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RELAP5/MOD3.2 Assessment Using INSC SP-V7
"VVER Core Heat Transfer"

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ABSTRACT

Assessments of the RELAP5/MOD3.2 computer code using critical heat flux data from three sets of experiments have been performed independently by analysts at the Electrogorsk Research and Engineering Center and the Idaho National Engineering and Environmental Laboratory. The experiments, performed at the KS-1 and V-200 facilities, investigated dryout at the top of rod bundles with geometry typical of VVER reactors. The two assessments were compared, investigating differences in the input models and explaining the resultant differences in the calculations. The differences between the two sets of calculations were generally much smaller than the differences between the calculations and the data. Both assessments found that the code calculations were in minimal agreement with the data, and recommended the development of a more applicable critical heat flux model for the code.

1 INTRODUCTION

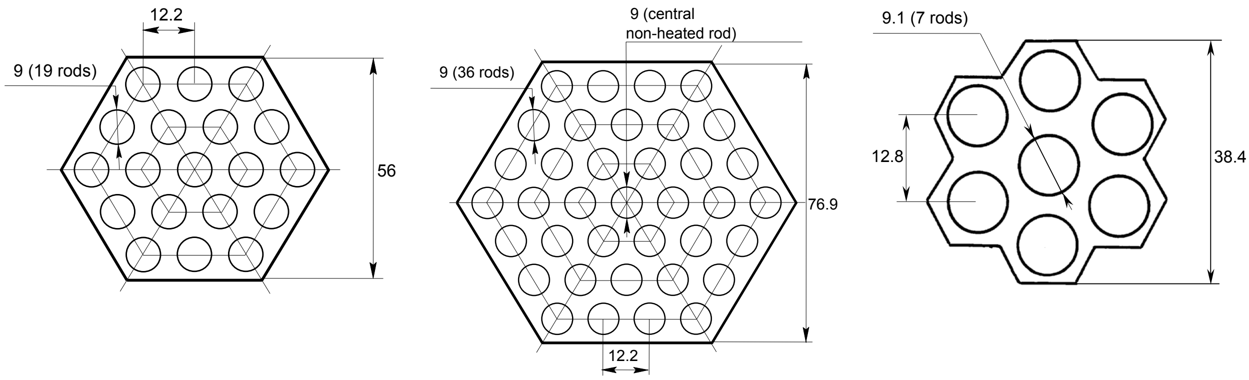
Calculations of VVER Standard Problem INSCSP-V7 [1] have been performed independently by analysts at the Electrogorsk Research and Engineering Center (EREC) and the Idaho National Engineering and Environmental Laboratory (INEEL). This standard problem represents rod bundle critical heat flux (CHF) data from three sets of experiments, two at the KS-1 facility at the Russian Research Center - Kurchatov Institute, and one set from the V-200 facility at the Institute of Physics and Power Engineering, Russia. These assessments aid in assessing the applicability of RELAP5/MOD3.2 for analyzing transients in VVER type reactors.

2 TEST AND FACILITY DESCRIPTIONS

The KS-1 is a semi-integral single loop model of the VVER primary system. The fuel assembly model for the core consists of full height electrically heated rods. Forced or natural circulation flow of the coolant can be modeled. For the standard problem INSC SP-V7, tests with 19- and 37-rod bundle were selected. These rod bundles are referred to as 3/- and 4/-, respectively. Their cross sections are shown in Figures 1a and 1b. Both bundles had heater rods with a 2.5-m heated length. The rods were made of stainless steel tubes with an outer diameter of 9.0 mm. The heater rod pitch was 12.2 mm. In the heated zone, there were 10 grid spacers. The grid spacer geometry is the same as that used in VVER-440 reactors. Both bundles had uniform axial and radial power profiles, and were enclosed in hexagonal working channels.

The V-200 test facility is a high-pressure, forced circulation circuit. It includes several loops with replaceable experimental working sections. The fuel assembly model for the core consists of seven reduced height electrically heated rods. The rods were made of stainless steel tubes with an outer diameter of 9.1 mm. The heater rod pitch was 12.8 mm. The heated length of the rod bundle was 800 mm. The rod bundle had three grid spacers. The grid spacer geometry is the same as that used in VVER reactors. This bundle had uniform axial and radial power profiles. The V-200 rod bundle was situated in the working channel with a complex hexagonal/triangular geometry. The cross section of the rod bundle is presented in Figure 1c.

