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Abstract

An innovative hybrid loop-pool design for the sodium cooled fast reactor (SFR) has been recently proposed with the primary objective of achieving cost reduction and safety enhancement. With the hybrid loop-pool design, closed primary loops are immersed in a secondary buffer tank. This design takes advantage of features from conventional both pool and loop designs to further improve economics and safety. This paper will briefly introduce the hybrid loop-pool design concept and present the calculated thermal responses for unproctected (without reactor scram) loss of forced circulation (ULOF) transients using RELAP5-3D. The analyses examine both the inherent reactivity shutdown capability and decay heat removal performance by passive safety systems.

1. Introduction

It has been well recognized that 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 (Todreas, 2007). Toward that end, an innovative hybrid loop-pool design for sodium cooled fast reactors (SFR) has been proposed (Zhao and Zhang, 2007). This design takes advantage of features from conventional both pool and loop designs to further improve the economics and safety of SFRs. This paper will briefly describe the hybrid loop-pool design concept and then present the calculated thermal responses for unprotected loss of forced circulation (ULOF) transients using RELAP5-3D (RELAP5-3D).

In the hybrid loop-pool design, closed primary loops are formed by connecting the reactor outlet plenum (hot pool), intermediate heat exchangers (IHX), primary pumps and the reactor inlet plenum with pipes. The closed primary loops are immersed in the cold pool (buffer pool). Modular Pool Reactor Auxiliary Cooling Systems (PRACS) are added to transfer heat from the hot pool to the buffer pool when the primary pumps stop running. Fig. 1 compares conventional pool design and the hybrid loop-pool design configurations. Under normal operations, 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 reject 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 (Peterson and Zhao, 2006). The cold pool is hydrodynamically decoupled from, but thermally coupled to, the primary cooling circuit and becomes a buffer pool.

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For normal power operations with forced 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 as well as the heat loss from the hot pool to the buffer pool through Redan (thermal baffle) is mainly removed by the DRACS to the environment so that the cold pool temperatures remain steady. Under loss of forced circulation cooling 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. Fig. 2 illustrates the flow pattern during loss of forced circulation (LOFC) transients. Decay heat removal mainly occurs through the PHX and DHX modules.

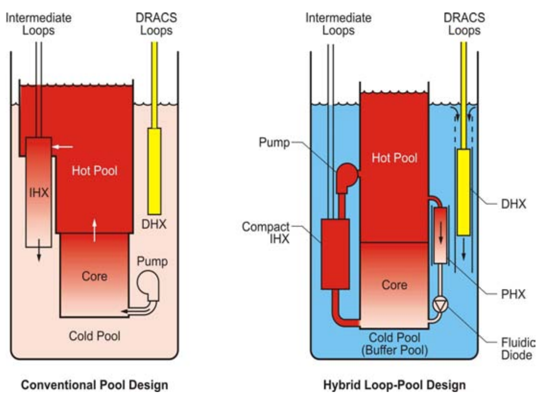

Fig. 1. Comparison of the hybrid loop-pool design and the conventional pool designs during LOFC conditions.

This innovative hybrid loop-pool design has the following major potential benefits (Zhao, et. al., 2008):

- Flexibility to optimize the 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 by improving economics and safety.
- Cost reduction and safety improvement: this design allows compact IHXs to be used which are smaller in size (and potentially less expensive) and could consequently reduce the buffer pool tank size and the containment building size. In addition, the primary sodium inventory is significantly reduced by decoupling the buffer pool from the primary cooling circuit. Hence, less radioactive sodium has to be purified during operations. The cold pool temperature can be set at a lower value than in a conventional pool design, which results in increased thermal inertia. Both the core inlet and outlet temperatures could be increased, which yields a higher thermal efficiency for electricity generation. This design provides an extra barrier to prevent sodium leakage and reduces the possibility a severe accident caused by the core uncovering.
- Improved in-service inspection capability: Due to liquid sodium's opacity, one disadvantage of the conventional pool designs is its difficulty to perform in-service inspection. The fuel assemblies have to be unloaded from the core and all the sodium has to be drained to perform detailed in-service inspection. With the hybrid loop-pool design, it becomes much easier. A potential procedure for in-service inspection is that the pumps can be operated at low power with minimum pump speed to remove decay heat after the reactor is shutdown. In addition, the hot pool provides large enough thermal inertia to absorb the heat from the reactor while the buffer pool can be drained to perform necessary inspection and repair.

