

Preliminary melcor input deck for steady state and turbine trip analysis of the NuScale US600 SMR

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Abstract: This paper models NuScale's SMR using MELCOR by relying on specifications in the latest Final Safety Assessment Report (FSAR) submitted to the U.S. Nuclear Regulatory Commission (USNRC). The input deck was crafted from scratch using only publicly available data and reasonable assumptions. Benchmark results for steady state and a turbine trip transient are presented, with the former showing excellent agreement with the reference values, while the latter produces shows slight deviations in the mass flow rates. As a preliminary study, the results are within acceptable limits and encourage further refinement to the model for use in other accident progression cases.

Keywords: MELCOR, small modular reactor, integral pressurized water reactor, safety analysis.

I. INTRODUCTION

The NuScale SMR is an integral Pressurized Water Reactor (iPWR) which was originally designed to generate up to 160MW of thermal power (50MWe) per NuScale Power Module (NPM), with NuScale recently applying to increase it to 250 MW (77MWe). Each NPM is driven by natural circulation whereby the coolant is heated by the core and flows upward through the riser, upper riser turns, and downward through the integrated steam generators and downcomer to the lower plenum to complete the cycle.

Unlike traditional PWRs, the NuScale NPM's main safety features are passively actuated. These include the Emergency Core Cooling System (ECCS), Decay-Heat Removal System (DHRS), and the Ultimate Heat Sink (UHS) in which the NPM is partially submerged [1]. In the event of an accident, the operation and interaction between these systems

will determine the accident progression. To model these phenomena, an input deck for the reactor was written from the ground up using the MELCOR Severe Accident Code [2]. The MELCOR Severe Accident Code has been extensively validated with many experiments throughout the years and has been used to model several nuclear power plants [3, 4, 5, 6].

II. METHODOLOGY

A. Melcor

This work was performed using MELCOR version 2.2.21402, a fully modular engineering level code for calculating the progression of severe accidents in mainly light water reactors (LWRs). MELCOR is "less mechanistic" and has "coarser nodalizations" than RELAP [7], which is able to model sophisticated thermal hydraulic behavior in reactor systems [8]. Broadly speaking, mechanistic codes are used to simulate in

greater detail the phenomena involved in particular accident phases, such as core damage, fission product release, or hydrogen combustion inside the containment.

Nevertheless, MELCOR is capable of giving best-estimate results for various severe accident scenarios, such as core degradation, effectiveness of engineered safety systems, and source term evaluation in good time [8]. Moreover, the code is also extensively validated and verified and currently serves as one of the USNRC's evaluation tools for severe accident progression not just for LWRs but also upcoming advanced reactors such as HTGRs and MSR using its updated functionalities to capture TRISO fuel and delayed neutron precursor behaviors respectively. In this article, the aim is to create an input deck based on non-proprietary information to facilitate collaboration among researchers. While standard practice demands that such models be fine-tuned to match those in existing literature, especially for cases with extensive studies, it is less straightforward for SMRs or advanced reactors with fewer benchmarks in the public domain. One then easily wades into the philosophical issue of striving to align with vendors' results or to report discrepancies, however unflattering it may be to the authors.

B. Nodalizations: control volumes, core, and heat structures

Technical data and specification about the NuScale NPM were taken mainly from the FSAR (Final Safety Analysis Report) and other publicly available sources [1]. These were used to create an input deck from scratch for the MELCOR Severe Accident Code. A simplified schematic of the model is given in Figure 1. The Control Volume Hydrodynamics package (CVH), Flow Path package (FL), Heat Structure package (HS), and Core package (COR) were used to model the bulk of the reactor in

MELCOR. The Engineered Safety Features package (ESF) was used to model the DHRS. The Radionuclide Package (RN1) was used to model the radionuclide transport pathways in the reactor. The Control Functions Package (CF) was also extensively used for fine-tuning and many purposes.

Nodalization of the NuScale SMR, the respective control volumes (CVs) and their indices are shown in Figure 1. Using the CVH package, the core was divided into 21 control volumes (3 radial rings and 7 axial levels) with the first and last rows reserved for the inactive fuel region and baffle plates. The upper and lower risers, upper riser turn, and pressurizer are each modeled by one CV.

