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Analysis of the Multi-Application Small Light-Water Reactor (MASLWR) design natural circulation phenomena.

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Abstract – *Today the use of advanced nuclear power plant, have an important role in the environment and economic sustainability of country energy strategy considering the capacity of a nuclear reactor of producing energy in safe and stable way contributing in cutting the CO2 emissions. In the last 20 years, in fact, the international community, taking into account the excellent operational experience of the nuclear reactors, starts the development of new advanced reactor designs, by including also the use of the natural circulation for the cooling of the core in normal and transient conditions. In this international framework, Oregon State University has constructed, under a U.S. Department of Energy grant, a system level test facility to examine natural circulation phenomena of importance to Multi-Application Small Light-Water Reactor (MASLWR) design. The MASLWR is a small modular integral pressurized water reactor relying on natural circulation during both steady state and transient operation, including an integrated helical coil steam generator. Starting from an experimental campaign in support of the MASLWR concept design verification, 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. Four tests have been performed at this facility in order to assess the thermal hydraulic behavior of the MASLWR design in normal and transient operation and to assess the passive safety system under transient condition. This paper illustrates a preliminary analyses, performed by TRACE code, aiming at the evaluation of the code capability in predicting natural circulation, heat exchange from primary to secondary side by helical steam generator in superheated condition and primary/containment coupling phenomena typical of the MASLWR design. The tests take into account for this analysis are the OSU-MASLWR-001, an inadvertent actuation of 1 submerged ADS valve and the OSU-MASLWR-002, a natural circulation test investigating primary system flow rates and secondary side steam superheat for a variety of core power levels and feed water flow rate. The analyses of the calculated data show that the TRACE code predicts the phenomena of interest of the selected tests.*

I. INTRODUCTION

Today considering the world energy demand increase, in order to fulfill an environment and economic sustainability, the energy policy of each country has to diversify the sources of energy and use stable and safe energy production options able of producing electricity in a clean way[1]. In the framework of the sustainable development, the use of advanced nuclear power plant, have an important role in the environment and economic sustainability of country energy strategy considering the capacity of a nuclear reactor of producing energy in safe and stable way contributing in cutting the CO2 emission.[2,3,4]

In the last 20 years, in fact, the international community, taking into account the excellent 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[5]. Design simplifications and increased design margins are included in the advanced Light Water Reactors (LWR)[6]. 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.[4,7,8]

In the current reactor designs, there is a good technical basis and some operational experience in relation with the natural circulation, because it has been used for removing decay heat if forced circulation becomes unavailable. In some existing reactors, like the VK-50 reactor in Russia, the Dodewaard reactor in the Netherlands, and the Humbholdt Bay 3 reactor in California, USA, the natural circulation has been used to remove the core heat under normal operation conditions. This operational experience can be applied to new reactor concepts.[4,7,8]

In the development process of these advanced nuclear reactors, the analysis of single and two phase fluid natural circulation in complex systems, under steady state and transient conditions, is crucial for the understanding of the physical and operational phenomena typical of these advanced designs. Besides, the use of experimental facility is fundamental in order to characterize the thermal hydraulic 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.[7]

Different computer codes have been developed to characterize two phase-flow systems [9], 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.[10]

The evaluation of a 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 characterizes 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 the best estimate thermal hydraulic systems codes, has been the objective of multiple international research programs[10].

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 MASLWR design. The MASLWR is a small modular integral PWR relying on natural circulation during both steady state and transient operation. It includes an integrated Steam Generator (SG) consisting of banks of vertical helical tubes contained within the Reactor Pressure Vessel (RPV)[10]. Four tests have been conducted in support of the MASLWR concept design verification in the experimental facility.

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 reactor plants and accompanying passive safety features as well[10]. An IAEA International Collaborative Standard Problem (ICSP)[11] on the "Integral PWR Design Natural Circulation Flow Stability and Thermo-Hydraulic Coupling of Containment and Primary System during Accidents" it is executing in the facility.

The validation and assessment of the thermal hydraulic system code TRACE, developed by USNRC to perform best estimate analysis for LWR, against the MASLWR natural circulation phenomena in steady and transient condition is the topic of the present research activity. Specifically the current research activity is focused on the assessment and validation of the TRACE code in predicting natural circulation, heat exchange from primary to secondary side by helical SG in superheated condition and primary/containment coupling phenomena typical of the MASLWR design.