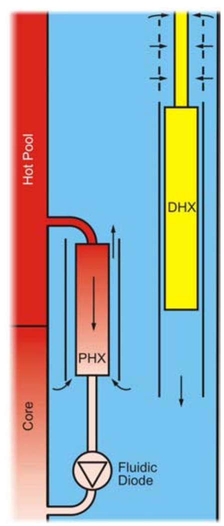

Fig. 2. Flow patterns illustration during loss of forced cooling transient.

2. RELAP5-3D Analysis

2.1. Summary on models

To verify the design ideas, especially how effective the passive safety systems are to transfer heat from the primary system to the buffer pool during transients such as LOFC with and without scram, a RELAP5-3D model of the loop-pool hybrid design was developed (Zhang, et al., 2008). RELAP5-3D is the latest version of an Idaho National Laboratory system analysis code that has been primarily used to simulate light water reactors. However, the code has a generalized capability to simulate a wide range of working fluids including sodium.

The Advanced Burner Test Reactor (ABTR) (Chang, et. al., 2006) developed by Argonne national laboratory (ANL) 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 some straightforward revisions, such as adding the PRACS and closing the primary loops as shown in Fig.1, a hybrid loop-pool design, SFR-hybrid, is obtained. The SFR-hybrid core design is the same as that for ABTR. 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 steady state buffer pool temperature is assumed the same as the reactor inlet temperature.

The results for the loss of forced circulation cooling with scram (or PLOF) have already been reported previously (Zhang, et. al., 2008). The results demonstrated that the PRACS can effectively transfer decay heat from the primary system to the buffer pool by natural circulation. In this paper, the analyses are extended to unprotected loss of forced circulation cooling (ULOF) transients.

The intent of the analyses carried out here is to demonstrate that with the hybrid loop-pool design, the inherent negative reactivity feedback of the core can safely shut down the reactor and the passive safety systems can effectively remove heat from the core to the environment, even in the hypothesized beyond-design-basis accidents with failure of the control rod system to scram the reactor.

The point kinetics model in RELAP5-3D was used in the ULOF calculations. The reactivity feedback mechanisms include radial expansion, axial expansion, Doppler, sodium density and control rod driveline expansion. The reactivity feedback coefficients were directly taken from ANL's ABTR design at the beginning of the equilibrium cycle condition (BEOC). The reactivity coefficients are shown in Table 1.

Table 1
Reactivity feedback coefficients used in the calculations

Reactivity Feedback Mechanisms	Reactivity Coefficient (cent/°C)
Radial expansion coefficient (α_r)	-0.59
Axial expansion coefficient (α_{ax})	-0.06
Doppler coefficient (α_D)	-0.10
Sodium density coefficient (α_{Na})	0.03

The control rod drivelines are immersed in outlet coolant from the core. A rise in core outlet temperature will cause the control rods to be inserted further into the core, providing a negative reactivity component. Since the control rod driveline feedback coefficient was not provided in the ABTR report, the reactivity feedback from control rod driveline expansion is ignored in this analysis.

A quasi-static model for the feedback reactivity is used in the calculations. The feedback reactivity is directly calculated from the respective temperature changes as:

$$\rho = \alpha_D \Delta T_f + \alpha_{Na} \Delta T_{Na} + \alpha_{ax} \Delta T_{cld} + \alpha_r \Delta T_r \quad (1)$$

Where ΔT_f is the fuel temperature change with respect to the steady state values, ΔT_{Na} the sodium temperature change, ΔT_{cld} the cladding temperature change and ΔT_r the temperature change for radial expansion reactivity calculation.

Radial expansion accounts for core dilation due to thermal expansion of the hexcan load pads as well as the core support grid plate. The radial core expansion reactivity feedback model used is a simple model (Ott, 1991). The equation is given as

$$\Delta T_r = \Delta T_{in} + \frac{XMC}{XAC}(\Delta T_{out} - \Delta T_{in}) \qquad (2)$$

where ΔT_{in} represents the reactor inlet temperature change and ΔT_{out} represents the reactor core outlet temperature change. *XMC/XAC* is the ratio of the core mid-plane (*XMC*) to the above core (*XAC*) load-pad elevation.

