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CORE COOLABILITY LIMITS IN LMFBR LOSS-OF-HEAT-SINK ACCIDENTS*

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SUMMARY

Considerable attention has been devoted to the loss-of-heat-sink (LOHS) accident after successful reactor shutdown in a liquid-metal fast breeder reactor (LMFBR) [1-5]. It has been estimated [2] that this accident accounts for more than 80% of the risk of hypothetical core disruptive accidents; therefore, attention should be focused on this specific contributor. In this paper, an analytical study has been performed to estimate the decay heat powers that can be safely removed by sustained free convection boiling without causing cladding damage for a complete range of inlet temperatures from nominal to saturation.

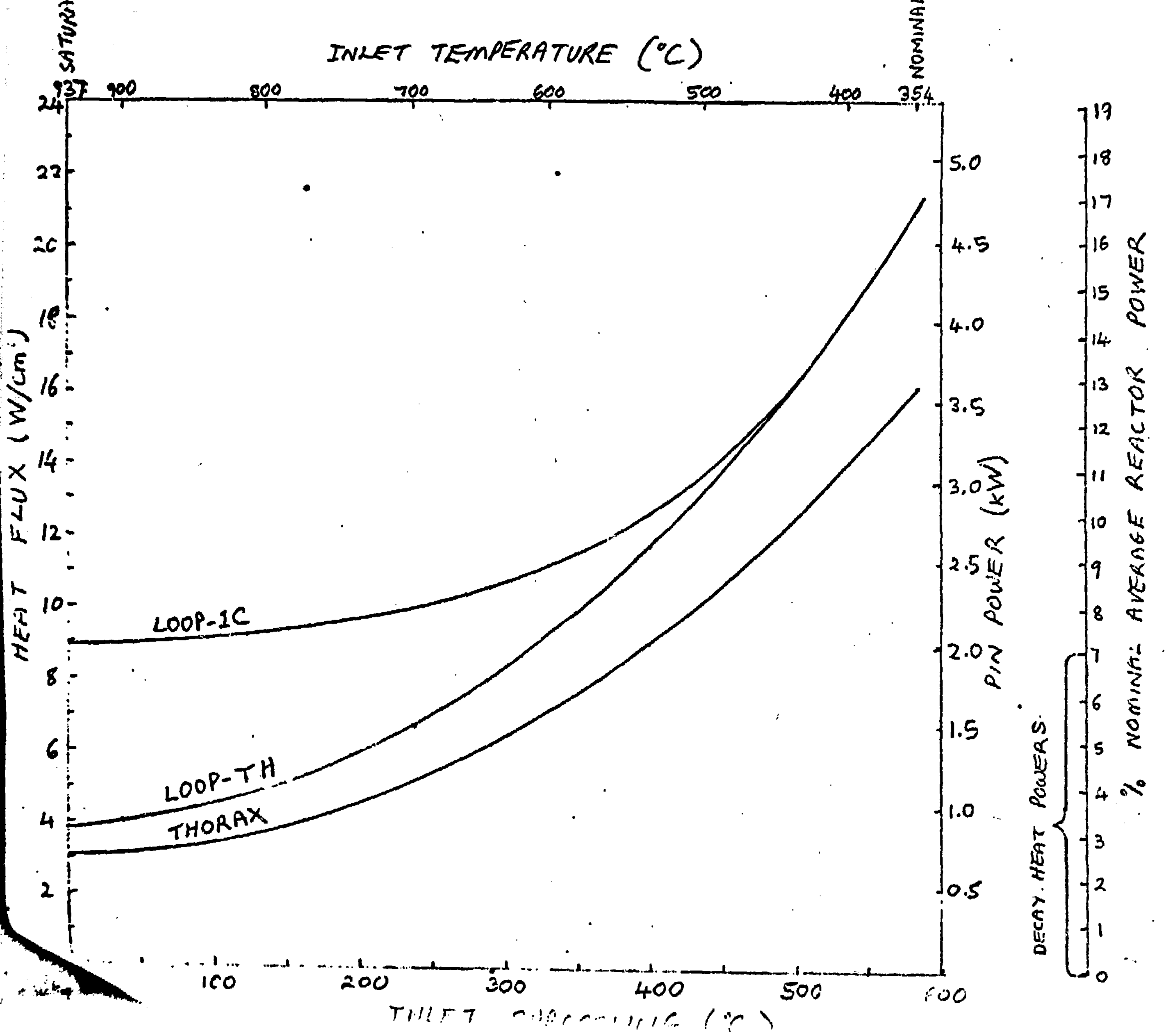
This study has been performed with the codes LOOP-1C [6], a one-dimensional (1-D) loop code; LOOP-TH [5], a two-dimensional (2-D) bundle code with 1-D loop modeling; and THORAX [7], a 2-D bundle code. The geometry used has been one 19-pin test bundle in parallel with a return leg connected by lower and upper plena. The dimensions of these legs were taken from the THORS (Thermal-Hydraulic Out-of-Reactor Safety) Facility SHRS (Shutdown Heat Removal System) Assembly 1 [5] at Oak Ridge National Laboratory (ORNL). This facility has been designed to determine hydraulic interactions between parallel simulated fuel assemblies under conditions similar to those that might occur during partial failure of the SHRS of an LMFBR of Large Scale Prototype Breeder Reactor (LSPB) design. Tests are being performed and data will be obtained at conditions similar to the ones modeled here.

