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Thermal-Hydraulics Characteristics during LOCA Transient for a CANDU-6 with CANFLEX-RU FUEL

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ABSTRACT

The thermal-hydraulics characteristics during the large loss-of-coolant accident (LOCA) have been analyzed for a CANDU-6 core fuelled fully with a CANFLEX-RU (Recovered Uranium) fuel bundle. The calculations were performed for three typical breaks: 55% pump suction break, 35% reactor inlet header break and 100% reactor outlet header break. The simulations for the thermal-hydraulics phenomena are done using the CATHENA code with a coupling of the physics code RFSP. From the simulation results, it is known that the power transient is largest for the 100% reactor outlet header break.

1. INTRODUCTION

As an advanced fuel for the Canada deuterium uranium (CANDU) reactor, a recovered uranium (RU) fuel has been developed. The carrier for the advanced fuel is CANDU flexible fueling (CANFLEX) fuel bundle. The nominal uranium enrichment of RU fuel is 0.92 wt% of U-235. Table I compares the channel parameters between standard 37-element fuel and CANFLEX-RU fuel bundle.

In this study, the thermal-hydraulics characteristics are analyzed during a large loss-of-coolant accident (LOCA) in a CANDU-6 core fuelled fully with a CANFLEX-RU fuel bundle. The coolant voiding after a large break in a primary circuit pipe increases rapidly. This is due to the loss of inventory, depressurization and increased boiling in the fuel channels due to a degraded fuel cooling. The decrease in coolant density will be most pronounced in the fuel channels of the broken loop downstream of the break. Coolant voiding in the core will introduce positive reactivity at a rate and depth for which the reactor regulating system could not compensate. This will lead to an increase in the reactor power.

The main thermal-hydraulics parameters such as coolant density, coolant flow, coolant temperature and loop powers are analyzed. Also, the fuel integrity threatened by the power pulse is assessed based on the power pulse.

2. CALCULATION PROCEDURE

A thermal hydraulic/reactor physics joint simulation for the safety analysis was performed with a thermal-hydraulic code CATHENA [1] and physics code RFSP. [2] The simulations were conducted for the cases such as the 55% pump suction break (PSB), 35% reactor inlet header (RIH) break and 100% reactor outlet header (ROH) break.

2.1 Thermal-hydraulic Modeling

The heat transport system model is based on the Wolsong 2/3/4/model [3]. The fuel channel and the associated heat structure model are changed for the Wolsong 2/3/4 model to consider the real geometry of the CANFLEX-RU fuel bundle. Since the power pulse mainly depends on the voiding rate of the channels downstream of the break (critical core pass), these channels are modeled in more detail than the others.

A two-loop network model of the heat transport system is used in the analysis. The core pass downstream of the break (critical pass) was modeled as 7 average channels with different powers, channel elevations and header/feeder connection elevations as follows (Fig. 1). The return pass of the broken loop (95 channels) is represented by channel group 3. The passes in the intact loop are represented by channel groups 1 and 2. The core region in each average channel is represented by 12 nodes. This is done to ensure sufficient accuracy in the prediction of the coolant density in the core region. Fuel dryout is prohibited during the transient to be conservative.

2.2 Reactor Core Conditions

For the conservative core conditions, the minimum allowable performance specification (MAPS) conditions are considered as follows:

- Crept pressure tube (2.5% creep in diameter),
- Coolant purity is 99.0 %,
- Startup the reactor from long shutdown during 12 hours, in which 1.8 ppm of boron is needed to maintain the criticality,

- The 8% side-to-side power tilt is supposed.

The detailed reactor physics conditions are described in Ref. 4.

3. RESULTS AND DISCUSSION

For the PSB55, the coolant density, coolant flow and fuel temperature of the critical pass of the broken loop (passes 4-1 to 4-7) are shown in Figs. 2, 3 and 4, respectively. For channel groups 1 to 3, the coolant density was practically unchanged. For the coolant flow of the critical pass, flow stagnation occurred shortly after the break, which was typical for the low flow rate, leading to a larger power pulse. The channel void fraction transient for the critical pass was consistent with the coolant density and flow. In case of the PSB55, the depressurization progresses very fast because the header is located nearest to the broken pump suction. Though the coolant temperature increased slowly, the average fuel temperature of the critical pass increased rapidly between 1.0 sec and 1.7 sec.

