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An Innovative Hybrid Loop-Pool SFR Design and Safety Analysis Methods: Today and Tomorrow

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INTRODUCTION

As discussed in Ref. 1, a major activity in fast reactor development is to reduce its capital cost sufficiently (about 20 to 40%) to compete with advanced light water reactors. Toward that end, we launched a thermal design and safety analysis effort to improve the economics and safety of SFRs. From the thermal design perspective, an innovative hybrid loop-pool SFR design has been proposed.[2, 3] This design takes advantage of the inherent safety of a pool design and the compactness of a loop design to further improve economics and safety. From the safety analyses perspective, we have initiated an effort to develop a high fidelity reactor system safety code.[4]

A HYBRID LOOP-POOL DESIGN

In the hybrid loop-pool SFR design, primary loops are formed by connecting the reactor outlet plenum (hot pool), intermediate heat exchangers (IHX), primary pumps and reactor inlet plenum with pipes. The primary loops are immersed in the cold pool (buffer pool). Modular Pool Reactor Auxiliary Cooling Systems (PRACS) transfer heat from the hot pool to the buffer pool when the primary pumps stop. Fig. 1 compares a conventional pool design and the hybrid loop-pool design configurations. Under normal operation, the primary loops operate in forced circulation driven by primary pumps which could be located either in the reactor hot leg or in the cold leg. The primary systems and the cold pool are thermally coupled by the PRACS, which is composed of PRACS heat exchangers (PHX), fluidic diodes and connecting pipes. Fluidic diodes are simple, passive devices that provide large flow resistance in one direction and small flow resistance in reverse direction. Direct reactor auxiliary cooling system (DRACS) heat exchangers (DHX) are immersed in the cold pool to transfer decay heat to the environment by natural circulation. Both DHX and PHX modules use conventional tube bundles to reduce flow resistance and are in baffles to enhance natural circulation as shown in Fig. 1. Similar design was used for liquid salt cooled advanced high temperature reactor (AHTR) systems.[5] The cold pool is hydrodynamically decoupled from, but thermally coupled to, the primary cooling circuit and becomes a buffer pool.

For normal power operation with forced circulation cooling, the primary loops transfer heat to modular IHXs located in the buffer pool. A small bypass with reactor inlet temperature flows upward through PHXs. This bypass flow heats up the buffer pool. This added heat is mainly removed by the DRACS to the environment so that the cold pool temperature remains constant. Under LOFC transients, reduced heat transfer in the reactor core causes the core temperatures to rise. Natural circulation establishes quickly and flow reversal happens through PRACS loops. Decay heat removal mainly occurs through the PHX and DHX modules.

This innovative hybrid loop-pool design has the following major potential benefits:

- **Flexibility to Optimize System Design**: reactor inlet and cold pool temperatures are decoupled, as are the primary loops and passive safety system. This provides more freedom to optimize the design, and offers the potential for better economics and safety.
- **Cost Reduction**: this design allows compact IHXs to be used which may offer the potential to reduce the buffer pool tank size. The safety improvement described below could allow

both core inlet and outlet temperatures to be increased, which yields a higher thermal efficiency for electricity generation.

- **Safety Improvement**: cold pool temperature can be set at a lower value than in a conventional pool design, which increases thermal inertia. Having the closed primary loops immersed in a secondary tank design provides an extra barrier to prevent sodium leakage and reduces the possibility a severe accident caused by uncovering the core.

