Preliminary Assessment of PHTS Pump Piping Break Accident of DSFR-600

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1. Introduction

As a part of Development of Regulatory Audit Technology for System Safety of Sodium-cooled Fast Reactors (SFR) , KINS is evaluating the applicability of TRACE [1] code for safety analysis of SFR Since 2012.

Based on the steady-state input deck for Demonstration Sodium Cooled Fast Reactor 600MW (DSFR-600) [2] component-wise specific modeling is developed for DSFR-600. In this study, one of representative design base accidents, PHTS pump pipe rupture accident is analyzed and modeling considerations is identified for the Residual Heat Removal System (RHRS) which plays key role in eliminating the decay heat from the core during accidents. [3]

2. System modeling and the accident scenario

TRACE code modeling and the steady-state condition of DSFR-600 used in the assessment of Primary Heat Transport System (PHTS) pump pipe rupture case is described.

DSFR-600 model includes PHTS, Intermediate Heat Transport System (IHTS) and RHRS. RHRS are composed of PDHRS and ADHRS and each system is composed of 2-FDRCs and 2-PDRCs in design. In the TRACE model they are modeled with two circuits with double capacity.

In the steady-state condition, the plant is operating at 100% power (1588.2MWt) and RHRS is standby with 4.93MW capacity to protect sodium freezing in the circuits. The rest of heat is transported to 2-SGs through 4-IHXs. Calculated steady state condition is compared with the design condition in Table I. PHTS and IHTS pump power was not modeled in the model.

Table I: TRACE code St.-St calculation result

	DESIGN K,MW/kg/s/,m	TRACE prediction
CORE I/O T. Power/flow	638.15/783.15 1548.2/8366.1	638.3/784.5 1548.2/8364.0
IHX I/O T. Q/flow	578.55/775.15 387.5/3073	578.44/776.77 385.83/3072.7
SG Q/flow	775/344.7	771.63/337.8
DHX Q	4.7	4.9

PHTS pump break scenario begins with double side break in one of pipe connections to the core inlet plenum coming from two PHTS pumps. [4] Overall accident sequence and assumptions are as followings;

1) During 100% power operation, one of PHTS pumps pipe break occurs during 5 seconds
2) Reactor Trip signal of Core exit Temp. High (set point: 847.15K) occurs
3) Reactor Trips after the signal with 1.6 sec. delay
4) Loss of offsite power occurs at 5 sec. after Reactor Trip. PHTS/IHTS Pumps Trip and SG feed isolation occurs
5) After 30 minutes, RHRS (PDRC/FDRC) activates with air damper control

In the code simulation, the pipe break occurs at 5 seconds in calculation time and before RHRS recover, air dampers are at normal operation, i.e. slightly opened for anti-freezing.

3. Calculation result and system response

Before the reactor trip, core power is decreased by Doppler to 1306.45 MW then decreased to the decay heat level. After reactor trip, core inlet temperature is decreased gradually before pump trip. Then core outlet temperature is increased before core inlet flow is increased by natural circulation.

Steam generator (SG) removed residual heat until 1,715 seconds by evaporation of water inventory within SGs after feed water isolation. As soon as SG is dried out, SG acts as a heat source. But its heat transfer rate was limited.

RHRS ran with 4.9MW capacity before break and its air damper is opened slightly until reactivation at 30 minutes after break. RHRS heat transfer rate was influenced by the coolant temperature and flow of DHX's shell side inlet.

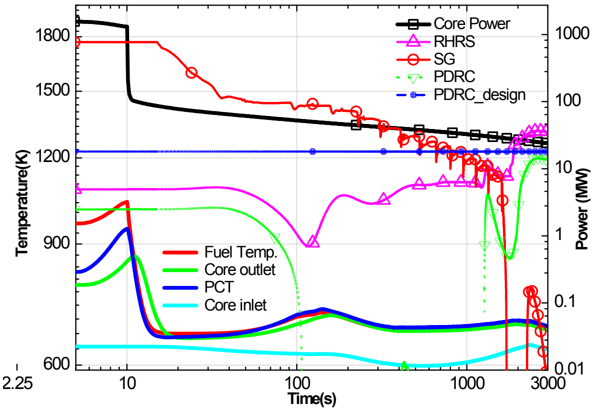

Fig. 1. System response for DSFR-600 PHTS pipe break

At 30 minutes after the accident, reactor core generates about 28MW decay heat, 1.8% of full power. After RHRS was reactivated, 35MW heat was removed from the hot pool, therefore the core inlet temperature decreased successively as shown in Fig 1.