II. MASLWR DESIGN

The MASLWR[12], Fig. 1, is a small modular integral PWR relying on natural circulation during both steady-state and transient operation.

It includes an integrated SG consisting of banks of vertical helical tubes contained within the RPV and located in the upper region of the vessel outside of the Hot Leg (HL) chimney. As it is shown in Fig. 1, the primary coolant flows outside the SG tubes, and the feed water is fully vaporized resulting in superheated steam at exit of the SG. The safety systems are designed to operate passively[10,12,13].

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 RPV can be depressurized using the Automatic Depressurization System (ADS), consisting of six valves discharging into various locations within the containment.[10,12,13]

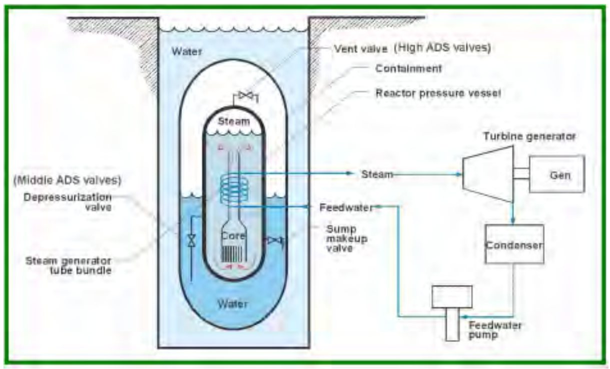

Fig. 1. MASLWR conceptual design layout [10,12,13].

The MASLWR has a net output of 35MWe. 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[14], recognizes it to be able to reach larger electricity market in developing and developed region.

III. OSU-MASLWR EXPERIMENTAL FACILITY

III.A. Experimental Facility Description

The OSU-MASLWR test facility[10,13,15], is scaled at 1:3 length scale and 1:254 volume scale and includes three major component packages consisting of the primary and secondary circuit and the containment structures. It is designed for full pressure (11.4 MPa) and full temperature (590 K) prototype operation. The Fig. 2 shows the experimental facility and details of the facility containment structures.

The primary circuit, consists of the RPV and the ADS blowdown lines, vent lines and sump recirculation lines. As it is shown in Fig. 2, the primary flow, exits the un-rodded Lower Plenum (LP) region, below the downcomer, radially inward into the rodded but unheated LP region, then upward into bottom of the core via a lower core flow plate. After leaving the core, the flow enters the chimney of the HL riser that creates a riser/downcomer configuration to enable natural circulation. After leaving the top of the HL riser, the flow enters the Upper Plenum (UP) that directs the flow radially outward and then down into the SG coil bundle of the SG primary section. After leaving the SG primary section, the flow continues downward into the CL downcomer region. This is an annular region bounded by the RPV wall on the outside and the HL riser on the inside. The flow exits the CL downcomer region into the LP to complete the primary flow circuit.[10]

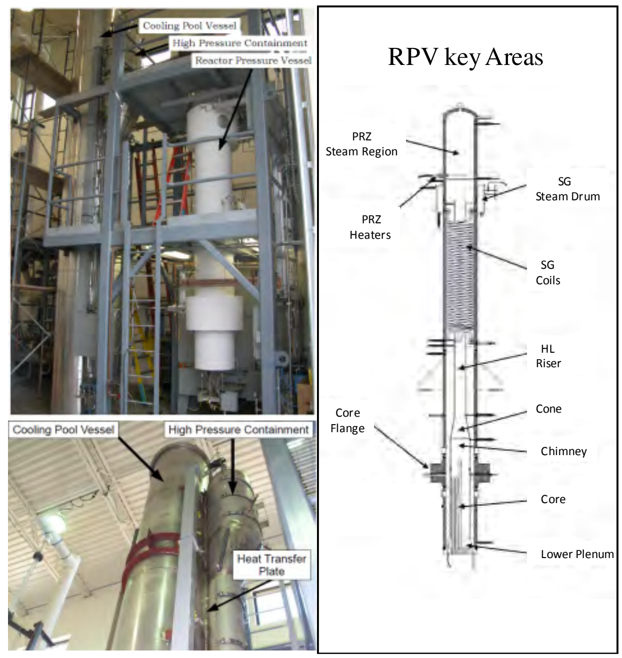

Fig. 2. OSU-MASLWR facility and RPV key areas [10,13,15].

