Preliminary Assessment of the Loss of Flow Accident for PGSFR

Andong SHIN*, Yong Won Choi, Moohoon BAE
Korea Institute of Nuclear Safety, 62 Gwahak-ro, Yuseong-gu, Daejeon, Korea
**Corresponding author: andrew@kins.re.kr*

1. Introduction

KINS is preparing licensing review for Prototype Generation IV Sodium cooled Fast Reactor (PGSFR) of 150MWe which is under developing.

TRACE code [1] have being considered as a candidate tool for SFR audit calculation for licensing review since 2012. On the basis of modeling and pre-calculation experience for the Demonstration Sodium cooled Fast Reactor (DSFR-600), TRACE code model for PGSFR was developed this year.

In this paper, one of representing Design Base Event (DBE), Loss of Flow (LOF) accident was pre-calculated and Locked Rotor (LR) case was compared with LOF case since it could be a possible limiting case for LOF representing DBE. Sensitivity calculation for the LR case was implemented for identifying major parameters for the scenario.

2. Steady State Conditions for Simulation

PGSFR is composed of Primary Heat Transport System (PHTS), Intermediate Heat Transport System (IHTS), Residual Heat Transport System (RHRS) and Power Conversion System (PCS). All of major systems are included in TRACE model except PCS such as Turbine.

2.1 TRACE code modeling

Main Coolant flow starts with two PHTS pumps suction from cold pool to the inlet plenum. Cold sodium flows to core through the inlet plenum. Core outlet coolant is collected in the hot pool. Due to Intermediate Heat Exchanger (IHX) is located between hot pool and cold pool, hot sodium is cooled by four IHXs shell side and returns into the cold pool.

Within IHTS, Sodium is heated in tube side of IHXs and transported to the Steam Generators (SG). IHTS flow is formulated by two Electro-Magnetic Pumps (EMP) located between SG and IHXs. In model, EMP was modeled by mechanical pumps without moment of inertia simulating no coast down characteristic of EMP during transient.

RHRS of PGSFR is composed of four circuits. Two for Active Decay heat Removal Circuits (ADRC) and others for Passive Decay heat Removal Circuits (PDRC). All of them are cooled by air. Decay Heat eXchanger (DHX) is submerged in the upper part of cold pool and outside of the redan structure that provides separation between cold and hot pool. Upper and lower cold pool was modeled with two volumes and connected with cross-flow junctions. Redan heat structure was also modeled to simulate heat transfer between hot and upper cold pool.