One of the most important factors in ensuring proper heat transfer from the primary side of the steam generators (1SGs) to the secondary side of the steam generators (2SGs) is to have a fine nodalization of both SG CVs. As each CV can only record one temperature change, multiple CVs stacked on top of the other will be able to display a temperature gradient, which is desirable when cold water from the bottom of the 2SGs absorbs heat from the 1SGs and transits from being subcooled and saturated water to superheated steam. Thus, in this paper, the primary and secondary steam generators each have 8 CVs. For the feedwater pipe and main steam line, instead of depicting them conventionally as a single vertical train, we have elected to adhere more closely to the actual physical representation of having the feedwater and steam entering and exiting through the top of the containment (see Figure 1).

The COR package of MELCOR is responsible for computing the thermal response of the core, including the lower plenum and lower head as well as their relocation during melting, yielding, and debris transport. Its

nodalization is finer than the CVH's but their overall geometries are consistent, with one exception: In order to avoid rounding issues leading to MELGEN consistency check errors, the CVH core volume is adjusted to be 0.01% larger than the COR volume.

For the core radial nodalization (see Figure 2), the core is radially discretized into 5 regions. The first region, outlined in blue, consists of 9 fuel assemblies (FAs); the second region, outlined in red, consists of the next 12 FAs; the third region, outlined in yellow, contains the final 16 FAs. The last two regions extend until the barrel and vessel inner walls as seen in the orange and blue outlines respectively. In MELCOR's COR package,

however, the nodalizations are recognized as rings in the COR_RP flag. Hence, the areas within the respective regions are treated as concentric circles to obtain their required radii.

With regard to the core axial nodalization in Figure 3, there consists of 10 active fuel segments, outlined in orange, and two inactive regions representing the bottom and top core plates, as well as the nozzle assemblies and fuel rod regions without fuel pellets. These are outlined in red. Comparing the COR package and CVH nodalizations (see Figure 1), each CV elevation (in rows 212 to 216) consists of two core axial nodes. By contrast, the inactive core regions occupy one entire CV (in rows 211 and 217).