The data selected for this standard problem represent a wide range of pressure (0.8–7.0 MPa), mass flux (210-2852 kg/m^{2}s), and inlet subcooling (0-211 K).



a) b) c)

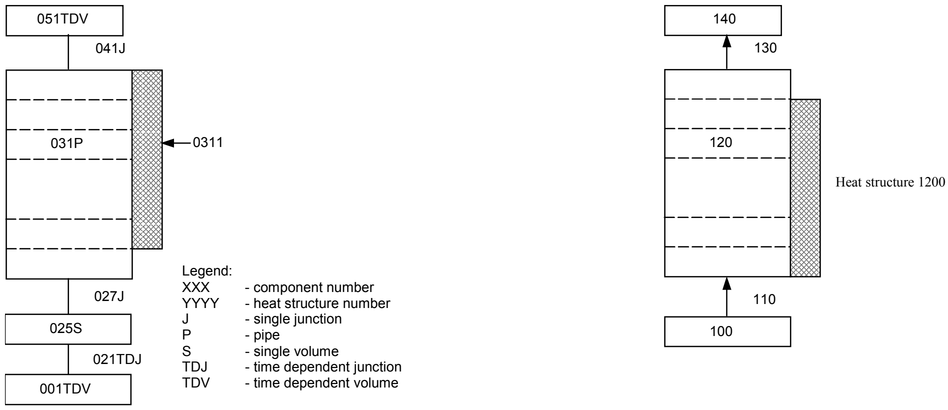
Figure 1: Rod bundle cross sections

The objective of all the tests was to measure parameters at the onset of the heater rod temperature excursion following CHF. Individual tests were started with a steady state condition at the desired pressure. The CHF was approached either by holding the flow rate constant and increasing the power, or by holding the power constant and reducing the flow rate or increasing the coolant inlet temperature.

3 RELAP5 INPUT MODEL DESCRIPTIONS

The general nodalization scheme for the EREC rod bundle input models is shown in Figure 2a [2]. The input model consists of seven hydrodynamic components and one heat structure. The bundle inlet temperature and flow rate are established by the inlet time dependent volume and junction, respectively. A lower plenum volume is used to check the inlet pressure. The rod bundle is simulated with a pipe component, with an attached heat structure modeling the heater rods. In all of the input decks, there was a one-to-one correspondence in the axial nodalization of the rod bundle hydrodynamic cells and the heat structure. At the top of the bundle, a single junction connected the pipe outlet to a time dependent volume, which established the pressure boundary condition. Because the data provided were for net heat flux, the outer bundle wall was not modeled. The number of axial nodes in the pipe modeling the bundle varied between the experiment sets. Loss coefficients of 0.26 were used for the grid spacers. The heater rods were modeled as cylindrical heat structures. The vertical rod bundle without crossflow heat transfer package was used. Default code options were used in most cases.

Similar input decks were used for the INEEL analysis of all three sets of experiments. The basic nodalization is shown in Figure 2b [3]. A time dependent volume and junction at the bottom of the bundle established the inlet flow and temperature. The single junction and time dependent volume at the top of the bundle established the pressure. The bundle region was modeled with a pipe, with the number of volumes changing for each facility. A heat structure was used to model the heater rods, with a one-for-one axial nodalization correspondence with the hydraulic volumes over the heated length, with an unheated volume at the top of the bundle. The outer wall of the bundle was not included in the base nodalization because the data provided were for the net heat flux. Grid spacer loss coefficients of 0.5 for KS-1 models and 0.1 for V-200 model were used in the heat structures only. The heater rods were modeled as cylindrical heat structures. The vertical rod bundle without crossflow heat transfer package was used. Default code options were used for nearly all of the junctions and volumes.



a) b)

