Analysis of Blowdown Event in Small Modular Natural Circulation Integral Test Facility

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INTRODUCTION

The design of a new generation of nuclear power reactors is underway with an emphasis on safety and passive systems. The NuScale Power, LLC (NuScale) small modular reactor (SMR) is a state of the art design with enhanced safety relying solely on passive systems and natural circulation driven flows. The design features a natural circulation primary system within which the core heats the coolant causing it to rise through a central hot leg after which it turns in an upper plenum and is cooled by a helical coil steam generator (HCSG). The lower temperature coolant then flows down the cold leg downcomer where it reaches the entrance of the core to complete the coolant flow circuit. The reactor pressure vessel (RPV) is housed within a submerged high pressure steel containment structure which serves not only as a radiation barrier but also a coolant flow path in the event of a loss-of-coolant-accident (LOCA). In the rare event that coolant escapes the RPV, the coolant will enter the lower pressure containment and flash to steam. Steam that is formed will condense on the cool containment shell, and heat will be transferred to the reactor building pool by conduction through the containment shell. The water level in the containment will rise as the RPV blows down into the containment. Once a specified level is established, emergency core cooling system (ECCS) valves located near the top of the reactor vessel and in the downcomer above the core are opened allowing a controlled flow path to form and liquid coolant to enter back into the RPV. Sizing of the containment vessel has been established such that heat transfer to the pool will exceed core decay heat production. This ensures that fuel damage cannot occur and that short and long term core coolability is maintained. NuScale will be submitting this reactor design for certification to the NRC and has performed scaled integral system experiments in support of this endeavor. The focus of the work presented here is to demonstrate the functionality of the NuScale Integral System Test (NIST) Facility as well the ability of the NRELAP system thermal-hydraulics code to simulate its behavior.

FACILITY DESCRIPTION

The NIST facility is a 1:3 length scale, 1:253 volume, scale and 1:1 time scale integral test facility built on the Oregon State University (OSU) campus. It was originally developed and constructed in the 2000–2003 timeframe by OSU in a joint program with the Idaho National Engineering and Environmental Laboratory, and NEXANT-Bechtel [1]. Since 2008, NuScale has had exclusive rights to use the facility and has made a number of modifications to bring the facility configuration into line with the current NuScale reactor design.

NIST, schematically shown in Fig. 1, includes three major component packages. The first is the primary circuit which includes the RPV with its internal components (electrically heated core, hot leg riser, cold leg downcomer, steam generator, and pressurizer) and ECCS vent and recirculation valves. The second component is the secondary circuit which includes the HCSG, feed water pump, and associated feed water and steam valves. The third component is the containment structure. This structure is modeled as two separate vessels. One vessel models the vapor volume, liquid volume, and condensation surface associated with the prototypic containment vessel. The second vessel models the heat capacity of the water-filled cooling pool within which the containment vessel is prototypically held. These two vessels are separated by a stainless steel heat transfer plate (HTP). This plate models the scaled heat transfer surface and conduction path between the containment vessel and the surrounding cooling pool. The containment vessel is connected to the RPV by four independent ECCS valves and corresponding lines. Two are vent lines that come off of the pressurizer and two are sump recirculation (core makeup) lines that return cool water to the RPV downcomer just above the simulated core. Flow through each of these lines is controlled via independent, automatically-operated valves controlled through the test facility control system.

The test facility is instrumented to capture the behavior of the facility during steady-state and transient operation. Examples of instrumentation available include: mass flow rates (feed water through HCSG coils), volumetric flow rates (main steam flow), differential pressures (across core, hot leg chimney, through HCSG, cold leg downcomer below HCSG), levels (RPV, pressurizer, containment and cooling pool), and temperatures (core inlet, core exit, primary loop through HCSG, transverse across HTP thickness). Controlled systems in the facility include: core heaters (including decay power modeling), main feed water pump, pressurizer heaters, and pressurizer water level.

EXPERIMENTAL DATA AND PREDICTIONS

The scaled experimental test facility was used to simulate the inadvertent opening of one reactor vent valve with subsequent ECCS valve actuation. A description of the experimental data and code analysis is given in the following subsections.