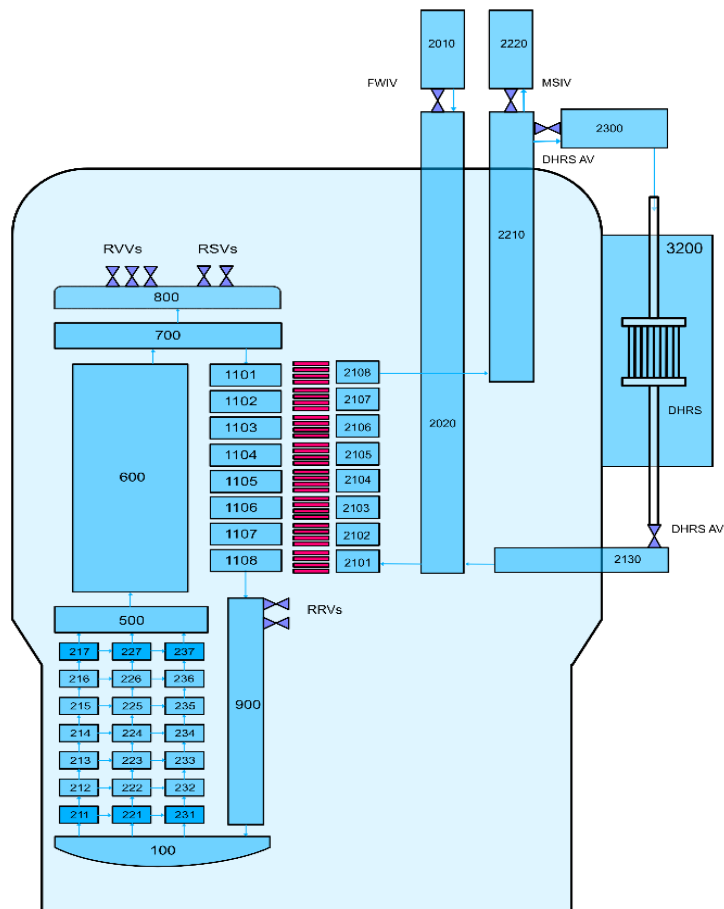


Fig. 1. MELCOR nodalization with DHRS and containment

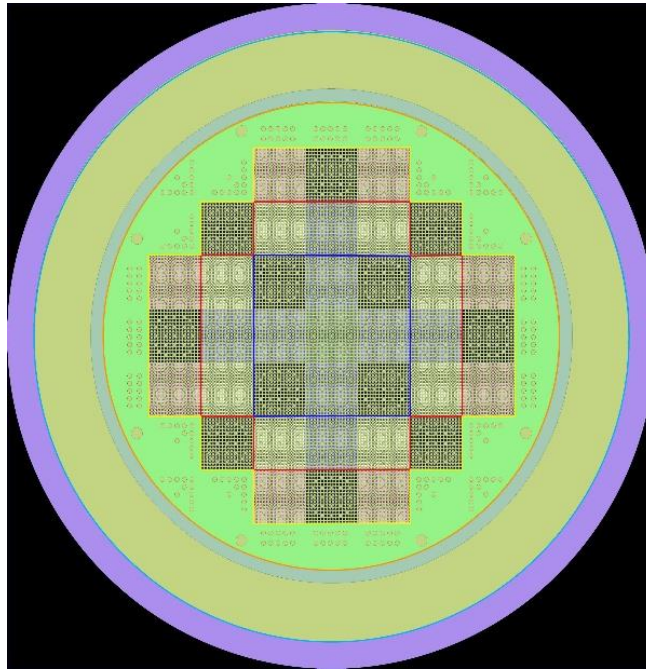


Fig. 2. MELCOR core radial nodalization

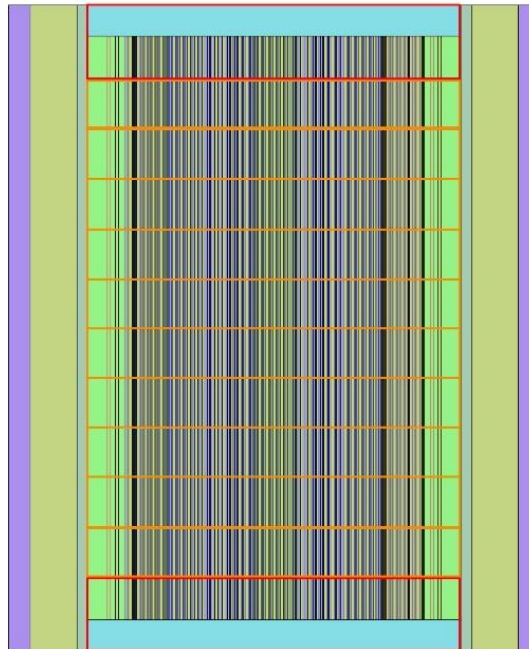


Fig. 3. MELCOR core axial nodalization

Heat structures are defined using the HS package. Each helical tube corresponding to the primary and secondary SGs is further divided into 4 heat structures in a manner as colored in

red in Figure 1. This is to ensure that as the water level in the 2SG changes, heat is transferred to the liquid or steam portion of the 2SG as appropriate.

C. Control functions

In setting up the input deck to eventually model transients, the Control Function (CF) package is key. CFs allow the user to define functions of variables that are declared in MELCOR. The implication is that time-dependent values of variables can be obtained; instead of coding if-else statements to trigger certain conditions and measure values of interest, CFs provide a layer of autonomy for users to design simulations according to their requirements with relative ease and minimal disarray.

D. Proportional Integral Controller

It was found that the temperature of the cold water from the downcomer to the lower plenum is highly sensitive to the mass flow rate of the coolant in the secondary circuit. Thus, to facilitate convergence of primary loop temperatures to the desired operating parameters, it is both necessary and realistic to adjust the secondary flow rate in response to the primary circuit temperatures. This was done using control functions to mimic a proportional integral controller (PIC) unit. As similar approach was also used to regulate the pressurizer sprays and heaters, in response to the pressurizer pressure and water level.

