The Fuel-Clad Transient Behavior of Uprated NuScale Using MELCOR

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INTRODUCTION

The occurrence of Fukushima-Daiichi accident in 2011, has significantly impacted the nuclear industry and the commissioning of nuclear power plants, hence, upgrading nuclear reactors has become necessary in three aspects: safety, flexibility, and economy. As a result of that, increasing research and development efforts are directed in the deployment of the Small Modular Reactor (SMR)[1]. SMR is meant to provide flexibility in establishing power generation unit, especially, for areas where it is difficult to access or transport fuel due to its lack of good infrastructure, offering economic option due to its modular design and size, featuring passive safety, and Long-life cycle and reduced need for refueling. That being said, many studies[2] have questioned the economic benefit of the promoted SMR plants, and concluded that the overall construction and operational costs per unit of electricity generation capacity would be more expensive compared to the traditional large Light Water Reactor (LWR) power plants. Additionally, Krall, *et al.*[3] concluded that SMRs potentially would have disadvantage in terms of nuclear waste and argued that the nuclear waste disposal would require more complex approaches that are more expensive than the available traditional ones.

Therefore, uprating the operating power of the SMRs is becoming a hot topic, in order to improve the economical aspect of it. Stewart, *et al.*[2] found that the overall cost can be reduced by ~15%, when uprating the power of an SMR by 20%. In fact, NuScale Power LLC submitted a Standard Design Approval to the Nuclear Regulatory Commission (NRC) for uprating the NuScale from 160 MWth to 250 MWth[4]. However, one of the main concerns when it comes to uprating a reactor is how the fuel-clad would behave beyond design-basis accident (BDBA) scenario. Since greater decay heat is required to be dealt with in case of the uprated operating power. To answer this question, this paper aims to explore for the NuScale reactor using preliminary calculations performed with the MELCOR code version 2024v[5].

METHODOLOGY

One of the most mature SMR designs is the NuScale reactor, which started development in 2003 from a collaboration between Oregon State University and Idaho National Engineering Laboratory, and which had its submission for the Design Certification in 2011[6]. NuScale plant is designed to include up to 12 small modules immersed in large water pool, each of the modules has a $160\ MWth$ reactor core housed with other primary system components in an integral reactor pressure vessel and surrounded by a steel containment. NuScale main advantage is its reliance on natural circulation for cooling the reactor core and elimination of pumps in the primary loop. The natural circulation loop can be described as follows; the heated water flow upwards due from the core and bypass to the riser, part of that flow goes to the steam generator and the smaller portion goes to the pressurizer. The steam generator exchange heat and cool the hot water as the water flows downwards through the downcomer (DC) to the lower plenum (LP), which feeds again the core and bypass channels with cold water. The two main passive safety systems in the NuScale SMR are the decay heat removal system (DHRS) and emergency core cooling system (ECCS). The ECCS is designed to reestablish natural circulation through the reactor pressure vessel (RPV) in the event of a loss of coolant accident (LOCA). The ECCS is composed of two Reactor Vent Valves (RVVs) at the top of the RPV and three Reactor Recirculation Valves (RRVs) on the side of the RPV. Modeling NuScale design using MELCOR code is based on the schematic view is presented in Fig. 1.

In the present work, the MELCOR 2024v is chosen to model the NuScale design and its transient scenario. MELCOR[5] is a fully integrated, engineering-level severe accident analysis code. It is developed at Sandia National Laboratories and maintained by the (NRC) as a nuclear plant risk assessment tool, commonly used for accident progression analyses. Moreover, many researchers had successfully simulated severe accidents in LWRs and SMRs with the MELCOR[7], making it well-suited for this study. Allaf *et al.*[8] modelled NuScale using MELCOR code 2.2 to explore the role of Cr-coating Zr cladding during accident progression. Hence, the model presented in this work is a modified version of the one presented in Ref.[8].

