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# Sensitivity analysis of the MASLWR helical coil steam generator using TRACE

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ABSTRACT

Accurate simulation of transient system behavior of a nuclear power plant is the goal of systems code calculations, and the evaluation of a code's calculation accuracy is accomplished by assessment and validation against appropriate system data. These system data may be developed either from a running system prototype or from a scaled model test facility, and characterize 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. The validation and assessment of the best estimate thermal hydraulic system code TRACE against the Multi-Application Small Light-Water Reactor (MASLWR) Natural Circulation (NC), helical coil Steam Generator (SG), Nuclear Steam Supply System (NSSS) design is a novel effort, and is the topic of the present paper. Specifically, the current work relates to the assessment and validation process of TRACE code against the NC database developed in the OSU-MASLWR test facility. This facility was constructed at Oregon State University under a U.S. Department of Energy grant in order to examine the NC phenomena of importance to the MASLWR reactor design, which includes an integrated helical coil SG. Test series have been conducted at this facility in order to assess the behavior of the MASLWR concept in both normal and transient operation and to assess the passive safety systems under transient conditions. In particular the test OSU-MASLWR-002 investigated the primary system flow rates and secondary side steam superheat, used to control the facility, for a variety of core power levels and Feed Water (FW) flow rates. This paper illustrates a preliminary analysis, performed by TRACE code, aiming at the evaluation of the code capability in predicting NC phenomena and heat exchange from primary to secondary side by helical SG in superheated condition and to evaluate the fidelity of various methods to model the OSU-MASLWR SG in TRACE. The analyses of the calculated data show that the phenomena of interest of the OSU-MASLWR-002 test are predicted by the code and that one of the reasons of the instability of the superheat condition of the fluid at the outlet of the SG is the equivalent SG model used to simulate the different group of helical coils. The SNAP animation model capability is used to show a direct visualization of selected calculated data.



## 1. Introduction

Accurate simulation of transient system behavior of a nuclear power plant is the goal of systems code calculations, and the evaluation of a code's calculation accuracy is accomplished by assessment and validation against appropriate system data. These system data may be developed either from a running system prototype or from a scaled model test facility, and characterize 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.

In this framework Oregon State University (OSU) has constructed, under a U.S. Department of Energy grant, a system-level

*Abbreviations:* ADS, Automatic Depressurization System; CHF, Critical Heat Flux; CL, Cold Leg; FW, Feed Water; 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; MS, Main Steam; NC, Natural Circulation; NPP, Nuclear Power Plant; NSSS, Nuclear Steam Supply System; OECD, Organization for Economic Cooperation and Development; OSU, Oregon State University; PARCS, Purdue Advanced Reactor Core Simulator; PKL, Primärkreisläufe (Test Facility); PRZ, Pressurizer; PWR, Pressurized Water Reactor; RHRS, Residual Heat Removal System; RELAP, Reactor Excursion and Leak Analysis Program; ROSA/LSTF, ROSA Large Scale Test Facility; RPV, Reactor Pressure Vessel; SESAR, Senior Group of Experts on Nuclear Safety Research; SETH, SESAR Thermal Hydraulics; SBLOCA, Small Break Loss of Coolant Accident; SG, Steam Generator; SNAP, Symbolic Nuclear Analysis Package; TRACE, TRAC/RELAP Advanced Computational Engine; UP, Upper Plenum; USNRC, U.S. Nuclear Regulatory Commission.

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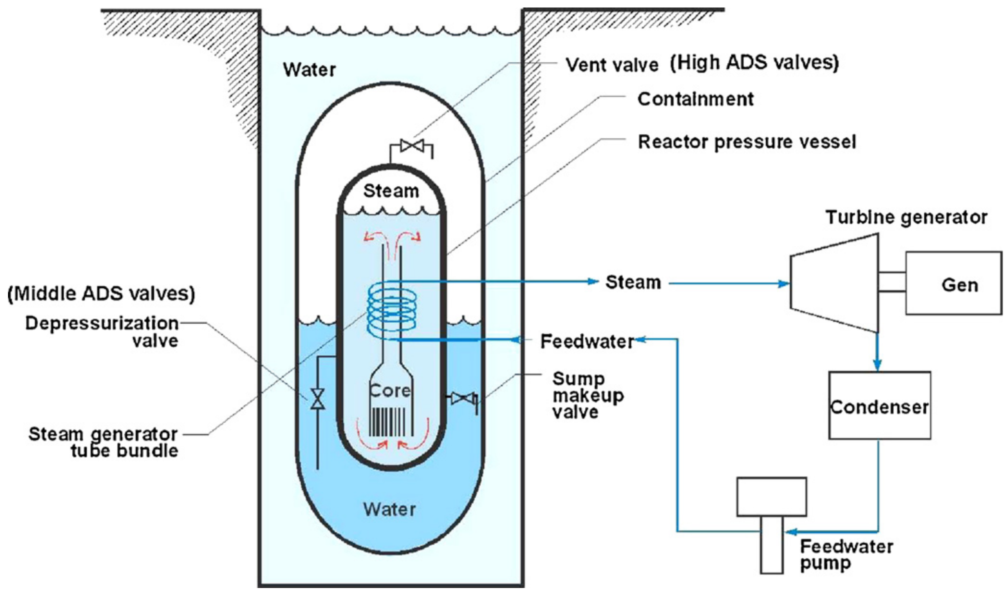


**Fig. 1.** MASLWR conceptual design layout (Modro et al., 2003; Reyes et al., 2007).

test facility to examine NC phenomena of importance to the MASLWR reactor design, developed by Idaho National Engineering and Environmental Laboratory, OSU and Nexant–Bechtel.

