Comparison of NRELAP5 to an ORNL THTF Test and the NuScale Critical Heat Flux Tests

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1.0 INTRODUCTION

The purpose of this summary is to illustrate the application of NRELAP5 for modeling prototypical critical heat flux (CHF) tests performed for the NuScale Power, LLC (NuScale) small modular reactor (SMR). This work was done in preparation for the development of a CHF correlation to be used in NRELAP5 and SCANR, a NuScale proprietary subchannel analysis code.

2.0 NUSCALE SMALL MODULAR REACTOR

NuScale has developed a new nuclear plant, which is a smaller, scalable version of pressurized water reactor technology, designed with natural circulation safety features.

The design features a natural circulation reactor coolant system within which the core heats the water causing it to rise through a central hot leg after which it turns in an upper plenum and heat is exchanged to the helical coil steam generator. The lower temperature water then flows down the cold leg downcomer where it reaches the inlet of the core to complete the water flow circuit.

The reactor pressure vessel (RPV) is inside a steel containment within a submerged reactor building cooling pool. The containment serves as a radiation barrier and a coolant flow path in the event of a small-break loss-of-coolant accident (SBLOCA), since large piping is not present in the NuScale design. In the rare event that water escapes the RPV, it 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 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.

3.0 CHF TESTING

As part of its design certification efforts to develop a new, safe and economical SMR, NuScale completed a major test program to obtain CHF data for its nuclear fuel using a full-length, full-power, electrically-heated fuel assembly mock-up with spacer grids. [1] It was tested over a wide range of natural circulation flow rates with both uniform and cosine shape power profiles.

The data is being used to define the limiting conditions for fuel performance and to validate NuScale's safety analysis computer codes. The test results show that the NuScale fuel has a significant safety margin under steady state natural circulation flow conditions.

In order to use the data, a CHF correlation was developed and is being implemented into the NRELAP5 and SCANR subchannel codes.

In order to validate the NRELAP5 code against rod bundle data, the ORNL THTF experimental tests were chosen for validation and as a template for the NuScale CHF test data comparisons. [2]

The Oak Ridge National Laboratory THTF Tests 3.07.9B, 3.07.9N, 3.07.9W were a series of experiments performed in the early 1980s carried out at the ORNL Thermal Hydraulic Test Facility (THTF) to investigate several heat transfer phenomena expected to occur during PWR loss of coolant accidents, including CHF and dispersed film flow boiling. Tests 3.07.9B, 3.07.9N, 3.07.9W are low mass flux runs. These experiments assess the CHF and film boiling heat transfer models for NRELAP5. The test chosen for this comparison was 3.07.9W.

The Stern Laboratories thermal hydraulic critical power experiments performed for NuScale were done using a five-by-five fuel assembly array installed in a vertical fuel channel representative of the NuScale small reactor design. It was designed to perform critical power tests in accordance with uniform and center peaked cosine reactor operating conditions to provide experimental data which is used to develop CHF correlations that cover the NuScale reactor operating envelope.

This paper discusses a group of uniform power tests for low, medium and high flow and power cases from among the test suite and the basic CHF correlation developed from the uniformly heated test data.

4.0 RELAP5 MODELS FOR THE ORNL THTF TESTS

The THTF is a non-nuclear pressurized water loop comprised of a pump, a vertical test section containing a pressurizer, a set of electrically heated rods, and a heat exchanger.

The test section contains sixty-four rods in an eight-by-eight bundle with a heated length of 3.66 m. Sixty of the rods are electrically heated and four rods are unheated with the geometry shown in Figure 1. Once the inlet flow was established, the power to the rods was increased until the dryout point reached a desired elevation in the test section. The test was run until the operating pressure and rod surface temperatures stabilized. The operating and boundary conditions for the 3.07.9W test used in the model are given in Table 1. The NRELAP5 model is shown in Figure 2.

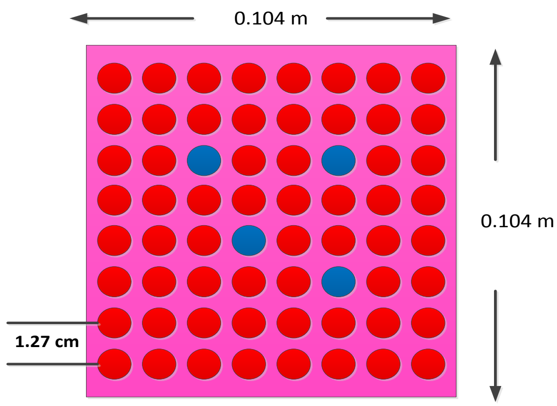

Fig. 1. ORNL bundle geometry

The NRELAP5 model consists of twenty four volumes with time dependent boundary cells at the inlet and outlet, along with a liquid mass flow boundary junction at the inlet. Since this is rod bundle geometry, certain prescriptions need to be followed to calculate the hydraulic and the heated diameters for such geometries. In addition, bundle interfacial drag models were activated at the internal junctions for this problem.

