CANDU 6 Small Break LOCA with High Power Channel of 5% or More Pressure Tube Creep

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1. Introduction

There are some aging phenomena which affect safety analysis in CANDU Nuclear Power Plant(NPP). These aging phenomena should be incorporated into thermal-hydraulic models of a system code like CATHENA(Canadian Algorithm for THErmal-hydraulic Network Analysis) for safety analysis. For example, roughness change of inlet/outlet feeders and inlet end fittings, inlet feeder orifice degradation, pressure tube(PT) creep, changes of boiler tube inside diameter and roughness, and PHT pump performance change are those aging phenomena in CANDU NPPs. Among them, PT creep is an important factor which slowly reduce operating power level of CANDU NPP in relation to reactor trip setpoint decrease of ROPS(Regional Overpower Protection System).

Generally, as PT creep increases, higher fuel sheath temperature is expected as a result of early fuel sheath heatup in an accident condition because of CHF(Critical Heat Flux) decrease. Slowly progressing accident like small break LOCA(Loss Of Coolant Accident) is affected the most by CHF decrease due to PT creep increase.

Recently, it was predicted fast progression of PT creep in some domestic CANDU 6 NPPs. In the past, PT creep was considered up to 5% in safety analysis. So, it is needed to consider more PT creep than 5% in safety analysis. In the present paper, some small break LOCA cases were analyzed assuming 5% or more PT creep in order to check its effect on fuel sheath heatup in an accident condition.

2. Analysis Methods

In small break LOCA cases considered in FSAR(Final Safety Analysis Report) Chap. 15 of domestic CANDU 6 NPPs, 2.5% RIH(Reactor Inlet Header) break is the severest break size that resulted in the highest fuel sheath temperature. The highest fuel sheath temperature is calculated by a high power channel named by O6mod. O6mod is a modified O6 channel of which channel power and bundle powers at 6th and 7th bundle locations are set to their LCOs(Limiting Condition for Operation), 7.3 MW and 935 kW, respectively. So, only O6mod channel in 2.5% RIH break case need to be analyzed in the present paper.

2.1 PT Creep Data Preparation

Bundle-wise PT creep data are directly inputted into CATHENA code[1,2]. 5% bundle-wise creep data of a O6mod channel were obtained from CATHENA models used in Wolsong NPP Unit 2/3/4 safety analysis. Then, the data were modified artificially to get 5.25%, 5.5%, 5.75%, 6.0%, 6.25%, 6.5% bundle-wise creep data without any physical considerations on PT creep phenomena in CANDU reactor, because it is a simple way of considering PT creep effect on safety analysis. Then the PT creep data were inputted into O6mod CAHTENA input files in order to make seven case decks. Like this, totally seven different PT creep cases were prepared.

2.2 CHF Correlation

CHF correlation of 37-element fuel used in domestic CANDU 6 NPP safety analysis can be applied up to 5.1% PT creep condition, because experimental data were obtained only up to 5.1% PT creep. However, in recent years, CHF experiments were performed up to 6.8% PT creep condition for the modified 37-element(37M) fuel, and a CHF correlation was also developed based on the experimental data[3]. So, the new CHF correlation which can be applied up to 6.8% PT creep condition was used in this study

2.3 CATHENA Models

In the present study, generic CATHENA models for domestic CANDU 6 NPPs were used, which reflected some aged PHTS(Primary Heat Transport System) conditions. The high power channel, O6mod used a minimum developing PDO(Post-Dryout) heat transfer correlation(DEV-PDO-7) for fuel sheath temperature calculation.

3. Analysis Results

3.1 Whole System Bahavior

For the whole system behavior, only one case of 2.5% RIH break case was analyzed. The event sequence was summarized in Table 1. At 44 seconds, the reactor was tripped by PHTS low pressure signal of SDS1, which is the second trip signal. LOCA signal occurred at 74 seconds, and ECC(Emergency Core

Cooling) injection started to broken loop at 129 seconds. The PHT pumps were tripped at 273 seconds.

Table 1 Event Sequence

Event	Time(sec)
Break occurs	0
High reactor building pressure signal (1st reactor trip signal)	9
Reactor trip signal(Low HTS pressure)	44
LOCA signal(5.25 MPa)	74
Loop isolation completed	94
Steam generator crash cooldown	104
ECC injection to broken loop	129
Automatic PHT pump trip signal	153
Broken loop refilled	213
ECC injection to intact loop begins	214
PHT pump trip	273
Intact loop refilled	271
Medium pressure ECC injection starts	988

3.2 High Power Channel Behavior

For checking the effect of PT creep on fuel sheath temperature in Small Break LOCA, totally seven cases with different maximum PT creeps were analyzed. Maximum fuel sheath temperatures for the cases were depicted in Figure 1. Before reactor trip, there was fuel sheath heatup over 500℃. As PT creep increases, intermittent sheath heatup occurred in a range of 5.0 ~ 6.0 PT creep condition, and then maximum fuel sheath temperature was continuously over 450℃ in a range of 6.0~0.6% PT creep condition. After the PHT pump trip, fuel sheath temperature was temporarily increased over 350℃ because of low channel flow. In the seven cases with different maximum creep, maximum sheath temperature behaviors were similar with each other after the reactor trip. There were only small differences in maximum sheath temperature between 300 and 400 seconds.

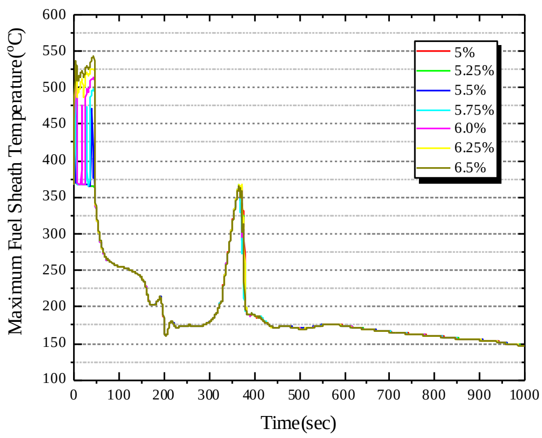

Figure 1. Maximum Fuel Sheath Temperature

4. Conclusions

It was expected that fuel sheath might heat up more if PT creep increased regardless of reactor trip in a case of small break LOCA. But, after the reactor trip, there were no big difference in maximum sheath temperature in the seven cases of O6mod channels with 5% or more PT creep. There seems to be sheath dryout before the reactor trip in the cases of PT creep over 6%. Thus, one should be careful that initial conditions in Small Break LOCA analysis will not have fuel sheath dryout.

REFERENCES

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