There was insufficient information on the performance of the pressurizer heaters and sprays to fully model the pressurizer. Instead, PIC logic was used to approximate the behaviour of the pressurizer during steady state. The pressure in the pressurizer is monitored in the PIC control function, and heat is directly injected or removed from the liquid portion of the pressurizer volume until the pressure matches the desired primary system operating

pressure. While this method allows instantaneous pressure response and will only be valid after steady state is achieved, the pressurizer is expected to fail very soon after the start of any transient or accident, and hence expected to have limited impact on the results. This method also has the added benefit of allowing much faster convergence of steady-state results.

E. Model limitations and assumptions

Many geometric and engineering parameters used in the model are based on estimates and engineering judgement, and validation is limited to what was described in the section above and is not yet comprehensive. The consequence of this is that our model may not describe the actual plant and its behaviors perfectly, especially for phenomena that have not yet been validated.

III. RESULTS AND DISCUSSION

A. Steady state validation

The input deck was run from -10000 seconds to 0 seconds to obtain steady state values at the rated power of 160MW, which were then benchmarked against the FSAR's NRELAP5 results, an input deck developed by NuScale based on the RELAP5 code. NRELAP5, used for calculating reactor coolant system (RCS) thermal hydraulics, reactor kinetics, and transport of non-condensable gases, is usually paired with SCDAP models to calculate core heatup and damage progression in an integrated fashion. Similarly, results for three other power levels of 120MW, 80MW, and 24MW were produced and compared with their respective FSAR values to ensure the robustness of the input deck. The results are presented in Table I.

Table I. Steady state parameters and their errors for the respective power levels

Power level	160MW		120MW		80MW		24MW	
Parameter	Results	Error (%)	Results	Error (%)	Results	Error (%)	Results	Error (%)
Core power [MWt]	160.19	0.12	120.24	0.20	80.29	0.36	24.36	1.50
Cold leg temp [K]	531.26	0.00	535.27	0.00	539.93	0.00	548.98	0.00
Hot leg temp [K]	583.15	-0.01	579.33	0.02	574.57	0.01	565.25	-0.04
2SG inlet temp [K]	422.00	0.02	422.01	0.03	422.01	0.03	422.02	0.03
2SG outlet temp [K]	578.20	-0.82	578.15	–	574.36	–	565.24	–
Core flow rate [kg/s]	588.59	0.27	521.54	-0.01	444.45	0.17	288.19	2.85
SG flow rate [kg/s]	67.78	1.01	50.86	–	34.10	–	10.45	–

B. Transients

A turbine trip or loss of external load (LOEL) is an event that causes the loss of heat removal from the hot leg due to decreasing steam intake through the turbine to the condenser when the turbine stop valves (TSVs) close [9]. As a result, energy from the steam generators cannot be dispelled, leading to an increase in temperature and pressure in the RPV. Table II shows the FSAR's sequence of events (Max RCS Pressure

Bias) together with the timings achieved in this study. On detection of high pressure, the reactor trip, secondary system isolation (SSI), and DHRS actuation signals are initiated. The power decreases to decay heat levels, while the SSI isolates the feedwater so that the DHRS can transfer the decay heat to the ultimate heat sink (UHS) or reactor pool. Given the loss of offsite power and DC supply, the reactor trip, DHRS actuation, and SSI occur simultaneously.