Given the pitch and the fuel diameter, the flow area for the bundle geometry can be calculated. The hydraulic perimeter for the fuel rod and the hydraulic diameter for the bundle can then be computed as well.

TABLE 1. Initial and Boundary Conditions

Test	Pressure (MPa)	Inlet Temperature (K)	Mass Flow (Kg/s)	Power (kW)
3.07.9W	12.0	567.0	1.58	2490.5

The cylindrical heat structures were modeled with the full convection heat transfer package, using the Groeneveld CHF default option in the code. [3] The standard materials and geometry used in the heated rods are shown in Figure 3. They include boron nitride (BN), Inconel and stainless steel 316 (SS316).

The problem was executed with the semi-implicit time integration scheme using a maximum time step of 0.05 seconds for fifty seconds.

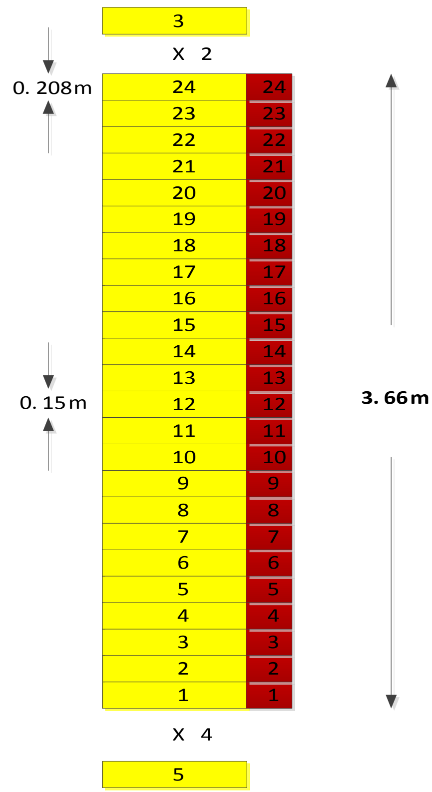

Fig. 2. NRELAP5 ORNL model

Figure 4 shows the results for the rod wall temperature for the 3.07.9 W test. The code calculated CHF point was lower by about 0.4 (m) than the experimental value. The comparisons between NRELAP5 and the test show acceptable agreement. Sensitivity studies were performed by increasing the number of cells with no change in results.

5.0 STERN LAB CHF TESTS

The Stern Lab CHF tests were prototypical experiments for the NuScale core performed over a wide range of conditions to form the basis for CHF correlations.

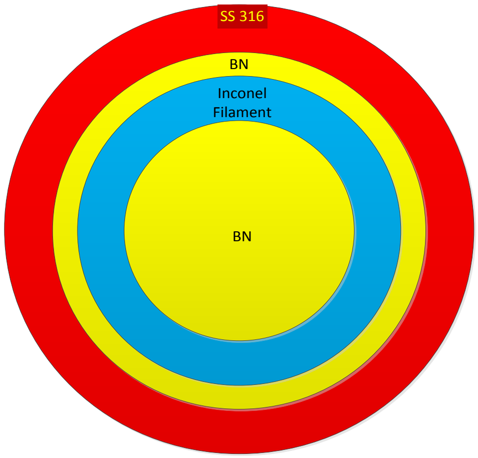

Fig. 3. Rod materials and geometry

The present correlation is proprietary for use in the NuScale core subchannel code SCANR and systems thermal hydraulics code NRELAP5.

The present NuScale CHF correlation has been formulated similarly to reference with additional unique features. [4] The correlation uses the Kelvin-Helmholtz wavelength, mass flux and sublayer thickness. The correlation at the present time is global in nature.

5.1 Stern Lab CHF NRELAP5 Model

For the model comparisons with NRELAP5, the THTF model was used as a template with modifications for the geometry, initial and boundary conditions to match those of the Stern Lab tests.

The geometry and material layout of the rods used for the Stern tests are similar to those of Figure 1 and Figure 3. However, the prototypical scaling and size are different along with the composition blend.

