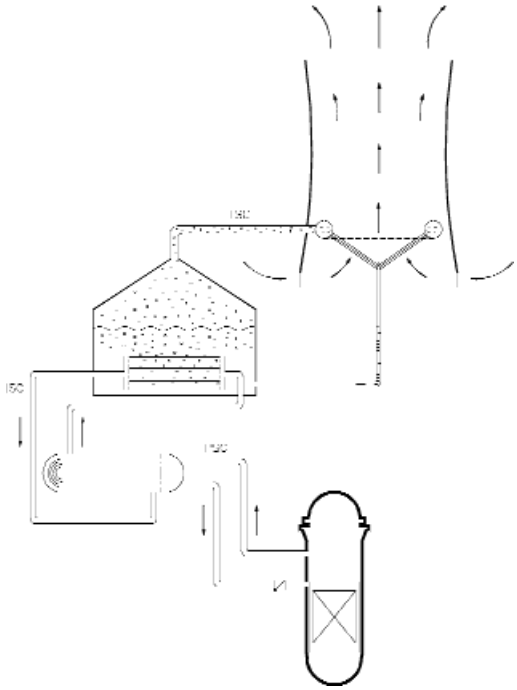


Fig. 3. Scheme of the safety core cooling system SCCS.

and start opening when this flow decreases under a set-point value); the operation of the system is completely passive.

The SCCS operation relies on the operation of special check valves able to automatically open, without any operator intervention and without the needs of energized systems, when the operating conditions require additional core cooling. These valves have a completely innovative design (Fig. 4). They are kept in closed position by means of the pressure difference between the reactor vessel inlet and outlet (that is roughly proportional to the square of the coolant flow-rate); when the flow-rate through the core goes to zero, the pressure difference decreases and when it is no longer sufficient to sustain the weight of the valve plug, this falls, and a complete flow area is opened, with a very low hydraulic resistance. Two valves, each one 100% capacity, are inserted in each SCCS train and, to increase the system availability to values that make its failure incredible, the additional two valves (in each loop, the second valve is of traditional design) are different in typology and mechanical construction.

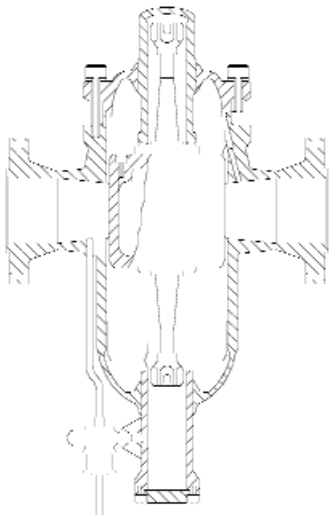


Fig. 4. SCCS special check valve.

When any of the four check valves is opened, the flow in the PSC, after a short transient phase, is assured by a difference in level of about 7 m between the vessel outlet nozzle and the primary heat exchanger and by the difference between inlet and outlet vessel temperatures.

A horizontal-axis U-tube heat exchanger (Fig. 5) transfers heat from the PSC to the ISC. Pressure in the ISC loop is slightly higher than 75 bar (controlled by a dedicated pressurizer); this value guarantees sub-cooled water conditions of the

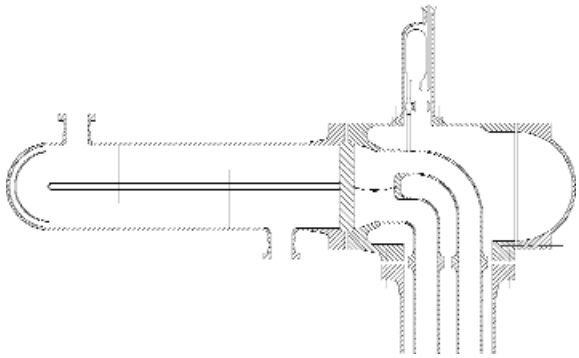


Fig. 5. SCCS primary heat exchanger.

fluid during any accidental situation or transient; the difference in level for natural circulation in the ISC loop is of about 10 m.

A second heat exchanger transfers the heat from the ISC circuit to the water of a reservoir. The steam produced in the reservoir is mixed with air initially present in the dome over the pool; pressure in the dome rises and this causes a flow of the air-steam mixture towards a small connection path with the atmosphere. Between the pool dome and the connection path with the atmosphere an inclined-tube heat exchanger is placed, where steam is partially condensed thanks to the action of external air drawn by a chimney.

The above mentioned choices introduced some constraints to the plant design; in particular the first limit, imposed by the functional requirements of the special emergency core cooling system, regards the rated thermal power, that cannot exceed approximately 1000 thermal MW, and in the solution herein described has been chosen equal to 600 MWth. Another characterizing parameter is the pressure in the primary system, chosen equal to 75 bar, which is different from the pressure values usually adopted in PWRs for the production of electric power (150-170 bar). This choice, that leads to a loss in thermodynamic efficiency of the plant because of the limitation of the higher isotherm in the steam cycle, has nevertheless allowed the adoption of the pressurized Containment for Primary loop Protection (CPP, the pressurized boundary that envelopes the primary cooling system and the emergency core cooling system), substantially eliminating the possibility of any type of loss-of-coolant accidents, including control rod ejection accident.