Table II. Sequence of events for the turbine trip [9]

Event	Time (s)	Reference value (s)
Event initiator - Turbine Trip and loss of FW flow	0	0
Turbine Stop Valves Fully Closed (assumption)	0	0
FW flow is secured (assumption)	0	0
Pressurizer heater power secured	0	0
CVCS Flow Secured	0	0
High Pressurizer Pressure analytical limit	10	10
Reactor Trip, SSI and DHRS signals issued	12	12
Secondary system isolation complete	12	12
RSV Lift Point (14.73 MPa /2137 psia)	15	15
Peak RCS Pressure	15	16
RSV reseats	26	24
DHRS valves open	42	42

As shown in Figure 4, when the turbine trip occurs at $t = 0s$, the reactor trip signal is issued at $t = 12s$. In about 29s after the transient, the reactor power decreases to about 9% of the total reactor power of 160MW. By contrast, the FSAR's results took 39s to reach similar power

levels. Both graphs converge to about 2% of total reactor power at 600s. This work replicates the FSAR power profile by modeling the decay power as an exponential decrease over time with a half-life of 4 seconds, which leads to some slight discrepancy as observed.

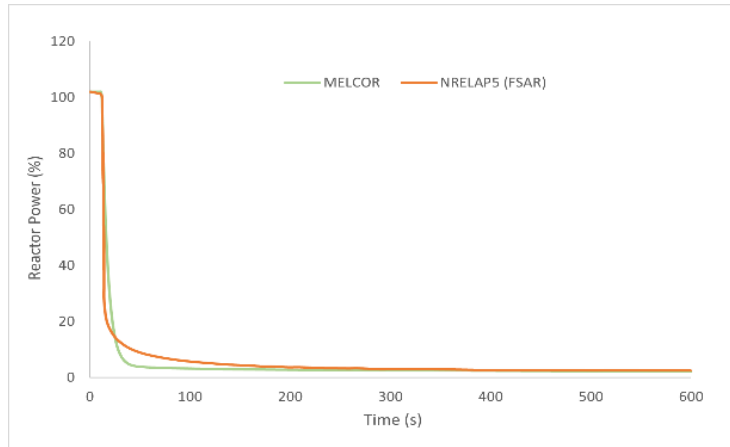


Fig. 4. Reactor power

The DHRS flow rate according to Figure 5 shows that it completely actuates at $t = 42s$. There is a spike in flow at about $t = 22s$, following which it settles to a level of about $4.5kg/s$. Similar patterns are observed with the FSAR but the flow rate is 2.4 times lesser. The difference can be attributed to the lack of sufficient information, such as the geometry and heat transfer model needed to render the DHRS accurately. It should be noted that the

initial flow rate in the FSAR was defined with a slight negative bias, whereas in this work the RCS flow rate is the nominal value since it is a calculation result and not an input.

The mass flow rate of the reactor coolant system or primary circuit is shown in Figure 6. As expected, the RCS flow rate decreases significantly since the start of the turbine trip to reach a minimum value of $104kg/s$ at $74s$. This is compared with a

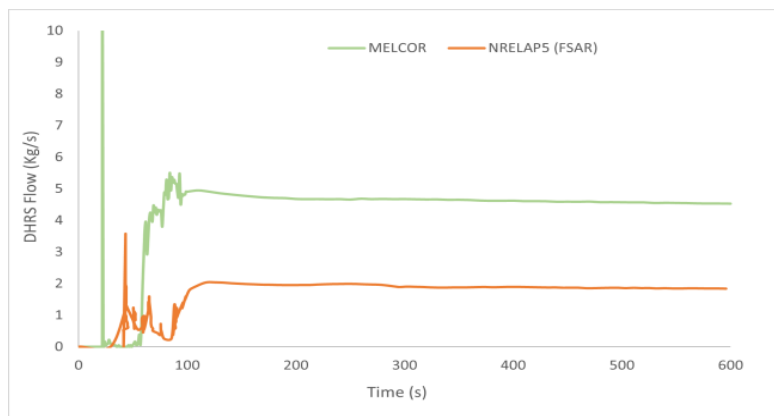


Fig. 5. DHRS flow

value of 36kg/s at 96s for the FSAR. While MELCOR's result stabilizes at around 190kg/s after 170s , the FSAR's has peaks or

oscillations from the 240s mark before tapering off at about 150kg/s at the end of the simulation.