5.2 Results

Figure 5 illustrates the one-dimensional NRELAP5 model geometry used for the Stern calculations, very similar to that of the THTF test.

Since the test procedure was to ramp the power slowly to avoid the burnout of heater elements over longer periods of transient time to steady state, the same procedure was followed during the simulation. A time step of 0.05-0.1 seconds was used during 3000.0-4000.0 seconds of transient to steady state using the semi-implicit integration scheme.

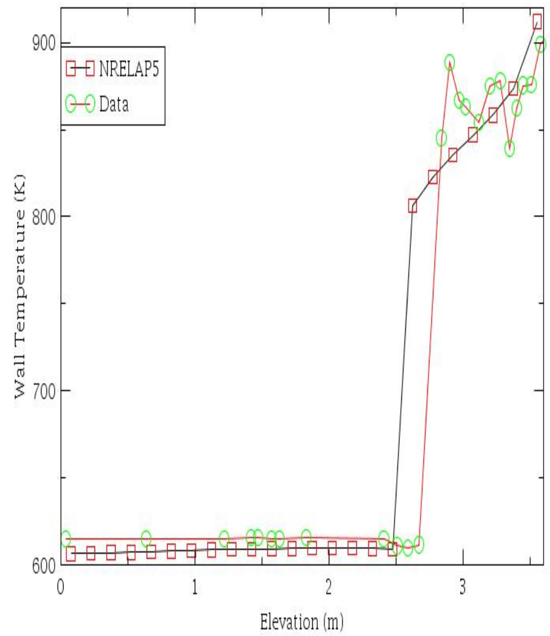

Fig. 4. THTF 3.07.9W rod wall temperature vs. elevation

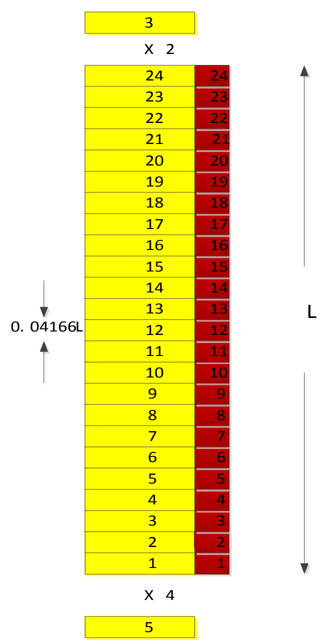

Fig. 5. Stern Lab NRELAP5 model

Table 2 shows the normalized values for the inlet pressure, temperature, and mass flow and the bundle power with respect to the maximums (M) obtained during the testing. As an example, pressures can range from 400.0-11000.0 MPa and temperatures from 363.15-603.15 Celsius for boundary conditions. It also shows the bundle power and CHF ratios, along with the NRELAP5 (N5) error predictions.

TABLE 2. Correlation Comparison to Test Data

Case	374	379	389	252	150	375	140	435
Pi/M	.13	.13	.13	1.0	.63	.13	.62	.25
Ti/M	.59	.59	.53	.94	.89	.59	.89	.64
Wi/M	.34	.59	.34	.45	.25	.44	.25	.27
Pwr/M	.43	.74	.45	.32	.32	.49	.30	.36
CHF/M	.44	.74	.46	.32	.33	.50	.32	.36
N5 % Error	13.	13.	12.	5.	7.	12.	18.	33.

6.0 CONCLUSIONS

The results from the comparisons of both models to the data indicate that:

1. The model has been implemented correctly.
2. The preliminary CHF correlation has expected results and trends. Higher flows and powers yield better results; lower flows and higher powers have higher errors. This is due to ~10% full scale instrumentation error at low flows. Further work is needed to examine better instrumentation at low flows.
3. At higher powers and lower flows the sublayer will separate. Further work is needed to characterize these heat transfer and flow regimes.

Future studies will examine the effects of grid spacer loss coefficients on ORNL tests and additional sets of data.

Additional work will also focus on local CHF correlations and their implementation into NRELAP5.

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NOMENCLATURE

CHF = critical heat flux (Kw)
D = diameter
H = hydraulic
Max = maximum value from test suite
N5 = NRELAP5
p = pitch
P = hydraulic (or wetted) perimeter
Pi = Inlet pressure (Pa)
Pwr = bundle power (Kw)
Ti = temperature (K)
Wi = inlet mass flow (Kg/s)
MPa = MegaPascals

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