The inclusion of the primary coolant system (average operating temperature: 234 °C) inside the low-enthalpy-water-filled pressurized containment (CPP, at the temperature of 70 °C) requires thermal insulation to reduce heat losses from the primary coolant system. An insulating system has been designed on the external side of the whole primary coolant boundary (only the lower head of the reactor vessel is thermally insulated in the internal part), through matrices of stainless steel

wiring, that cause the presence of semi-stagnant water and can resist to high pressure and to fast pressure gradients, with acceptable shape modifications. This system limits heat losses to about 0.3% of the reactor thermal power.

The Experimental Facility NICOLE to Test the Innovative Safety Core Cooling System

To experimentally analyze the transient performance of the MARS emergency decay heat removal system, the experimental facility NICOLE (Naturally Induced circulation COoling Loop for Emergency) was designed and built.

The NICOLE plant essentially includes:
- a heat generator, which simulates the PSC/ISC heat exchanger;
- a pool (heat sink), which simulates the ISC/TSC heat exchanger;
- a water circulating loop, which includes a hot leg, a cold leg and which simulates the intermediate circulating loop of MARS emergency decay heat removal system.

The heat generator is a steel cylinder, having a 1.7 m diameter and a 2.6 m height. It is shown in the Fig. 6. A volume of diathermic oil (up to 6 m3) simulates the primary fluid in the PSC/ISC heat exchanger. The oil inventory can be varied to simulate different primary fluid thermal capacities.

Thermal power (up to 50 kW) is generated by 8 electrical heaters immersed in the oil volume. Oil has been used being its vapour pressure extremely low even at high temperatures (up to 200 °C).

The heat generator includes two 20 mm OD tube bundles, to transfer heat to the water circulating loop. Each tube bundle can be isolated by a check valve, to vary the heat transfer surface of the heat exchanger. A variable speed mixer allows to vary the oil heat transfer coefficient and to maintain the oil temperature uniform.

NICOLE pool is shown in the Fig. 7. In the pool a volume of water (up to 2.8 m3) simulates the TSC pool fluid in the MARS emergency decay heat removal system. Pool water inventory can be varied to modify its thermal capacity.

A heat exchanger, realized with a $^1/_2$" BWG tubes bundle, is immersed in the water pool. The number of operating tubes can be varied by isolation valves, to allow variations of the heat transfer surface. The initial pool water temperature can be varied through electrical heaters that are immersed in the pool water.

The circulating loop is realized by 1" OD steel tubes. It includes a hot leg, connecting the heat generator tubes bundles outlet with the pool tubes bundle inlet and a cold leg, between the pool HX outlet with the heat generator tube bundles inlet. A regulation valve is located on the cold leg, to impose further pressure drops in the loop and to simulate, in this way, different circulating loop configurations. A nitrogen pressurizer is placed on the hot leg of the circulating loop to limit its pressure variations.

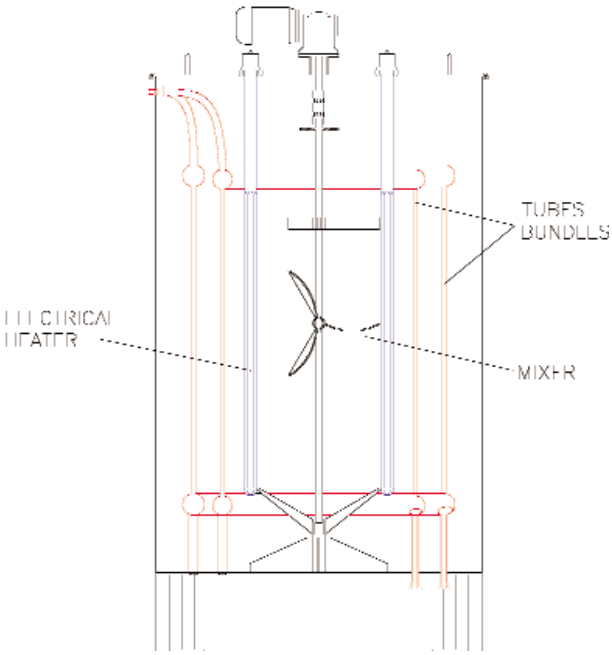


Fig. 6. Heat generator of the NICOLE experimental plant.

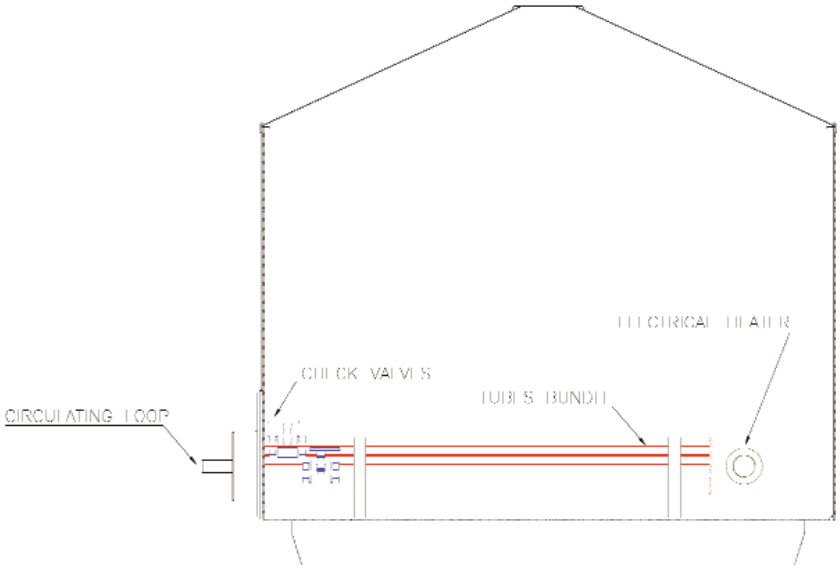


Fig. 7. Pool of the NICOLE experimental plant.