The SG of the facility is a once through heat exchanger consisting of three separate parallel coils tubes. The outer and middle coils consist of five tubes each while the inner coil consists of four tubes. A total of 14 tubes, of 0.0159 m outside diameter with a total heated length of 86 m, are present in the facility.

A High Pressure Containment vessel (HPC) 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 MASLWR containment structure, in which the RPV sits, as well as the cavity within which the containment structure is located.[10,13,15]

III.B. OSU-MASLWR Experimental Campaign

The first experimental test campaign conducted at the OSU-MASLWR facility[10,12,13] were in support of the MASLWR concept design verification. 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-. The tests analyzed in this paper are the OSU-MASLWR-001 and 002.

The purpose of the test OSU-MASLWR-001, a design basis accident for MASLWR concept design, was to determine the behavior of the RPV and containment pressure following an inadvertent actuation of one middle ADS valve. In order to minimize the rise in containment pressure the normal open sequence used in the MASLWR for the ADS valves is: 1) the submerged lines, 2) the high containment lines and 3) the sump recirculation lines. In this way a large fraction of the energy transferred to the containment is direct into the subcooled containment coolant.[10,12,13]

By opening the submerged line ADS 1 a subcooled blowdown, characterized by a rapid pressure decrease, takes place. This period is followed by a saturated blowdown, characterized by chocked two-phase flow condition and a decrease of the depressurization rate. When the PRZ pressure is equal to the saturation pressure a single phase blowdown, characterized by an increase of the depressurization rate, takes place. When the vent valves, located in the upper part of the RPV, are opened the equalization of the RPV and HPC takes places. When the sump recirculation lines are opened, starts the refill of the core. Then the blowdown of the RPV terminates and the refill determines the end of the reverse core flow, due to the blowdown, starting a normal core direction flow period. The RPV level water never fell down the upper part of the core during the execution of the test 1 and the refill period of the MASLWR was demonstrated and thermal-hydraulically characterized.[10,13] The phenomena[16] of interest in this test are the single and two-phase natural circulation, the heat transfer in covered core, the blowdown and the refill.

The test OSU-MASLWR-002 investigated the primary system flow rates and secondary side steam superheat for a variety of core power levels and feed water flow rate. OSU-MASLWR-002 test stepped power level incrementally up to 165 kW, varying feed water flow rate at each power level. Six core power steps and seven feed water steps has been used. Since the slope of the main steam superheat curve increases if the value of the core power increases and decreases if the value of the feed water flow rate increases, the target of this test is to acquire primary system flow rate and secondary side steam superheat for different core power and feed water flow rate. The phenomena[16] of interest in these tests are the single phase natural circulation, the heat transfer in covered core, the heat transfer in SG primary and secondary side and the superheating in secondary side for a variety of primary and secondary operation conditions.[10]

III. TRACE CODE

In order to analyze the thermal hydraulic behavior of LWR reactors, the USNRC has maintained four codes, the RAMONA, the RELAP5, the TRAC-B and the TRAC-P.[10,17,18] In the last years the NRC is developing an advanced best estimate thermal hydraulic system code, by merging, among other things, the capability of the previous codes into a single code called TRAC/RELAP Advanced Computational Engine or TRACE.[10,18,19]

It is a component-oriented system thermal hydraulic code designed to perform best estimate analysis for LWR and based on two fluid, two phase field equations. This set of equations 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.[10] TRACE code can be used together with a user-friendly front end, Symbolic Nuclear Analysis Package (SNAP)[20], able to support the user in the development and visualization of the code model and to show a direct visualization of selected calculated data and their time evolution using its animation model capability[10].

IV. OSU-MASLWR TRACE MODEL

The present OSU-MASLWR TRACE model [10,21,22,23], shown in Fig. 3, is developed in order to evaluate the TRACE code capability 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 coupling primary/containment. TRACE nodalization has been developed by using SNAP.

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

The primary circuit, consists of the RPV and the ADS blowdown lines, vent lines and sump recirculation lines. The RPV comprises the core, the HL riser, the SG and the PRZ. UP is divided in two thermal hydraulic regions connected to the PRZ. The PRZ is modeled with two pipes in order to allow natural circulation/convection phenomena. The PRZ heaters are modeled. The internal shells between the fluid ascending the HL and the fluid descending the CL are modeled with heat structures thermally-coupled with these two different hydraulic regions. This permits to simulate the direct heat exchange between them.[10] The RPV shell and the connected insulation are modeled.