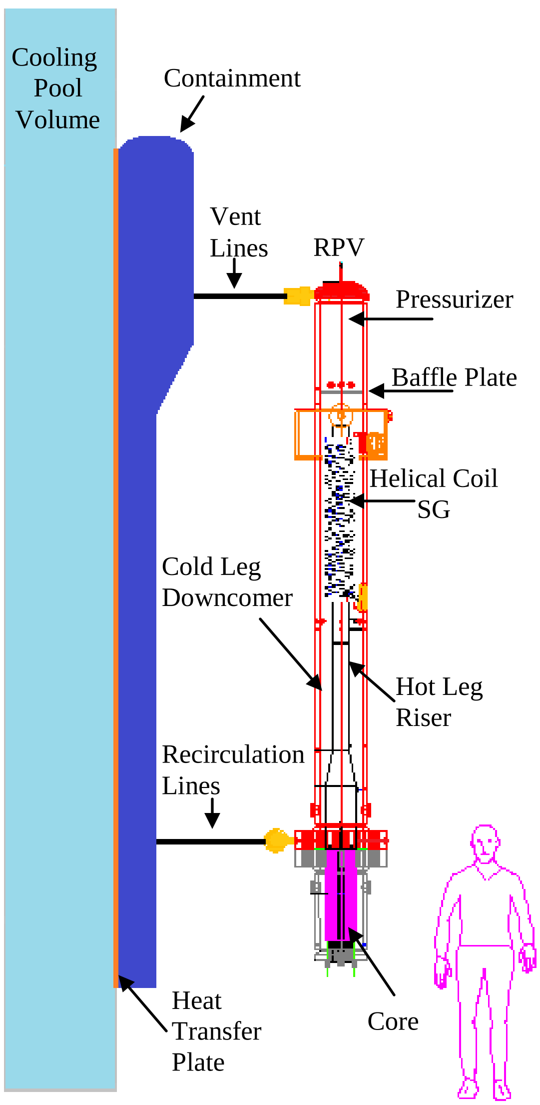

Fig. 1 Schematic of the NuScale Integral System Test (NIST) facility and major components.

Experiment

The experimental data shown in Fig. 2 and Fig. 3, were obtained by opening one reactor vent valve during steady state operation (0-834 s). A sequence of events is shown in Table I. The core heater power then trips to decay power mode and the steam generator feed water and steam lines are isolated. The pressurizer heaters are tripped soon after steam generator isolation. Reactor pressure decays quickly due to flow into containment which simultaneously causes the containment pressure to rise. Peak containment pressure is achieved at approximately 1600 seconds, after which it decreases at a rate similar to that of the RPV. During this period a slight pressure difference between the RPV and containment exists due to flow losses through the vent line and valve. At 6737 seconds, the remaining vent valve and two recirculation valves are opened to create a circulation path between the RPV and containment. Vapor generated in the RPV rises and passes to the containment through the vent lines where it is condensed on the cool containment shell. The condensed volume flows to the bottom of the containment where it passes through the recirculation lines back into the RPV. The level in the RPV remains well above the top of the core during the entire transient. Soon after recirculation valve opening, the level in the RPV slowly rises indicating flow from the containment to the RPV. In the long term (not shown), core decay heat will diminish and RPV and containment level will collapse to a single equilibrium value between the two end points shown.

Table I. Sequence of events

EVENT	Time (s)
Steady State Operation	0-834
1st Reactor Vent Valve Opens	834
Reactor Trip to Decay Power Mode	839
Isolate SG Feed water	842
Isolate SG Steam Line	851
Trip PZR Heaters	854
2nd Reactor Vent Valve Opens	6736
Both Recirculation Valves Open	6736

System Code Analysis

The NuScale thermal-hydraulic system code NRELAP5 was used to predict the behavior of the described experimental transient. The model input is composed of a detailed volume based nodalization which includes all components of the test facility. Steady state conditions are achieved using control systems emulating that of the facility. Valve openings and core trip to decay power mode are accomplished using trip signals with timing equivalent to the experimental actuations. The Henry-Fauske choked flow model is applied at valve locations.

The initial steady state follows that of the experimental data as seen in Fig. 2. At the time of the first vent valve opening, the pressure in the RPV quickly falls and choking phenomena is predicted in the vent line. The predictions trend well with the data over the entire range of the experiment. Peak containment pressure is slightly over predicted but the timing to peak pressure is near the same as that of the data. Long term, after both the reactor vent and recirculation valves are opened there is a slight offset between the data and predictions. This offset can be attributed to model parameters that are slightly out of alignment with the test such as flow loss coefficient through the recirculation valve and heat loss to the environment.

