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Analysis of Primary/Containment Coupling Phenomena Characterizing the MASLWR Design During a SBLOCA Scenario

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1. Introduction

Today considering the world energy demand increase, the use of advanced nuclear power plants, have an important role in the environment and economic sustainability of country energy strategy mix considering the capacity of nuclear reactors of producing energy in safe and stable way contributing in cutting the CO_2 emission (Bertel & Morrison, 2001; World Energy Outlook-Executive Summary, 2009; Wolde-Rufael & Menyah, 2010; Mascari et al., 2011d). According to the information's provided by the "Power Reactor Information System" of the International Atomic Energy Agency (IAEA), today 433 nuclear power reactors are in operation in the world providing a total power installed capacity of 366.610 GWe, 5 nuclear reactors are in long term shutdown and 65 units are under construction (IAEA PRIS, 2011).

In the last 20 years, the international community, taking into account the operational experience of the nuclear reactors, starts the development of new advanced reactor designs, to satisfy the demands of the people to improve the safety of nuclear power plants and the demands of the utilities to improve the economic efficiency and reduce the capital costs (D'Auria et al., 1993; Mascari et al., 2011c). Design simplifications and increased design margins are included in the advanced Light Water Reactors (LWR) (Aksan, 2005). In this framework, the project of some advanced reactors considers the use of emergency systems based entirely on natural circulation for the removal of the decay power in transient condition and in some reactors for the removal of core power during normal operating conditions (IAEA-TECDOC-1624, 2009; Mascari et al., 2010a; Mascari et al., 2011d). For example, if the normal heat sink is not available, the decay heat can be removed by using a passive connection between the primary system and heat exchangers (Aksan, 2005; Mascari et al., 2010a, Mascari, 2010b). The AP600/1000 (Advanced Plant 600/1000 MWe) design, for

example, includes a Passive Residual Heat Removal (PRHR) system consisting of a C-Tube type heat exchanger immersed in the In-containment Refueling Water Storage Tank (IRWST) and connected to one of the Hot Legs (HL) (IAEA-TECDOC-1391, 2004; Reyes, 2005c; Gou et al., 2009; Mascari et al., 2010a). A PRHR from the core via Steam Generators (SG) to the atmosphere, considered in the WWER-1000/V-392 (Water Moderated, Water Cooled Energy Reactor) design, consists of heat exchangers cooled by atmospheric air, while the PRHR via SGs, considered in the WWER-640/V-407 design, consists of heat exchangers immersed in emergency heat removal tanks installed outside the containment (Kurakov et al., 2002; IAEA-TECDOC-1391, 2004; Gou et al., 2009; Mascari et al., 2010a). In the AC-600 (Advanced Chinese PWR) the PRHR heat exchangers are cooled by atmospheric air (IAEA-TECDOC 1281, 2002; Zejun et al., 2003; IAEA-TECDOC-1391, 2004; Gou et al., 2009; Mascari et al., 2010a) and in the System Integrated Modular Advanced Reactor (SMART) the PRHR heat exchangers are submerged in an in-containment refuelling water tank (IAEA-TECDOC-1391, 2004; Lee & Kim, 2008; Gou et al., 2009; Mascari et al., 2010a). The International Reactor Innovative and Secure (IRIS) design includes a passive Emergency Heat Removal System (EHRS) consisting of an heat exchanger immersed in the Refueling Water Storage Tank (RWST). The EHRS is connected to a separate SG feed and steam line and the RWST is installed outside the containment structure (Carelli et al., 2004; Carelli et al., 2009; Mascari, 2010b; Chiovaro et al., 2011). In the advanced BWR designs the core water evaporates, removing the core decay heat, and condenses in a heat exchanger placed in a pool. Then the condensate comes back to the core (Hicken & Jaegers, 2002; Mascari et al., 2010a). For example, the SWR-1000 (Siede Wasser Reaktor, 1000 MWe) design has emergency condensers immersed in a core flooding pool and connected to the core, while the ESBWR (Economic Simplified Boiling Water Reactor) design uses isolation condensers connected to the Reactor Pressure Vessel (RPV) and immersed in external pools (IAEA-TECDOC-1391, 2004; Aksan, 2005; Mascari et al., 2010a).

The designs of some advanced reactors rely on natural circulation for the removing of the core power during normal operation. Examples of these reactors are the MASLWR (Multi-Application Small Light Water Reactor), the ESBWR, the SMART and the Natural Circulation based PWR being developed in Argentina (CAREM)(IAEA-TECDOC-1391, 2004; IAEA -TECDOC-1474, 2005; Mascari et al., 2010a). In particular the MASLWR (Modro et al., 2003), figure 1, is a small modular integral Pressurized Water Reactor (PWR) relying on natural circulation during both steady-state and transient operation.

In the development process of these advanced nuclear reactors, the analysis of single and two-phase fluid natural circulation in complex systems (Zuber, 1991; Levy, 1999; Reyes & King, 2003; IAEA-TECDOC-1474, 2005; Mascari et al., 2011e), under steady state and transient conditions, is crucial for the understanding of the physical and operational phenomena typical of these advanced designs. The use of experimental facilities is fundamental in order to characterize the thermal hydraulics of these phenomena and to develop an experimental database useful for the validation of the computational tools necessary for the operation, design and safety analysis of nuclear reactors. In general it is expensive to design a test facility to develop experimental data useful for the analyses of complex system, therefore reduced scaled test facilities are, in general, used to characterize them. Since the experimental data produced have to be applicable to the full-scale prototype, the geometrical characteristics of the facility and the initial and boundary

conditions of the selected tests have to be correctly scaled. Since possible scaling distortions are present in the experimental facility design, the similitude of the main thermal hydraulic phenomena of interest has to be assured permitting their accurate experimental simulation (Zuber, 1991; Reyes, 2005b; Reyes et al., 2007; Mascari et al., 2011e).

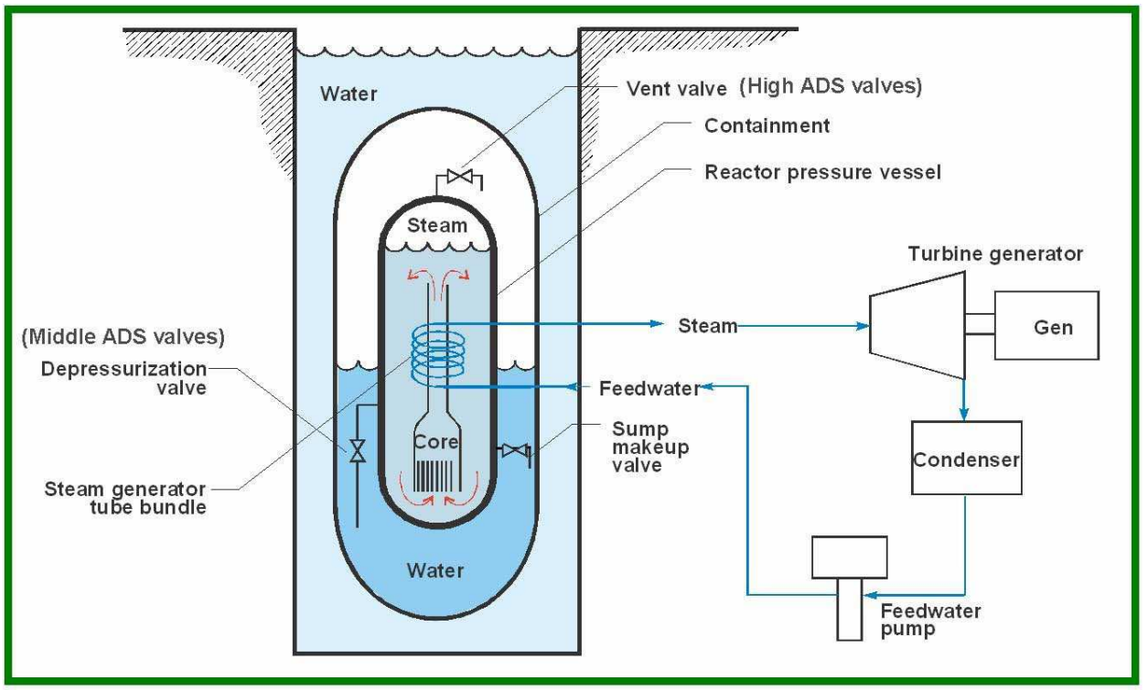


Fig. 1. MASLWR conceptual design layout (Modro et al, 2003; Reyes et al., 2007; Mascari et al., 2011a).

Different computer codes have been developed to characterize two-phase flow systems, from a system and a local point of view. Accurate simulation of transient system behavior of a nuclear power plant or of an experimental test facility is the goal of the best estimate thermal hydraulic system code. The evaluation of a thermal hydraulic system code's calculation accuracy is accomplished by assessment and validation against appropriate system thermal hydraulic data, developed either from a running system prototype or from a scaled model test facility, and characterizing the thermal hydraulic phenomena during both steady state and transient conditions. The identification and characterization of the relevant thermal hydraulic phenomena, and the assessment and validation of thermal hydraulic systems codes, has been the objective of multiple international research programs (Mascari et al., 2011a; Mascari et al., 2011c).

In this international framework, Oregon State University (OSU) has constructed, under a U.S. Department of Energy grant, a system level test facility to examine natural circulation phenomena of importance to the MASLWR design. The scaling analysis of the OSU-MASLWR experimental facility was performed in order to have an adequately simulation of the single and two-phase natural circulation, reactor system depressurization during a blowdown and the containment pressure response typical of the MASLWR prototype (Zuber, 1991; Reyes & King, 2003; Reyes, 2005b). A previous testing program has been

conducted in order to assess the operation of the prototypical MASLWR under normal full pressure and full temperature conditions and to assess the passive safety systems under transient conditions (Modro et al. 2003; Reyes & King, 2003; Reyes, 2005b; Reyes et al., 2007; Mascari et al., 2011e). The experimental data developed are useful also for the assessment and validation of the computational tools necessary for the operation, design and safety analysis of nuclear reactors.

For many years, in order to analyze the LWR reactors, the USNRC has maintained four thermal-hydraulic codes of similar, but not identical, capabilities, the RAMONA, RELAP5, TRAC-B and TRAC-P. In the last years, the USNRC is developing an advanced best estimate thermal hydraulic system code called TRAC/RELAP Advanced Computational Engine or TRACE, by merging the capabilities of these previous codes, into a single code (Boyac & Ward, 2000; TRACE V5.0, 2010; Reyes, 2005a; Mascari et al., 2011a). The validation and assessment of the TRACE code against the MASLWR natural circulation database, developed in the OSU-MASLWR test facility, is a novel effort.

This chapter illustrates an analysis of the primary/containment coupling phenomena characterizing the MASLWR design mitigation strategy during a SBLOCA scenario and, in the framework of the performance assessment and validation of thermal hydraulic system codes, a qualitative analysis of the TRACE V5 code capability in reproducing it.

2. MASLWR conceptual design and SBLOCA mitigation strategy

The MASLWR (Modro et al., 2003; Reyes et al. 2007, Mascari et al., 2011a; Mascari et al., 2011e), figure 1, is a small modular integral PWR of 35 MWe developed by Idaho National Engineering and Environmental Laboratory, OSU and Nexant–Bechtel. During steady state condition the primary fluid, in single phase natural circulation, removes the core power and transfers it to the secondary fluid through helical coil SG. In transient condition the core decay heat is removed through a passive primary/containment coupling mitigation strategy based on natural circulation. The use of natural circulation reduces the number of active components simplifying the configuration of nuclear steam supply system. The reactor core and a helical coil SG are both located within the RPV. The integrated SG consists of banks of vertical helical tubes located in the upper region of the vessel outside of the HL chimney.