The secondary circuit comprises the common inlet header, the SG coils and the steam drum. The SG coils are simulated with three different oblique equivalent group of pipes in order to simulate the three separate parallel coils banks of tubes.

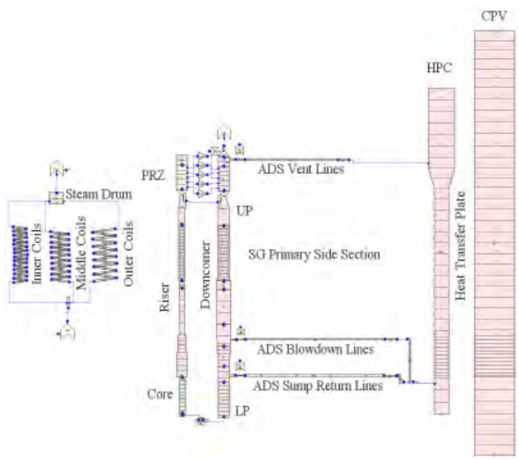

Fig. 3. OSU-MASLWR TRACE model.[10]

The HPC and the CPV are modeled by two hydraulic region thermally coupled by an heat structure simulating the heat transfer plate. The HPC and CPV shell and the related insulation are modeled. In the simulation of the OSU-MASLWR-001 test the HPC has been modeled again and has been divided in two hydraulic regions connected by single junctions in order to allow natural circulation/ convection phenomena during the primary/ containment coupling following the blowdown.

In order to improve the capability of the code to reproduce natural circulation phenomena the “slice nodalization” technique, consisting in realizing the same dimension in nodes of different zones of the nodalization simulating zones of the plant at the same elevation, is adopted.[10]

The qualification process of the OSU-MASLWR TRACE nodalization is still in progress considering the facility characterization conducted in the IAEA ICSP framework.[24] Therefore the current results are preliminary and should not be used for the code assessment, but are able to show the TRACE capability in reproducing the thermal hydraulic phenomena typical of the MASLWR design.[10] The procedure considered in the validation process is reported in [25].

V. ANALISIS OF TRACE CALCULATED DATA

V.A. Analysis of the OSU-MASLWR-002 test

The calculated data, here presented, are focused on the analysis, performed by TRACE V5.0 Patch 01 (plot reference P1) and Patch 02 (plot reference P2) code, aiming at the evaluation of the code capability in predicting natural circulation phenomena and heat exchange from primary to secondary side by helical SG in superheated condition by simulating the OSU-MASLWR-002 test. A third calculation (plot reference P2_HL), made by using TRACE V5.0 Patch 02, in which the heat losses of the facility are incremented, is here reported as well.

The analyses of the calculated data, shows that the phenomena characterizing this test are qualitatively predicted by the code. The primary system flow rates and the secondary side steam superheat, for a variety of core power levels and feed water flow rates, are collected by the TRACE analysis and compared with the experimental data.[10]

The primary single phase natural circulation and the heat transfer in covered core phenomena are qualitatively predicted by the code as it is shown by the inlet/outlet fluid core temperatures, Fig. 4, the primary volumetric flow rate, Fig. 5, and the difference between the core inlet and outlet fluid temperature.

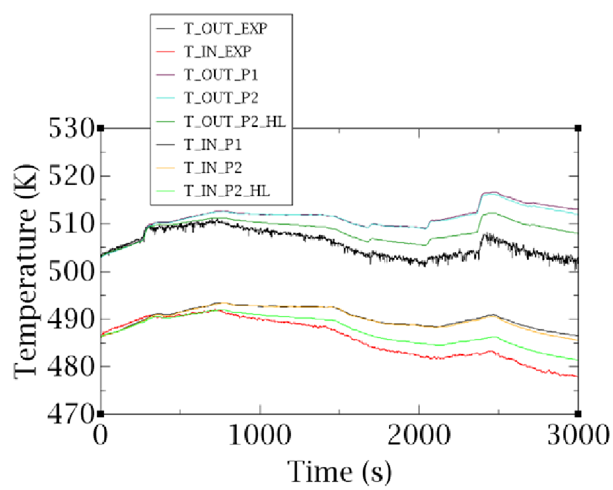

Fig. 4. Experimental data versus code calculations for fluid temperature at the core outlet/inlet for the OSU-MASLWR-002 test.

