September 17, 2018 Docket No. PROJ0769

U.S. Nuclear Regulatory Commission
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One White Flint North
11555 Rockville Pike
Rockville, MD 20852-2738

SUBJECT: NuScale Power, LLC Submittal of Changes to “Loss-of-Coolant Accident Evaluation Model Topical Report”, TR-0516-49422

REFERENCES

1. NuScale Power, LLC, “Response to NRC Request for Additional Information No. 15.00.02-15 (eRAI 9513) on the NuScale Design Certification Application,” dated August 28, 2018 (ML18240A379)
2. NuScale Topical Report, “Non-Loss-of-Coolant Accident Methodology,” TR-0516-49416, Revision 1, dated August 10, 2017 (ML17222A827)
3. NuScale Topical Report, “Loss-of-Coolant Accident Evaluation Model,” TR-0516-49422, Revision 0, dated December 30, 2016 (ML17004A122)

On August 28, 2018 NuScale submitted a response to RAI 15.00.02-15 from eRAI No. 9513 (Reference 1) which included changes to the Non-Loss-of-Coolant Accident Methodology topical report (Reference 2). This letter submits corresponding changes to the Loss-of-Coolant Accident Evaluation Model topical report (Reference 3). The Enclosure to this letter provides the report pages incorporating revisions to the Loss-of-Coolant Accident Evaluation Model topical report, in redline/strikeout format. NuScale will include this change as part of a future revision to the NuScale Loss-of-Coolant Accident Evaluation Model topical report.

Enclosure 1 is the nonproprietary version of the Loss-of-Coolant Accident Evaluation Model topical report revision. No changes were made to the proprietary information on the revised pages.

This letter makes no regulatory commitments or revisions to any existing regulatory commitments.

If you have any questions, please feel free to contact Paul Infanger at 541-452-7351 or at pinfanger@nuscalepower.com.

Sincerely,

Zackary W. Rad
Director, Regulatory Affairs
NuScale Power, LLC

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Enclosure 1: "Changes to Loss-of-Coolant Accident Evaluation Model, TR-0516-49422," nonproprietary version

Enclosure 1:

"Changes to Loss-of-Coolant Accident Evaluation Model, TR-0516-49422," nonproprietary version

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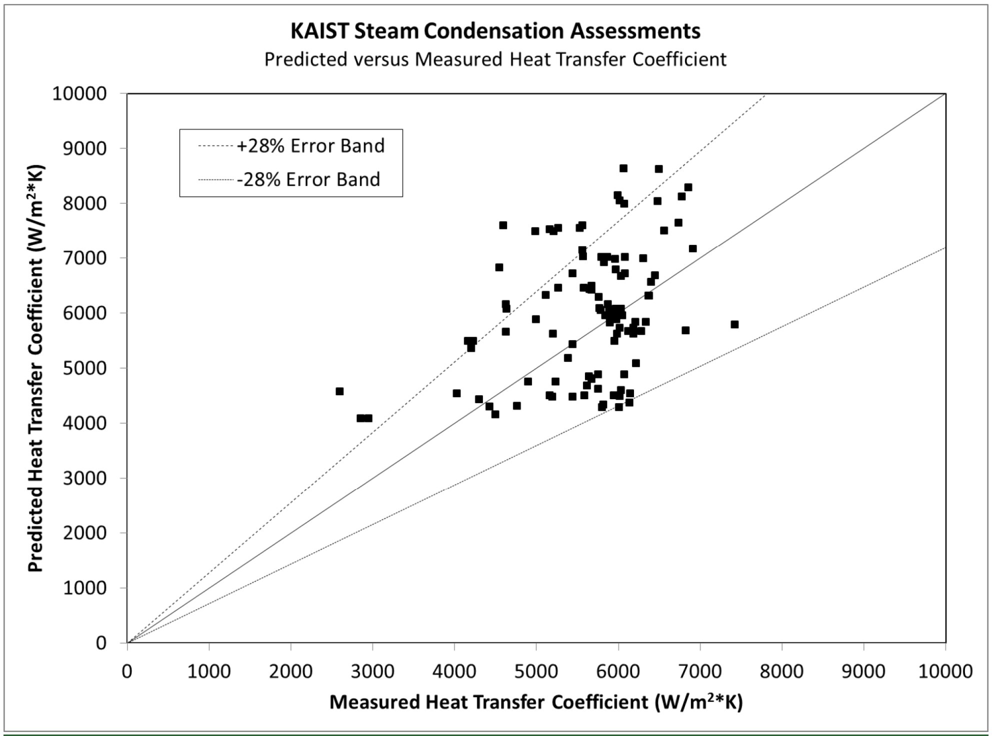

Figure 7-16. Measured versus predicted heat transfer coefficient

Figure 7-17 through Figure 7-19 present heat transfer coefficient, temperature and mass flow rate versus test section elevation. The majority of the predicted values (all but one) lie within the uncertainty range of the data.