A further loop can be connected to the pool, with an air-cooled heat exchanger to condense the steam eventually produced and to maintain the liquid inventory in the pool.

NICOLE plant also includes the following auxiliary systems:
– a water charging/discharging system for the pool and the circulating loop;
– an oil charging/discharging system for the heat generator;
– a nitrogen charging/discharging system for the pressurizer;
– a compressed air circuit for the air pressure-actuated valves;
– a safety relief system on the pressurizer.

The experimental facility is instrumented with the following control instruments:
– 30 temperature transmitters;
– 5 pressure transmitters;
– 3 level transmitters;
1 mass flow transmitter.

The NICOLE plant is controlled by a computerized system, by which also transient conditions in the power generation or circulating loop pressure drops can be imposed. A simplified scheme of the experimental plant and its P&ID are shown in Figure 8.

The NICOLE plant has been also conceived as a scaled model of the intermediate loop of MARS emergency decay heat removal system.

The scaling procedure that has been used is the so called "time preserving volumetric" (Ishii and Kataoka, 1984). This procedure is particularly suitable for the scaling of natural circulation systems, because it allows to reproduce, through the employment of a scaled model, both the transient temporal evolution and the fluids temperature and pressure profiles.

With reference to the application of time preserving volumetric procedure, the parameters of the intermediate loop of MARS emergency decay heat removal system are:

H = vertical height of the emergency system intermediate loop = 10 m;
P = thermal power transferred in the PSC/ISC heat exchanger and in the ISC/TSC heat exchanger = 6 MW (decay heat = 1% of full power);
A = hot leg and cold leg flow area = 9.86×10-2 m^2 (16" tubes);
V = pool water volume = 290 m^3;
$(US)_{PSC/ISC}$ = 3.7×105 W/K;
$(US)_{ISC/TSC}$ = 3.49×105 W/K;
$\sum_j \left[\left(\frac{f_m \times L}{D_h} \right)_j + K_j \right]$ = equivalent friction factor of the intermediate loop = 56.5.

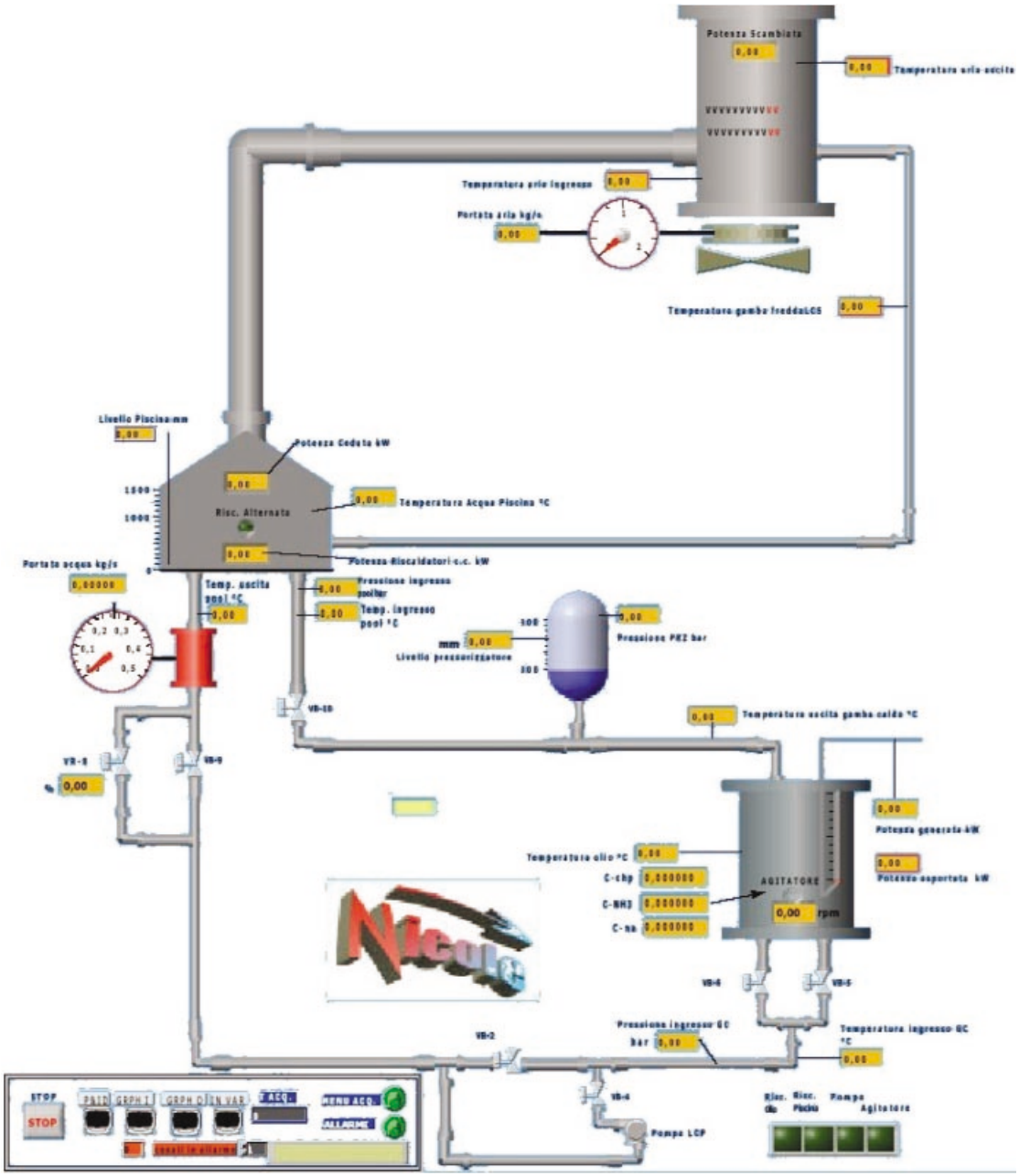


Fig. 8. Simplified scheme of the NICOLE plant.

