Sensitivity analysis of the MASLWR helical coil steam generator using TRACE

Fulvio Mascari, Giuseppe Vella  
Dipartimento di Ingegneria Nucleare, Università degli Studi di Palermo, 
Viale delle Scienze, Edificio 6, 90128 Palermo

Brian Woods  
Department of Nuclear Engineering and Radiation Health Physics, Oregon State University 
128 Radiation Center Corvallis, OR 97331-5902

Kent Welter, Jason Pottorf, Eric Young  
NuScale Power Inc. 
201 NW Third Street Corvallis, Oregon 97330

Martina Adorni, Francesco D’Auria  
San Piero a Grado Nuclear Research Group (SPGNRG) 
University of Pisa, Italy

ABSTRACT

Accurate simulation of transient system behavior of a nuclear power plant is the goal of systems code calculations, and evaluation of a code's calculation accuracy is accomplished by assessment and validation against appropriate system data. These system data may be either from a running system prototype or scaled model test facility, and are comprised of the relevant phenomena during both steady state and transient conditions. The identification and characterization of the relevant data, 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 TRACE systems code against the Multi-Application Small Light-Water Reactor (MASLWR) natural circulation, helical coil steam generator, 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 the system thermal-hydraulic TRACE code against the natural circulation database developed in the OSU-MASLWR test facility. This facility was constructed at Oregon State University under a Department of Energy grant in order to examine the natural circulation phenomena of importance to the MASLWR reactor design, which includes an integrated helical coil steam generator. This paper illustrates a preliminary sensitivity analysis to evaluate the fidelity of various methods to model the OSU-MASLWR steam generator in TRACE. The results of these analyses show that oscillations observed in a previous study by Mascari (2008) are related to the particular steam generator nodalization used. 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 evaluation of a code's calculation accuracy is accomplished by assessment and validation against appropriate system data. These system data may be either from a running system prototype or scaled model test facility, and are comprised of the relevant phenomena during both steady state and transient conditions. The identification and characterization of the relevant data, 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 Department of Energy grant, a system-level test facility to examine Natural Circulation (NC) phenomena of importance to the MASLWR reactor design, developed by Idaho National Engineering and Environmental Laboratory, OSU and NEXANT–Bechtel. The MASLWR, figure 1a, is a small modular pressurized light water reactor relying on NC during both steady-state and transient operation, which includes an integrated helical coil Steam Generator (SG). Each module of the prototypical MASLWR has a net output of 35MWe [1].

The planned work of OSU, related to the OSU-MASLWR test facility, will be of value not only to specifically investigate the MASLWR concept design further but advance the broad understanding of integral NC reactor plants and accompanying passive safety features as well. Furthermore an IAEA International Collaborative Standard Problem (ICSP) on the OSU-MASLWR test facility is envisaged [2].

In the framework of the performance assessment and validation of thermal hydraulic codes this paper illustrates a preliminary sensitivity analysis, performed by TRACE code (TRACE V5.0 Patch 01), to evaluate the fidelity of various methods to model the OSU-MASLWR SG.

2 DESCRIPTION OF THE OSU-MASLWR FACILITY

2.1 OSU-MASLWR test facility overview

The OSU-MASLWR test facility [3] is scaled at 1:3 length scale, 1:254 volume scale and 1:1 time scale, is constructed entirely of stainless steel, and is designed for full pressure (11.4 MPa) and full temperature (590 K) prototype operation. The test facility includes three major component packages. The first is the primary circuit which includes the RPV with its internal components (core, Hot Leg (HL) riser, SG, pressurizer (PRZ)) and Automatic Depressurization System (ADS) blowdown lines, vent lines and sump recirculation lines. The facility RPV key areas are shown in figure 1b. The second is the secondary circuit which includes the SG (internal to vessel), feed water pump, and associated feed water and steam valves. The third is the containment structure. The test facility models the containment structure in which the RPV sits as well as the cavity within which the containment structure is located. This modelling is accomplished by using two vessels, a high pressure containment vessel and a cooling pool vessel, with an heat transfer surface between them to establish the proper heat transfer area. In addition to the physical structures that comprise the test facility, there is an instrumentation and control system.

2.2 OSU-MASLWR SG overview

In the MASLWR concept design, the primary coolant is circulated around the outside of the SG tubes. The test facility tube bundle, figure 2, is a helical coil consisting of fourteen tubes of 1.59 cm OD with a total heated length of 86.0 m. This SG is a once through heat exchanger and is located within the pressure vessel in the annular space between the HL riser and the inside surface of the RPV. There are three separate parallel sections (coils) of stainless
steel tubes. The outer coil and middle coils consist of five tubes each while the inner coil consists of four tubes. Each coil is separated from the others but joined at a common inlet header to ensure pressure equilibrium within the coil.