The analysis of the data related to the flow temperature after the SG primary side section and the core inlet temperature, show that the direct heat exchange, through the internal shell, between the fluid ascending the HL and the fluid descending the CL, determines a fluid temperature increase along the down comer region. The phenomenon is qualitatively predicted by the TRACE model used[10]. Besides, the TRACE code is qualitatively able to predict the temperature decrease along the riser due to the same phenomenon. Fig. 6. shows the RPV temperature profile of the TRACE model, developed by using SNAP.

The level of the integrated PRZ, shown in Fig. 7, is qualitatively predicted by the code. Previous PRZ pressure discrepancies, Fig. 8, predicted by the TRACE V5 patch 01, are now not predicted by the patch 02 that shows, in general, a more stable prediction of PRZ pressure and level.

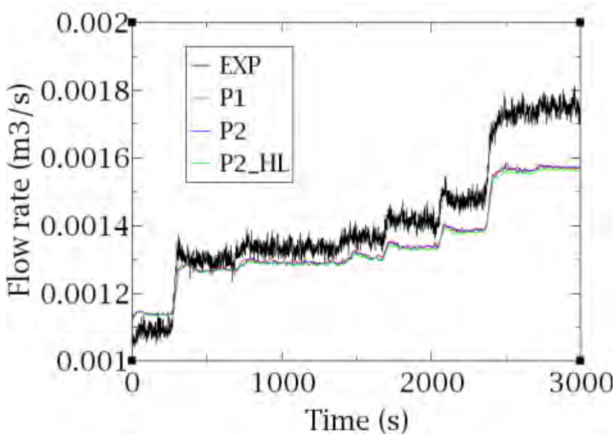

Fig 5: Experimental data versus code calculations for primary volumetric flow rate for the OSU-MASLWR-002 test.

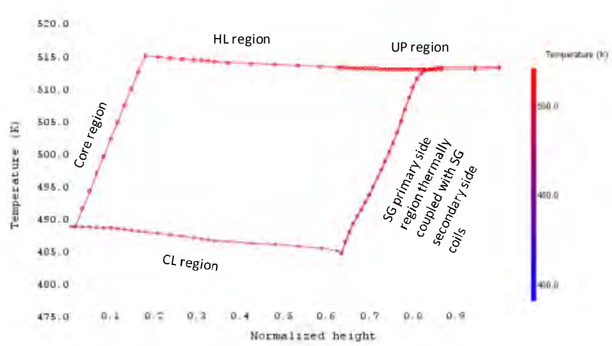

Fig 6: TRACE RPV temperature profile (P2) for the OSU-MASLWR-002 test (2605 s after the SOT).

The heat exchange from primary to secondary side by helical SG in superheated condition is predicted by the TRACE code as it is shown in Fig. 9. As in the experimental data the slope of the main steam superheat curve increases if the value of the core power increases and decreases if the value of the feed water flow rate increases.[10]

Fig. 10, developed by using the SNAP, shows the fluid temperature along the inner helical coil cells and the fluid temperature profile along the SG primary side section. From this figure it is possible to identify the subcooled, saturated and superheat region of the inner equivalent helical coil. In agreement with the experimental data, the steam will leave the SG superheated.

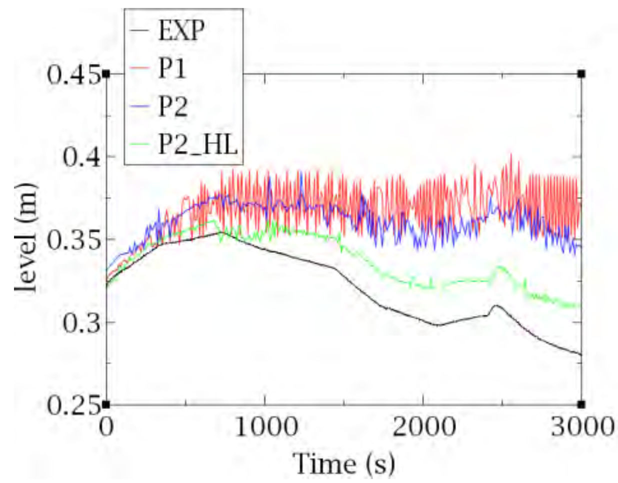

Fig. 7: Experimental data versus code calculations for PRZ level for the OSU-MASLWR-002 test.

