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Preliminary Assessment of Transient of Over Power Accident for DSFR-600

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1. Introduction

The conceptual design of Demonstration Sodium cooled Fast Reactor (DSFR) of 600MWe capacity has been done by KAERI. And Prototype Generation IV Sodium cooled Fast Reactor (PGSFR) of 150MWe is under development targeting licensing application by 2017.

One of activities of Development of Regulatory Audit Technology for System Safety of SRP is to prepare the audit calculation tool for safety analysis.

TRACE code [1] was selected as one of candidates for audit code, so sodium properties and heat transfer model in the code was verified first.[2] On the basis of MARS-LMR code input, DSFR-600 TRACE model was developed and applied to PHTS tube rupture case, one of design base events (DBE) of DSFR-600.[3]

In this study, Transients of Over Power (TOP) event is assessed using TRACE code as one another case of DBEs of DSFR-600 for preparation of audit calculation of PGSFR.

2. System modeling and the accident scenario

TRACE code model was developed with the reference of Conceptual Design Report [4] and Safety Evaluation Report [5] of DSFR.

Code model is composed of Primary Heat Transport System (PHTS), intermediate loops, steam generators and Residual Heat Removal Systems (RHRS). In PHTS, reactor core, 2-primary sodium pumps, 4-Intermediate Heat eXchangers (IHX) and 2-double capacity Decay Heat eXchangers (DHX) were modeled. Two intermediate loops are composed of 2 IHXs and Steam Generator (SG). RHRS was modeled with double capacity loops of Active Decay Heat Removal Circuit (ADRC) and Passive Decay Heat Removal Circuit (PDRC). Fig. 1 shows the nodding diagram of DSFR-600.

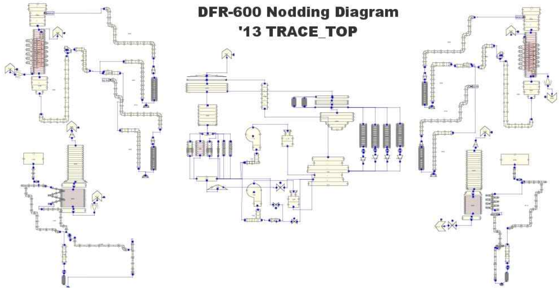

Fig. 1. TRACE code nodding diagram of DSFR-600

The calculated steady state conditions are compared with design conditions in Table 1. Major difference of the design and calculated condition is the primary and intermediate pumps heat. In modeling, these pumps heat was neglected, so the transported heat to SGs was lower than the design about 4MW each. RHRS heat removal was 4.9MW compared to the 4.7MW of the design to protect from coolant freezing in RHRS loops. DBA evaluation by KAERI did not consider the reactivity feedback of core and control rod thermal expansion. In TRACE code simulation also these reactivity feedbacks were omitted for conservative purpose.

Table I: St.-St condition comparison between design and calculation

	DESIGN K,MW/kg/s	TRACE prediction
CORE I/O T. Power/flow	638.15/783.15 1548.2/8366.1	638.15/784.31 1548.2/8366.1
IHX I/O T. Q/flow	578.55/775.15 387.5/3073	578.55/776.86 385.82/3073
SG Q/flow	775/344.7	771.58/338.64
DHX Q	4.7	4.9

TOP scenario starts with 30 cent reactivity insertion from 5 sec. to 20 sec. [5] Overall accident sequence and assumptions are as followings;

1) During 100% power operation, 30 cents reactivity is inserted during 15 seconds.
2) Reactor Power increase up to 111.7% of the high power set-point.
3) Reactor Trips after the signal with 0.6 sec. delay
4) Loss of offsite power occurs at 5 sec. after the reactor trip. PHTS/IHTS pumps are start to cost-down and SG feed isolation occurs
5) After 30 minutes, RHRS (2PDRC/ 1FDRC) are activated with air damper control

3. Calculation result and system response

Due to the inserted reactivity, reactor power increased up to 1733MWt from 5s until 9.9s. Then high power set-point of 111.7% full power activated the reactor trip. PHTS and intermediate pumps were still operating at the moment until loss of off-site power after reactor trip. So, reactor power and other system temperature decreased sharply with reactor scram. From 5s after the reactor tip, i.e. after 14.9s, PHTS, intermediate pumps and steam generator feed were started to decrease by loss of off-site power. Primary flow decreased to 140kg/s until 114s by primary pumps cost-down. Due to the decreased system flow to the minimum level, the system temperature increased. Finally, second peak of system temperature was appeared around 145s. System flow increased slightly up 177kg/s at that time and began to oscillate above 200kg/s. Until the depletion of SGs inventory, SG played role as a heat sink, the core inlet temperature showed dependency on the heat removal of SGs through intermediate loop.

