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# PRETEST PREDICTIONS FOR DEGRADED SHUTDOWN HEAT-REMOVAL TESTS IN THORS-SHRS ASSEMBLY 1*


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## SUMMARY

The recent modification of the Thermal-Hydraulic Out-of-Reactor Safety (THORS) facility at ORNL [1] will allow testing of parallel simulated fuel assemblies under natural-convection and low-flow forced-convection conditions similar to those that might occur during a partial failure of the Shutdown Heat Removal System (SHRS) of an LMFBR. An extensive test program has been prepared and testing will be started in September 1983. THORS-SHRS Assembly 1 consists of two 19-pin bundles in parallel with a third leg serving as a bypass line and containing a sodium-to-sodium intermediate heat exchanger. Testing at low powers will help indicate the maximum amount of heat that can be removed from the reactor core during conditions of degraded shutdown heat removal. The thermal-hydraulic behavior of the test bundles will be characterized for single-phase and two-phase conditions up to dryout. The influence of interassembly flow redistribution including transients from forced- to natural-convection conditions will be investigated during testing. A degradation of the SHRS system of a reactor could lead to an elevated inlet temperature, and so several tests will be performed with inlet temperatures up to 538$^{o}$C.

In order to provide guidance in conducting this test program, computer codes have been utilized to perform pretest calculations. Predictions have been performed with 1) THORAX [2], a 2-D bundle code, 2) LOOP-1 [3], a 1-D loop

code, and 3) LOOP-TH, a combination of these with 2-D bundle modeling and 1-D loop modeling. Code results reported here are for several forced- and free-convection tests in which the power will be gradually increased until boiling occurs and then increased further until dryout occurs. At a nominal inlet temperature of 354$^{o}$C, the best estimate of the power to achieve free-convection boiling is in the range between 3.5 and 4.7 kW/pin (cross-hatched region in Figure 1). For an elevated inlet temperature of 538$^{o}$C, free-convection boiling is predicted to occur in the range between 1.9 and 2.6 kW/pin (lightly shaded region in Figure 1). These predictions indicate an inherent safety margin because the lower end of this second power range is the maximum decay heat power of approximately 7% of the nominal average reactor power of 28 kW/pin.

The bundle power required to achieve boiling is predicted to be approximately a linear function of the test section inlet flow for forced-convection flow up to approximately 1 L/s. The heavy solid and dashed lines in Figure 1 show the power required for boiling as a function of the test section inlet flow for the nominal and elevated inlet temperatures of 354$^{o}$C and 538$^{o}$C. The light solid lines show the augmentation of isothermal flows of 0.38 L/s and 0.88 L/s for gradually increasing power up to boiling. Dryout occurs at a slightly higher power represented by the width of the heavy lines, and the flow is shown to drop to zero. Calculations for tests in which parallel bundles are operated give very similar results to single bundle tests for the powers required for boiling and dryout (Figure 2). For both forced- and free-convection tests in which the power is gradually increased, dryout is estimated to occur at a power approximately 0.1 to 0.4 kW/pin higher than that which produces boiling. The extent of this power range will be thoroughly investi-

In conclusion, degraded shutdown heat removal tests will be performed at ORNL in the THORS-SHRS Assembly 1 sodium loop which contains two 19-pin bundles in parallel. Pretest predictions with loop and bundle codes indicate the power required for boiling inception and dryout at nominal and elevated temperatures for natural- and forced-convection conditions. The results of the pretest predictions indicate that the decay heat can be removed from a reactor core during a partial failure of the shutdown heat removal system.

## REFERENCES


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2. J. F. Dearing and S. D. Rose, "Two-Dimensional Modeling of Sodium Boiling in the W-1 Sodium Loop Safety Facility Experiment," *Trans. Amer. Nucl. Soc.*, 39, 1067 (1981)

3. J. J. Carbajo, "Comparison of One- and Two-Dimensional Sodium Boiling Models," *Trans. Amer. Nucl. Soc.*, 44, 319 (1983)

## Figure Captions

Fig. 1. Graphical summary of SHRS Assembly 1 pretest predictions of heater pin power to achieve boiling vs test section inlet flow for free- and forced-flow conditions at nominal and elevated inlet temperatures. (Dryout is estimated to occur at a power approximately 0.1 to 0.4 kW/pin higher than that which produces boiling for tests in which the power is gradually increased to boiling and then dryout.)