Its small size considered the prototypical MASLWR relatively portable and thus well suited for employment in smaller electricity grids but take into account its design simplicity, its simplified parallel construction, the consequent reduction of the capital costs, reduction of construction time, reduction of finance and operation cost, recognizes it to be able to reach larger electricity market in developing and developed regions (Modro et al., 2003; Reyes & Lorenzini, 2010; Mascari et al., 2011d).

As it is shown in figure 1, the primary coolant flows outside the SG tubes, and the Feed Water (FW) is fully vaporized resulting in superheated steam at exit of the SG. The safety systems are designed to operate passively. The RPV is surrounded by a cylindrical containment, partially filled with water. This containment provides pressure suppression and liquid makeup capabilities and is submerged in a pool of water that acts as the ultimate heat sink. The MASLWR steady-state operating conditions are reported in the table 1.

Reactor Thermal Power	150 MW
Primary Pressure	7.60 MPa
Primary Mass Flow Rate	597 kg/s
Reactor Inlet Temperature	491.80 K
Reactor Outlet Temperature	544.30 K
Primary Side Saturation Temperature	565 K
Secondary Side Steam Pressure	1.50 MPa
Secondary Side Steam Outlet Quality	1
Secondary Side Steam Temperature	481.40 K
Secondary Side Saturation Temperature	471.60 K
Feedwater Temperature	310 K
Feedwater Flowrate	56.10 kg/s

Table 1. MASLWR steady-state operating conditions (Modro et al., 2003; Reyes & King, 2003; Reyes, 2005b).

The RPV, figure 1, can be depressurized using the Automatic Depressurization System (ADS), consisting of six valves discharging into various locations within the containment. In particular two independent vent valves (high ADS valves), two independent depressurization valves (middle ADS valves) and two independent sump recirculation valves are considered in the MASLWR design.

The integral arrangement of the plant allows avoiding pressurized primary components outside the RPV eliminating the possibility of large break Loss of Coolant Accident (LOCA) and reducing the Small Break LOCA (SBLOCA) initiating event. Of particular interest is the SBLOCA mitigation strategy typical of the MASLWR design. Following, for example, an inadvertent opening of an ADS valve, a primary side blowdown into the pressure suppression containment takes place. The RPV blowdown causes a primary pressure decrease and a consequent containment pressure increase causing a safety injection signal. It automatically opens, figure 1, the high ADS valves, the middle ADS valves and the sump recirculation valves. As the primary and the containment pressures become equalized, the blowdown is terminated, and a natural circulation flow path is established. Infact, when the sump recirculation valves are opened the vapor produced in the core goes in RPV upper part and through the high ADS valve goes to the containment where it is condensed. At this point through the sump recirculation lines and the down comer the fluid goes to the core again. The pressure suppression containment is submerged in a pool that acts as the ultimate heat sink. This mechanism, based on natural circulation, permits the cooling of the core (Modro et al., 2003; Reyes & King, 2003; Reyes, 2005b; Reyes et al., 2007; Mascari et al., 2011).

The MASLWR concept design and its passive safety features were tested in a previous test campaign developed at the OSU-MASLWR experimental facility (Modro et al., 2003; Reyes & King, 2003; Reyes et al., 2007; Mascari et al., 2011a; Mascari et al., 2011e), figure 2. The planned work related to the OSU-MASLWR test facility will be not only to specifically

investigate the MASLWR concept design further but also advance the broad understanding of integral natural circulation reactor plants and accompanying passive safety features as well. Furthermore an IAEA International Collaborative Standard Problem (ICSP) on the "Integral PWR Design Natural Circulation Flow Stability and Thermo-Hydraulic Coupling of Containment and Primary System During Accidents" is being hosted at OSU and the experimental data will be collected at the OSU-MASLWR facility. The purpose of this IAEA ICSP is to provide experimental data on single/two-phase flow instability phenomena under natural circulation conditions and coupled containment/reactor vessel behavior in integral-type reactors (Woods & Mascari, 2009; Woods et al.; 2011).

3. OSU-MASLWR test facility

The OSU-MASLWR test facility (Modro et al., 2003; Reyes & King, 2003; Reyes et al., 2007; Galvin, 2007; Mascari et al., 2011a, 2011b, 2011c, 2011d, 2011e), figure 2, is scaled at 1:3 length scale, 1:254.7 volume scale and 1:1 time scale, is constructed entirely of stainless steel and it is designed for full pressure and full temperature prototype operation.

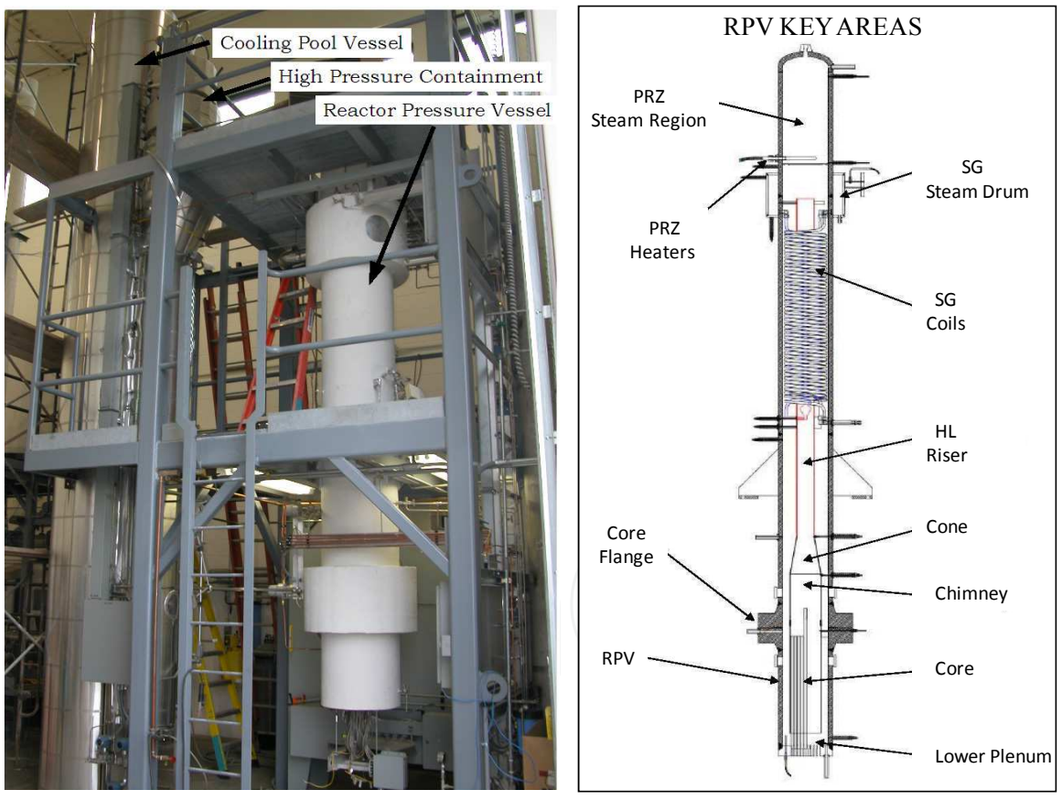


Fig. 2. OSU-MASLWR test facility layout (Reyes et al., 2007; Mascari et al., 2011a; Mascari et al., 2011e).

The facility includes the primary and secondary circuit and the containment structures. Two vessels, a High Pressure Containment (HPC) vessel and a Cooling Pool Vessel (CPV) with an heat transfer surface between them to establish the proper heat transfer area, are used to model the containment structures, in which the RPV sits, as well as the cavity within which the

containment structure is located. The two middle ADS lines, two high ADS lines and the two ADS sump recirculation lines are modelled separately. In addition to the physical structures that comprise the test facility, there are data acquisition, instrumentation and control systems.

The facility is instrumented for capturing its thermal hydraulic behaviour during steady and transient conditions; in particular thermocouples are used to measure fluid, wall and heater temperature; pressure transducers are used to measure pressure; differential pressure cells are used to measure water level, pressure loss and flow rate; flow Coriolis meters are used to measure mass flow rate; vortex flow meters are used to measure steam mass flow rate, pressure and temperature; power meters are used to measure heater power.

In the previous testing program four tests have been conducted: the OSU-MASLWR-001 -inadvertent actuation of 1 submerged ADS valve-; the OSU-MASLWR-002 -natural circulation at core power up to 210 kW-; the OSU-MASLWR-003A- natural circulation at core power of 210 kW (Continuation of test 002)-; the OSU-MASLWR-003B -inadvertent actuation of 1 high containment ADS valve-.

Since the target of the OSU-MASLWR-001 test was to determine the pressure behavior of the RPV and containment following an inadvertent actuation of one middle ADS valve, it gives a wide number of informations about the primary/containment coupling phenomena characterizing the MASLWR design. Therefore it is the test chosen for this analysis.

3.1 OSU-MASLWR RPV description

The internal components of the RPV, figure 2, are the core, the HL riser, the Upper Plenum (UP), the Pressurizer (PRZ), the SG primary side, the Cold Leg (CL) downcomer and the Lower Plenum (LP). The RPV shell is covered by Thermo-12 hydrous calcium silicate insulation.

The core is modelled with 56 cylindrical heater rods distributed in a 1.86 cm pitch square array with a 1.33 pitch to diameter ratio. The nominal power of each heater rod is 7.1 kW resulting in a maximum core power of 398 kW. The diameter of the core rod is 1.59 cm.

A lower core flow plate, figure 3, contains 76 auxiliary flow holes of 0.635 cm of diameter, arranged at 1.86 cm pitch square array, and 57 core rod flow holes. In order to create a flow annulus between the flow plate and the core rod, the holes of the rodded lower core flow plate are oversized at 1.72 cm.

The core is shrouded, figure 4, to ensure all flow enters the core via the bottom and travels the entire heated length. The shroud is shaped to partially block the primary coolant flow through the outermost auxiliary flow holes in order to ensure that each heated rod receives approximately equal axial coolant flow. The amount of blockage is dependent on the number and location of heated rods adjacent to each auxiliary flow hole. At mid elevation a core grid wires is considered in order to maintain the radial alignment of the core rods. Four thermocouples for measuring the core inlet temperatures are located at the bottom CL entering the core. The core heater rod temperatures are measured. Six thermocouples vertically spaced every 15.24 cm measuring water temperatures, are located in the center of core thermocouple rod. The pressure loss in the core is measured; the power to the core heater rod bundles is measured.

The HL riser, figure 2, consists of a lower region, an upper region and a transition region. The lower region consists of a pipe with an outside diameter of 20.32 cm, an inside diameter of 19.71 cm and a wall thickness of 0.305 cm. The upper region consists of a pipe with an outside diameter of 11.43 cm, an inside diameter of 10.23 cm, and a wall thickness of 0.602 cm. The transition region consists of a cone with a 0.305 cm thickness and an half angle of 20.61°. The pressure loss between core top and riser cone, the pressure loss in the riser cone and the pressure loss in the chimney, from the exit of the transition cone to the UP, are measured. Along the riser a thermocouple measures the water temperature inside chimney below SG coil and another one measures the water temperature at top of chimney. The flow rate within the HL chimney is measured with a differential pressure cell used to measure flow. The primary containment water level is measured as well.