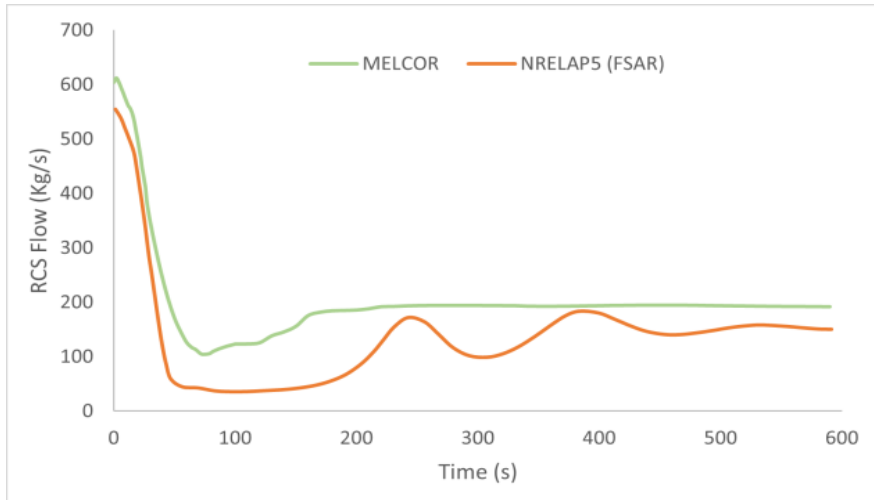


Fig. 6. RCS flow

The RPV pressure as shown in Figure 7 first experiences a spike in pressure due to the turbine trip which prevents heat transfer through the secondary circuit into the condenser. Peak pressure occurs around 14s in MELCOR, which coincides with the timing reported in the FSAR. The RSV actuation lift point is triggered at 14.73 MPa and this allows steam to escape into the containment

to lower the pressure. As the RSV reseats at 24s , the pressure rises again as evidenced by the slight bump in pressure at the 40s mark, and similarly at the 42s in the FSAR. As the DHRS isolation valves fully open and steam begins to be dumped into the DHRS condenser to transfer heat to the UHS, the pressure then gradually decreases to 11.18MPa and 11.80MPa respectively.