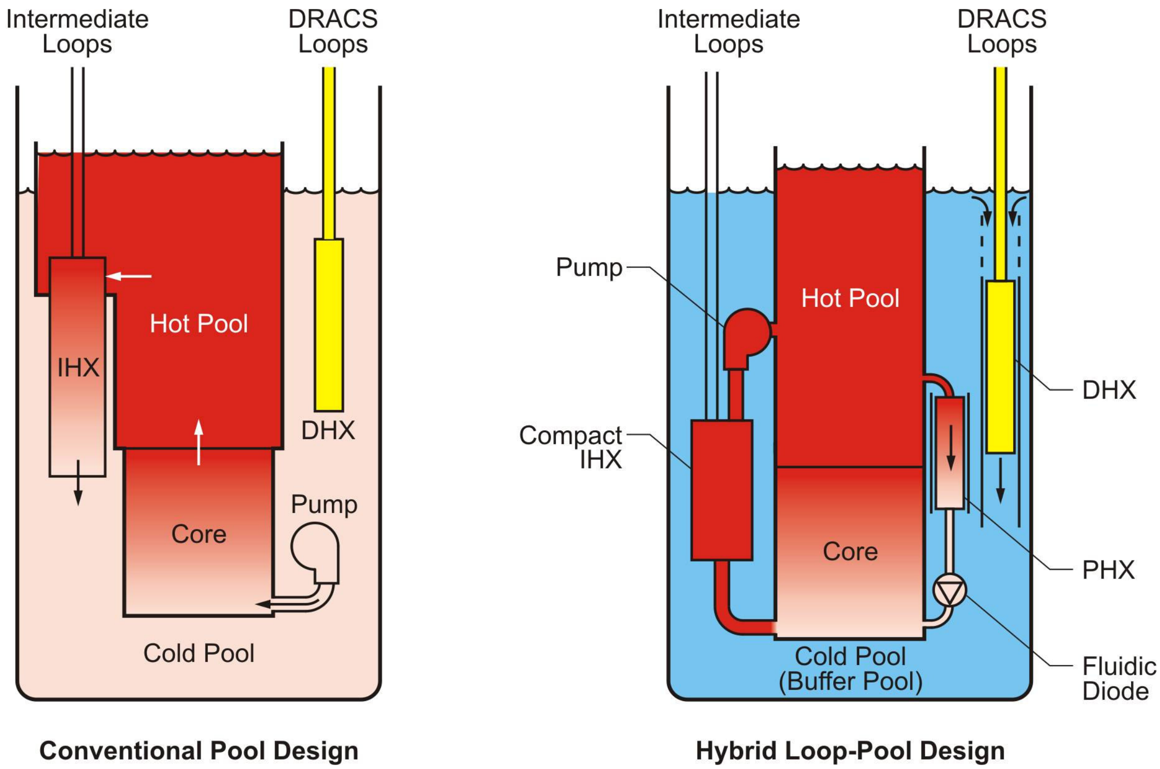

Fig. 1. Comparison of conventional pool design and hybrid loop-pool design for SFR.

RELAP5-3D ANALYSIS

To verify the design ideas, especially how the passive safety systems behave during transients such as LOFC with/without scram, a RELAP5-3D model of the loop-pool hybrid design was developed.[6] The Advanced Burner Test Reactor (ABTR)[7] developed by Argonne National Llaboratory was used as the reference reactor core and primary loop design. ABTR is a 250 MW thermal power conventional pool type SFR with the reactor inlet temperature at 355 °C and the outlet temperature at 510 °C. With minor revisions, such as adding the PRACS and closing the primary loops as shown in Fig.1, a hybrid loop-pool design, ABTR-hybrid, is obtained. The ABTR-hybrid core design is the same as that for ABTR. The reference ABTR-hybrid design assumes that the steady state buffer pool temperature is the same as the reactor inlet temperature. The PHX heat transfer area is sized to match decay heat generation approximately 2 to 3 hours after the LOFC occurs. The DRACS heat removal systems are sized to match decay heat generation approximately 4 to 6 hours after the LOFC occurs.

The core power distribution from ABTR neutronics calculations was used in the RELAP5-3D model to calculate the steady state conditions. The simulation shows a bypass flow rate through the PRACS of about 1% of the total flow rate. The LOFC transient with scram calculations assumed that the decay heat was removed only through PRACS from the hot pool to the buffer pool. No credit was taken for heat removal through the IHXs during the transient. Flow reversal happens in the PHX modules from the steady state upward bypass flow to the natural circulation downward flow at later stage of pump coast-down. The cold sodium coming out of PHXs is mixed with the hot sodium coming out of the IHXs at the reactor inlet and then feeds into the reactor core. Fig. 2 shows key temperatures during the LOFC with scram. The peak clad, hot pool, cold pool temperatures are shown on the figure. The results show that the PRACS can effectively transfer decay heat from the primary system to the buffer pool by natural circulation.

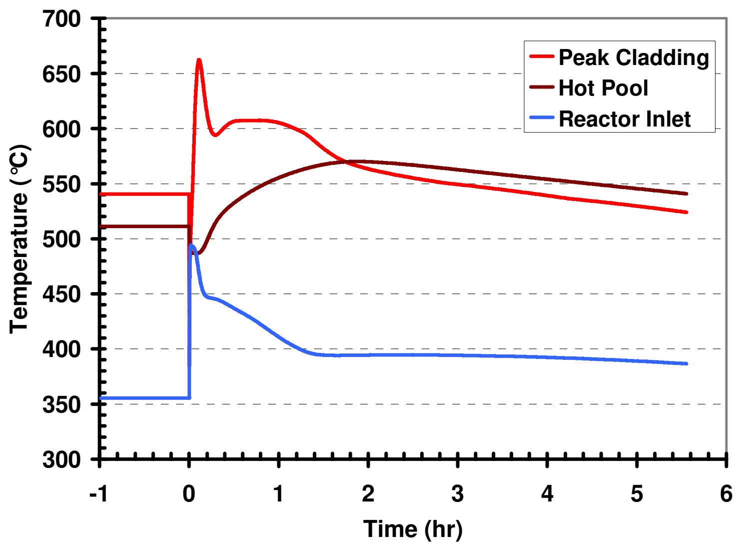

Fig. 2: Key temperatures during LOFC with scram transient

HIGH-FIDELITY SYSTEM SAFETY ANALYSES TOOLS DEVELOPMENT

The SFR economics can also be improved by reducing safety analysis uncertainty and the associated requirements for over-conservatism in design through high fidelity, multi-scale, multi-physics thermal hydraulic analysis methods. Current reactor system analysis and neutronics codes have been successfully used to analyze existing nuclear power plants and are being used to design next generation nuclear power systems. However, additional improvements may be possible when advances in numerical methods, computer science, and physical models during the past three decades are incorporated into the new generation of system analysis codes.