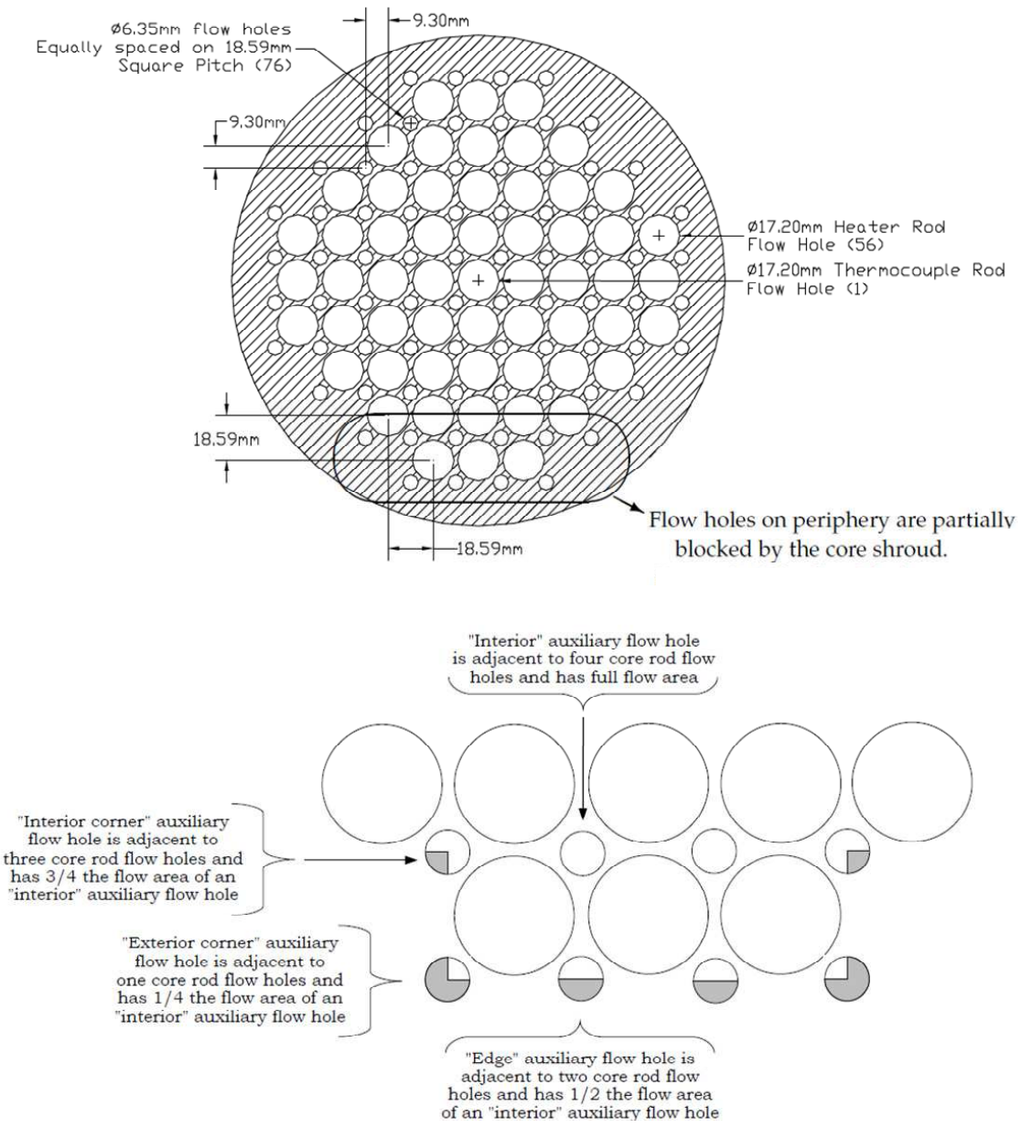


Fig. 3. Lower core flow plate layout and auxiliary flow hole blockage by core shroud (Galvin, 2007; Mascari et al., 2011e).

The UP is separated from the heated upper PRZ section by a thick baffle plate having eight 2.54 cm holes, spaced uniformly around the baffle plate periphery, which allow free communication of the PRZ to the remainder of the RPV during normal operation and for volume surges into and/or out of the PRZ due to transients.

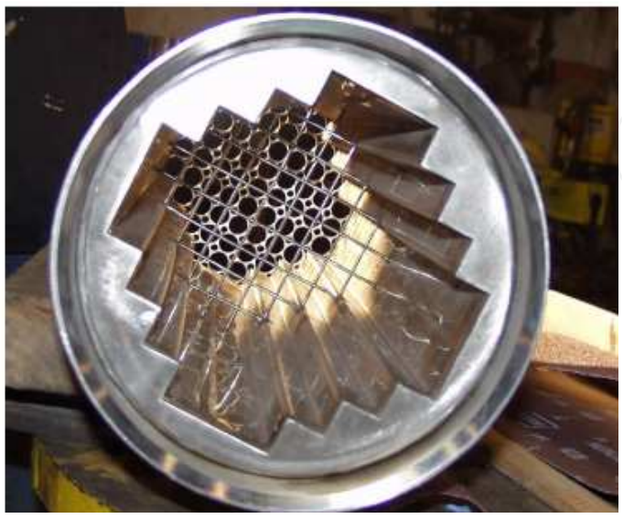


Fig. 4. Core shroud photo (Galvin, 2007; Mascari et al., 2011e).

The PRZ is integrated in the RPV and is located in its upper part. In the PRZ are located three heater elements, each 4 kW, that are modulated by the test facility control system to maintain nominal primary system pressure at the desired value. The PRZ steam temperature and pressure are measured. The PRZ level is measured as well.

The CL downcomer region, figure 2, is an annular region bounded by the RPV wall on the outside and the HL riser on the inside, and the flow area reduces at the HL riser cone. In the SG primary side section is inserted the SG helical coil bundle. Thermocouples are located in the CL downcomer region to measure water down flow temperatures after SG coils. The primary side pressure loss across SG and the pressure loss in the annulus below SG are measured.

The SG of the facility is a once through heat exchanger and is located within the RPV in the annular space between the HL riser and the inside surface of the RPV. The tube bundle, is a helical coil consisting of fourteen tubes with a total heated length of 86 m. There are three separate parallel coils of stainless steel tubes. The outer and middle coils consist of five tubes each while the inner coil consists of four tubes. The value of the degree of the steam superheat is changed in order to control the facility. A thermocouple is located at the exit of each helical coil. The main steam pressure and temperature are measured. The main steam mass flow rate is measured with a vortex flow meter. The FW supply in the SG outer, middle and inner coil mass flow rate are measured with flow Coriolis meters. The related pressures are measured. The FW temperature is measured as well.

3.2 OSU-MASLWR containment structures description

The HPC, figure 5, consists of a lower cylindrical section, an eccentric cone section, an upper cylindrical section and an hemispherical upper end head. For scaling reasons, in order to

have an adiabatic boundary condition in all the wall of the HPC except through the heat transfer plate wall where the condensation has to take place, four groups of containment heaters have been installed permitting the heat transfer takes place only between the CPV and HPC containment. These heaters are located in the exterior surface of the HPC, under the insulation, and above the containment water level. The temperatures of heaters located on the walls of the HPC and the temperatures within the walls of the HPC, between the heaters and the water, are measured. The entire HPC is covered by Thermo-12 hydrous calcium silicate insulation. The HPC level and pressure are measured.

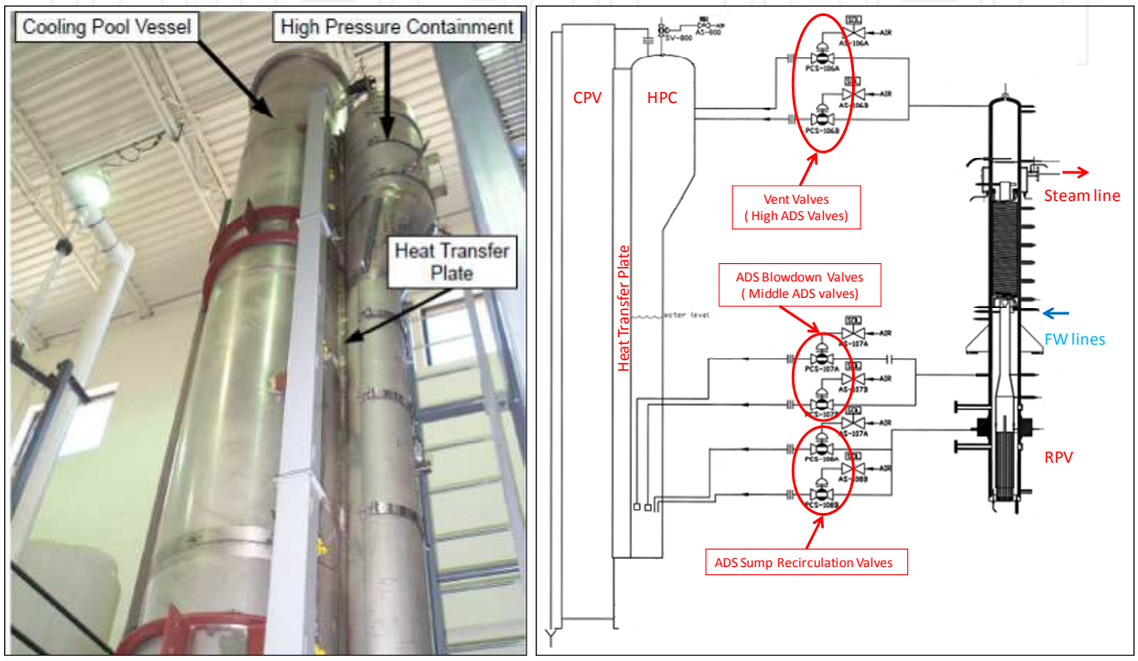


Fig. 5. OSU-MASLWR containment structures (Reyes et al., 2007; Mascari et al., 2011d; Mascari et al., 2011e).

The CPV consists of a tall right cylindrical tank covered by Thermo-12 hydrous calcium silicate insulation. The CPV level and water temperature are measured. One disk rupture is connected between the HPC and the CPV.

The heat transfer plate, having the same height of the HPC without the hemispherical head, provides the heat conduction between the HPC and CPV. The heat transfer plate is scaled in order to model the heat transfer area between the MASLWR design high pressure containment vessel and the cooling pool in which it sits.

Five thermocouples are located at six different elevations to measure the temperature distribution from the HPC to the CPV. In particular one group of thermocouples measures the water temperatures located inside the HPC near the heat transfer plate, one measures the water temperatures located inside the CPV near the heat transfer plate, one measures the wall temperatures at the midpoint of the heat transfer plate between the CPV and the HPC, one measures wall temperatures within the heat transfer plate between the CPV and the HPC nearest to the HPC and one measures wall temperatures within the heat transfer plate between the CPV and the HPC nearest to the CPV.

3.3 OSU-MASLWR ADS lines description

The high ADS lines, figure 6, are horizontally oriented and connect the PRZ steam space with the HPC. A pneumatic motor operated globe valve is located in each line. Downstream from each isolation valve is a transition piece with an internal square-edge orifice. The two high ADS lines enter the HPC above the waterline, penetrate it and then terminate with a sparger.