Where:

L = component length in the fluid flow direction [m];
D_h = component hydraulic diameter [m];
f_m = Moody friction factor [-];
K = concentrated friction factor [-];
S = heat transfer surface area [m^2];
U = global heat transfer coefficient [W/m^2K];

The experimental plant parameters are:

H = plant vertical height = 9 m;
P = thermal power to be generated by electrical heaters in the heat generator = 20 kW;
A = flow area of hot leg and cold leg = 6.93 × 10-4 m^2;
V = pool water volume = 1 m^3;
$(US)_{GC}$ = 1.25 × 103 W/K;
$(US)_{POOL}$ = 1.17 × 103 W/K;

$$\sum_j \left[\left(\frac{f_m \times L}{D_h} \right)_j + K_j \right] = \text{equivalent friction factor} = 58.7.$$

Therefore, using 20 mm ID tubes for the hot leg and for the cold leg (at the moment they are realized through 1" tubes OD) and a 520 mm water level in the pool, NICOLE plant can be considered as a scaled model, by a scaling factor of approximately 1/300, of the intermediate loop of MARS emergency decay heat removal system.

Simulation Analysis

The experimental plant NICOLE has been preliminary tested in more than 30 experiments in transient conditions. The aim of these tests was to verify the plant functionality and operability and the accuracy of the control and data acquisition systems. The experimental data were also used to validate the RELAP5 nodalization, by which it has been possible to simulate with high accuracy different transients during the pre-test phases.

In Fig. 9, a typical comparison between the experimental data from the NICOLE plant and the RELAP5 results is shown. This example refers to the flow rate in natural circulation during a cooling phase of the system.

Experimental Analysis on the Natural Draft – Dry Cooling Tower

The goal of the experimentation is to investigate a dry cooling tower system in order to find the most efficient technical solutions to remove the core residual heat decay in passive safety conditions. The investigation regards the following points:

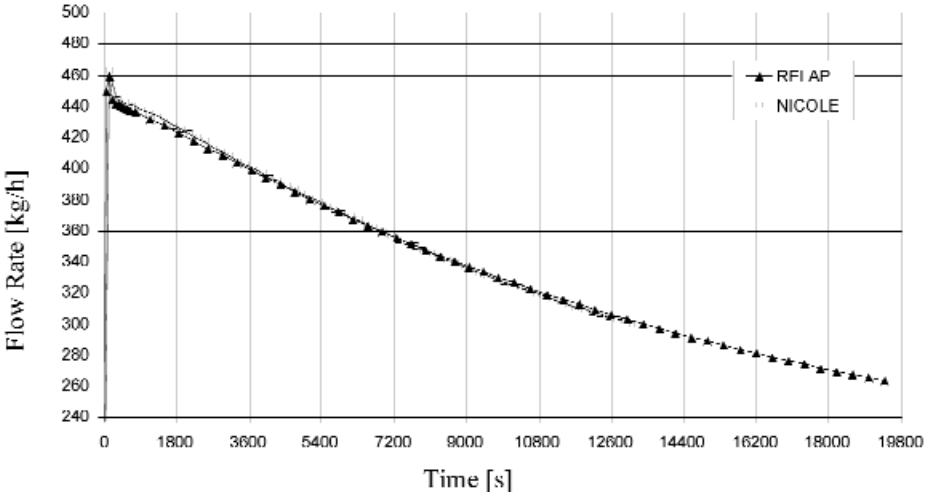


Fig. 9. Flow rate in a natural circulation test. Experimental data and RELAP results comparison.

– the influence of the heat exchanger arrangement: vertical and horizontal;
– the shape of the tower which depends on the angle inclination;
– the effects of the wind.

The experimental apparatus includes (see the picture in Figure 10):
– two different arrangements of the electrical heaters which simulate the heat exchanger;
– the general hyperboloidal shape of the tower reproduced by two conical pieces coupled in order to make up the throat. There are three different angle solutions for the lower cone: for each of them there are three or four different upper cones;
– an air fan to create the wind conditions.

The instrumentation used to perform the experimental analysis is:
– an air velocity transducer to measure the air velocity at the inlet just behind the heaters and at the throat;
– hot wire anemometers to measure the air temperature at the outlet;
– thermoresistences to measure the external air temperature and the temperature at the throat;
– a voltmeter/amperometer for measuring the values of the power given to the heaters.

The efficiency was evaluated by introducing a thermal parameter and a geometrical parameter.

The thermal parameter is defined as the ratio between the inlet-outlet temperature difference over the external temperature, $Ft = \Delta T/T_e$;

Fig. 10. The experimental apparatus.

