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Coupled Reactor Physics and Coolant Dynamics of Heavy Liquid Metal Coolant Systems

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INTRODUCTION

Cooling of advanced nuclear designs with heavy liquid metals such as lead or lead-bismuth eutectic offers the potential for plant simplifications and higher operating efficiencies compared to previously considered liquid metal coolants such as sodium or NaK. Such applications would however also introduce additional safety concerns and design challenges, therefore necessitating a verifiable computational tool for transient design-basis analysis of heavy liquid metal coolant (HLMC) systems. This capability would enable analysts to compare operational and safety characteristics of design alternatives, and to evaluate relative performance advantages with a consistent, deterministic measure.

COMPUTATIONAL CAPABILITIES

No existing computer code provides all the computational capabilities needed for systems analysis of HLMC designs. However, the SASSYS-11 computer code, long used for analysis of sodium-cooled designs, has been adapted to provide a scoping reactor kinetics/thermal-hydraulics analysis capability and a framework for future development. Revision of the SASSYS-1 reactor kinetics, thermophysical properties, convective heat transfer, and flow momentum transfer models has permitted preliminary design and safety assessments of HLMC systems.
PRELIMINARY RESULTS

The reactor system analyzed is a 300 MWt lead cooled, nitride fueled fast reactor with an average power density of 0.1 MW/liter, a coolant inlet temperature of 425°C, a coolant temperature rise of 125°C, a fuel element diameter of 12 mm, and a square fuel element pitch-to-diameter ratio of 1.34. The choice of large fuel elements and a loose pitch was made to enhance coolant natural circulation. The coolant circuit is arranged in a pool configuration with the steam generator in the primary coolant stream. The thermal separation distance of the steam generator and the reactor core is about 6 m.

Figure 1 presents the results from a SASSYS-1 analysis of an unprotected (without scram) loss-of-flow/loss-of-heat-sink (ULOF/ULOHS) accident sequence in this design. In this analysis, the only reactivity feedback considered was fuel Doppler (T dk/dT = -0.005), power to all pumps was lost at 10s, and the heat rejection rate was reduced to 10% at 20s. Figure 1 shows that the reactor power is reduced by the prompt negative reactivity feedback from fuel heating associated with the flow coastdown and the eventual inlet temperature rise. Natural circulation flow in the faulted condition is predicted to be around 30% of the initial flow. The equilibrium coolant outlet temperature is less than 700°C, or more than 1000°C below the boiling point and is also below structural element degradation thresholds.

In a second application, the 300 MWt reactor system was reconfigured to operate as a subcritical system driven by a source of fast neutrons, such as might be provided by a spallation reaction of accelerated protons with lead. The target system was arranged to have an initial subcritical reactivity of -18$ (keff = 0.94). For the transient, the spallation source was reduced to zero in the first 2.5 s, the heat rejection at the steam generator was reduced to 3.5% of nominal power in the first 2 s, and the primary coolant pumps were tripped at 15 s. Figure 2 shows that the power drops rapidly to decay heat levels upon termination of the neutron source. The two normalized
flows plotted in Fig. 2 are for fueled subassemblies operating at 2.9 MW and 2.0 MW, respectively. The significance of the depicted subassembly flow transients is the flow reversal in the lower power subassembly (negative flow in channel 2 after 50 s), indicating flow recirculation within the core and the need for multidimensional core flow hydraulics.

These results demonstrate the complex coupled thermal, hydraulic, and kinetics phenomena that interact in HLMC design and safety transients, and the need for detailed computational capabilities.

REFERENCES

Fig. 1. Pb-Cooled Reactor Power and Flow Histories for ULOF/ULOHS Accident.

Fig. 2. Power and Flow Histories for Beam Shutdown in the Accelerator Driven System.