Sodium from the break pipe flows into the cold pool directly and some of coolant in-flow from intact pump piping exits to the cold pool after break. This resulted in decrease of core inlet flow. Calculated core outlet temperature peaked with 863.9K after break and fuel cladding and centerline temperature also peaked with 947 K and 1,036K respectively.

Preliminary calculation result for PHTS pump piping break using TRACE code showed all of these temperature safety criteria were satisfied .

4. Review on RHRS design parameters

RHRS including FDRC and PDRC were carefully modeled and verified with the design parameters. Each model of DHX, AHX, and FDHX were tested with design parameters such as the heat capacity on desired coolant conditions. Based on the specific component model, FDRC and PDCR models also checked those performances. For PDRC, its performance showed identical value between components case and assembled circuit. For FDRC, combined with DHX and FDHX with piping, calculated heat removal capacity was higher than the design value.

RHRS removes the decay heat from the hot pool through the Decay Heat Exchanger (DHX). DHX are dipped in the hot pool and inflow is developed in passive manner. It was also identified that the performance of RHRS is strongly depends on the liquid sodium temperature and induced inflow through DHX. The performance of PDRC and FDRC was checked by sensitivity study for the sodium conditions into the DHX for the estimation of performance during transient. Sensitivity result showed RHRS's heat removal depends on the sodium temperature strongly than inlet flow and RHRS's performance can be degraded by coolant condition.

Actual DHX shell side flow in the hot pool initiated by hot sodium near the inlet of DHX and density difference of coolant between inlet and outlet of DHX gives driving force for DHX shell side flow. So, main contributor of DHX flow is the temperature of sodium near the inlet of DHX.

In 1-dimensional modeling of the hot pool, DHX, IHX and pumps, DHX mass flow decided by K-factor adjusted at the inlet, to a portion of main flow from the hot pool node to the IHX inlet derived by pump inertia or natural circulation between hot and cold pool.

Comparison between the performance of PDRC by sensitivity and the transient calculation showed that the transient performance of RHRS can be checked by pre-calculation and/or test in terms of inlet coolant condition of the DHX.

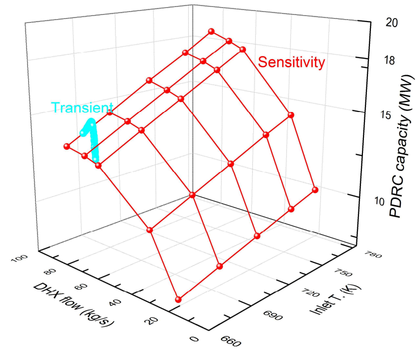

Fig. 2. PDRC capacity predicted by sensitivity and transient

The pre-calculation result of FDRC model showed that FDRC's heat removal capacity could be deteriorated with its air damper control.

FDRC is treated as an active component in design and some of information is yet decided in design such as the characteristic of the Electro Magnetic FDRC Pumps and FDHX tube side pressure drop at low flow. In modeling, EM pump is assumed as rotary pump with zero speed control at pump trip. So its modeling still needs validation with reasonable tests.

5. Conclusions

Preliminary analysis was performed with TRACE code for DSFR-600 PHTS pump piping break accident. The calculation result showed that the calculated safety parameters are conforms to the design criteria for DBA accidents.

RHRS design of DSFR-600 and its performance during transient was also reviewed by sensitivity study on the effect of sodium condition to the transient decay heat removal capability of RHRS. Following insights are identified. These should be considered in improving the design also in licensing review of SFR safety analysis.

1) The transient performance of RHRS might differ from the component's design capacity.
2) RHRS's transient performance also should be included in the design documents and validated with reasonable test and/or analysis with consideration of the variation of coolant conditions during transient.
3) The analytic model used for safety analysis should consider 3-D effect of vessel pool and its uncertainty with reasonable conservatism.

REFERENCES

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[4]Safety Evaluation for Transients of Demonstration SFR, KAERI/TR-4288/2011/, Feb. 2011