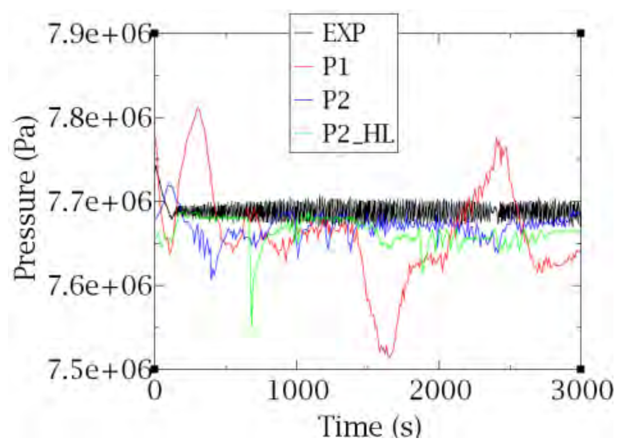

Fig. 8: Experimental data versus code calculations for PRZ pressure for the OSU-MASLWR-002 test.

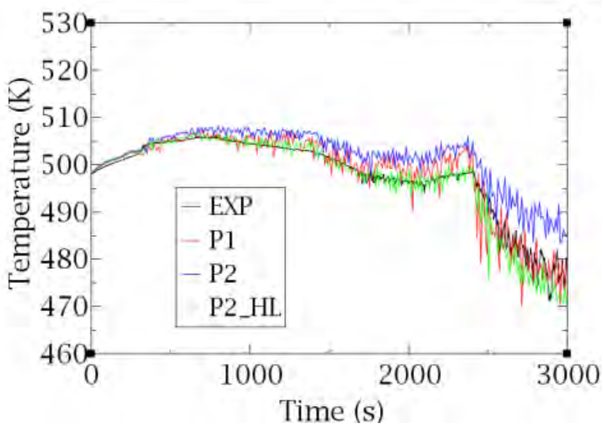

Fig. 9. Experimental data versus code calculations for the fluid temperature at the SG coil outlet for the OSU-MASLWR-002 test.

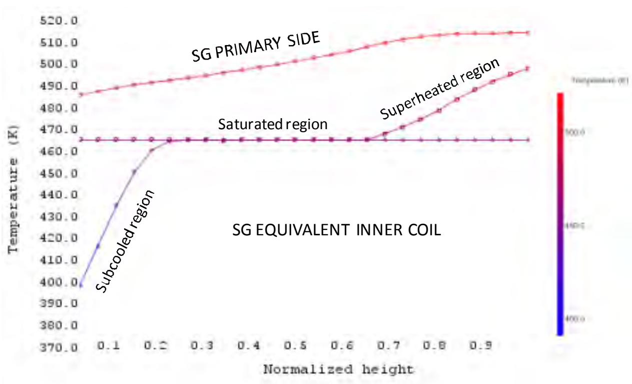

Fig. 10. SG primary side and equivalent inner coil temperature diagram for the OSU-MASLWR-002 test (2605 s after SOT).

The inlet/outlet fluid core temperature, show a qualitative agreement but a general overestimation compared with the experimental data. This could be related to SG primary and secondary side heat transfer. One of the reason could be an underestimation of the helical coil heat transfer coefficient during the different phase of the test. No specific helical coil models has been used during the simulation. However, in order to quantitative evaluate the capacity of the TRACE code to simulate OSU-MASLWR natural circulation phenomena, a qualification of the TRACE nodalization against a heat losses experimental characterization at different primary side temperature is a necessary step. Fig. 4 shows the behavior of inlet/outlet core temperature by increasing the heat losses of the TRACE model (P2_HL). A general quantitative improvement of the calculated data has been showed by P2_HL calculation.[21]

As it is shown by Fig. 5, the primary volumetric flow rate shows an underestimation compared with the experimental data in the last part of the transient. A pressure drop experimental characterization at different mass flow rate is necessary in order to qualify the TRACE nodalization.

A previous analysis, reported in[10], and performed by using the TRACE V5 patch 01, show that one of the reasons of the instability of the superheat condition of the fluid at the outlet of the SG, observed in a previous studies[21] as well, is the equivalent SG model used to simulate the different group of helical coils. In particular, if the helical coils are modelled by only one "equivalent" vertical tube a more stable fluid temperature at the outlet of the helical tubes is predicted by the code.[10]

V.B. Analysis of the OSU-MASLWR-001 test

The purpose of the OSU-MASLWR-001 test, is to determine the behavior of the RPV and containment pressure following an inadvertent actuation of one middle ADS valve located below the HPC and RPV water level before the SOT.[13] The phenomena of interest in this test are single and two-phase natural circulation, heat transfer in covered core and primary/containment coupling following a blowdown.