The geometrical parameter is defined as the ratio between the throat diameter and the outlet diameter of the upper cone $Fg = D_t/D_o$.

The tests were performed at four different electrical heaters temperature (130°, 110°, 90° and 70°C) for each configuration (cone shapes). Then the power, the mass flow rate and the thermal heat transfer coefficient were measured in function of the thermal and the geometrical factor. By observing the results at the different heaters temperatures and in several ambient temperatures, it seems that a configuration that best performs the heat exchange does exist (in Figure 11 the mass flow rate is plotted versus of the geometrical factor, for a fixed throat diameter and at the heaters temperature of 110°C). This trend also appears by the CFD analysis (Fig. 12).

Back-Up Temperature-Actuated Scram System (ATSS)

General description

The core of the MARS reactor is optionally equipped with two different control rod systems. The first one, active type, is quite similar to classic PWR control

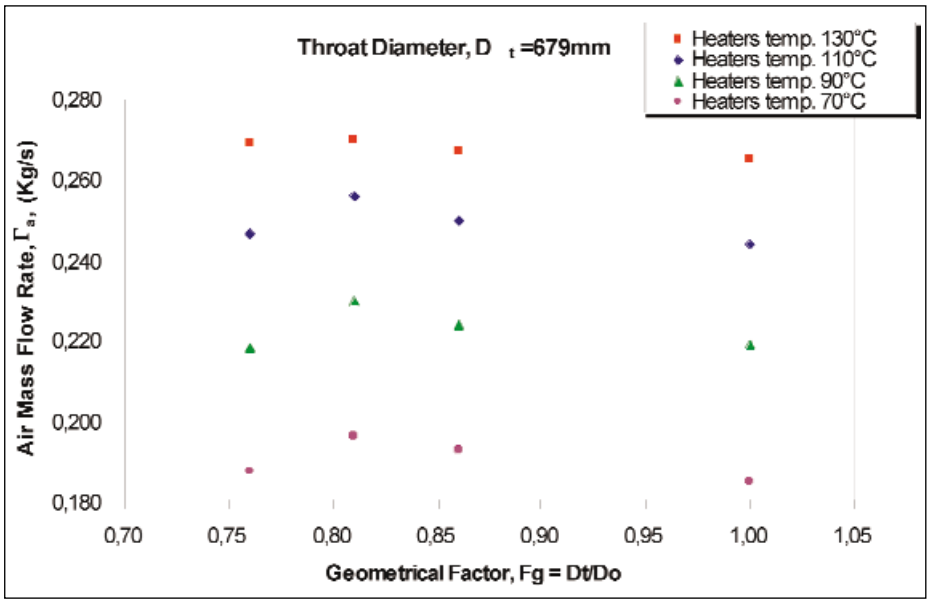


Fig. 11. Mass flow rate versus of the geometrical parameter.

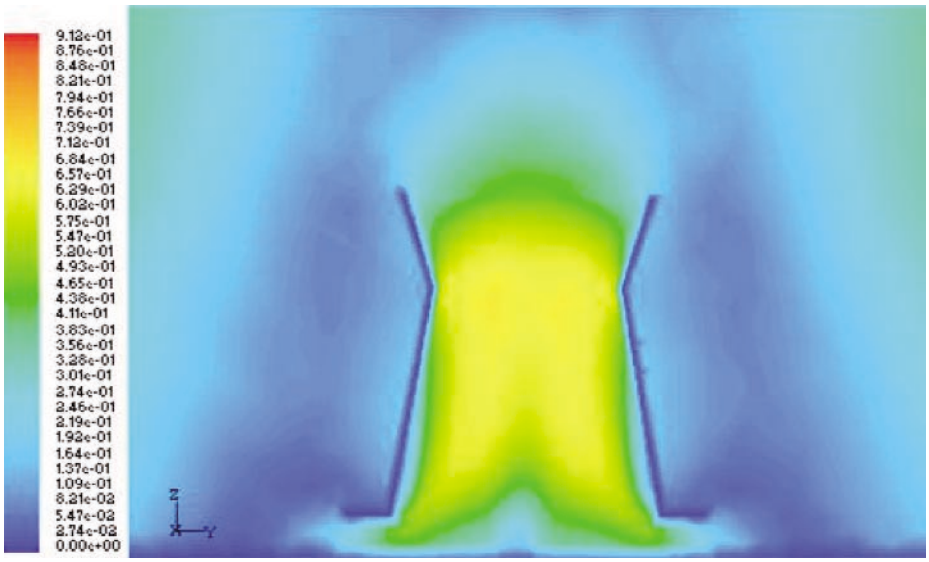


Fig. 12. The axial contours of the velocity magnitude by a CFD analysis for the tower with a throat diameter of 679 mm and outlet diameter of 888 mm at the electrical heaters temperature of 130°C.

rod systems; the second one, passive type, causes the control rods insertion into the core when the core coolant temperature reaches a selectable set value.

The overall number of fuel assemblies is 89 and the core diameter is 228 cm, compatible with the dimensions of the reactor vessel hypothesized during the preliminary design of the plant.

The core includes three zones with different enrichment (1/3 core strategy loading). The control rod clusters are distributed according to Fig. 13. They are grouped as follows:

4 blue clusters (belonging to the active system);
8 red clusters (belonging to the active system);
8 green clusters (belonging to the active system);
16 purple clusters (belonging to the active system);
9 yellow clusters (belonging to the passive system).