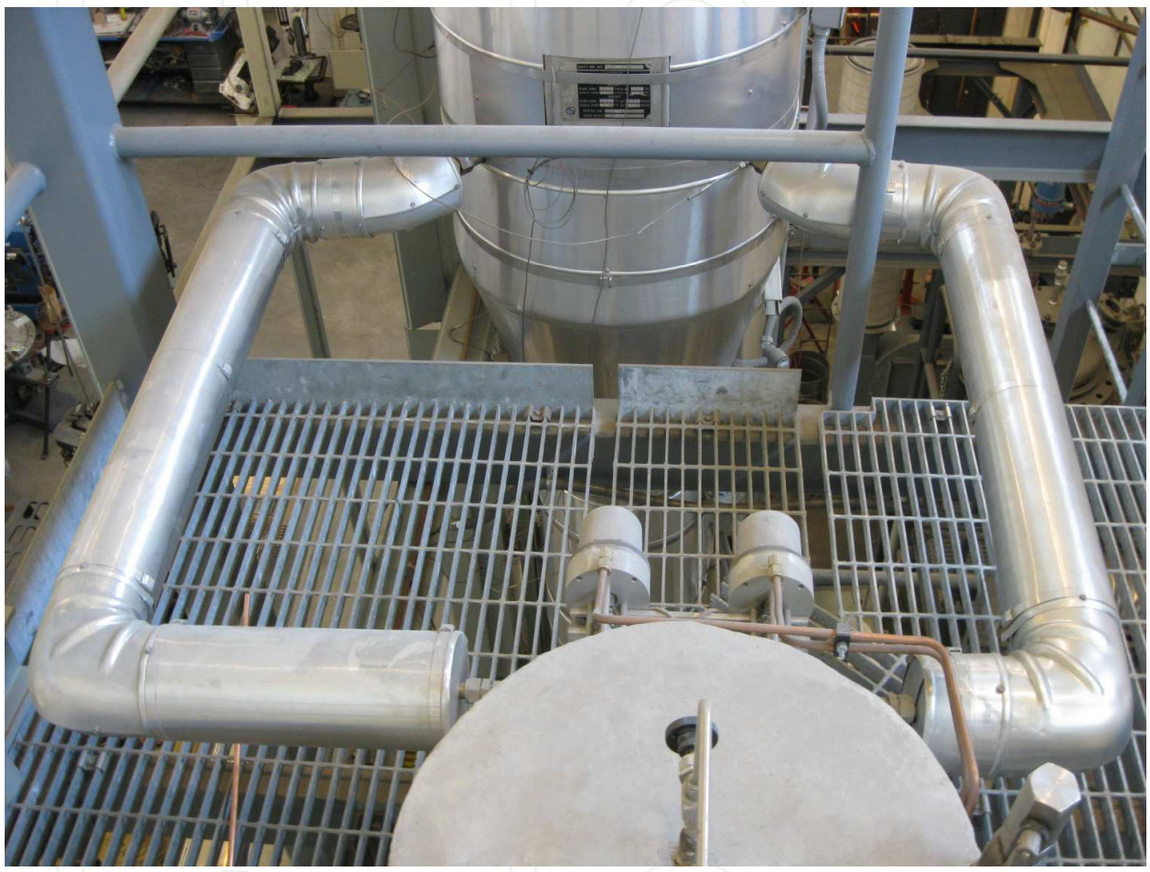


Fig. 6. High ADS lines photo.

The middle ADS lines are horizontally oriented and connect the RPV CL to the HPC. A pneumatic motor operated globe valve is located in each line. Downstream from each isolation valve is a transition piece with an internal square-edge orifice. These two lines enter the HPC, penetrate it and then turn downward before terminating below the HPC waterline. A sparger is considered at the end of these lines.

The ADS sump recirculation lines are horizontally oriented and connect the RPV lower CL to the HPC. A pneumatic motor operated globe valve is located in each line. Downstream from each isolation valve is a transition piece with an internal square-edge orifice. These two lines enter the HPC, penetrate it and then turn downward before terminating below the HPC waterline. No sparger is considered for these lines.

The water temperatures inside the ADS lines outside of the HPC are measured.

4. OSU-MASLWR-001 test description

The purpose of the OSU-MASLWR-001 test (Modro et al. 2003; Reyes et al., 2007; Pottorf et al., 2009; Mascari et al., 2011e), a design basis accident for MASLWR concept design, was to determine the pressure behavior of the RPV and containment following an inadvertent actuation of one middle ADS valve. The test successfully demonstrated the blowdown behavior of the MASLWR test facility during one of its design basis accident.

Following the inadvertent middle ADS actuation the blowdown of the primary system takes place. A subcooled blowdown, characterized by a fast RPV depressurization, takes place after the Start Of the Transient (SOT). A two-phase blowdown occurs when the differential pressure, at the break location, results in fluid flashing. A choked two-phase flow condition prevails and a decrease in depressurization rate of the primary system is experimentally observed. When the PRZ pressure reaches saturation, single phase blowdown occurs and the depressurization rate increases. The RPV and HPC pressure and the primary saturation temperature are shown in figure 7. The P_{sat}, saturation pressure, is based on the temperature at the core outlet.

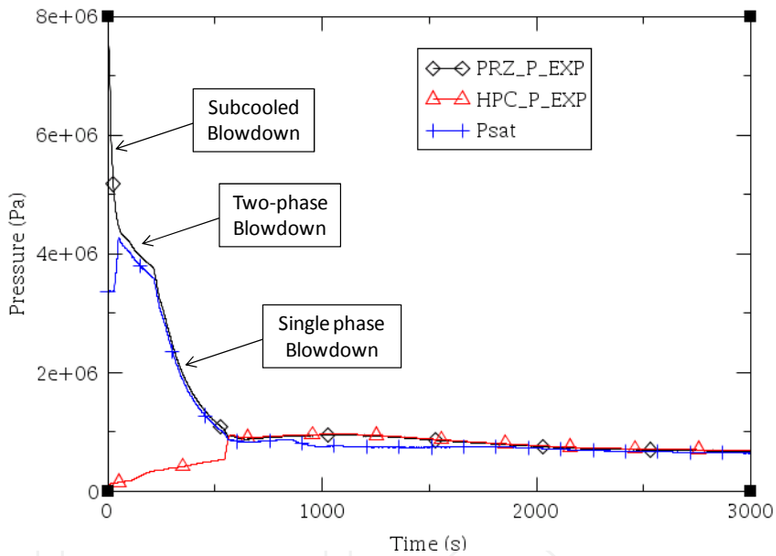


Fig. 7. RPV and HPC pressure behaviour during the OSU-MASLWR-001 test (Modro et al., 2003; Reyes et al., 2007; Mascari et al., 2011e).

At 539 s after the SOT the pressure difference between the RPV and the HPC reaches a value less than 0.517 MPa, one of the high ADS valve is opened and, with approximately 10 s of delay, the other high ADS valve is opened equalizing the pressure of the primary and HPC system.

At 561 s after the SOT the pressure difference between the RPV and the HPC reaches a value less than 0.034 MPa, one of the sump recirculation valve is opened and, with approximately 10 s of delay, the other sump recirculation valve is opened terminating the blowdown period and starting the refill period. The refill period takes place for the higher relative coolant height in the HPC compared to the RPV. Figure 8 shows the RPV level evolution experimentally detected during the test. The RPV water level never fell below the top of the core during the execution of the test 1.

During the saturated blowdown period, the inlet and the outlet temperature of the core are equal each other assuming the saturation temperature value. A core reverse flow and a core coolant boiling off at saturation is present in the facility during this period. When the refill takes place the core normal flow direction is restarted and a delta T core is observed depending on the refill rate and core power, figure 9.

When the refill of the reactor takes place the level of the coolant reaches the location of the flow rate HL measurement point, therefore an increase of the RPV flow rate is detected for this phenomenon, figure 10.

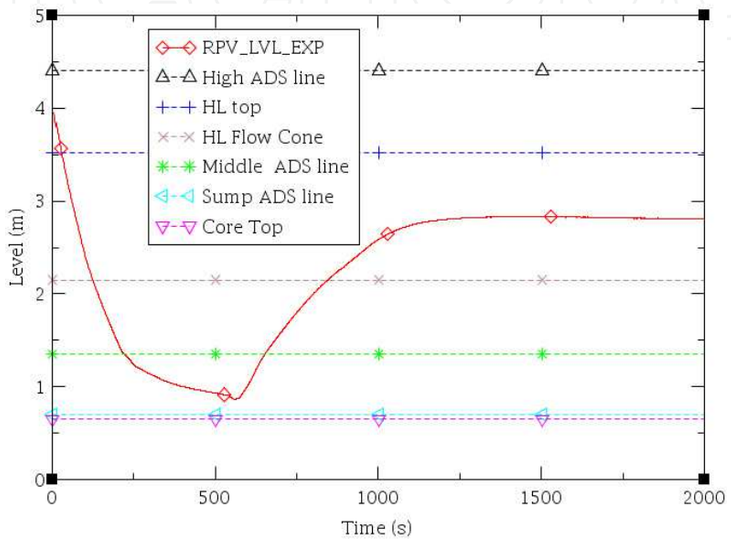


Fig. 8. RPV water level inventory behaviour during the OSU-MASLWR-001 test (Modro et al., 2003; Reyes et al., 2007; Mascari et al., 2011e).

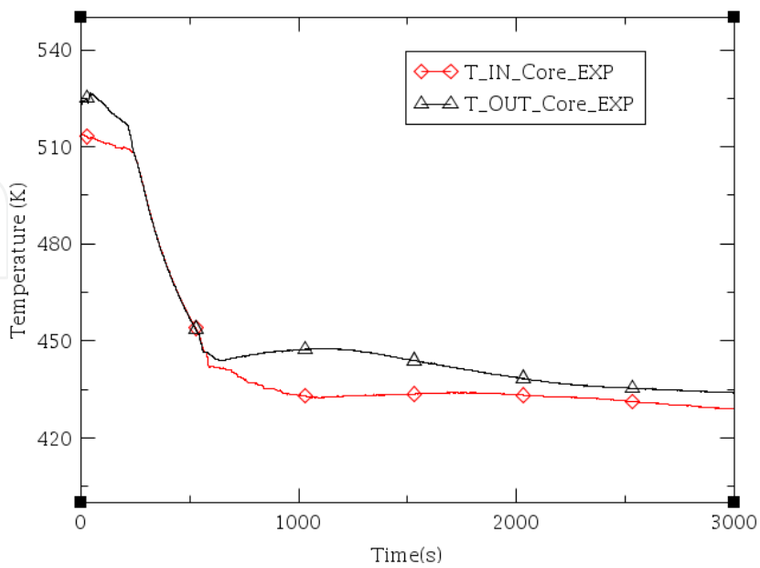


Fig. 9. Inlet/outlet core temperature behavior during the OSU-MASLWR-001 test (Modro et al., 2003; Mascari et al., 2011e).

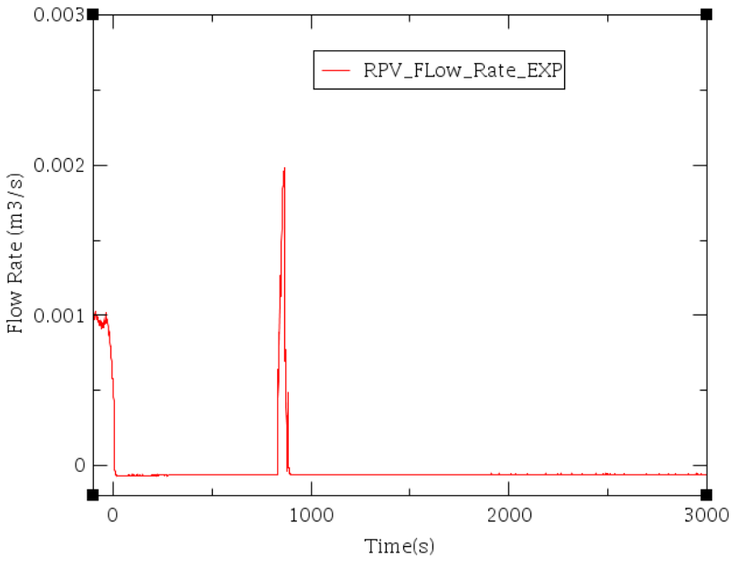


Fig. 10. RPV flow rate during the OSU-MASLWR-001 test (Modro et al., 2003; Mascari et al., 2011e).