The MASLWR (Modro et al., 2003), Fig. 1, is a small modular pressurized light-water reactor relying on NC during both steady state and transient operation, which includes an integrated SG consisting of banks of vertical helical tubes contained within the Reactor Pressure Vessel (RPV) and located in the upper region of the vessel outside of the Hot Leg (HL) chimney. The primary coolant flows outside the SG tubes, and the FW is fully vaporized resulting in superheated steam at the SG exit. MASLWR's safety systems are designed to operate passively. The RPV is surrounded by a cylindrical containment, partially filled with water, which provides pressure suppression and liquid makeup capabilities. The RPV can be depressurized using the Automatic Depressurization System (ADS), which consists of six valves discharging into various locations within the containment. The entire containment vessel is submerged in a pool of water that acts as the ultimate heat sink. The MASLWR has a net output of 35 MWe. Its small size makes the prototypical MASLWR relatively portable and thus well suited for employment in smaller electricity grids. These smaller electricity grids may be found in developing or remote regions.

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 NC reactor plants and accompanying passive safety features as well. Furthermore an IAEA International Collaborative Standard Problem (ICSP) (Woods and Mascari, 2009) on the "Integral PWR Design Natural Circulation Flow Stability and Thermo-Hydraulic Coupling of Containment and Primary System during Accidents" will be executed in the facility. The purpose of this IAEA ICSP is to provide experimental data on single/two-phase flow instability phenomena under NC conditions and coupled containment/reactor vessel behavior in integral-type reactors. This data can be used to assess thermal hydraulic codes for reactor system design and analysis.

In order to analyze the thermal hydraulic behavior of LWR reactors, the USNRC has maintained four codes: RAMONA, RELAP5, TRAC-B and the TRAC-P (Boyack and Ward, 2000). In the last years the NRC has developed an advanced best estimate thermal hydraulic system code, called TRAC/RELAP Advanced Computational Engine or TRACE (Cheng et al., 2009; Reyes, 2005; TRACE V5.0, 2008), to perform best estimate analysis for LWR. Different studies using the TRACE code have been developed in the recent years. A TRACE model of Almaraz NPP has been used to study a loss of RHRS at midloop operation (Queral et al., 2008). A TRACE model of the Maanshan PWR NPP has been used to evaluate its effectiveness by simulating a turbine trip and load reduction transients and comparing the results with Maanshan NPP data (Wang et al., 2009). A TRACE model of the PKL test facility has been used to simulate the PKL III E3.1 test (Jasiulevicius, 2005) – loss of RHRS in midloop operation with the reactor coolant system closed – in the framework of OECD SETH/PKL benchmark on test E3.1 (Bucalossi, 2006). A TRACE model of ROSA/LSTF test facility has also recently been used to simulate a RPV upper head SBLOCA test (Freixa and Manera, 2010). Furthermore, the analysis of a test of "inadvertent actuation of 1 submerged ADS valve" (OSU-MASLWR-001 test), performed in the OSU-MASLWR test facility, has been developed using TRACE, RELAP5/Mod3.3, and RELAP5-3D code (Pottorf et al., 2009).

In the framework of the performance assessment and validation of thermal hydraulic codes, this paper illustrates a preliminary analysis, performed by TRACE code (TRACE V5.0 Patch 01), aiming at the evaluation of the code capability in predicting NC phenomena and heat exchange from primary to secondary side by helical SG in superheated condition and to evaluate the fidelity of various methods to model the SG in TRACE by simulating the OSU-MASLWR-002 test.

## 2. Description of the OSU-MASLWR facility

### 2.1. OSU-MASLWR test facility overview

The OSU-MASLWR test facility (Galvin, 2007; Reyes et al., 2007), shown in Fig. 2, is scaled at 1:3 length scale and 1:254 volume scale, is constructed entirely of stainless steel, and it is designed for full pressure (11.4 MPa) and full temperature (590 K) prototype operation. The facility includes the primary and secondary circuit and the containment structure. The primary circuit includes the RPV and the ADS blowdown lines, vent lines and sump recirculation lines. The internal components of the RPV, Fig. 3, 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 secondary circuit includes the FW treatment and storage system, the main FW pump, the main FW system supply lines, the SG sec-