The axial expansion or contraction is assumed to be controlled by the expansion or contraction of the cladding.

Table 1 indicates that the radial expansion has the largest reactivity feedback coefficient. Hence a good SFR design would want to maximize the radial expansion reactivity feedback to provide sufficient negative reactivity to quickly reduce the reactor fission power and bring the reactor to the decay heat power level.

2.2. *ULOF with no heat loss through intermediate heat transport system (IHTS)*

The transient being studied here is the same as one major beyond-design-basis event – Unprotected Loss of Cooling – analyzed for PRISM design (Salerno, et. al., 1988). The postulated transient sequence for the unprotected loss of cooling analyzed for PRISM is the loss of all cooling by the intermediate heat transport system and loss of primary pump power without scram.

The basic accident sequence analyzed here is the loss of normal power to the reactor primary and intermediate coolant pumps, with failure of the emergency power supplies. The result is an immediate loss of forced flow in the primary and intermediate coolant circuits. It is also assumed that the reactor safety system fails to insert the control rods, and the loss of forced flow proceeds at full power. In addition, it is assumed that heat removal to the intermediate heat transport system (IHTS) ceases when the transient is initiated, so that the only heat removal path is through the PRACS and DRACS. With the postulated ULOF transient analyzed here, the hybrid loop-design provides the mechanism to allow the core inlet temperature to rise rapidly in the early phase of the transients. The initial rapid rise in the core inlet temperature causes the reactor core to expand and introduces a sharp reduction in reactivity and a corresponding reduction in power subsequently.

Fig. 3 shows the normalized values of the reactor power, decay power, total flow rate and PHX flow rate after the flow reversal occurred at the early time of the transients. Fig. 4 shows the reactor power, total flow rate and PHX flow rate at the extended time of the transients. As can be seen from Fig. 3, the natural circulation flow in PHXs is fully established before the primary pump coastdown stopped. This provides a smooth transition for the reactor flow going from forced circulation to natural circulation and prevents the reactor flow from going down too low as might encountered for conventional pool type design SFRs. Fig. 5 shows the flow rate through PHX following the initiation of the ULOF. The peak cladding, reactor outlet, hot pool and reactor inlet temperatures responses are shown in Figs. 6 and 7 at the early time and the extended time of the transients, respectively. As indicated in Fig. 6, the hot sodium from the hot pool quickly enters the reactor inlet after the initiation of ULOF. The hot sodium coming into the inlet plenum heats up the core grid support plate and the load pads and dilute the core, which introduces negative reactivity feedback. The rapid rise of the inlet temperature and the large amount of negative reactivity the radial expansion introduces as shown in Fig. 8 quickly shuts down the reactor. As shown in Fig. 8, the radial expansion feedback reactivity is the dominating reactivity feedback mechanism. The hybrid loop-pool design is able to provide sharp negative reactivity feedback through enhanced radial expansion capability.

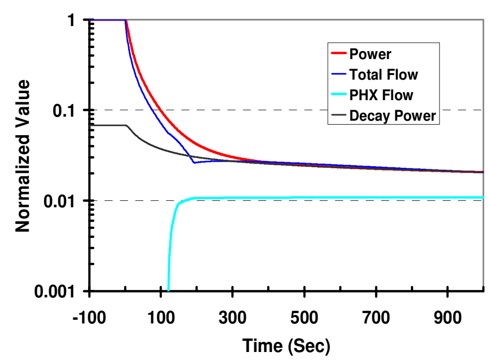

Fig. 3. The normalized power, decay power, reactor flow rate and the PHX flow rate during ULOF, at early time.

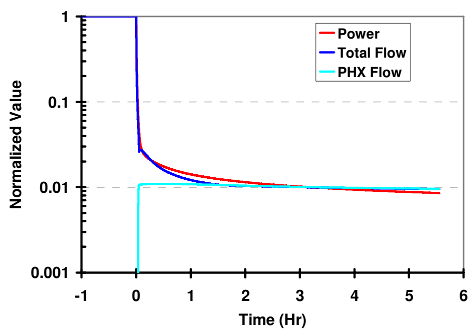

Fig. 4. The normalized power, reactor flow rate and PHX flow rate during ULOF, at extended time.