The traditional reactor system analysis approach is to separately develop a neutronics code and a thermal hydraulics code and then to loosely couple them together. Whenever nonlinearly coupled physics are linearized or operator split, there are new first order truncation errors that are introduced into the simulation. These truncation errors affect the physics of the simulation. In the operator split approach of coupled physics, one relies heavily on engineering judgment and experimental data to assess the importance of the truncation errors. Often solutions can be “tuned” to match experimental data by accounting for the numerical errors through modifying physical models. For some new reactor designs, there does not exist a large experimental database. Therefore, the accuracy of the simulation becomes more critical. Higher fideelity simulation tools may offer the potential to predict the behavior of the reactor when experimental data are limited and to predict local parameters that cannot reasonably be measured with instruments in experiments. Hence, more accurate simulations with quantified uncertainties can be used to help in the decision making process.

A pilot system analysis code (called System Analysis for Reactor Applications with High Fidelity, SARAH) is being developed at INL to implement the ideas above. The goal of this work is to improve the accuracy of nuclear reactor safety transient analyses by developing a single code that uses a single nonlinear solution method to solve the nonlinearly coupled equations of thermal hydraulics and neutronics simultaneously. By having a tightly coupled simulation code that has no linearizations or operator splitting and is second order accurate in space and time, one can address the importance of the truncation errors on the safety transient analyses. Fig.3 summarizes the major aspects of SARAH code development.

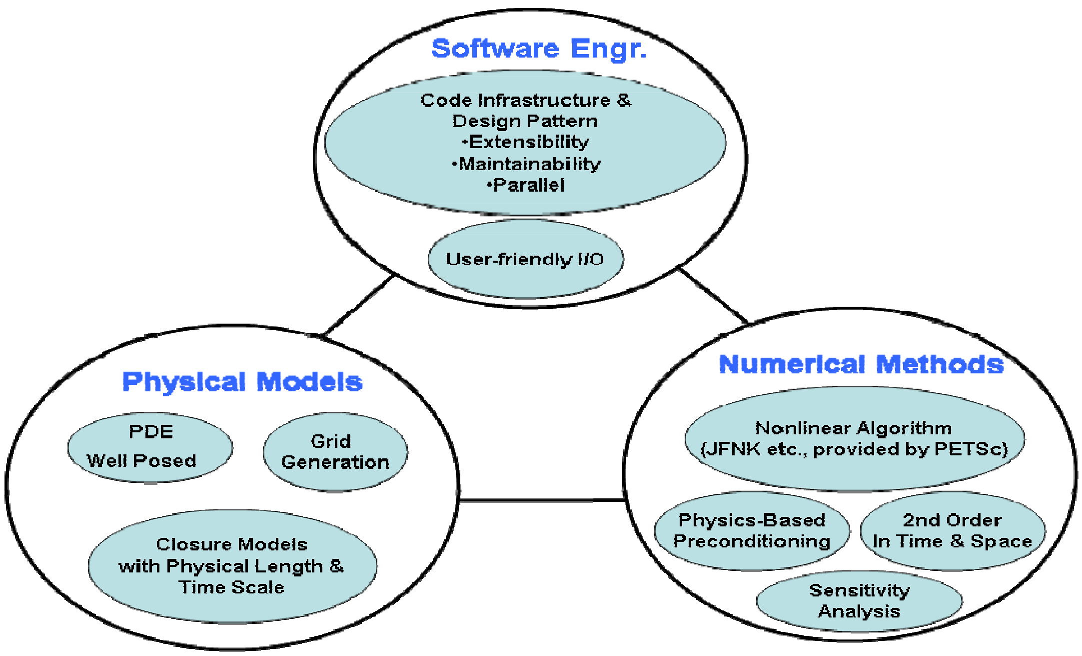

Fig. 3. The paradigm for SARAH development

CONCLUSIONS

Preliminary evaluations suggest that features incorporated in a hybrid loop-pool SFR design have the potential to improve the economics and safety of the SFR. Initial results from transient analyses show that the fuel clad and sodium temperatures during LOFC are within acceptable SFR temperature limits and that the potential exists to increase the reactor outlet temperature to improve economics over conventional pool designs. Initial evaluations also indicate that inherent safety characteristics of hybrid loop-pool design are enhanced by the large thermal inertia of sodium within the hot and buffer pools and innovative passive safety systems design. Development of high-fidelity safety analysis tools may provide additional opportunities to further improve the economics and safety of SFRs.

ACKNOWLEDGEMENTS

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