5. Code application

5.1 TRACE code

TRACE (TRACE V5.0, 2010) is a component-oriented code designed to perform best estimate analyses for LWR. In particular this code is developed to simulate operational transients, LOCA, other transients typical of the LWR and to model the thermal hydraulic phenomena taking place in the experimental facilities used to study the steady state and transient behavior of reactor systems (Mascari et al., 2011a).

TRACE is a finite volume, two fluid, code with 3D capability. The code is based on two fluid, two-phase field equations. This set of equations consists of the conservation laws of mass, momentum and energy for liquid and gas fields (Reyes, 2005a):

- Mixture mass conservation equation:

$$\frac{\partial}{\partial t}[\rho_v \alpha + (1-\alpha)\rho_l] + \nabla \cdot [\rho_v \vec{v}_v \alpha + \rho_l \vec{v}_l (1-\alpha)] = 0$$

- Vapor mass conservation equation:

$$\frac{\partial}{\partial t}(\rho_v \alpha) + \nabla \cdot (\rho_v \vec{v}_v \alpha) = \Gamma_v$$

- Liquid momentum conservation equation:

$$\frac{\partial \vec{v}_l}{\partial t} + \vec{v}_l \cdot \nabla \vec{v}_l = -\frac{1}{\rho_l}\nabla p + \frac{c_i}{(1-\alpha)\rho_l}(\vec{v}_v - \vec{v}_l)\,|\,\vec{v}_v - \vec{v}_l\,| - \frac{\Gamma_{cond}}{(1-\alpha)\rho_l}(\vec{v}_v - \vec{v}_l) + \frac{c_w}{(1-\alpha)\rho_l}\vec{v}_l\,|\,\vec{v}_l\,| + \vec{g}$$

- Gas momentum conservation equation:

$$\frac{\partial \vec{v}_v}{\partial t} + \vec{v}_v \cdot \nabla \vec{v}_v = -\frac{1}{\rho_v}\nabla p + \frac{c_i}{\alpha \rho_v}(\vec{v}_v - \vec{v}_l)\,|\,\vec{v}_v - \vec{v}_l\,| - \frac{\Gamma_{Boiling}}{\alpha \rho_v}(\vec{v}_v - \vec{v}_l) + \frac{c_{wv}}{\alpha \rho_v}\vec{v}_v\,|\,\vec{v}_v\,| + \vec{g}$$

- Mixture energy conservation equation:

$$\frac{\partial}{\partial t}[(\rho_v \alpha e_v + \rho_l(1-\alpha)e_l] + \nabla \cdot [\rho_v \alpha e_v \vec{v}_v + \rho_l(1-\alpha)e_l \vec{v}_l] = -p\nabla \cdot [\alpha \vec{v}_v + (1-\alpha)\vec{v}_l] + q_{wl} + q_{dlv}$$

- Vapor energy conservation equation:

$$\frac{\partial}{\partial t}(\rho_v \alpha e_v) + \nabla \cdot (\rho_v \alpha e_v \vec{v}_v) = -p\frac{\partial \alpha}{\partial t} - p\nabla \cdot (\alpha \vec{v}_v) + q_{wv} + q_{dv} + q_{iv} + \Gamma_v h_v$$

The resulting equation set is coupled to additional equations for non-condensable gas, dissolved boron, control systems and reactor power. Relations for wall drag, interfacial drag, wall heat transfer, interfacial heat transfer, equation of state and static flow regime maps are used for the closure of the field equations. The interaction between the steam-liquid phases and the heat flow from solid structures is also considered. These interactions are in general dependent on flow topology and for this purpose a special flow regime dependent constitutive-equation package has been incorporated into the code.

TRACE uses a pre-CHF flow regime, a stratified flow regime and a post-CHF flow regime. In order to study the thermal history of the structures the heat conduction equation is applied to different geometries. A 2D (r and z) treatment of conduction heat transfer is taken into account as well.

A finite volume numerical method is used to solve the partial differential equations governing the two-phase flow and heat transfer. By default, a multi-step time-differencing procedure that allows the material Courant-limit condition to be exceeded is used to solve the fluid-dynamics equations.

TRACE can be used together with a user-friendly front end, Symbolic Nuclear Analysis Package (SNAP), able to support the user in the development and visualization of the nodalization, to show a direct visualization of selected calculated data, and accepts existing RELAP5 and TRAC-P input. The TRACE/SNAP architecture is shown in figure 11.

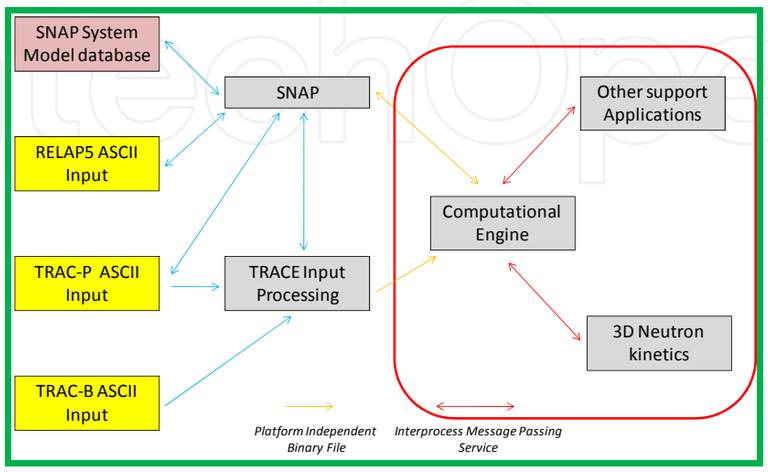


Fig. 11. TRACE/SNAP environment architecture (Staudenmeier, 2004; Mascari et al., 2011a).

SNAP (SNAP users manual, 2007) is a suite of integrated applications including a "Model Editor", "Job Status", the "Configuration Tool" client applications and a "Calculation Server". In particular, the "Model Editor" is used for the nodalization development and visualization and for the visualization of the selected calculated data by using its graphical and animation model capability. The codes currently supported in SNAP are CONTAIN, COBRA, FRAPCON-3, MELCOR, PARCS, RELAP5 and TRACE.

5.2 OSU-MASLWR TRACE model

An OSU-MASLWR TRACE model (Mascari et al., 2008, 2009a, 2009b, 2010b, 2011a, 2011b, 2011c, 2011d) is developed in order to evaluate the TRACE code capabilities in predicting the thermal hydraulic phenomena typical of the MASLWR design as natural circulation, heat exchange from primary to secondary side by helical SG in superheated condition and primary/containment coupling during transient scenario.

The TRACE nodalization, developed by using SNAP, models the primary and the secondary circuit. The containment structures consisting of the HPC, CPV and heat transfer plate are modeled as well, figure 12.

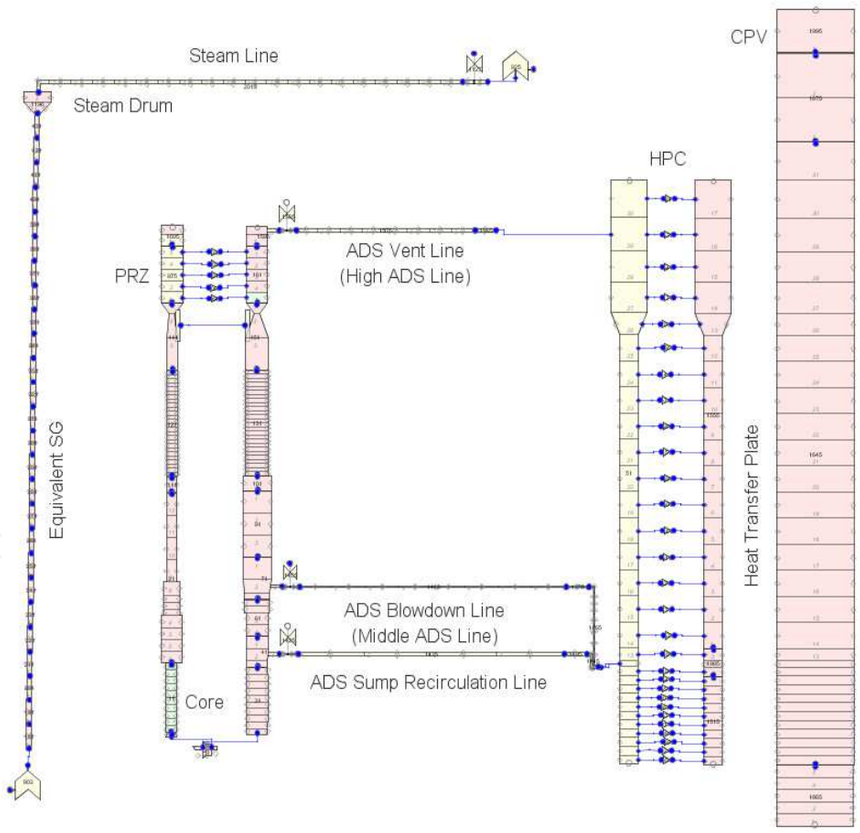


Fig. 12. OSU-MASLWR TRACE model.

The "slice nodalization" technique is adopted in order to improve the capability of the code to reproduce natural circulation phenomena. This technique consists in realizing the mesh cells of different nodalization zones, at the same elevation, with the same cell length (Mascari et al.,

2011a). In this way it is avoided the error due to the position/elevation of the cell nodalization center that can influences the results of the calculated data when natural circulation regime is present. If the "slice nodalization" technique is not used, this error has to be taken into account and its effect increases if larger nodalization cells are used. In this case it can be reduced by using a "fine nodalization". In general, its effect on the results is less important when forced circulation regimes are simulated. However the "slice nodalization" technique could require nodes of small length increasing the numerical error and the computational time. The "code user" has to take into account these disadvantages during the nodalization development.

The primary circuit of the TRACE model, comprises the core, the HL riser, the UP, the PRZ, the SG primary side, the CL down comer and the LP. After leaving the top of the HL riser, the flow enters the UP divided in two thermal hydraulic regions connected to the PRZ. After living the UP, the flow continues downward through the SG primary section and into the CL down comer region. The core is modeled with one thermal hydraulic region thermally coupled with one equivalent active heat structure simulating the 56 electric heaters. The PRZ is modeled with two hydraulic regions, connected by different single junctions, in order to allow potential natural circulation/convection phenomena. The three different PRZ heater elements are modeled with one equivalent active heat structure. The thick baffle plate is modeled as well. The direct heat exchange by the internal shell between the hotter fluid, in the ascending riser, and the colder fluid, in the descending annular down comer, is modeled by heat structures thermally coupled with these two different hydraulic regions. SG coils are modeled with one "equivalent" group of pipes, in order to simulate the three separate parallel helical coils. The steam line of the facility is modeled as well.

In order to simulate the OSU-MASLWR-001 test the HPC is divided in two thermal hydraulic regions, connected by single junctions, in order to allow the simulation of possible natural circulation phenomena. The ADS lines are modeled.

The RPV, HPC and CPV shell and the connected insulation are modeled.

5.3 TRACE model qualification process

A nodalization, representing an actual system (integral test facility or nuclear power plant), can be considered qualified when it has a geometrical fidelity with the involved system, it reproduces the measured nominal steady state conditions of the system, and it shows a satisfactory behavior in time-dependent conditions.