Figure 2: General nodalization schemes

4 RESULTS

The test series with the KS-1 37-rod bundle 4/- included 63 data points at varying pressures, temperatures, and flow rates. A comparison of the base case calculated results of EREC and INEEL for this bundle is presented in Figure 3. This figure shows deviations between the measured CHF and values calculated by EREC and INEEL for all data points of the bundle 4/- test series. The deviations are defined as:

$$Deviation = \frac{CHF_{measured} - CHF_{calculated}}{CHF_{measured}} \times 100\% \tag{1}$$

All of the measured CHF values were overpredicted by the code in both teams' calculations. The predictions were generally worse at low pressure and higher mass flux, with no apparent dependence on the inlet water temperature or subcooling. The pressure was a more dominant influence than the mass flow.

The test series with the KS-1 19-rod bundle 3/- included 54 data points at varying pressures, temperatures, and flow rates. A comparison of the base case calculated results of EREC and INEEL for this bundle is presented in Figure 4. All of the measured CHF values were overpredicted by the code in both teams' calculations. Both teams found that the deviation depends on the pressure. The larger deviations were at lower pressures. There were no apparent dependencies on the mass flux and water subcooling.

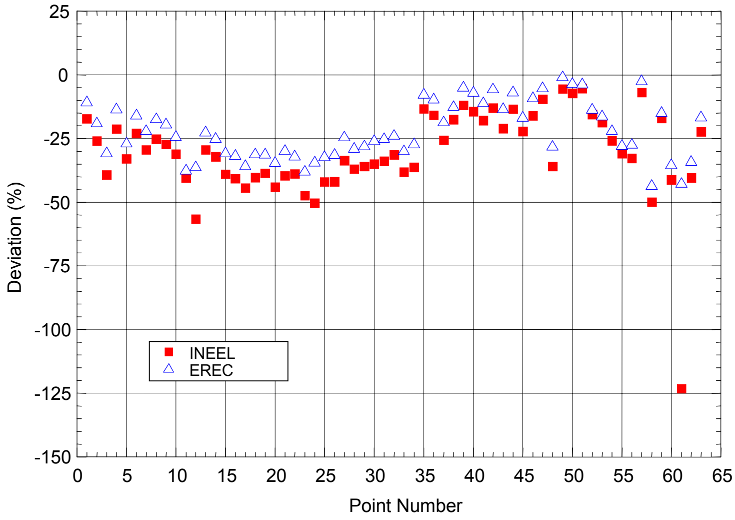


Figure 3: Comparison of the calculated results for KS-1 37-rod bundle 4/

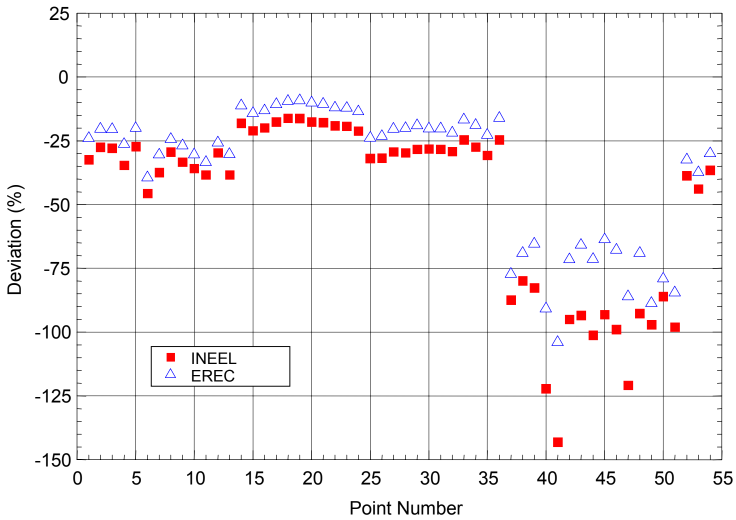


Figure 4: Comparison of the calculated results for KS-1 19-rod bundle 3/

The test series with the V-200 7-rod bundle included 25 data points at the same pressure, with varying temperatures and flow rates. A comparison of the base case calculated results of EREC and INEEL for this test series is presented in Figure 5. All of the measured CHF values were overpredicted by the code in the INEEL base case calculations. In the EREC base case calculations, three points were slightly underpredicted; the others were overpredicted. In both teams' calculations there was a noticeable effect of mass flux on the predicted CHF, with the overprediction being greater at higher mass fluxes.