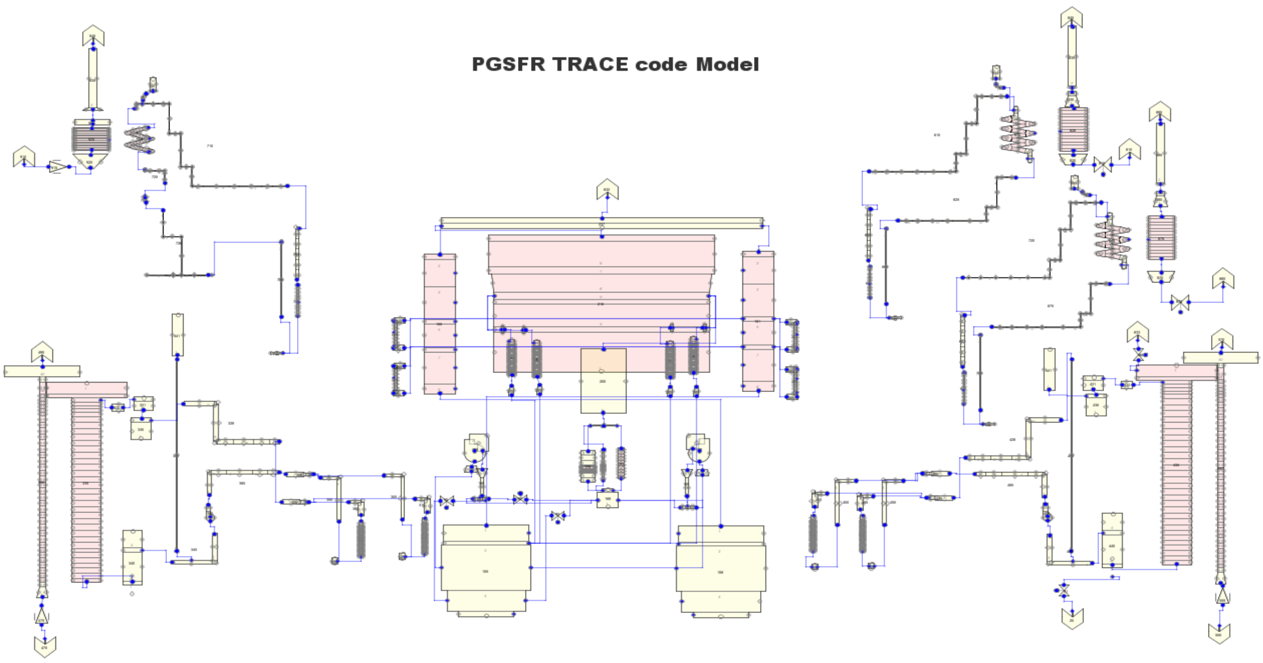

Fig. 1. TRACE nodding diagram for PGSFR

ADRC is configured with DHX, EM pump and blower for Forced Heat eXchanger (FHX). PDRC is composed with DHX and Air Heat eXchanger (AHX) and the air damper.

In modeling, four DHX, one ADRC and two PDRCs is modeled because one of active components is assumed inoperable during accident analysis. One DHX without connection with RHRS was assumed its heat removal capacity is maintained at normal heat transfer rate (0.295MWt)

PGSFR core is designed with 112 fuel drivers, 78 reflectors, 114 shields and 9 control rod assemblies. In modeling total 313 assemblies are categorized with average, hot and non-fuel divers. Non-fuel drivers are reflectors, shields and control rod assemblies and they do not have wire-wrapped pins within assemblies. TRACE code PGSFR modeling is as Fig. 1.

2.2 Simulated Steady State condition

Modeled steady state condition is 102% of power condition. The assembly wise coolant flow and power is considered in decision of flow and power of hot assemblies i.e. the hot assembly flow and power condition is chosen for highest lank in the power/flow fraction. Used power peaking fraction and flow for hot assembly was 1.6 and 24.56 kg/s. Reactor vessel surface heat loss was neglected in simulation.

Overall Plant Configuration and the designed normal operation condition (100% power) and the simulated 102% power are showed in Fig. 1. Due to the difference of core power simulated system conditions showed high IHTS and SG feed flow, core outlet and IHTS hot-leg temperature. But major parameters in transient condition such as core inlet temperature, PHTS pump flow was maintained as designed value.

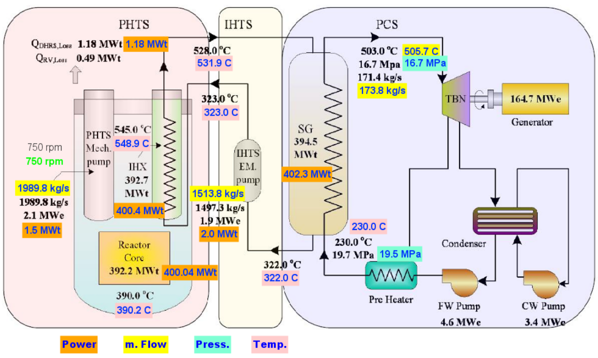

Fig. 1. Designed (100% power) and simulated (102%) PGSFR condition

3. Review on LOF Accident Scenario

Loss of Flow is defined as the loss of core cooling capacity due to a pumping failure of the primary pump. The cause of LOF is failure of mechanical PHTS pump or power supply loss resulted from loss of off-site power. Designer's representing initiating event for LOF was all of the PHTS pump failure. [2]

LOF accident is also assessed for Light Water Reactors (LWR) and Locked Rotor (LR) accident is also assessed as one of limiting case of LOF accident. LR accident is defined as PHTS pump rotor stuck accident. During LR accident, only one pump could be used to cool the core. Therefore LR case is also need to be assessed for PGSFR

Most important component for LOF accident is PHTS pump. Pump Modeling is based on the design specification. [3] Rated pump parameters were decided but the Momentum of Inertia (MOI) and homologue curve were not published yet. Current model uses DSFR-600 pump's MOI and homologue curve.