The OSU-MASLWR nodalization qualification process (Bonuccelli et al., 1993) is still in progress, because the facility experimental characterization will be conducted in the framework of the current IAEA ICSP. In particular several important facility operational characteristics, like pressure drop along the primary loop, at different primary side mass flow rates, and heat losses, at different primary side temperatures, determined to be of importance during the planned ICSP experiments, will be evaluated and distributed to the ICSP participants. Besides, some nodalization models, here presented, are still preliminary because some geometrical data and the complete instrument characterization and location will be delivered in the ICSP framework as well. Therefore the current results are preliminary and should not be used for the code assessment, but are able to show the TRACE capability to reproduce the primary/containment coupling phenomena typical of the MASLWR prototypical design (Mascari et al., 2009b; Mascari et al., 2011c).

5.4 Analysis of the OSU-MASLWR-001 TRACE calculated data

Starting from the calculated data developed in previous analyses (Pottorf et al., 2009; Mascari et al., 2009b; Mascari et al., 2011c) the target of this section is to give an expanded revised analyses, after a first review of the TRACE nodalization, of the TRACE V5 patch 2 code capability in predicting the primary/containment coupling phenomena typical of the MASLWR prototypical design.

The analysis of the OSU-MASLWR-001 calculated data shows that the TRACE code is able to qualitatively predict the primary/containment coupling phenomena characterizing the test. The blowdown phenomena, the refill of the core and the long term cooling, permitting of removing the decay power, are predicted by the code.

In particular, following the inadvertent middle ADS actuation, the blowdown of the primary system takes place. A subcooled blowdown, characterized by a fast RPV depressurization, is predicted by the code after the SOT.

When the differential pressure in the facility at the break location results in flashing, a two-phase blowdown, qualitatively predicted by the code, occurs. A decrease in depressurization rate of the primary system is then observed, in agreement with the experimental data.

When the PRZ pressure reaches saturation, single phase blowdown occurs and the depressurization rate increases again, in agreement with the experimental data. The RPV and HPC pressure versus code calculations are shown in figure 13.

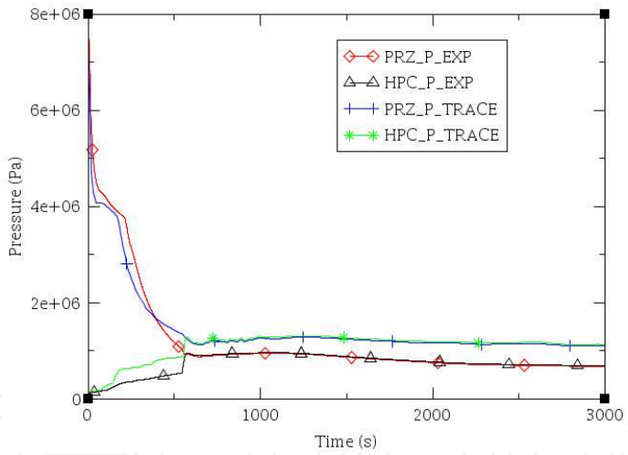


Fig. 13. Experimental data versus code calculation for PRZ and HPC pressure.

In agreement with the experimental data, when the pressure difference between the RPV and the containment reaches a value less than 0.517 MPa, the high ADS valves are opened which equalizes their pressure.

When the pressure difference reaches a value less than 0.034 MPa, the sump recirculation valves are opened and the refill period begins. The refill phenomenon is predicted by the code. As in the experimental data, the refill period takes place for the higher relative coolant height in the HPC compared to the RPV.

Figure 14 shows the RPV level evolution experimentally detected during the test versus the calculated data. In agreement with the experimental data, the RPV water level never fell

below the top of the core. Figure 15 shows the HPC level versus code calculation during the test. The qualitative behavior is well predicted by the TRACE code.

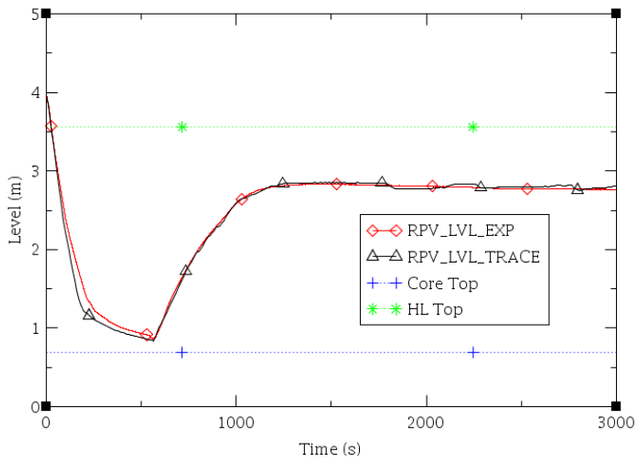


Fig. 14. Experimental data versus code calculation for RPV level.

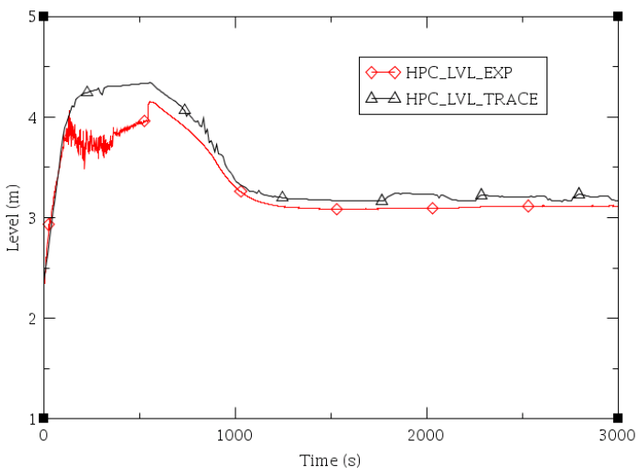


Fig. 15. Experimental data versus code calculation for HPC level.

In agreement with the experimental data, during the saturated blowdown period the inlet and the outlet temperature of the core are equal each other assuming the saturation temperature value. A core reverse flow and a core coolant boiling off at saturation is predicted by the code. When the refill takes place, the core normal flow direction is restarted. Figure 16 shows the experimental data versus code calculation for outlet core temperature.

In agreement with the experimental data when the sump recirculation valves are opened the vapor produced in the core goes in the upper part of the facility and through the high ADS valve goes to the HPC where it is condensed. At this point through the sump recirculation line and the down comer the fluid goes to the core again (Mascari et al., 2011c, 2011d). Figure 17 shows the long term cooling flow path typical of the MASLWR design.

Figure 18 shows, by using the SNAP animation model capabilities, the fluid condition of facility, 976 s after the SOT, predicted by the TRACE code.

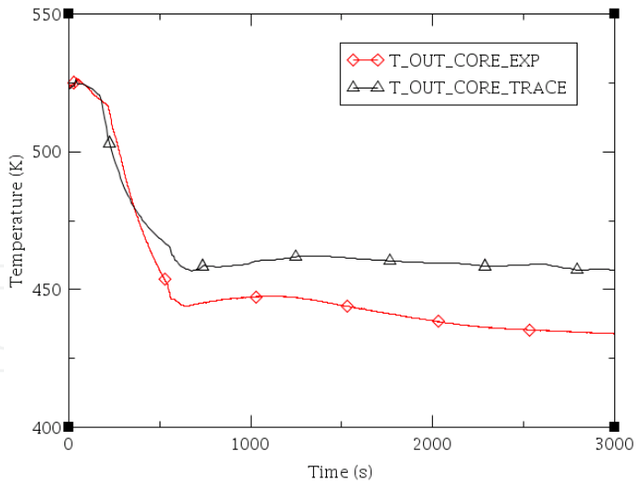


Fig. 16. Experimental data versus code calculation for outlet core temperature.

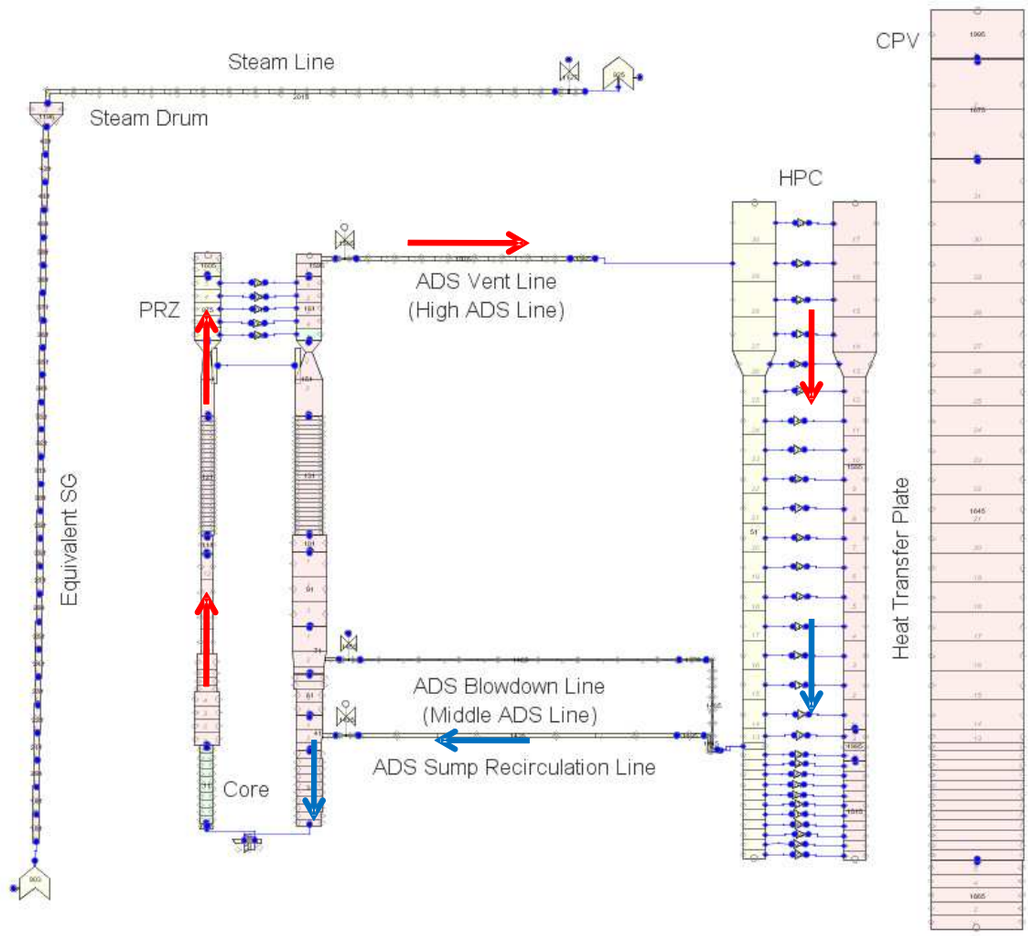


Fig. 17. Long term cooling flow path typical of the MASLWR design.