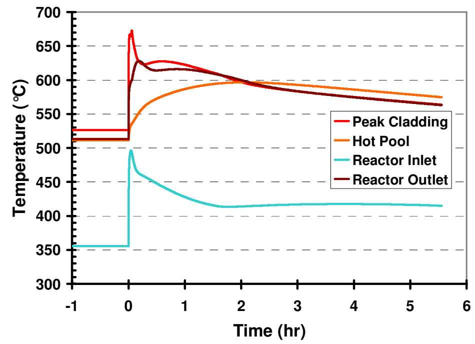

Fig. 7. The key temperature response during ULOF, at extended time.

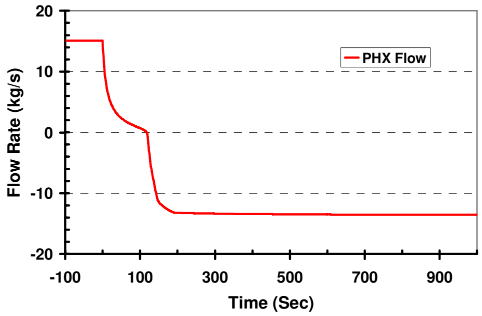

Fig. 5. The flow rate in PHX during ULOF

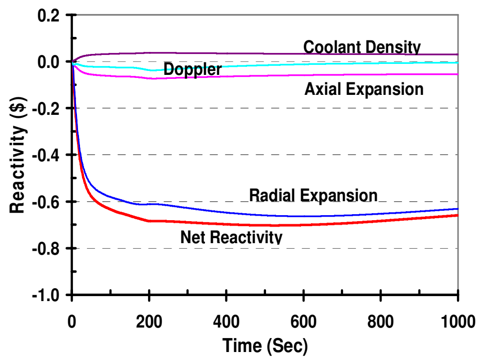

Fig. 8. Transient reactivity feedback.

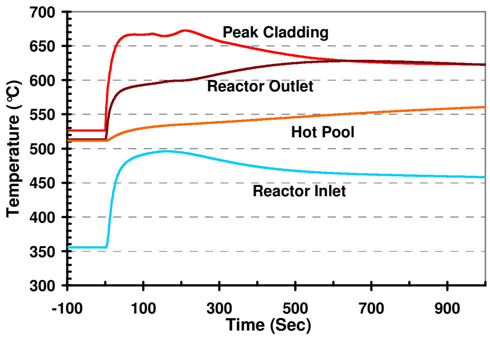

Fig. 6. The key temperature responses during ULOF, at early time.

The ULOF transient can be roughly subdivided into three phases. The first phase is the forced cooling and thermal mixing at the early stage of the primary pump coastdown. At this stage the reactor flow rate drops off at a faster rate than that of the reactor power as shown in Fig. 3. The power to flow ratio is larger than that for the rated power steady state condition. Since it is assumed that there is no heat removal to the intermediate heat transport system and the PHX heat removal from the primary system to the buffer pool is relatively small comparing with the total power, most of the heat generated in the reactor will stay in the primary system. The primary system would heat up and the temperatures would rise as shown in Fig. 6. During this stage the reactor inlet temperature rises rapidly in the first tens of seconds following the initiation of the transient. This rapid rise in the inlet temperature introduces large amount of negative reactivity due to the core radial expansion. This large negative reactivity feedback along with the negative feedback reactivity introduced by the axial expansion and Doppler shuts down the reactor quickly.

The second phase is the transition from the forced cooling provided by the pump coastdown to fully established natural circulation cooling. As the pumps coast down, the upward flow rate through PRACS decreases and gradually approaches zero. Toward the later stage of the pump coastdown, the buoyancy force would overcome the pump head driving force and force the flow to reverse in the PHX. As the pumps head becomes weaker when the pumps coast down further, the buoyancy force drives more flow through the PHX and the natural circulation fully establishes before pumps coastdown stops. The hot sodium is sucked out of the hot pool and transfers heat to the buffer pool through PHXs. The cold sodium coming out of PHXs is then mixed with the hot sodium coming out of the IHXs in the reactor inlet plenum. The inlet temperatures are trending down during this phase.