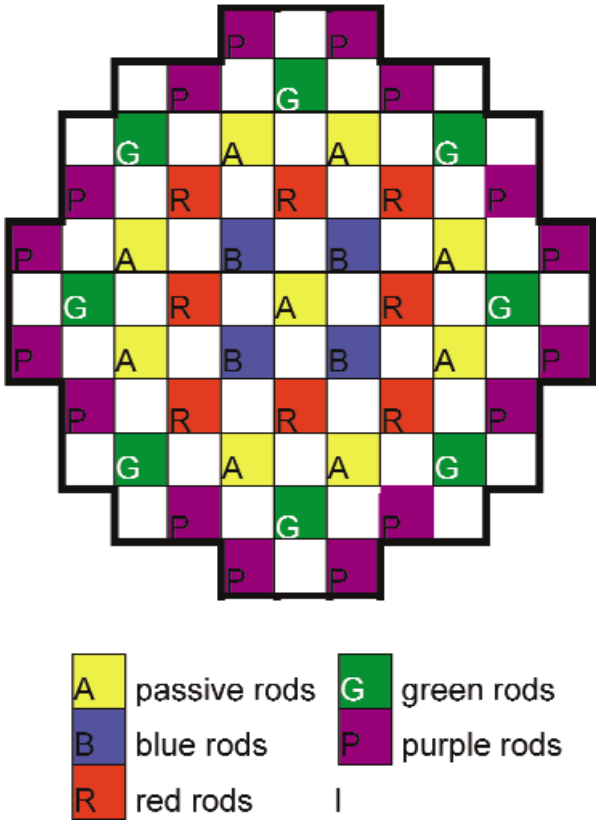


Fig. 13. Control rod cluster distribution.

The new scram system (ATSS) was conceived and designed to provide an automatic and safe shutdown of the reactor (scram) as soon as the fluid temperature in the core rises above a selected set point.

This special scram system also eliminates the occurrence of ATWS accidents.

The reactivity control by the ATSS is obtained through the action of control rod clusters with the same geometrical and physical characteristics as the traditional scram system. The innovative features of the ATSS are in the type of actuator selected. Each ATSS control rod cluster is controlled by a special actuator, based on a simple physical principle: the thermal expansion of a rod, due to the variation of temperature of the core coolant, which leads to the disconnection of hooks holding the control rod cluster.

From a mechanical point of view, the ATSS actuation principle is based on two rods made with materials with very different thermal expansion coefficients, which are connected in the lower extremities. If the upper extremity of rod B is fixed on the structure of the reactor vessel internals and that of rod A is maintained free, when a temperature variation in the core occurs, the A upper edge will exhibit a displacement referred to the B edge. This displacement is used to cause the hook disconnection (Fig. 14).

The ATSS two-metal sensor is manufactured using a concentric, internal rod (rod B) connected to a 9.6 O.D. mm hollow rod (with the dimensions of a fuel rod). This allows the insertion of the mechanism into the fuel element. The sensor is inserted into the center of a fuel assembly which hosts the control rod cluster.

The differential expansion is transmitted to the upper extremity, where the control rod disconnection device is placed: the mechanical force developed in the sensor itself directly releases the hook sustaining the ATSS control rod cluster (Fig. 15).

In the preliminary design, the ATSS sensor is made of two coaxial cylinders:

- the internal cylinder is made of INVAR (Fe 63.5%, Ni 36%, Mg+C bal.): $\alpha_1 = 4.5 \cdot 10^{-6}\ °C^{-1}$;
- the external hollow cylinder is made of AISI 316: $\alpha_2 = 17 \cdot 10^{-6}\ °C^{-1}$.

If L is the total length of the two cylinders, the corresponding differential displacement Dl is:

$$\Delta l = \Delta l_1 - \Delta l_2 = \alpha_1 \bullet L_1 \bullet \Delta T - \alpha_2 \bullet L_2 \bullet \Delta T = (\alpha_1 L_1 - \alpha_2 L_2) \bullet \Delta T$$

For instance, with $L_1 = L_2 \approx 5$m, an increase ΔT=30°C leads to a displacement of about 1.6 mm (Fig. 16). Even taking into account the compression due to friction forces, the expected "output" differential displacement is about 1.2 mm. Detailed calculations through specific computer programs have been performed to analyze the time response of the new device and its geometrical and thermomechanical behaviour, with particular reference to the force to be developed on the hook.

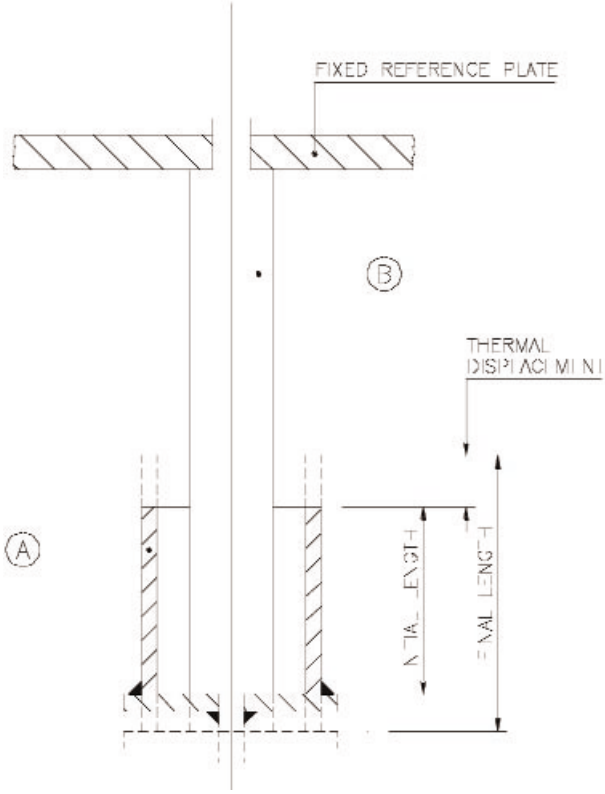


Fig. 14. Operation scheme of ATSS.