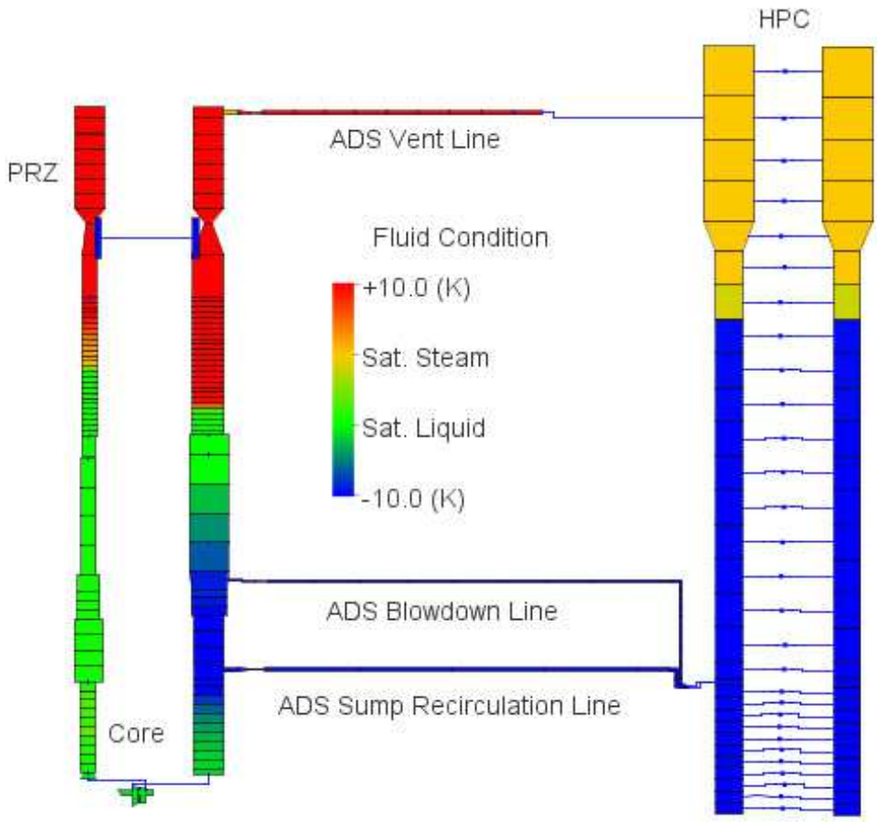


Fig. 18. SNAP animation model used to analyze the OSU-MASLWR-001 test (976s after the SOT).

From a quantitative point of view the results of the calculated data show a general over prediction compared with the experimental data. It is thought that this could be due to a combination of selection of vent valve discharge coefficients and condensation models applied to the inside surface of the containment (Pottorf et al., 2009; Mascari et al., 2011c).

However, in order to quantitatively evaluate the capacity of the TRACE code to simulate OSU-MASLWR-001 primary/containment coupling phenomena, a qualification of the TRACE nodalization against an heat losses experimental characterization at different primary side temperature is necessary. Figures 19 and 20 show the behavior of PRZ pressure and outlet core temperature respectively by increasing the heat losses of the TRACE model (TRACE_HL). A general quantitative improvement of the calculated data has been showed by TRACE_HL calculation. Therefore in order to quantitatively evaluate the capability of the TRACE code to simulate the OSU-MASLWR phenomena, and therefore use the calculated data for the TRACE code assessment, is necessary a TRACE nodalization qualification against several facility operational characteristics like heat losses at different primary side temperatures and pressure drop at different primary mass flow rate. Currently the TRACE model qualification process is in progress considering the facility characterization that will be disclosed during the current IAEA ICSP.

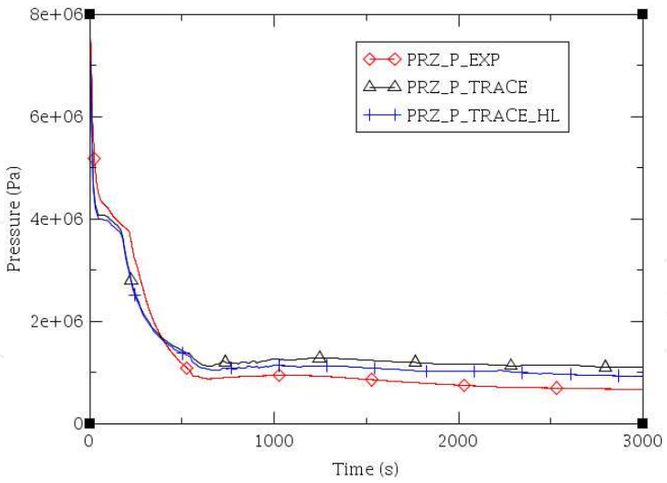


Fig. 19. Experimental data versus code calculations for PRZ pressure.

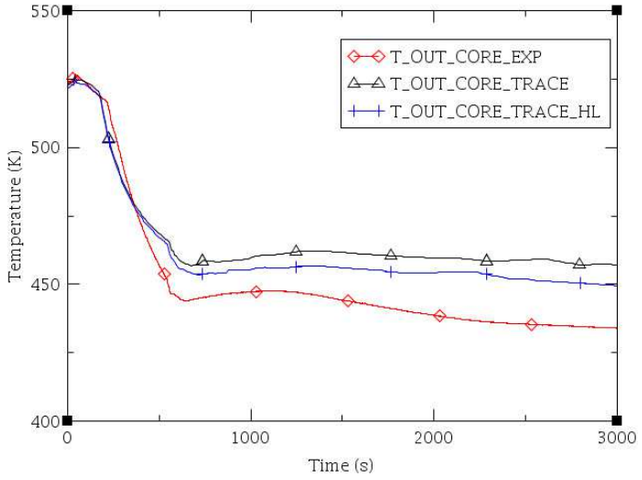


Fig. 20. Experimental data versus code calculations for outlet core temperature.

Considering the importance of the containment/reactor vessel thermal hydraulic coupled behavior for the advanced integral-type LWR, in the IAEA ICSP framework a further test, simulating a loss of FW transient with subsequent ADS actuation and long term cooling, will be executed in the OSU-MASLWR facility (Woods & Mascari, 2009; Woods et al.; 2011).

6. Conclusions

The MASLWR is a small modular integral PWR relying on natural circulation during both steady-state and transient operation. During steady state condition the primary fluid, in single-phase natural circulation, removes the core power and transfers it to the secondary fluid through helical coil SG. In transient condition the core decay heat is removed through a passive primary/containment coupling mitigation strategy based on natural circulation.

The experimental analysis of the primary/containment coupling phenomena, characterizing the MASLWR design, has been developed in a first testing program at OSU-MASLWR

experimental integral test facility. In particular the OSU-MASLWR-001 test determined the pressure behavior of the RPV and containment following an inadvertent actuation of one middle ADS valve. The test successfully thermal hydraulically demonstrated the passive primary/containment coupling typical of the MASLWR design SBLOCA mitigation strategy.

In the last years the USNRC has developed the advanced best estimate thermal hydraulic system code TRACE in order to simulate operational transients, LOCA, other transients typical of the LWR and to model the thermal hydraulic phenomena taking place in the experimental facilities used to study the steady state and transient behavior of reactor systems. The validation and assessment of the TRACE code against the MASLWR passive primary/containment coupling mitigation strategy is a novel effort and it is the topic of this chapter. Since the qualification process of the OSU-MASLWR TRACE nodalization is still in progress, considering the facility characterization conducted in an IAEA ICSP on "Integral PWR Design Natural Circulation Flow Stability and Thermo-Hydraulic Coupling of Containment and Primary System during Accidents", the current results are preliminary and should not be used for the code assessment, but are able to show the TRACE capability to reproduce the thermal hydraulic phenomena typical of the MASLWR primary/containment coupling SBLOCA mitigation strategy.

The analysis of the OSU-MASLWR-001 calculated data shows that the TRACE code is able to qualitatively predict the primary/containment coupling phenomena characterizing the MASLWR design. The sub-cooled blowdown, two-phase blowdown and single phase blowdown, following the inadvertent middle ADS actuation, are qualitatively predicted by the code. The refill phenomenon is qualitatively predicted by the code as well. In general the results of the calculated data show an over prediction compared with the experimental data. It is thought that this could be due to a combination of selection of vent valve discharge coefficients and condensation models applied to the inside surface of the containment. In agreement with the experimental data, the RPV water level never fell below the top of the core. However, in order to quantitatively evaluate the capability of the TRACE code to simulate the OSU-MASLWR phenomena, and therefore use the calculated data for the TRACE code assessment, is necessary a TRACE nodalization qualification against several facility operational characteristics like pressure drop at different primary mass flow rates and heat losses at different primary side temperatures. Currently the TRACE model qualification process is in progress considering the facility characterization conducted during the IAEA ICSP.

7. Abbreviations

ADS	Automatic Depressurization System;
AP600/1000	Advanced Plant 600/1000 MWe;
CAREM	Natural Circulation based PWR being developed in Argentina;
CHF	Critical Heat Flux;
CL	Cold Leg;
CPV	Cooling Pool Vessel;
EHRS	Emergency Heat Removal System;
ESBWR	Economic Simplified Boiling Water Reactor;
FW	Feed Water;

HL	Hot Leg;
HPC	High Pressure Containment;
IRWST	In-containment Refueling Water Storage Tank;
IAEA	International Atomic Energy Agency;
ICSP	International Collaborative Standard Problem;
IRIS	International Reactor Innovative and Secure;
LOCA	Loss of Coolant Accident;
LP	Lower Plenum;
LWR	Light- Water Reactor;
MASLWR	Multi-Application Small Light Water Reactor;
OSU	Oregon State University;
PRHR	Passive Residual Heat Removal;
PRZ	Pressurizer;
PWR	Pressurized Water Reactor;
RPV	Reactor Pressure Vessel;
SMART	System Integrated Modular Advanced Reactor ;
SBLOCA	Small Break Loss of Coolant Accident;
SG	Steam Generator;
SOT	Start of the Transient;
SNAP	Symbolic Nuclear Analysis Package;
SWR	Siede Wasser Reaktor;
TRAC	Transient Reactor Analysis Code;
TRACE	TRAC/RELAP Advanced Computational Engine;
UP	Upper Plenum;
USNRC	U.S. Nuclear Regulatory Commission;
WWER	Water Moderated, Water Cooled Energy Reactor.

8. Nomenclature

e	Total energy;
g	Gravitational force;
p	Pressure;
T	Temperature;
v_k	Phase velocity.

9. Greek symbols

α	Vapor void fraction;
Γ_k	Mass generation rate per unit volume;
ρ	Density.

10. References

Aksan, N. (2005). Application of Natural Circulation Systems: Advantage and Challenges II, In: *IAEA TECDOD-1474, Natural Circulation in Water Cooled Nuclear Power Plants*, IAEA, pp.101-114, ISBN 92–0–110605–X, Vienna, Austria, 2005

Bertel, E.; Morrison, R. (2001). Nuclear Energy Economics in a Sustainable Development Prospective, *NEA News 2001* - No. 19.1

Bonuccelli, M.; D'Auria, F.; Debrecin, N.; Galassi, G. M. (1993). A Methodology For the Qualification of Thermalhydraulic Code Nodalizations, *Proc. of NURETH-6 Conference*, Grenoble, France, October 5–8, 1993

Boyack, B. E.; Ward, L.W. (2000). Validation Test Matrix for the Consolidated TRAC (TRAC-M) Code. Los Alamos National Laboratory, LA-UR-00-778, February 2000

Carelli, M. D.; Conway, L. E.; Oriani, L.; Petrovi´c, B.; Lombardi, C. V.; Ricotti, M. E.; Barroso, A. C. O.; Collado, J. M.; Cinotti, L.; Todreas, N. E.; Grgi´c, D.; Moraes, M. M.; Boroughs, R.D.; Ninokata, H.; Ingersoll, D.T.; Oriolo F. (2004). The Design And Safety Features of the IRIS Reactor. *Nuclear Engineering and Design* 230 (2004) 151–167

Carelli, M.; Conway, L.; Dzodzo, M.; Maioli, A.; Oriani, L.; Storrick, G.; Petrovic, B.; Achilli, A.; Cattadori, G.; Congiu, C.; Ferri, R.; Ricotti, M.; Papini, D.; Bianchi, F.; Meloni, P.; Monti, S.; Berra, F.; Grgic, D.; Yoder, G.; Alemberti A. (2009). The SPES3 Experimental Facility Design for the IRIS Reactor Simulation. Hindawi Publishing Corporation, *Science and Technology of Nuclear Installations,* Volume 2009, Article ID 579430, 12 pages doi:10.1155/2009/579430

Chiovaro, P.; Di Maio, P. A.; Mascari, F.; Vella, G. (2011). Analysys of The SPES-3 Direct Vessel Injection Line Break By Using TRACE Code, *Proceeding of the XXIX Congresso UIT sulla Trasmissione del calore,* ISBN: 978- 88467- 3072- 5, Torino, June 20-22, 2011.