The third phase is the long-term decay heat removal phase. During this phase, natural circulation through PHX removes reactor decay heat from the primary system to the buffer pool. The reactor inlet temperature is gradually decreasing. The peak cladding temperature reaches its maximum value and starts to decrease. During this phase, all temperatures evolve very slowly due to the large thermal inertia of the hot pool and the buffer pool sodium. The normalized decay heat power will be lower than the normalized reactor flow rate as shown and the reactor stays safely shutdown.

2.3. *A parametric study – reducing the radial expansion coefficient*

Since the radial expansion is the dominating reactivity feedback mechanism, in this parametric study the radial expansion reactivity coefficient is reduced by 30% from -0.59 cent/$^{\circ}$C to -0.413 cent/$^{\circ}$C. Other parameters remain unchanged. The motivation for this study is that for large scale commercial SFRs, the radial expansion feedback will be weaken. As indicated by the previous IFR work (Wade and Hill, 1997), the radial expansion coefficient could be reduced by 30% for large commercial SFRs.

Fig. 9 shows the normalized power for the original case and the current case (Case 1) with 30% reduction in the radial expansion reactivity feedback coefficient. The reduction in the radial expansion coefficient yielded less radial expansion reactivity feedback as shown in Fig. 10 when compared to Fig. 8. The reactor power stayed higher during the ULOF transient, which added more heat to the system and consequently yielded higher system temperatures. Fig. 11 shows the key temperatures during the transient. The maximum peak cladding temperature went up about 30°C when compared to the original case.

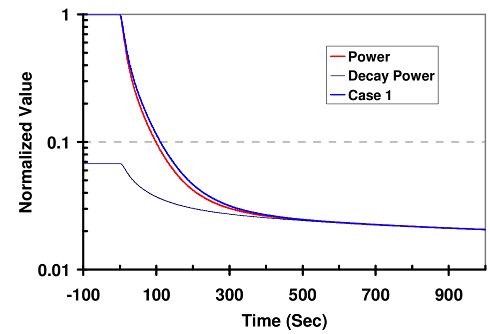

Fig. 9. The comparison of the normalized power for Case 1 and the original case.

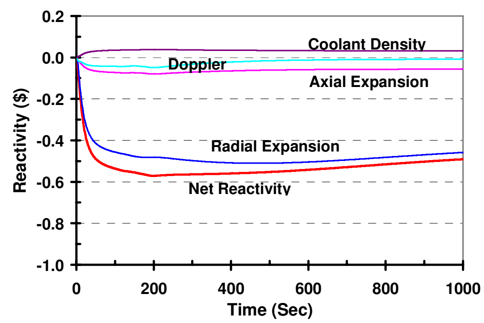

Fig. 10. Reactivity feedback for case 1.

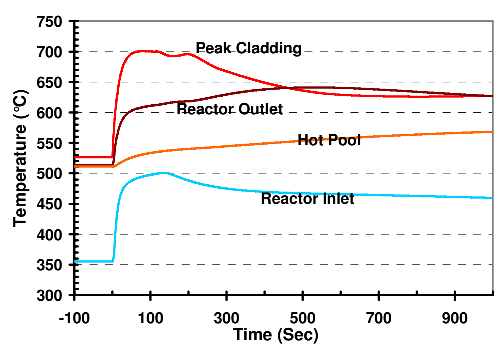

Fig. 11. The key temperatures for case 1.

2.4. *ULOF with the ITHS heat removal turned on*

In this run the intermediate heat transport loop heat removal is turned on. The ABTR design used printed circuit heat exchangers (PCHE) as the sodium-to-CO2 heat exchangers for the power conversion system. Through these PCHE modules, heat is transferred from the sodium in the intermediate heat transport system to the CO2 in the supercritical CO2 Brayton cycle. The metal mass for the PCHEs is about 110 tonnes, comparing with the total IHX mass of about 80 tonnes. This large amount of metal can absorb quite significant amount of heat during transients and consequently could impact the sodium temperatures exiting the IHXs as well as the reactor inlet.