In the transient calculation, SG dry-out occurred at 2371s. Before SG dry-out, primary coolant temperature increased by insufficient heat removal through SGs and RHRS. In 1810s, actuation of damper of two PDRCs and one ADRC supplied sufficient heat removal from the reactor pool. Fuel temperatures decreased by the end of calculation as Fig. 1.

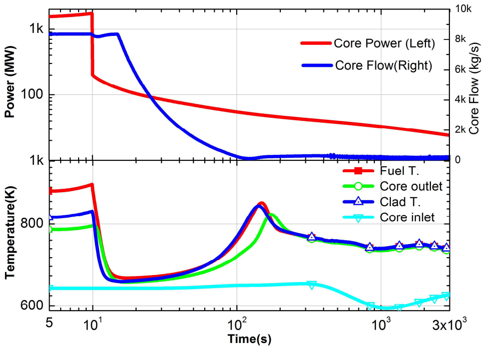

Fig. 1. DSFR response at transients of over power accident

Peak of fuel temperature was appeared at 9.92s, just before reactor trip. Calculated peak fuel temperature was 22K higher than the steady-state condition. Whereas TOP assessment using MARS-LMR showed that PCT was calculated to be 21K higher than the steady-state temperature. [4] TRACE code prediction of peak fuel temperature showed reasonable value. However in second peak temperature of fuel and clad between two codes, TRACE code calculated 817.82K, which is higher than MARS-LMR code value of 750.15K. This difference mainly came from heat removal of SGs during transients, i.e. inventory of SGs and natural circulation characteristics during transient. Therefore, SG inventory during normal operation and natural circulation characteristics of reactor pool should take into account in DBE analysis.

4. Pre-calculation of UTOP

One of Beyond Design Base Events (BDBE), Unprotected Transients of Over Power (UTOP) was pre-calculated and compared with the above TOP case. Major difference between TOP and UTOP is whether the reactor protection system is activated or not during transients. In UTOP case, reactor does not scrammed by trip signals such as over power or over temperatures etc.

Conservative assumption was also used in this BDBE pre-calculation, in DBE calculations only the reactivity feedbacks by the doppler and coolant density are used without feedback from the core radial, fuel and control rod axial expansion. Those feedbacks are known as largest negative feedback in SFR. In Fig. 2, the reactor power of UTOP case increased up to the unaffordable condition without additional negative feedback.

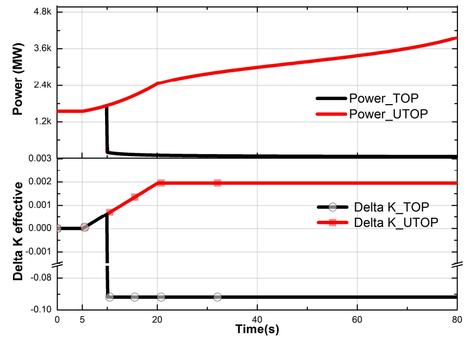

Fig. 2. Calculated power comparison in TOP and UTOP case without core axial and control rod reactivity feedback

The above pre-calculation implies that the assumption of DBE assessment without core radial, fuel and control rod axial thermal expansion reactivity feedback is conservative and these feedbacks will play important role in BDBE assessment.

5. Conclusions

One of the design base events, transients of over power of Demonstration Sodium cooled Fast Reactor was simulated using TRACE code.

Predicted fuel temperature showed that the peak fuel temperature occurs when the reactor scrammed and predicted temperature was similar to the MARS-LMRs assessment by KAERI. In this study, it is found that the second peak of fuel temperature is influenced by the inventory of steam generator and the natural circulation characteristic of the reactor vessel pool. Pre-calculation of the unprotected transients of over power with conservative reactivity assumption showed that this assumption is conservative in design base even assessment. However the method of measurement and applying the core radial, fuel and control rod axial expansion reactivity feedback is crucial in BDBE assessment of SFR.

REFERENCES

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