Figure 1: a) MASLWR conceptual design layout and b) OSU-MASLWR RPV key areas.

Cold Main Feed Water (MFW) enters at the bottom of the SG and boils off after travelling a certain length in the SG. This boil off length is a function of both core power and MFW flow rate and can be adjusted by varying core power, feedwater flow rate or both. Nominally, the boil off length is approximately 40% shorter than the actual length of the SG tubes so the steam will leave the SG superheated. The value of the degree of the steam superheat is changed in order to control the facility. In general 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 decreases. The difference between the main steam saturation temperature and the measured main steam temperature is used to estimate the value of the main steam superheat. Each SG coil exhausts the superheated steam into a common steam drum from where it is subsequently exhausted to atmosphere via the main steam system.

Figure 2: Photographs of SG coil bundle arrangement

The OSU-MASLWR-002 test (NC operation with core power up to 210 kW) and the OSU-MASLWR-003A (NC operation at 210 kW) investigated the primary system flow rates and secondary side steam superheat for a variety of core power levels and feedwater flow rates. OSU-MASLWR-002 stepped power levelincrementally to 210 kW, varying feed water
flow rate at each power level, and OSU-MASLWR-003A was an extended 210 kW steady
test establishing initial conditions for the OSU-MASLWR-003B, inadvertent high
containment ADS vent line actuation [1,3,4].

3 CODE APPLICATION

3.1 TRACE code

In order to analyze the behaviour of the PWR and BWR reactors, the NRC has
maintained four thermal-hydraulic codes, the RAMONA, the RELAP5, the TRAC-BWR and
the TRAC-PWR. The RELAP5 code produces a one-dimensional representation of the flow
field in the analysis of the LWR reactor. It includes a point reactor kinetics models. The
TRAC-P code provides a modeling of multidimensional flows, in the analysis of the PWR
reactor. It is used in particular for the LBLOCA transients. The RAMONA code produces a
very simple ID representation of the flow field in the analysis of the BWR reactor. It also
includes a three-dimensional reactor kinetics model. The TRAC-B code produces a detailed
representation of the flow field in the study of the BWR.

In the last years the USNRC is developing a modern advanced best estimate reactor
system code, by merging the capability of the previous codes into a single code. The new
code is called the TRAC/RELAP Advanced Computational Engine or TRACE, and is a
component-oriented code designed to analyze reactor transients for PWR and BWR.

It is a finite-volume, two-fluid, code with 3-D capability. It is possible to model heat
structures and control systems that interact with the component models. It can be run coupled
with the three dimensional reactor kinetics code PARCS.

TRACE has been used together with an user-friendly front end, Symbolic Nuclear
Analysis Package (SNAP), able to support the code user in the development and visualization
of the model [5,6,7,8].

3.2 OSU-MASLWR TRACE model

The present OSU-MASLWR TRACE model [9] made by using SNAP and shown in
figure 3, is developed in order to evaluate the TRACE capability in predicting NC phenomena
and heat exchange from primary to secondary side by helical SG in superheat condition. This
preliminary nodalization models the primary and the secondary circuit. The third major
facility component package is not present in this study. A complete preliminary nodalization
of the facility is used in the simulation of the test OSU-MASLWR-001, an inadvertant
actuation of 1 submerged ADS valve, presented in the REF[10]. The core and the PRZ heaters
are simulated and are in operation during the simulation of the 002 transient.

This paper illustrates a preliminary sensitivity analysis in order to evaluate the fidelity
of various methods to model the OSU-MASLWR SG in TRACE by using three different
equivalent helical coils model. In the first model (REF) the SG coils are simulated with three
different equivalent oblique group of pipes in order to simulate the three separate parallel
sections (coils) of tubes. In the second model (SEN1) the SG coils are simulated with three
different equivalent vertical group of pipes. In the third model (SEN2) the SG coils are
simulated with only one equivalent vertical group of pipes thermally coupled with only one
equivalent heat structure. In all the different equivalent models the secondary circuit
comprises the common inlet header, the SG coil and the steam drum.
3.3 TRACE model qualification process

A nodalization, representing an actual system (integral test facility or nuclear power plant), can be considered qualified when: a) it has a geometrical fidelity with the involved system, b) it reproduces the measured nominal steady-state conditions of the system, and c) it shows a satisfactory behavior in time dependent conditions [9].

The OSU-MASLWR test facility nodalizations qualification process is still in progress, because the facility experimental characterization will be conducted in the framework of the envisaged ICSP in order to evaluate several important facility operational characteristics and other characteristics determined to be of importance during the planned ICSP experiments. Some nodalization model are still preliminary because some physical structure facility characteristics and the complete instrument characterization and location will be delivered in the ICSP framework as well.