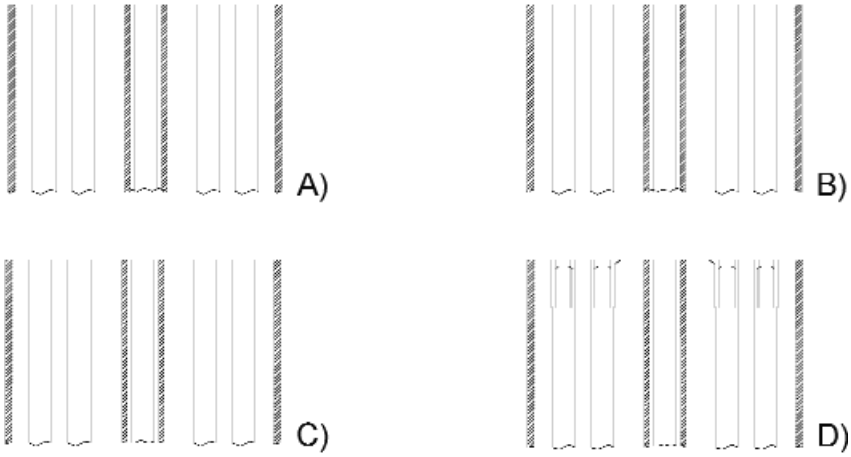


Fig. 15. ATSS operation sequence.

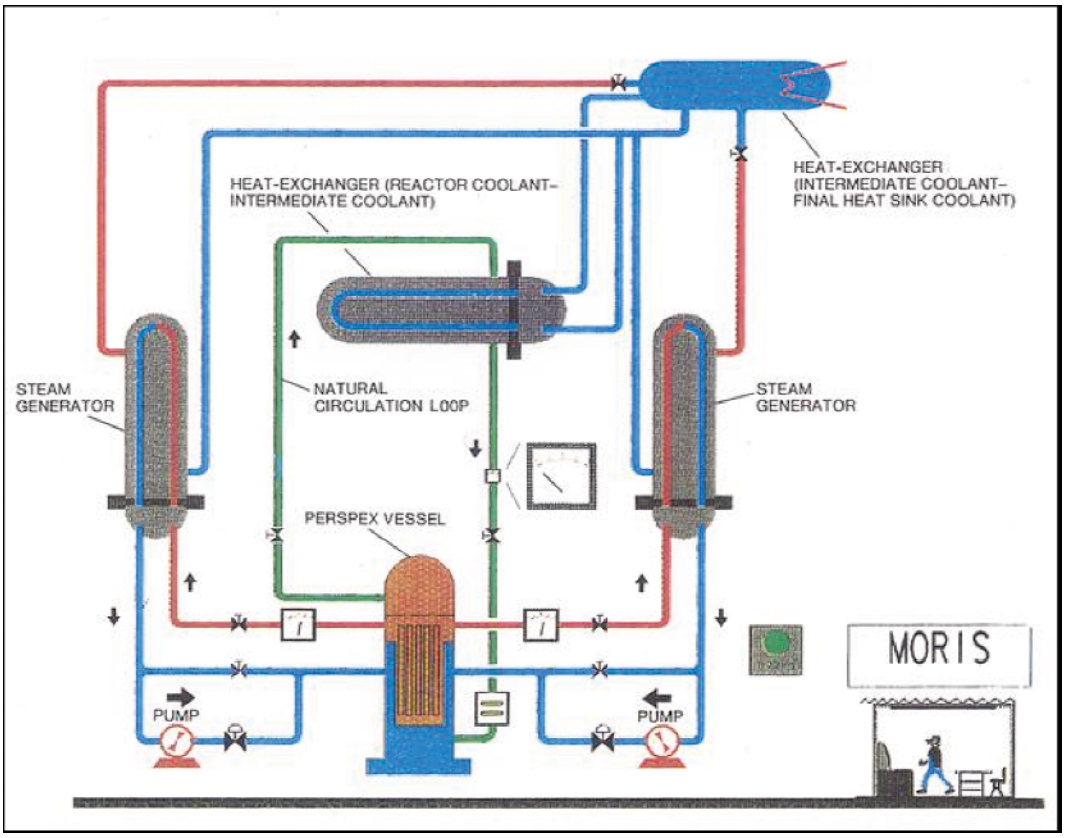


Fig. 16. Simplified scheme of the MORIS Experimental Facility.

Actually, new analyses based on different materials are being performed to obtain greater displacement in less than 2 seconds (assuming a similar increase of the average temperature in the core as in the preliminary design). Two special alloys have been identified: AISI 72, characterized by an expansion coefficient of about $27 \cdot 10^{-6}$ °C^{-1}, and Alloy 39, with $\alpha_1 = 3.2 \cdot 10^{-6}$ °C^{-1}. Coupling these two materials, an effective expansion of about 2.9 mm in 2 seconds can be obtained.