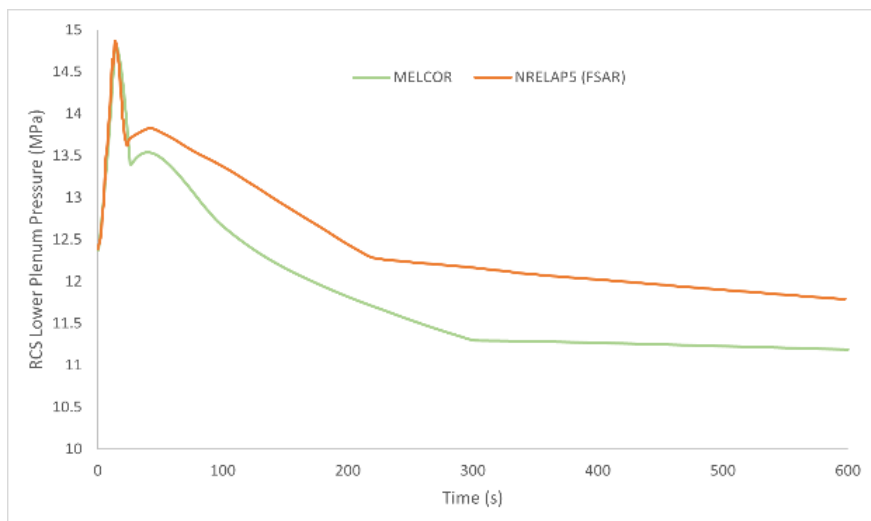


Fig. 7. RPV pressure

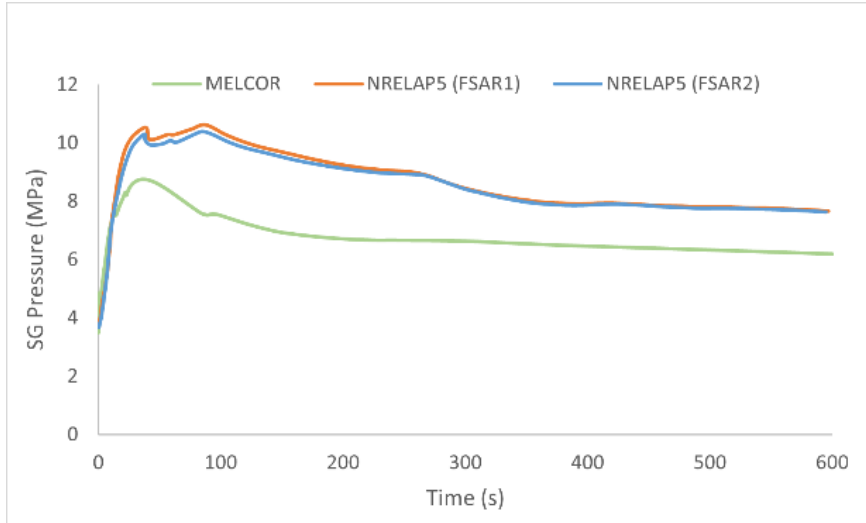


Fig. 8. 2SG pressure

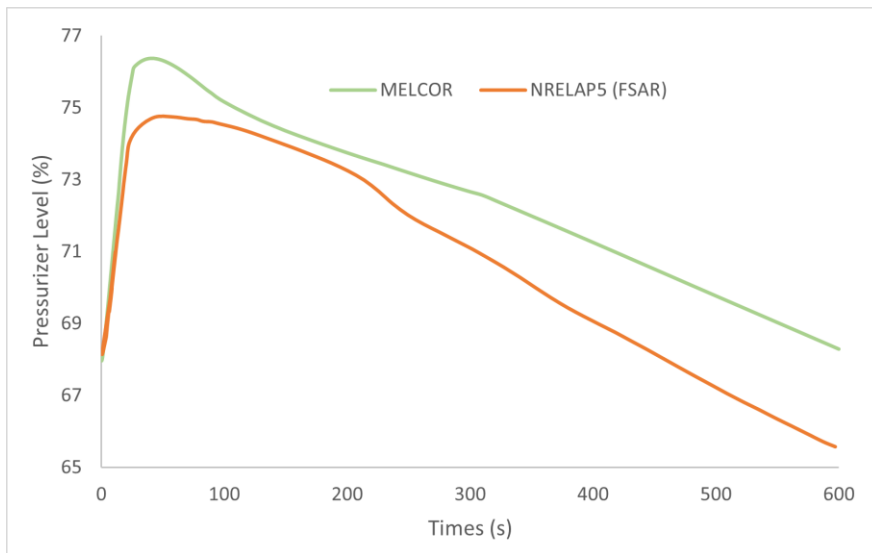


Fig. 9. Pressurizer water level

The secondary circuit SG pressure is shown in Figure 8. The peak pressure in the steam generator tubes occurs at 37s after the turbine trip before gradually settling down to a pressure of about 6.2MPa. Although the FSAR gives a value of about 7.6MPa, they have similar patterns, and the pressures are well within the design pressure of 14.48MPa.

Finally, the pressurizer water level is shown in Figure 9. In both instances, the water level increases from 68% to 76% for the first 44s. By contrast, the water level in the FSAR rose by about 4% before ending at about 2.7% lower at 600s. In both cases, the water level first increases then decreases as the temperature first increases then steadily decreases.

IV. CONCLUSION

This study has presented a preliminary analysis of NuScale's 160MWth SMR by first conducting a steady state run followed by a turbine trip transient using MELCOR. While the input deck lacks proprietary values such as the exact geometric information of the DHRS and heat transfer coefficients, the steady state results generated using publicly available data and reasonable assumptions have managed to align with the FSAR's NRELAP5 results to a satisfactory extent.

The turbine trip simulation results are also comparable with the FSAR values in terms of the trend. Some of the more notable differences include deviations in the DHRS and RCS flow rates. Estimations and assumptions used in lieu of missing geometry data in the FSAR may have contributed to these differences. Work is currently ongoing to fine-tune the model, especially for the DHRS where the discrepancy is largest due to insufficient data on the condensers.

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