Overall, the results show that NRELAP5 calculations are in excellent agreement with the KAIST measured experimental data. These results validate NRELAP5 for prediction of key thermal-hydraulic phenomena associated {{

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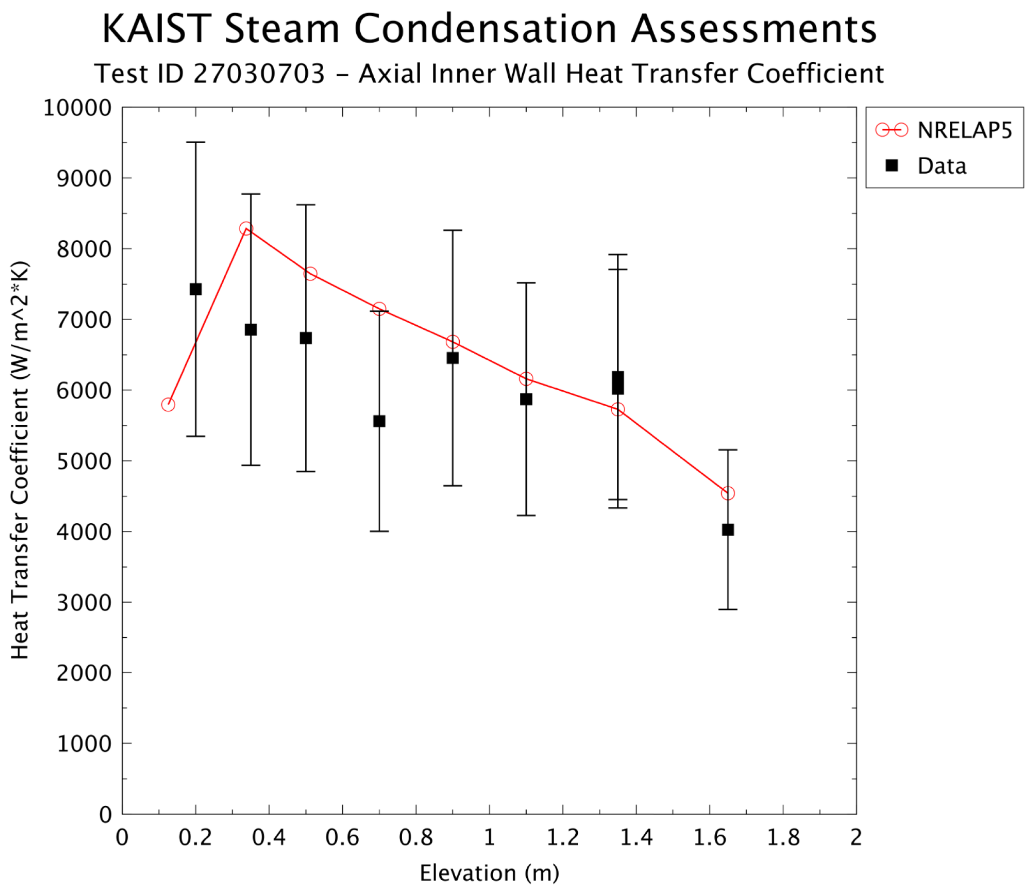