Other experimental activities

In the specific case of the MARS reactor plant, the experimental activities regard only the performance demonstration of fluid circuits and of a few innovative mechanical components. In fact, this nuclear plant is based on consolidated, well proven, traditional engineered solutions for all the "nuclear" system. The innovative parts of the MARS design regard the performance of hydraulic circuits whose

operation is based on simple physical laws, according to plant solutions which are very well consolidated and well proven in generalized engineering applications.

Since the safety behavior of the MARS nuclear plant and, in particular, of its emergency cooling system, is easy-to-understand and may be conceptually shown through easy experimental tests, an experimental activity has been nevertheless carried out.

In the following, some of additional experimental tests carried out are briefly summarized.

An experimental facility (called MORIS) was built at the ENEA research centre of Casaccia, to simulate the general thermal-hydraulic behavior of the MARS primary cooling system and of the emergency core cooling system (see Fig. 16, Palazzi et al., 1988). The tests were successful.

In the same laboratories, the innovative check valve developed and designed for application in parallel with a traditional-type clapet valve in each line of the two parallel trains of the primary loop of the safety core cooling system (SCCS) was successfully tested, in the experimental facility CIVAP (see Figs 17 and 18, Ferrari et al., 1997), in an important scale and with clean and dirty water.

Fig. 17. CIVAP Experimental Facility.

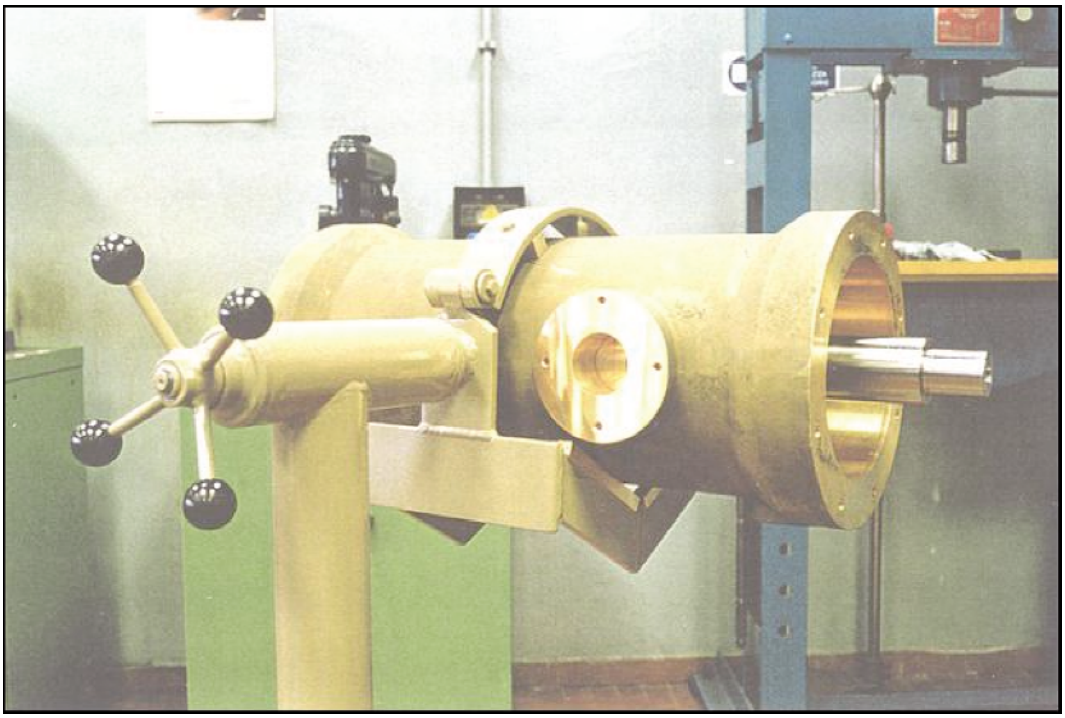


Fig. 18. Innovative MARS SCCS check valve during assembling before testing.

Among the additional experimental activities carried out at the University of Rome "La Sapienza", in the laboratories of the "Nuclear engineering and energy conversion" Department, concerning thermal-hydraulic phenomena and components, we mention:

– natural circulation in steady state and in transient conditions, simulating the operation conditions of the SCCS;

– boiling in pools with water at atmospheric pressure, with inclined tubes, simulating the pool heat exchangers of the SCCS;

– condensation of steam with low and high concentration of non-condensable gases inside inclined tubes, simulating the atmospheric condenser of the SCCS.

The tests have shown a very good performance and a very high reliability of the components and systems of the MARS nuclear plant.

REFERENCES

Cumo M., Naviglio A., Caira M., Sorabella L. 1997. Design principles, targets and criteria for nuclear reactor plants with improved safety features The Second Int. Conf. on Advanced Reactor Safety - ARS '97, Orlando, Florida, Nov. 1997.

Ishii M. and Kataoka I. 1984. Scaling laws for thermal-hydraulic systems under single phase and two-phase natural circulation. Nuclear Engineering and Design, 81, 411-425.

G. Palazzi, D. Savelli, "Heat Transfer and Fluid Flow Experimental Researches Applied to Power Plants" - ENERGIA NUCLEARE n° 3, Sept. 1988.

T. Ferrari, P. Incalcaterra, C. Kropp, "Attività Sperimentali su Prototipo di valvola di non ritorno ad azionamento passivo" - ENEA - EIHE 97037 - 1997.