MELCOR is composed of a number of different packages, the most fundamental ones to simulate the thermal-hydraulics behavior are 1- Control Volume Hydraulics (CVH), which models the free control volume that is occupied by the fluid, 2- Flow Path (FL), that models the flow direction and its losses due to friction and the 3- Heat

Structures (HS), that models the heat transfer due to conduction and radiation from the core to the ultimate heat sink. The 4- Core (COR) package is utilized to model the core structure and power distribution, which communicate with the other components through the CVH, FL, and HS packages. That together model the major systems of a reactor plant and their generally coupled interactions. In addition to these packages, the Control-Function (CF), the Decay Heat (DCH) and the Radio-Nuclides (RN) packages are implemented for the simulation of the transient behavior.

Fig. 2 describes the control volumes and the flow direction as developed using the CVH and FL packages, whereas, the conduction heat transfer from the core to the ultimate heat sink is modelled using the HS package as described in Fig. 3. The COR package is utilized to model the core structure and power distribution, which communicate with the other components through the CVH, FL, and HS packages. Fig. 4 shows the nodalization approach adopted for this work and the different components considered in the core.

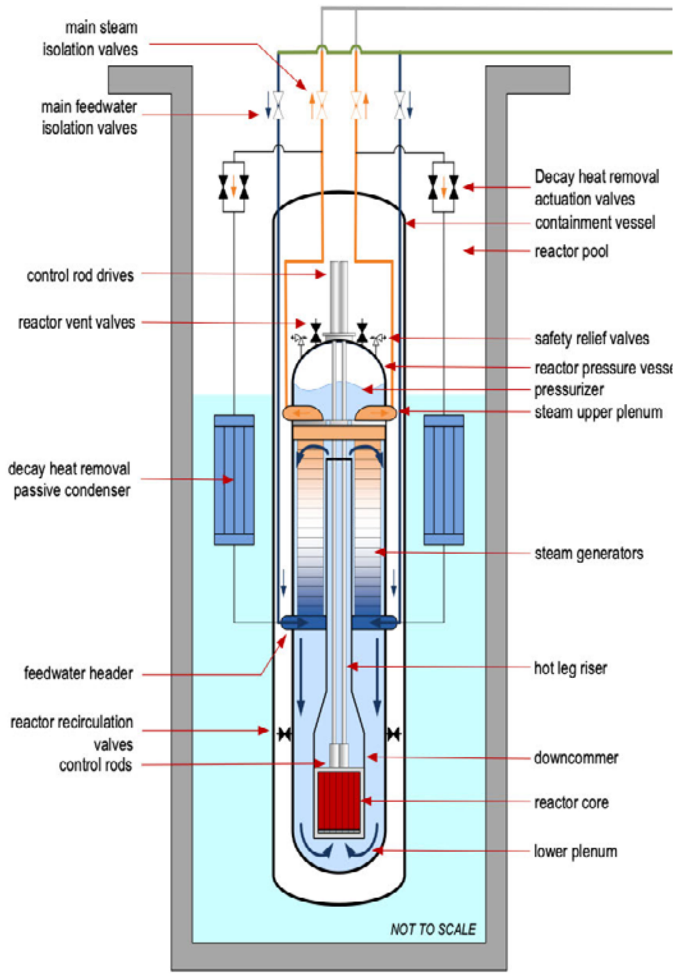

Fig. 1. Conceptual Schematic View of the NuScale Power Module[6].