Fig. 2. LOOP-TH predictions of bundle inlet flow for a natural-convection single and parallel bundle tests with increasing power to dryout in SHRS Assembly 1 at an elevated inlet temperature of 538°C.

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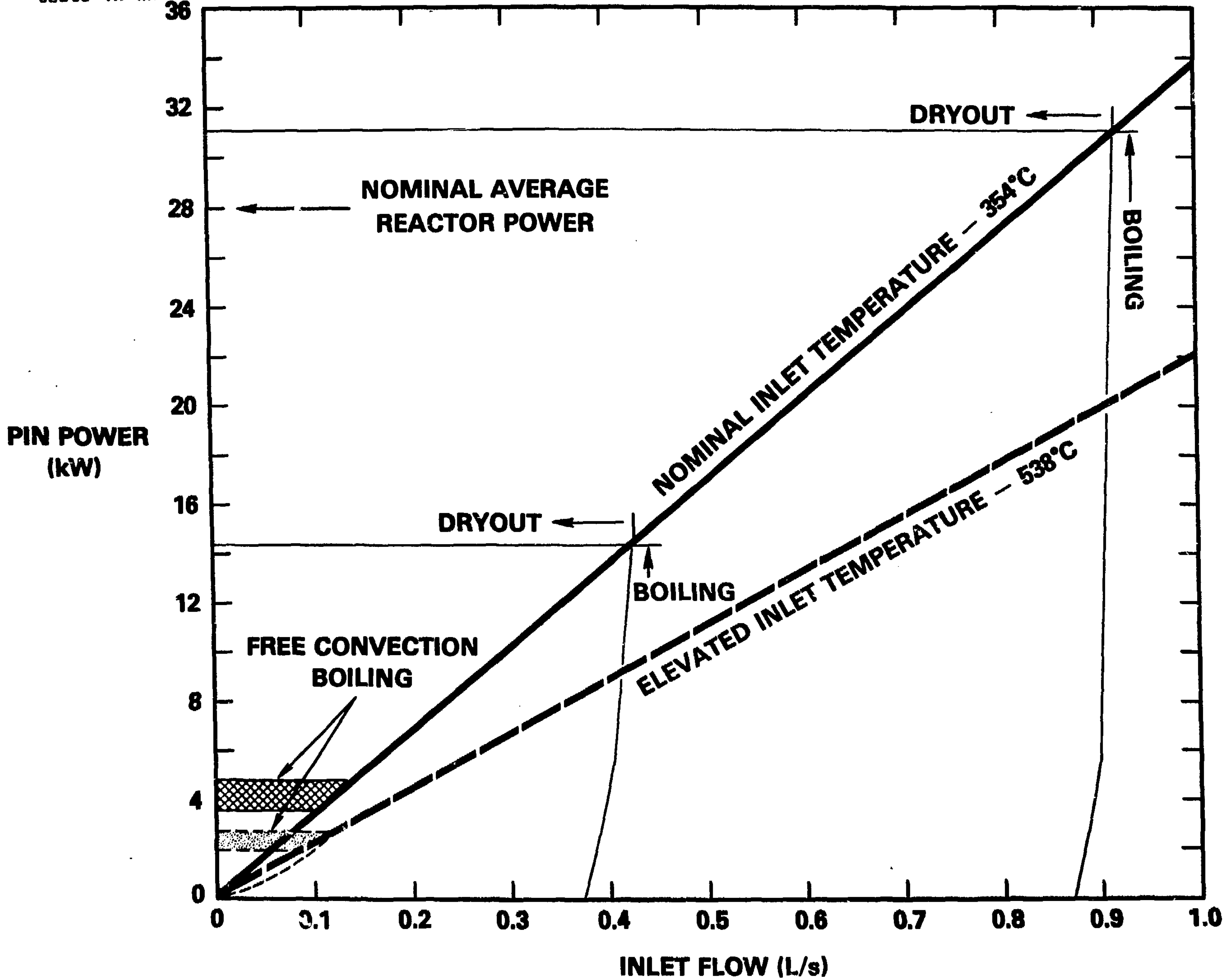

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