The saturation temperature downstream of the bundle was assumed to be 937°C, a value that was believed representative of experimental operating

conditions. Calculations were performed for sodium inlet temperatures up to 936°C (near saturation). These high temperatures may occur when the decay heat power cannot be adequately removed as a result of a LOHS accident. In the calculations, the bundle power was gradually increased until boiling occurred and then increased further until dryout was predicted. The calculations model the flow which is driven by natural convection conditions due to the density difference between the test section leg in which sodium is boiling and the return leg where there is single phase sodium. A heat exchanger is located at the top of the return leg in order to maintain the sodium at the specified inlet temperature. These conditions simulate the core of an LMFBR during free convection conditions in which liquid sodium would circulate from the upper plenum, down through cold blanket assemblies and then return through the hot driver assemblies.

Figure 1 gives the predicted dryout heat flux as a function of the inlet subcooling. The dryout power is found to be sensitive to the subcooling value. For lower inlet temperatures (i.e., high subcooling) more power is necessary to achieve dryout. For low inlet subcooling (e.g., ~~40~~ 37°C, inlet temperature of 900°C), free convection boiling at powers below dryout is very stable and continuous. For higher inlet subcooling (e.g., ~~240~~ 237°C, inlet temperature of 700°C), boiling at powers below dryout is predicted to be intermittent. As shown in Fig. 1 at a nominal inlet temperature of 354°C, dryout is predicted by the three codes in the range 3.6 to 4.75 kW/pin or 16 to 21 W/cm^2 (approx. 13 to 17% nominal average reactor power). For zero inlet subcooling conditions, LOOP-1C predicts dryout at an approximate power of 2.0 kW/pin (9 W/cm^2). The result from the LOOP-TH code is approx. 0.8 kW/pin (4 W/cm^2), and the THORAX code predicts dryout at approx. 0.7 kW/pin (3 W/cm^2). (THORAX only models the bundle, therefore less buoyancy-induced natural convection flow is predicted to occur and consequently a lower dryout power is calculated). The 0.7 kW/pin value, ~~is~~ equivalent to 3 W/cm^2 ~~or~~ is

FIG. 1 DRYOUT HEAT FLUX AS A FUNCTION OF INLET SUBCOOLING UNDER FREE CONVECTION CONDITIONS AS PREDICTED BY THE CODES. THORAX, LOOP-TH AND LOOP-1C.



approximately 2% of the nominal average reactor power. In an LMFBR, the decay heat power drops to 2% in less than 1 hour after shutdown. Ishii and Fauske [8] estimate a power of 8 W/cm^2 that can be safely removed by natural convection conditions with low inlet subcooling. This value is between the values of 3 W/cm^2 (0.7 kW/pin) and 9 W/cm^2 (2.0 kW/pin) calculated here.

The time interval from the LMFBR reactor shutdown until the sodium coolant becomes saturated depends on the reactor geometry, sodium inventory, and heat losses from the reactor components and piping. Typically, values from 10 to 12 hours have been calculated [3]. By the time saturation conditions have been reached, the decay heat power levels are so low that they can be removed safely by sodium boiling. Although sustained boiling will eventually deplete the sodium inventory, several hours of additional time should be available in order to take corrective measures. Fauske [4] states that it may take 100 hours for core uncovery to occur after reactor shutdown. Some structural damage may also occur because of these high temperatures; but with remedial actions, this accident sequence should not lead to core damage.

In conclusion, the limits of coolability in an LMFBR LOHS accident after successful reactor shutdown have been estimated. Calculations have determined that a range of decay heat powers can be safely removed by free convection boiling without cladding damage for inlet temperatures from nominal to saturation. At the limiting condition of zero subcooling, a heat flux of 3 W/cm^2, corresponding to a decay heat level approx. 2% nominal average reactor power, can be safely removed. This analysis strongly supports the argument that there is a margin of many hours to take corrective measures during a LOHS accident.

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FIGURE CAPTION

FIG. 1 Dryout Heat Flux as a Function of Inlet Subcooling Under Free Convection Conditions as Predicted by the Codes THORAX, LOOP-1C and LOOP-TH