Figure 7-17. KAIST and NRELAP5 axial heat transfer coefficient

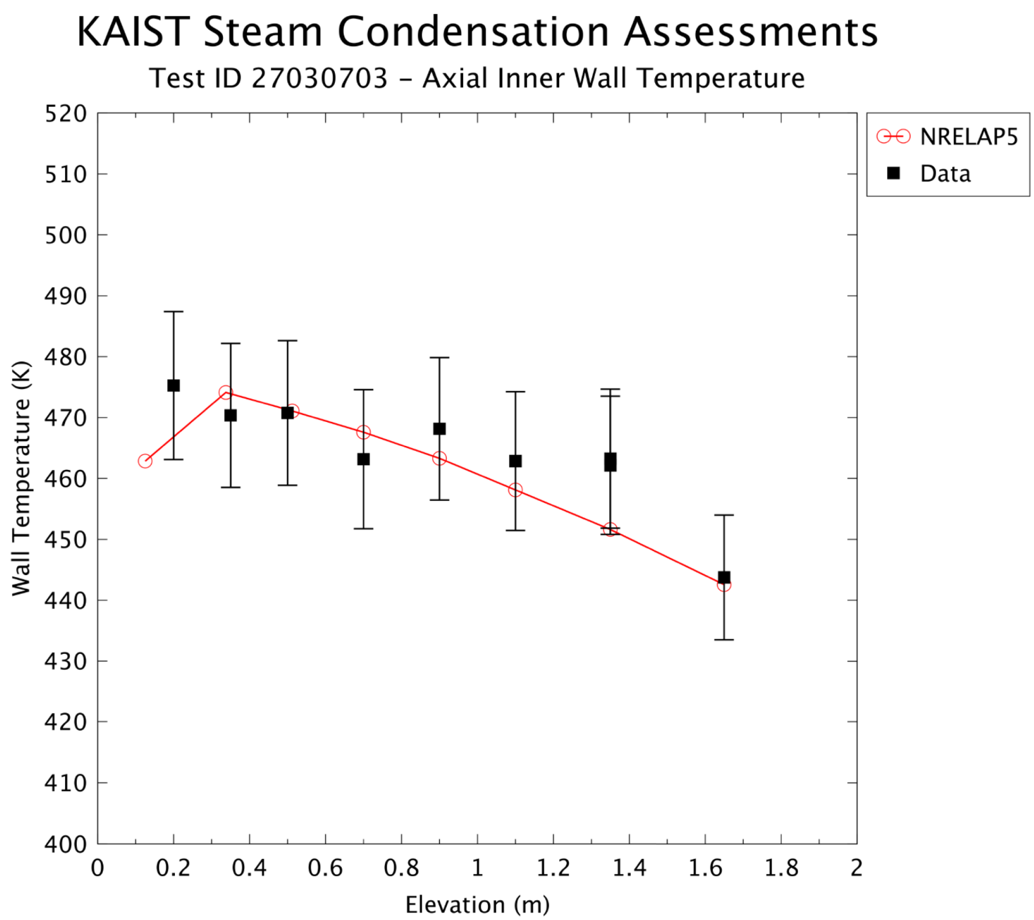

Figure 7-18. KAIST and NRELAP5 axial inner wall temperature

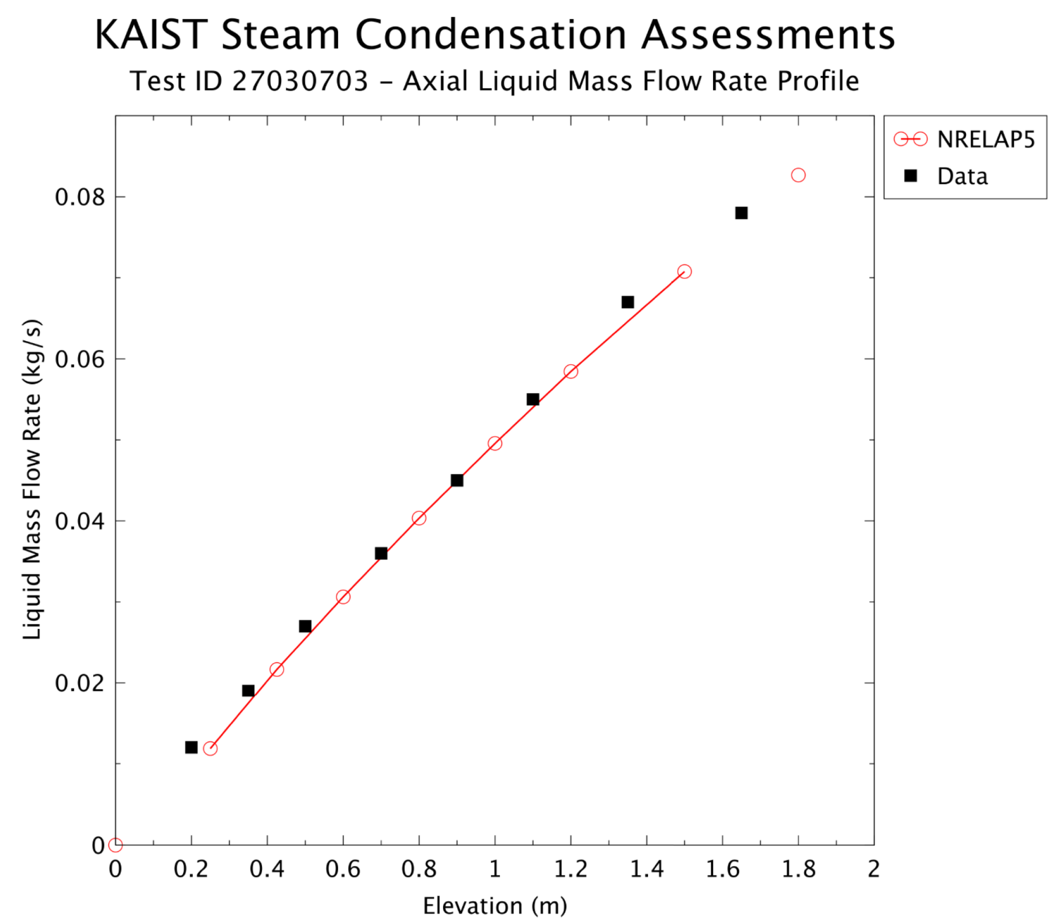

Figure 7-19. KAIST and NRELAP5 axial liquid mass flow rate

7.2.5 FRIGG

The FRIGG loop tests for the Marviken boiling heavy water reactor project were executed in four phases by ASEA-ATOM during the years 1967-1970 (Reference 61). These experiments included measurements of axial and radial void distribution, single-phase and two-phase pressure drop, natural circulation mass velocity, stability limits as well as detailed dynamic characteristics, and burnout in natural and forced circulation.

The axial and radial void distribution data as a function of mass flow, inlet sub-cooling, system pressure, and thermal power provide an excellent data set for evaluating the NRELAP5 interphase drag and heat transfer models under two-phase flow conditions. The FRIGG phase 4 (FRIGG-4) tests applied both a non-uniform radial and axial thermal power profile on the heated rod bundle best simulating the power profiles associated with a typical operating reactor core. As such the FRIGG-4 tests are used to assess NRELAP5 performance.