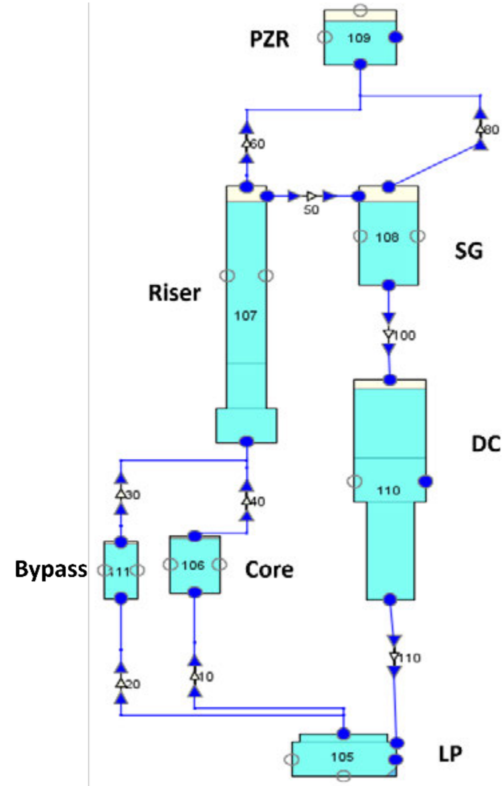

Fig. 2. Representation of the control volume of each component (CVH) and the direction of the flow (FL) inside the reactor pressure vessel, where the core is represented as one control volume[8].

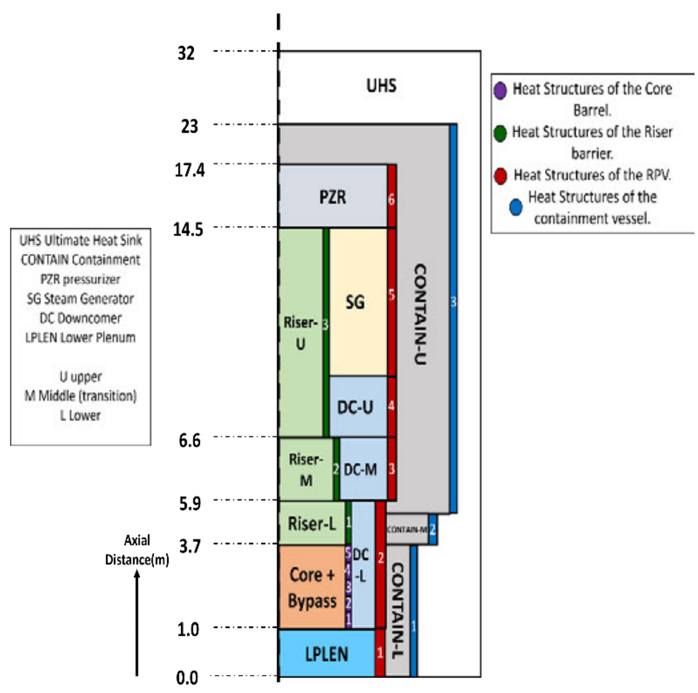

Fig. 3. Symmetric-Axial view of the NuScale model and its different CVH and HS components[8].

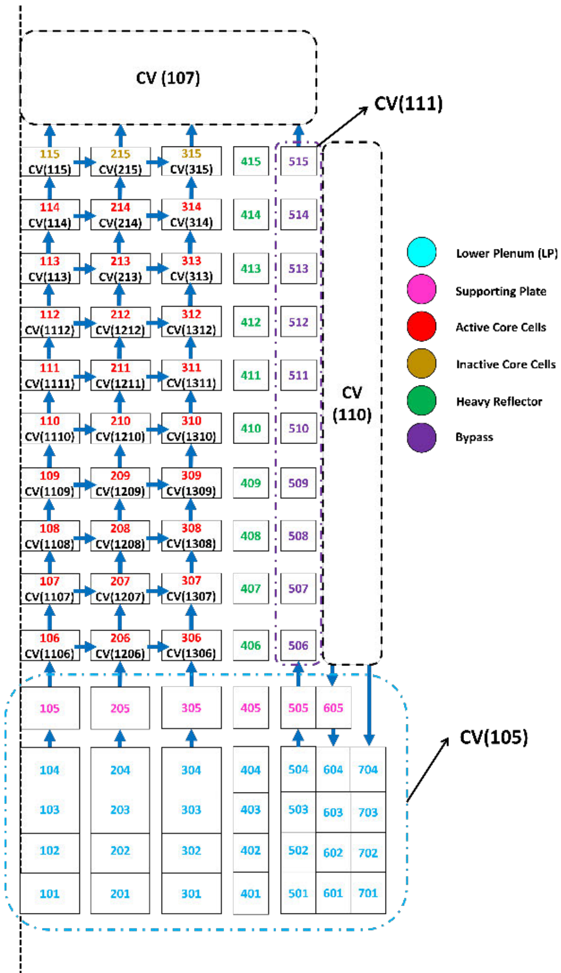

Fig. 4. Nodalization of the NuScale core model using the COR package, besides, the associated control volumes (CV) and the flow paths directions (represented by arrows) are presented.