The analyses of the calculated data, developed by using the TRACE V5 patch 02 (a previous analyses by using the patch 01 is presented in[23]), show that the phenomena of interest in the test are qualitatively predicted by the code. Following the inadvertent middle ADS actuation the blowdown of the primary system takes place. The following subcooled, saturated and a single phase blowdown are predicted by the code. Consequentially, as it is shown in Fig. 11, different depressurization rate are predicted by the code, in agreement with the experimental data.

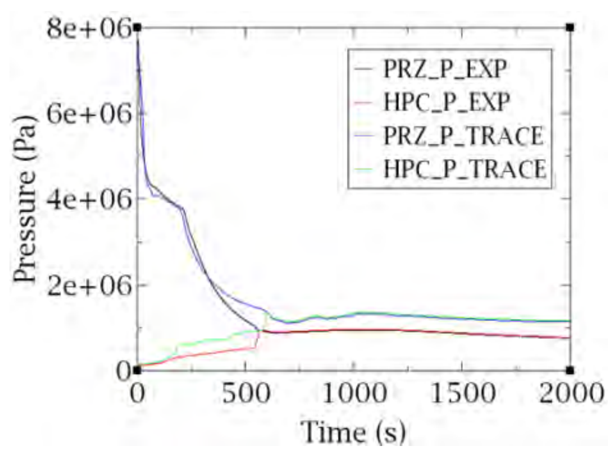

Fig. 11. Experimental data versus code calculations for the PRZ and HPC pressure for the OSU-MASLWR-001 test.

When the pressure difference between the RPV and the HPC reaches a value less than 0.517 MPa, the vent valves are opened which equalizes their pressure in agreement with the experimental data, Fig. 11. When the pressure difference reaches a value less than 0.034 MPa, the sump valves are opened and the refill period begins in agreement with the experimental data. When the sump valves are opened the vapor produced in the RPV goes in the upper part of the facility and through the vent valve goes to the HPC where it is condensed. At this point through the sump line the fluid go to the core again. This mechanism permits the cooling of the core with a two-phase natural circulation. The refill period of the MASLWR is qualitatively predicted by the TRACE code as it is shown by the RPV level, Fig. 12. In agreement with the experimental data, during the simulation, the RPV water level never fell down the upper part of the core.

The PRZ pressure, the HPC pressure, the RPV level and the HPC level are qualitative predicted by the code. In particular the code over predicts the final PRZ pressure. The HPC pressure is over predicted during the transient as well. 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.[22] The topological modeling of the HPC and CPV influences the code prediction. A splitting of the CPV in two thermal hydraulic regions connected by single junctions could allow the possible natural circulation/ convection phenomena inside. The inlet/outlet core temperature is qualitatively predicted by the code and show a general overestimation compared with the experimental data.

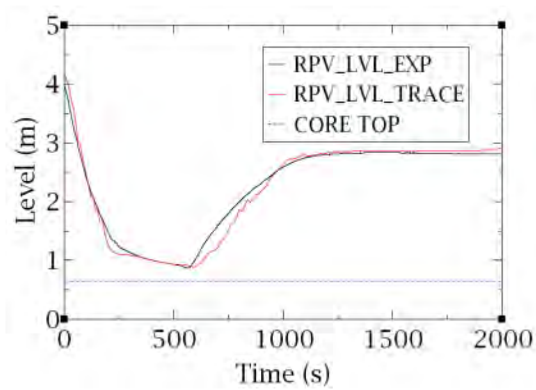

Fig. 12. Experimental data versus code calculations for RPV level for the OSU-MASLWR-001 test.

As it is reported in the[22], in which a comparison between the RELAP5, RELAP3D and TRACE has been performed, the codes tends to over predict important safety related parameters.[22]

Currently the OSU-MASLWR TRACE model has been updated[26], to simulate the IAEA ICSP tests, taking into account the updated facility characterization conducted in the ICSP framework.[24]

A depressurization phase following an ADS blowdown, during one of the phase of one the ICSP test, is shown in Fig. 13.