3.4 OSU-MASLWR-002 test reference and sensitivity calculation results

The OSU-MASLWR-002 test reference calculated results show a qualitative agreement with the experimental data for a number of main important variables as the core inlet/outlet and the SG outlet temperature shown in figure 4 a and b, the core mass flow and the delta T core shown in figure 5 a and b. The core flow calculated data show an underestimation compared to the experimental data in the last 500 s of the transient, pressure drop calibration is required, as shown in figure 5 a. The core delta T (related to the core mass flow, core power and the thermophysical condition of the primary fluid) is well predicted during the first 2500 s and it shows an overestimation in the last 500 s of the transient, figure 5 b.

The average temperature at the outlet of the helical coils shows a qualitative agreement compared to the experimental data during the first 2500s but, after about the first 350s of the transient, show a continuous oscillation thought the secondary fluid is always in superheat condition. In the last 500s, the fluid at the outlet of the SG helical coils is almost in saturated condition, figure 4 b. In the simulation, the time dependence of the pressure at the outlet of the SG is fixed as a boundary condition. With this condition, the inlet/outlet SG pressure show agreement compared to the experimental data. A detailed model of the main steam line (hydraulic volume, heat structures and related logic used to control the main steam pressure) is necessary to further investigate the SG outlet pressure and temperature.

Figure 3: OSU-MASLWR TRACE model (REF)
The PRZ level is qualitatively predicted by the code but shows a continuous oscillation during all the transient, figure 6 a. The PRZ pressure behavior shows discrepancies compared with the experimental data, figure 6 b. More investigation need the PRZ nodalization (hydraulic volume and heat structure) and the logic used to control the PRZ pressure by using the PRZ heaters.

A detailed analysis of the pre-test phase, used to reach the Boundary Initial Condition (BIC), is suggested in order to have a more accurate simulation of the selected transient.

The results of the sensitivity analyses show that the stability of the superheat condition of the fluid at the outlet of the SG is related to the equivalent SG model used to simulate the different group of helical coils. If the helical coils are modelled by only one equivalent vertical tube (SEN2) a more stable temperature at the outlet of the helical tubes is predicted by the code, figure 4 b. However the experimental flow regime development in the SG helical coil, that influences the degree of superheat, is unknown therefore more investigation need the fidelity of various methods to model the OSU-MASLWR SG in order to simulate the possible flow regime development in the SG helical coil.

The comparison of the reference results produced by the TRACE V4225 and the TRACE V5.0 Patch 01 show that the TRACE V 5.0 predicts a more stable superheat condition at the outlet of the SG and there is not the tendency of the SG outlet temperature to oscillate between the saturated and superheat condition. A correct qualitative behavior of the core inlet/outlet temperature and PRZ level in the last 500s of the transient is predicted by using the TRACE V5.0
Patch 01. A previous analysis of the OSU-MASLWR 002 test, performed by using the TRACE V4225, is presented in the REF[9].

The figure 7 shows a SNAP animation model used to have a direct visualization of a selected parameter in all the volumes of the nodalization.

Figure 6: Experimental and calculated data for a) PRZ Level and b) PRZ pressure

Figure 7: SNAP animation model used to analyse the OSU-MASLWR-002 test

4 CONCLUSION

The calculated data here presented are focused on a preliminary analyses of the capability of the TRACE V5.0 Patch 01 code to correctly predict NC phenomena and heat exchange from primary to secondary side by helical SG by evaluating the fidelity of various methods to model the OSU-MASLWR SG.

The OSU-MASLWR-002 test reference calculated results show a qualitative agreement with the experimental data for a number of main important parameters like inlet/outlet core temperature, PRZ level, core mass flow rate, delta T core and SG outlet temperature. The PRZ pressure behavior shows discrepancies compared with the experimental data. In general the comparison between the experimental and calculated data show that primary circuit stores more energy compared with the experimental data. A detailed analysis of the pre-test phase, used to reach the BIC, and of the PRZ and steam line nodalization (hydraulic volume, active and not
active heat structure, related logic used to control the PRZ and main steam pressure respectively) is suggested in order to have a more accurate simulation of the selected transient.

The results of the sensitivity analyses show that the different equivalent SG helical coils models show a different stability condition at the outlet of the SG. In particular by simulating the three SG coil groups with only one equivalent tube a more stable superheat condition is reached at the SG outlet. More investigation need the model with three different oblique or vertical tubes in order to study the possible instability conditions predicted by the code.

However in order to evaluate the real capability of the code in predicting heat exchange from primary to secondary side by helical SG in superheat condition, is necessary, after the release of the experimental data, the complete qualification of several important operational characteristics of the facility including operational system/component heat losses and pressure drops (forward/reverse and single/two phase).

REFERENCES