3.1 Loss of Flow Accident Simulation

LOF accident was simulated with the scenario proposed by the designer. At the beginning of the transients all of PHTS pumps were tripped for loss of PHTS power supply and stared coast down. At 1.69s, fraction of power to flow exceeded 111.7% of set-point and reactor was tripped. 5 seconds after reactor trip, intermediates pumps SG feed were tripped and RHRS is actuated at 5 s

Overall system heat removal is showed in Fig. 1. Core power was sharply decreased by reactor trip about 2s. Before SG water inventory was dried out near 200 s, Decay heat was removed by IHTS. After 16870s RHRS heat removal exceeded the decay heat.

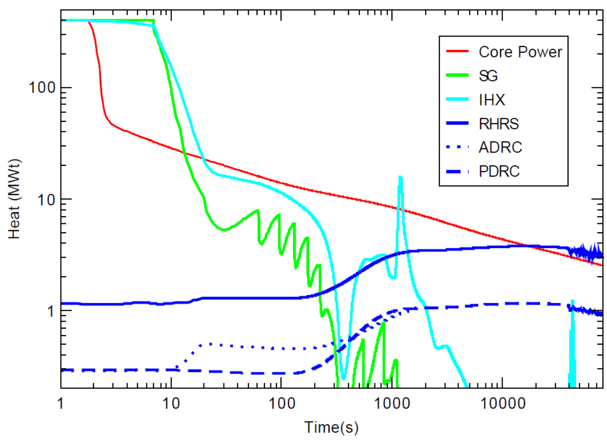

Fig. 2. Heat transfer response for the loss of flow accident case

Temperature response for LOF case showed that first temperature peak was occurred near reactor trip point, maximum fuel and inner surface temperature was 994.64K and 897.3K each. At second peak inner surface clad temperature was 894.7 K as shown in Fig. 3 Peak fuel and inner surface clad temperature was satisfied the design criteria.

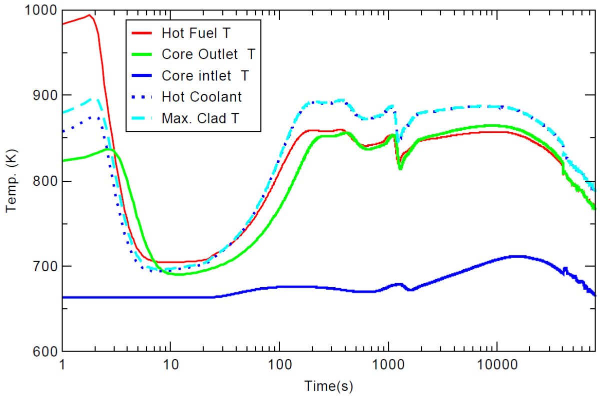

Fig. 3. Temperature Response for LOF Accident

3.2 Locked Rotor Case comparison

Locked Rotor accident is initiated with stuck one of PHTS pump rotor. As soon as one of pumps was stuck, power to flow fraction exceeded 111.7% of set-point at and reactor was tripped within 0.04 seconds. 5 seconds after reactor trip, intact pump, intermediate pumps and SG feed were tripped and RHRS is actuated at 5 s.

Due to rotor stuck, one PHTS pump supply about 52% of normal core flow into the core until the pump trip.

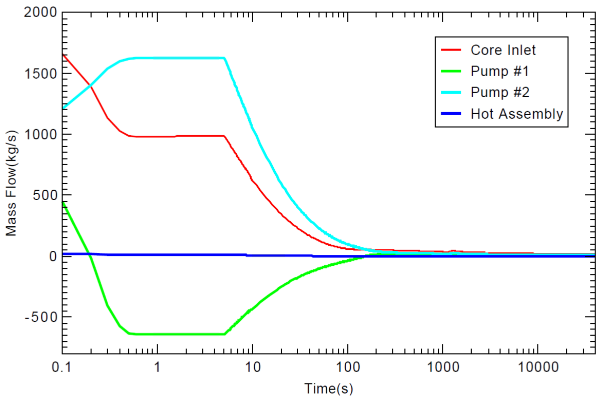

Fig. 4. Flow response for LR case

Clad Temperature estimation for LR case showed in Fig 5. First temperature peak was occurred with 893 K and second peak also occurred at 144s with 897.4K for clad inner surface, which is slightly higher than LOF case. First temperature peak is drop rapidly with the benefit of pump coast down. But second peak needed longer time to decrease.