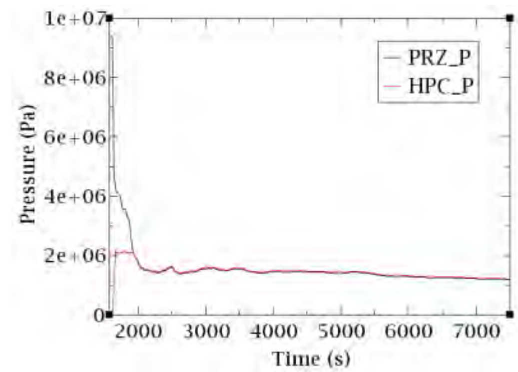

Fig. 13. PRZ and HPC pressure calculated data for ICSP test.[26]

Fig. 14 shows an animation of the calculated facility fluid conditions during a two phase natural circulation at reduced primary side mass inventory.

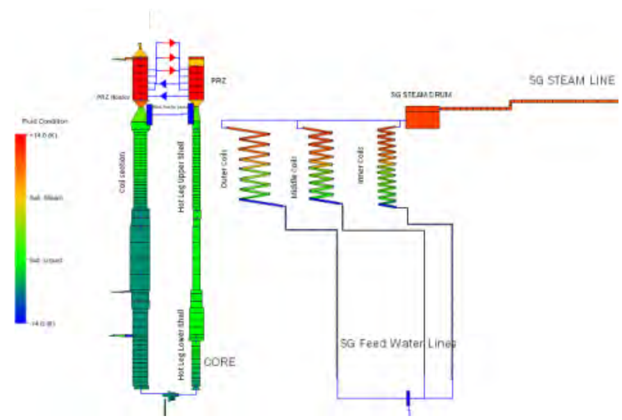

Fig. 14. Calculated facility fluid condition during natural circulation at reduced primary side inventory for ICSP test.[26]

CONCLUSIONS

The validation and assessment of the best estimate thermal hydraulic system code TRACE against the MASLWR natural circulation phenomena in steady and transient condition is the topic of the present research activity. The experimental database has been developed in a first experimental campaign at OSU-MASLWR experimental integral test facility, constructed, under a U.S. Department of Energy grant, at Oregon State University. The OSU-MASLWR-001 test, an inadvertent actuation of 1 submerged ADS valve, and the OSU-MASLWR-002 test, a natural circulation test, have been selected and analyzed by using the TRACE V5 code.

Since the qualification process of the OSU-MASLWR TRACE nodalization is still in progress, considering the facility characterization conducted in the IAEA ICSP framework, 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 design.

The analyses of the OSU-MASLWR-002 test show that the TRACE code is able to qualitatively predict natural circulation phenomena and heat exchange from primary to secondary side by helical SG in superheated condition. The subcooled, saturated and superheated region of the SG secondary side are predicted by the code resulting in steam superheat at the SG exit. An overestimation of the inlet/outlet core temperature is predicted by the code. One of the reasons could be an underestimation of the helical coil heat transfer coefficient during the different phase of the transient. TRACE model heat losses calibration against an experimental heat losses characterization is necessary.

The analysis of the OSU-MASLWR-001 test show that the TRACE code is able to qualitative predict the single and two-phase natural circulation and primary/containment coupling phenomena characterizing the test. The subcooled saturated and single phase blowdown is predicted by the code. The refill of the core, permitting its cooling, is predicted as well. In agreement with the experimental data the RPV level water never fell down the upper part of the core during the blowdown following the middle valve opening. The results of the calculated data show a general over prediction compared with the experimental data.

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 characteristic 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.[27]

NOMENCLATURE

ADS, Automatic Depressurization System;
CHF, Critical Heat Flux;
CL, Cold Leg;
CPV, Cooling Pool Vessel;
HL, Hot Leg;
HPC, High Pressure Containment;
IAEA, International Atomic Energy Agency;
ICSP, International Collaborative Standard Problem;
LOCA, Loss of Coolant Accident;
LP, Lower Plenum;
LWR, Light- Water Reactor;
MASLWR, Multi-Application Small Light-Water Reactor;
NSSS, Nuclear Steam Supply System;
OSU, Oregon State University;
PRZ, Pressurizer;
PWR, Pressurized Water Reactor;
RPV, Reactor Pressure Vessel;
SBLOCA, Small Break Loss of Coolant Accident;
SG, Steam Generator;
SOT, Start of the Transient;
SNAP, Symbolic Nuclear Analysis Package;
TRACE, TRAC/RELAP Advanced Computational Engine;
UP, Upper Plenum;
USNRC, U.S. Nuclear Regulatory Commission.

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