For the power uprate, the overall system design is expected to be similar for any fuel design based on the NPM-160 MWth system[9]. The model has been verified against the benchmark results presented in the NuScale Final Safety Analysis Report (FSAR)[9], for steady operation and a LOCA scenario, and the VOYGR report[4]. Table 2 lists the compared parameters, where very good agreement is obtained with a relative error less than 2.5%.

Transient Accident Scenario

To investigate the fuel-clad failure progression, the 160 MWth and the 250 MWth are assumed to go through a severe accident scenario. The severe accident in this study is a LOCA into the containment event initiated by the spurious actuation of a single RVV opening, with success of the reactor trip system, whereas the DHRS is assumed unavailable. Besides, the ECCS partially actuates when the level of water reaches the level of RRV, this occurs when the remaining two RVVs open while both RRVs fail to open. This scenario aims to simulate a rapid vapor-space LOCA with an intact containment, and further information related to the timeline events can be found in[9]. Keeping in mind, the assumed decay heat for the 250 MWth is around 1.56 times larger than the 160 MWth.

Table I. Table Name

Power	160 MWth		250 MWth	
Parameter	Ref.[9]	MELCOR model	Ref.[4]	MELCOR model
Mass Flow Rate (kg/s)	587	598.8	Greater	695.9
Pressure (MPa)	12.8	12.7	13.8	13.83
Hot Leg Temp. (K)	583.5	577.5	589	590
Cold Leg Temp. (K)	531	525.8	521	521

RESULTS & DISCUSSION

For comparing the fuel-clad transient behavior for the two NuScale models, the first fuel-clad segment to fail is investigated (note: the failure point is set at 2500 K[5]). The temperature results show that the fuel-clad in case of the 160 MWth is heating up at ~8.5 hours (Fig. 5), whereas the 250 MWth starts heating up earlier at ~4.5 hours (Fig. 6). This is expected due to the higher decay heat generated in case of the 250 MWth, However, the fuel-clad fails at ~12 hours after the initiation of the accident in case of the 160 MWth, whereas in case of the 250 MWth, it fails at ~21.5 hours. Besides, the temperature history in the latter case features increase in temperature, followed by drop/fluctuation in temperature to almost having a steady-behavior, and finally increase in temperature up to failure point.

Based on the conducted preliminary investigation, this significant delay to failure and the remarkable changes in the behavior found to be caused by the natural circulation that is established from the LP through the core; the riser, the DC, back to the LP. This natural circulation is initiated by the evaporation of the water in the LP. Furthermore, when the LP dries out, the fuel-clad found to heat up again leading to failure.

The evaporation of the LP's water also occurs in case of the 160 MWth, but by investigating the vapor velocity from the DC to LP, it can be understood that it takes longer time to accelerate (bottom of Fig. 5). On the other hand, the vapor flow in the 250 MWth case starts accelerating just after the evaporation of the liquid in the LP (Fig. 5). Moreover, by comparing the vapor velocities in the core channel, the 250 MWth case is ~1.5 times more than in the 160 MWth case, as can be seen in Figs. 5 and 6. This indicates that the established convective heat transfer is better in case of the 250 MWth.