The RPV and containment levels are also well predicted and only slight offsets between data are seen in Fig. 3. The offset in the RPV level near the time of the vent valve opening is due to predicted liquid hold-up in the pressurizer at the baffle plate. This liquid hold-up was not seen in the experiment. Therefore improved facility modeling or improved counter current flow limitation (CCFL) modeling in NRELAP5 will be required to achieve better RPV level predictions while the pressurizer drains. The containment level rises as the RPV is discharging through one vent valve into the containment. Near 4000 seconds the predicted containment level trends just below the data. This behavior is most likely due to heat loss to the environment being over predicted by the model. If calculated heat loss through the containment walls is too large, then the density of fluid and therefore level will be slightly lower than that of the experiment. The same is true for pressure after the recirculation valves are opened.

CONCLUSIONS

NuScale Power has been very active in performing experimental studies which can be used to validate system thermal-hydraulics codes. The NIST facility was used to study system behavior during a LOCA event resulting from the inadvertent opening of a reactor vent valve. The ability of the NRELAP5 code to predict steady state and transient natural circulation behavior for the NuScale SMR design is evident in the calculations. Results show that the pressure in the RPV decreases throughout the experiment while collapsed liquid level is maintained above the heated core. Level trends also show that opening the recirculation valves allows for level in the RPV to recover due to flow from the containment to the RPV.

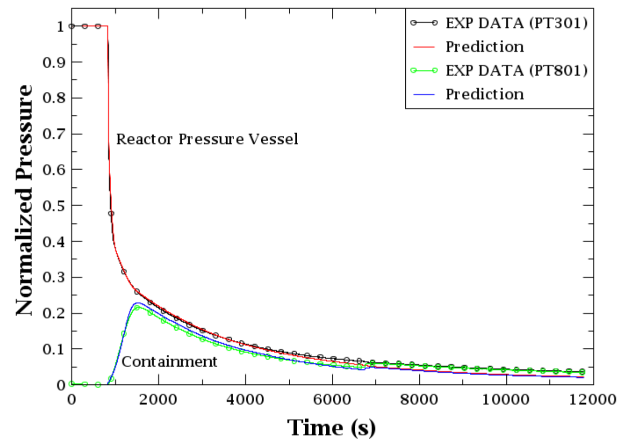

Fig. 2 Comparison of normalized experimental RPV and containment pressure with NRELAP code predictions for blowdown event.

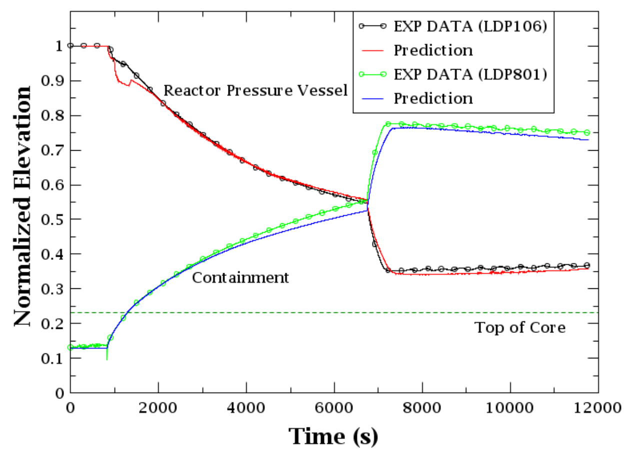

Fig. 3 Comparison of normalized experimental RPV and containment collapsed liquid level with NRELAP code predictions for blowdown event.

NOMENCLATURE

ECCS = emergency core cooling system
EXP = experimental
HTP = heat transfer plate
LOCA = loss of coolant accident
NIST = NuScale Integral System Test
OSU = Oregon State University
RPV = reactor pressure vessel
SMR = small modular reactor
HCSG = helical coil steam generator

REFERENCES

1. S. M. Modro, J. E. Fisher, K. D. Weaver, J. N. Reyes Jr, J. T. Groome, P. Babka and T. M. Carlson, "Multi-Application Small Light Water Reactor Final Report," Idaho National Engineering and Environmental Laboratory, 2003.