This case study would quantify the effect of heat absorption of the thermal inertial in the IHTS on the peak cladding temperature during ULOF. In this case study, it is assumed that CO2 flow rate goes to zero immediately following the transient and therefore there will be no heat rejection to the environment from the power conversion system. The only heat loss to the IHTS is the heat absorption by the metals in the intermediate loops. The intermediate sodium pumps are electromegmatic pumps and a much shorter pump coastdown time is assumed in the transient calculations. The intermediate sodium pumps' coastdown time is about 36 seconds versus about 200 seconds for the primary pumps.

Fig. 12 shows the normalized power for the original case (without heat loss to the IHTS) and the case with heat loss to IHTS considered. With heat loss to IHTS, the reactor inlet temperature increase is reduced in the first several hundred seconds as shown in Fig. 13 when compared to Fig. 6. Hence the negative radial expansion effect is weakened as shown in Fig. 14. The reactor power stayed higher than the original case as shown in Fig. 12. However, with the additional heat removal to the IHTS, the maximum peaking cladding temperature hardly changed. After the maximum peak cladding temperature is reached, the peak cladding temperature drops a lot faster than the original case with no heat loss to the ITHS. Please note that the peaking cladding temperatures shown on the figures are for the clad in the fuel region only. Above the fuel region, the ABTR fuel pin design has 120 cm gas plenum region. The cladding materials in the gas plenum region are heated up during the early stage of the transient. During the period between 200 second and 750 second, the stored energy in the cladding materials in the gas plenum region is adding heat to the sodium coolant. Hence the reactor outlet temperature is actually higher than the peak cladding temperature in the fuel region.

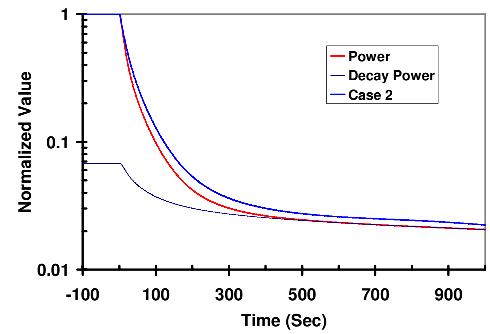

Fig. 12. The comparison of the normalized power for Case 2 and the original case.

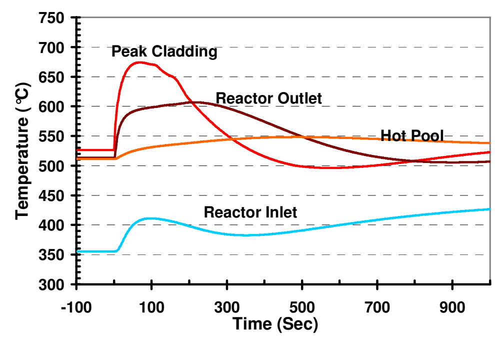

Fig. 13. Key temperatures during ULOF for case 2.

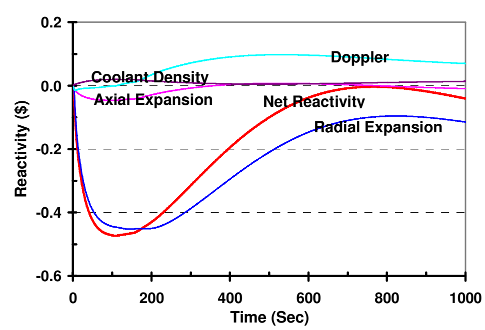

Fig. 14. Reactivity feedback for case 2.

3. Conclusions and Future Work

The hybrid loop-pool design has great potential to improve the economics and safety of the SFRs. Transient analyses show that the fuel clad and sodium temperature responses are very favorable during ULOF for this non-optimized hybrid design. The inherent neutronic and thermal-hydraulic performance characteristics of the hybrid design provide self-protection in the postulated accidents to limit consequences without activation of engineered systems or operator actions. Large potential exists to further increase the reactor outlet temperature to improve the economics over the conventional pool designs. Inherent safety characteristics of hybrid loop-pool design are ensured by large thermal inertia of sodium within the hot and buffer pools, and innovative passive safety system.

Future work will also be extended to analyze loss of secondary heat sink without scram accidents, loss of coolant accidents (LOCA), and further optimization analyses such as finding the optimal buffer pool, reactor inlet and outlet temperatures for large power hybrid designs with and without the use of compact IHXs.

Acknowledgement

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