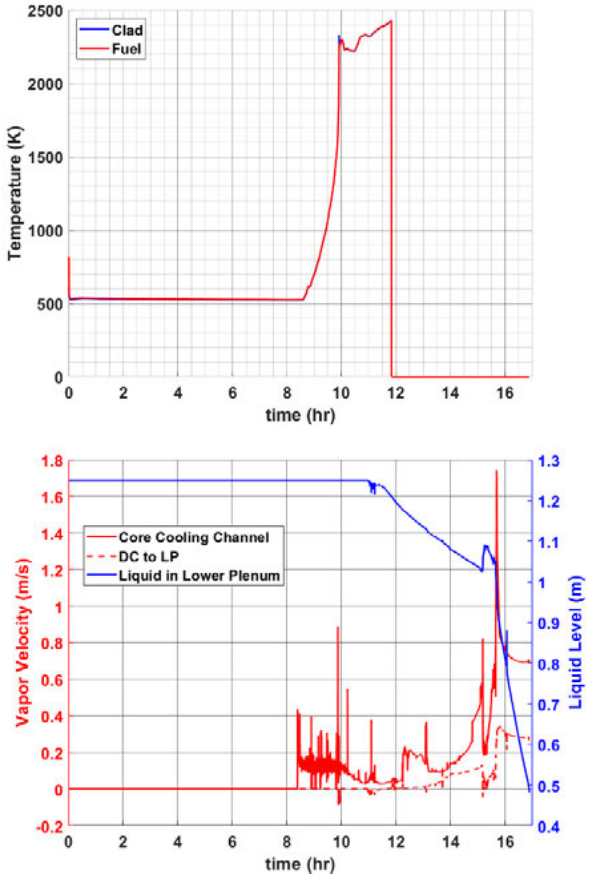

Fig. 5. Transient behavior for the 160 MWth case: Top-figure: Temperature history of the first fuel-clad segment to fail. Bottom-figure: vapor velocity and liquid level change over time.

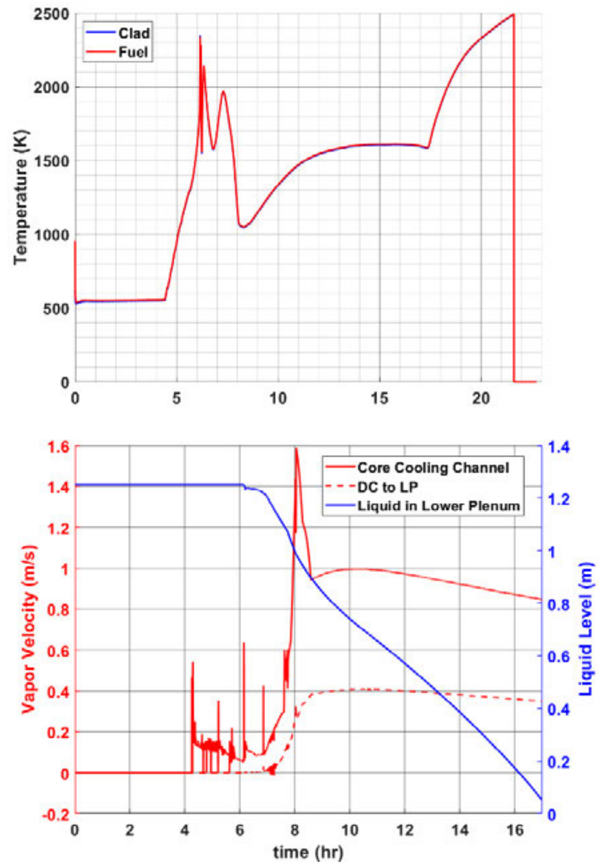

Fig. 6. Transient behavior for the 250 MWth case: Top-figure: Temperature history of the first fuel-clad segment to fail. Bottom-figure: vapor velocity and liquid level change over time.

CONCLUSION

With the increasing interest in uprating SMRs, this research explores the impact of uprating on the transient behavior of the NuScale reactor from 160 MWth to 250 MWth. This is achieved by simulating the transient of a LOCA scenario using the MELCOR 2024v code. The preliminary investigation shows that the first segment of the fuel-clad failure is delayed by almost 10 hours with the 250 MWth. Although the uprate causes increase in the decay heat generated by 1.5 times, the greater decay heat has benefited in enhancing and establishing the natural circulation leading to significant delay for the fuel-clad to fail. Further sensitivity analyses are being conducted to develop more confidence on the reliability of the results, and to have better understanding of the observed phenomenon.

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