All CHF values calculated by INEEL had larger deviations than those of EREC.

5 DISCUSSION OF DIFFERENCES

Two teams have independently developed input decks to analyze this standard problem. As a result there are differences between the RELAP5 models, which lead to differences in the calculated results.

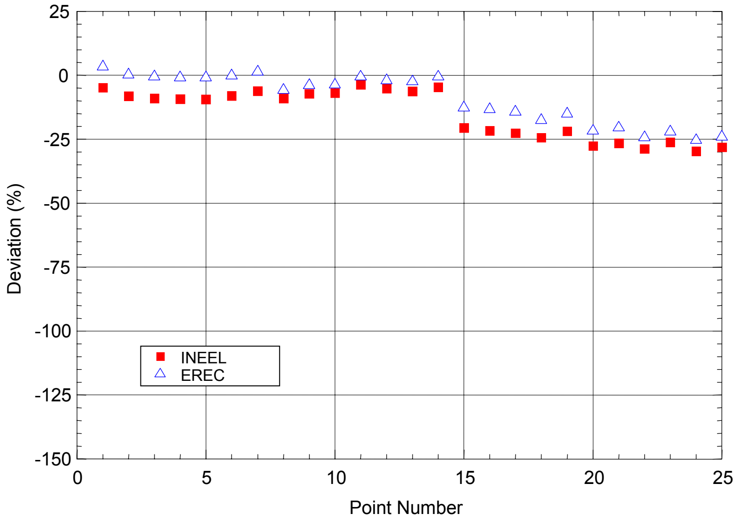


Figure 5: Comparison of the calculated results for V-200 7-rod bundle

Sensitivity calculations were performed to determine the principal contributors to the differences in the calculated CHF. The approach was to change the parameters in each input deck to the value used in the other input deck. With all of the modeling changes included, the sensitivity calculations were very close to the other model's base calculation. This shows that there are not other modeling differences that are unaccounted for.

For the KS-1 experiment series, differences in the axial nodalization of the working channel, the grid loss coefficient value in the heat structure, and the heated equivalent diameter value all contributed about equally to the differences in the calculations. For the V-200 experiments, most of the differences in the predicted CHF were caused by the heated equivalent diameter and the elevation of the middle grid spacer (the closest one upstream of the CHF location).

6 CONCLUSIONS

The differences between the two teams' calculations were generally small compared to the deviations between the calculated and measured critical heat fluxes.

Overall, the agreement between RELAP5/MOD3.2 calculations and the experiment results is judged to be minimal. The CHF was overpredicted in nearly every case. For the V-200 test series at low mass flux, the calculations were within the uncertainty band of the data, which is reasonable or excellent agreement. However, the calculations were outside the stated uncertainty for most of the other tests. There was no evidence of a bundle size effect. The code CHF predictions were generally worse at low pressure.

These assessments indicate that a CHF model more applicable to the VVER core geometry is needed in the RELAP5/MOD3.2 code, because the user cannot work around this problem by changing the facility input model.

REFERENCES

[1] L. L. Kobzar, R. S. Pomet'ko, and M. V. Davydov, "Standard Problem INSCSP-V7 Definition Report: VVER Core Heat Transfer" (Deliverable V7-1 of Work Order 984216426 under Joint Project 6 with the U. S. International Nuclear Safety Center), International Nuclear Safety Center of Russian Minatom, 2000.

[2] M. V. Davydov and O. I. Melikhov, “Final Analysis Report for Standard Problem INSCSP-V7: VVER Core Heat Transfer” (Deliverable V7-3 of Work Order 984216426 under Joint Project 6 with the U. S. International Nuclear Safety Center), International Nuclear Safety Center of Russian Minatom, 2001.

[3] Paul D. Bayless, “RELAP5/MOD3.2 Assessment Using CHF Data from the KS-1 and V-200 Experiment Facilities”, INEEL/EXT-01-00782, Idaho National Engineering and Environmental Laboratory, 2001.