D'Auria, F.; Modro, M.; Oriolo, F.; Tasaka, K. (1993). Relevant Thermal Hydraulic Aspects of New Generation LWRs. *Nuclear Engineering and Design* 145, 241-259 (1993)

Galvin, M. R. (2007). OSU-MASLWR Test Facility Modification Description Report. IAEA Contract Number USA-13386, Oregon State University (November 19, 2007)

Gou, J.; Qiu, S.; Su, G.; Jia D. (2009). Thermal Hydraulic Analysis of a Passive Residual Heat Removal System for an Integral Pressurized Water Reactor. Hindawi Publishing Corporation, *Science and Technology of Nuclear Installations*, Volume 2009, Article ID 473795, 12 pages, doi: 10.1155/2009/473795

Hicken, E. F.; Jaegers, H. (2002). Passive Decay Heat Removal From the Core Region, In: *IAEA-TECDOC-1281, Natural Circulation Data and Methods For Advanced Water Cooled Nuclear Power Plant Designs,* IAEA, pp. 227-237, ISSN 1011–4289, Vienna, Austria, April 2002

IAEA-TECDOC-1281. (2002). *Natural Circulation Data and Methods for Advanced Water Cooled Nuclear Power Plant Designs*. IAEA, ISSN 1011–4289, Vienna, Austria, April 2002

IAEA-TECDOC-1391. (2004). *Status of Advanced Light Water Reactor Designs 200*4. IAEA, ISBN 92–0–104804–1, Vienna, Austria, May 2004

IAEA-TECDOD-1474. (2005). *Natural Circulation in Water Cooled Nuclear Power Plants*. IAEA, ISBN 92–0–110605–X, Vienna, Austria, November 2005

IAEA-TECDOC-1624. (2009). *Passive Safety Systems And Natural Circulation In Water Cooled Nuclear Power Plants*. IAEA, ISBN 978–92–0–111309–2, Vienna, Austria, November 2009

IAEA Power Reactor Information System (PRIS). http://www.iaea.org/programmes/a2/ 2 December 2011

IAEA – TECDOC DRAFT. *Natural Circulation Phenomena and Modelling for Advanced Water Cooled Reactors*. In preparation

Kurakov, Y.A.; Dragunov, Y. G.; Podshibiakin, A. K.; Fil, N. S.; Logvinov, S. A.; Sitnik, Y. K.; Berkovich, V.M.; Taranov, G. S. (2002). Development and Validation of Natural Circulation Based Systems for new WWER Designs, In: *IAEA-TECDOC-1281*, IAEA, pp.83-96, ISSN 1011–4289, Vienna, Austria, April 2002

Lee, K. Y.; Kim M. H. (2008). Experimental and Empirical study of Steam Condensation Heat Transfer with a Noncondensable Gas in a Small-Diameter Vertical Tube. *Nuclear Engineering and Design* 238 (2008) 207-216

Levy, S. (1999). Two-Phase flow in complex systems. *A Wiley-Interscience Publication*, John Wiley & Son, Inc. (1999)

Mascari, F.; Woods, B. G.; Adorni, M. (2008). Analysis, by TRACE Code, of Natural Circulation Phenomena in the MASLWR-OSU-002 Test, *Proceedings of the International Conference Nuclear Energy for New Europe 2008*, ISBN 978-961-6207-29-4, Portoroz, Slovenia, September 8-11, 2008

Mascari, F.; Vella, G.; Woods, B.G.; Welter, K.; Pottorf, J.; Young, E.; Adorni, M.; D'Auria, F. (2009). Sensitivity Analysis Of The MASLWR Helical Coil Steam Generator Using TRACE, *Proceedings of International Conference Nuclear Energy for New Europe 2009*, ISBN 978-961-6207-30-0, Bled, Slovenia, September 14-17

Mascari, F.; Vella, G.; Woods, B.G.; Adorni, M.; D'Auria, F. (2009). Analysis of the OSU-MASLWR Natural Circulation Phenomena Using TRACE Code, IAEA, organized in cooperation with the OECD Nuclear Energy Agency and the European Commission, Technical meeting on "*Application of Deterministic Best Estimate Safety Analysis*", University of Pisa, Pisa, Italy, September 21–25, 2009

Mascari, F.; Vella, G; Buffa, P.; Compagno, A.; Tomarchio, E. (2010). Passive Safety Systems in view of Sustainable Development. *Final Report on the Round Tables*, Erasmus Intensive Programme Project (IP) ICARO Intensive Course on Accelerator and Reactor Operation 28/02/10 – 12/03/10– Sicilia, Italia –

Mascari, F. (2010). PhD Thesis on "Natural Circulation and Phenomenology of Boron Dilution in the Pressurzied Water Reactors (Circolazione Naturale e Fenomenologie di Boron Dilution in Reattori ad Acqua in Pressione). University of Palermo, 2010

Mascari, F.; Vella, G.; Woods, B.G.; Welter, K.; Pottorf, J.; Young, E.; Adorni, M.; D'Auria, F. (2011). Sensitivity Analysis Of The MASLWR Helical Coil Steam Generator Using TRACE. *Nuclear Engineering and Design* 241 (2011) 1137–1144

Mascari, F.; Vella, G. (2011). IAEA International Collaborative Standard Problem on Integral PWR Design Natural Circulation Flow Stability and Thermo-hydraulic Coupling of Containment and Primary System during Accidents" Double Blind Calculation Results. Dipartimento dell'Energia, Università degli Studi di Palermo. Prepared for IAEA, Vienna, Austria , February 2011

Mascari, F.; Vella, G.; Woods, B. G.; D'Auria, F. (2011). Analysis of the Multi-Application Small Light-Water Reactor (MASLWR) Design Natural Circulation Phenomena, *Proceedings of ICAAP 2011*, Nice, France, 2-5 May, 2011

Mascari, F.; Vella, G.; Woods, B.G. (2011). TRACE Code Analyses For The IAEA ICSP On Integral PWR Design Natural Circulation Flow Stability And Thermo-Hydraulic Coupling Of Containment And Primary System During Accidents, *Proceedings of*

the *ASME 2011 Small Modular Reactors Symposium*, Washington, DC, USA, September 28-30, 2011

Mascari F.; Vella, G.; Woods, B.G.; D'Auria F. (2011) Analyses of the OSU-MASLWR Experimental Test Facility. *Science and Technology of Nuclear Installations*, Hindawi, In Progress

Modro, S.M.; Fisher, J. E.; Weaver, K. D.; Reyes, J. N., Jr.; Groome J. T.; Babka, P.; Carlson, T. M. (2003). Multi-Application Small Light Water Reactor Final Report, DOE Nuclear Energy Research Initiative Final Report, Idaho National Engineering and Environmental Laboratory, December 2003

Pottorf, J.; Mascari, F.; Woods, B. G (2009). TRACE, RELAP5 Mod 3.3 and RELAP5-3D Code Comparison of OSU-MASLWR-001 Test. *Transactions of the ANS*, Volume 101, 2009, ISSN 0003-018X

Reyes, J. N., Jr.; King, J. (2003). Scaling Analysis for the OSU Integral System Test Facility. Department of Nuclear Engineering Oregon State University, NERI Project 99-0129, Prepared For U.S. Department of Energy

Reyes, J. N., JR. (2005). Governing Equations in Two-Phase Fluid Natural Circulation Flows, In: *IAEA TECDOD-1474, Natural Circulation in Water Cooled Nuclear Power Plants*, IAEA, pp.155-172, ISBN 92–0–110605–X, Vienna, Austria, November 2005

Reyes, J. N., Jr. (2005). Integral System Experiment Scaling Methodology, In: *IAEA TECDOD-1474, Natural Circulation in Water Cooled Nuclear Power Plants*, IAEA, pp.321-355, ISBN 92–0–110605–X, Vienna, Austria, November 2005

Reyes, J.N. Jr. (2005). AP 600 and AP 1000 Passive Safety System Design and Testing in APEX, In: *IAEA TECDOD-1474, Natural Circulation in Water Cooled Nuclear Power Plants*, IAEA, pp.357-381, ISBN 92–0–110605–X, Vienna, Austria, November 2005

Reyes, J.N., JR.; Groome, J.; Woods, B. G.; Young, E.; Abel, K.; Yao, Y.; Yoo, J. Y. (2007). Testing of the Multi Application Small Light Water Reactor (MASLWR) Passive Safety Systems. *Nuclear Engineering and Design 237*, 1999-2005 (2007)

Reyes, J.N., Jr.; Lorenzini, P. (2010). NusScale Power: A Modular, Scalable Approach to Commercial Nuclear Power. *Nuclear news*, ANS (June 2010)

Staudenmeier, J. (2004). TRACE Reactor System Analysis Code, MIT Presentation, Safety Margins and Systems Analysis Branch, Office of Nuclear Regulatory Research, U.S. Nuclear Regulatory Commission (2004)

Symbolic Nuclear Analysis Package (SNAP), 2007. Users Manual. Applied Programming Technology, Inc., Bloomsburg, PA

TRACE V5.0. *Theory and User's manuals*. 2010. Division of System Analysis, Office of Nuclear Regulatory Research, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001

Wolde-Rufael, Y.; Menyah, K. (2010). Nuclear Energy Consumption and Economic Growth in Nine Developed Countries. *Energy Economics* 32, 550-556, (2010)

Woods, B. G.; Mascari, F. (2009). Plan for an IAEA International Collaborative Standard Problem on Integral PWR Design Natural Circulation Flow Stability and Thermo-hydraulic Coupling of Containment and Primary System during Accidents. Department of Nuclear Engineering and Radiation Health Physics, Oregon State

University. Prepared for IAEA, Vienna, Austria. Manuscript completed: March 2, 2009. Data Published: April 8, 2009

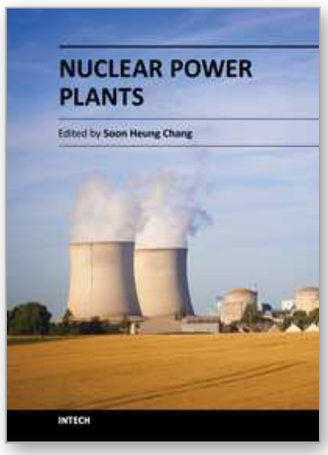
Woods, B. G.; Galvin, M. R.; Jordan, B. C. Problem specification for the IAEA International Collaborative Standard Problem on Integral PWR Design natural circulation flow stability and thermo-hydraulic coupling of containment and primary system during accident, Department of Nuclear Engineering and Radiation Health Physics, Oregon State University. Prepared for IAEA, Vienna, Austria. DRAFT

World Energy Outlook, Executive Summary. IEA - International Energy Agency-, 2009

Zejun, X.; Wenbin, Z.; Hua, Z.; Bingde, C.; Guifang, Z.; Dounan, J. (2003). Experimental Research Progress on Passive Safety Systems of Chinese Advanced PWR. *Nuclear Engineering and Design* 225 (2003) 305–313

Zuber, N. (1991). Appendix D: Hierarchical, Two-Tiered Scaling Analysis, An Integrated Structure and Scaling Methodology for Severe Accident Technical Issue Resolution, U.S Nuclear Regulatory Commission, Washington, D.C. 20555, *NUREG/CR-5809, November 1991*

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