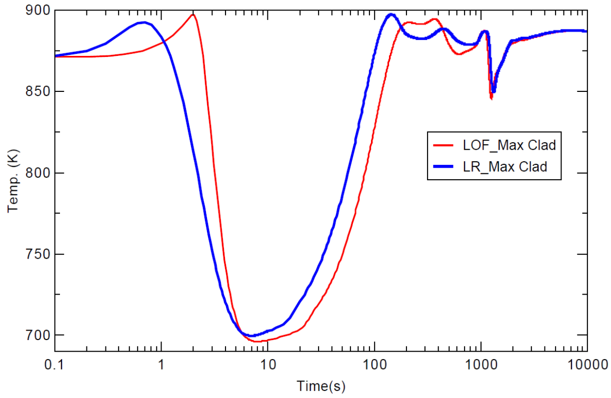

Fig. 5. Clad Temperature Comparison with LOF and LR case

3.3 Sensitivity Study for Locked Rotor Accident

LOF and LR case study showed that clad temperature peaks could be influenced assumed parameters such as pump coast down characteristics, reactor trip delay time, RHRS actuation time and etc.

Variance of Pump coast down characteristics in the Motion of Inertia (MOI) from 2000 to 6000, for reactor trip delay time up to 1.6s and for RHRS actuation time of 5 seconds to 30 minutes and hot assembly power peaking factor of 2.07234 cases were studied.

Sensitivity result for the fuel centerline and inner surface clad temperature is showed at Fig. 6

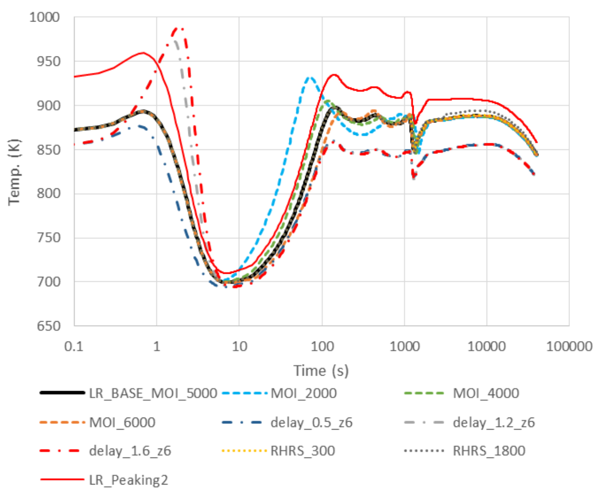

Fig. 6. Sensitivity Calculation for LR in MOI, Trip Delay and RHRS Actuation time

Result of sensitivity calculation for the locked rotor accident in terms of the motion of inertia, reactor trip delay, RHRS actuation time and hot assembly power peaking factor showed that the reactor trip delay is a major parameters for first clad peak and the low pump motion of inertia results in increase of second temperature peak and increased hot assembly peaking factor elevated clad temperature response during entire transient. And RHRS actuation time showed no impact on the temperature peak. First peak in occurred at the peak node and the second peak at slug top node.

3. Conclusions

For the preparation of the review of licensing application for PGSFR, TRACE model for the PGSFR was developed and the loss of flow accident was pre-calculated. The locked pump rotor case was also calculated as a possible bounding case for the loss of flow scenario. Pre-calculation showed that the locked rotor case was similar or worst case to the loss of flow accident. Therefore, the locked rotor case should take into account in design base accident assessment of PGSFR.

Sensitivity calculations for the rocked rotor case also studied for identification of unfixed design parameters influencing to estimation of inner surface temperature. Sensitivity result showed that the first temperature peak was largely influenced by reactor trip delay and second peak mostly influenced by pump coast down characteristic. Hot assembly power peaking determined by the nuclear design impacted both temperature peaks so it should be considered as one of major safety assessment parameter.

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REFERENCES

[1] TRACE V5.0 Theory Manuel, USNRC, 2010
[2] K.L. Lee et al, Analysis of design base events in the conceptual design of a PGSFR using MARS-LMR, KAER/TR-4971/2013, KAERI, 2013
[3] Training Program for SFR system, KINS/TR-254, Feb. 2014.