The thermal-hydraulic behaviors of the RIH35 are similar to those of the PSB55 because the coolant is not normally fed in both cases. The coolant density, channel flow and the fuel temperature of the critical pass of the RIH435 are shown in Figs. 5, 6 and 7, respectively. The fuel temperature increases slightly earlier compared to the PSB55

For the channel coolant density of the ROH100 shown in Fig. 8, the transient behavior is very similar to the RIH35, except that the rapid boiling of the coolant in the critical pass is delayed by ~0.5 sec. This delay was mainly caused by the fact that the break location of the ROH100 was further away from the critical core pass than the RIH35 break location was. In case of the ROH100, the onset of rapid coolant boiling occurs in the critical pass. However the average fuel temperature, shown in Fig. 9, increases rapidly after 1.0 sec, changes slowly after ~2 sec.

The total energy deposition up to 3 sec is shown in Table II. The total energy added to the bundle was 5.70 MWsec for the PSB55, 5.72 MWsec for the RIH35, and 7.20 MWsec for the ROH100, respectively. This is equivalent to an energy content of 604, 605 and 698 J/g for the hot pin of a bundle initially at the licensing limit of 935kW. These values are 236, 235, and 142 J/g below the conservative limit (840 J/g) for the fuel breakup and correspond to margins of 28%, 28% and 17% to the fuel breakup, respectively.

4. SUMMARY

For the CANFLEX-RU fuelled CANDU reactor, the thermal-hydraulic characteristics have been investigated during the large LOCA. The calculations were performed for three typical breaks: 55% pump suction break, 35% reactor inlet header break and 100% reactor outlet header break. The main results and conclusions of the analysis are summarized as follows:

- In the 55% PSB, the coolant density and the fuel temperature increase rapidly after the break, while the coolant temperature increases slowly.
- The thermal hydraulic characteristics of the 35% RIH break are similar to those of the 55% PSB.
- The boiling of the 100% ROH break is delayed by 0.3~0.4 sec, which the largest power pulse compared to the other breaks.
- For the 5th bundle in the M-4 channel, the maximum value of total energy deposition up to 3 sec was 698.4 J/g, which has ~17% margin for the fuel break up threshold.

From the above results, it is expected that there is no fuel break up during the LOCA transient in a CANDU-6 reactor with CANFLEX-RU fuel.

ACKNOWLEDGEMENT

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3. S. Lam et al., "Analysis Report: Large Loss of Coolant Accident (Large LOCA) Wolsong NPP 2, 3, 4", 86-03500-AR-029, Rev. 1, (1995).

Table I Comparison of Channel Parameters between 37-Element and CANFLEX-RU Bundle

Parameters	Standard 37-element Bundle	CANFLEX-RU Bundle
Element Number	37	43
Sheath Radius (mm)	6.55	6.75(large) 5.75(small)
Sheath Thickness (mm)	0.4	0.39(large) 0.36(small)
Pellet Radius (mm)	6.1	6.335(large) 5.365(small)
Pressure Tube Average Inner Radius (mm)	51.7	51.7
Pressure Tube Average Thickness (mm)	4.343	4.343
Calandria Tube Average Inner Radius (mm)	64.5	64.5
Calandria Tube Average Thickness (mm)	1.397	1.397
Pitch Circle Radius (mm) for : Outer Elements Intermediate Elements Inner Elements	 43.31 27.53 14.88	 43.84 30.75 17.34

Table II Margin to Fuel Breakup Threshold

Break	Initial-Power Seconds (MW.s)	Pulse Energy (J/g)	Total Energy Deposition (J/g)	Margin to Breakup (%)
55% PSB	6.922	358.0	604.4	28.0
35% RIH	6.943	359.0	605.4	27.9
100% ROH	8.739	452.0	698.4	16.9

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19	20	21	22
A									8	3	8	1	2	1								
B						8	3	8	3	8	3	2	1	2	1	2	1					
C					8	3	8	3	8	3	8	1	2	1	2	1	2	1				
D				8	3	8	3	8	3	4	3	2	1	2	1	2	1	2	1			
E			8	3	8	3	8	3	4	3	5	1	2	1	2	1	2	1	2	1		
F			3	8	3	4	3	4	3	4	3	2	1	2	1	2	1	2	1	2		
G		3	9	3	9	3	4	3	4	3	4	1	2	1	2	1	2	1	2	1	1	
H		9	3	9	3	4	3	4	3	5	3	2	1	2	1	2	1	2	1	2	2	
J	9	3	9	3	5	3	5	3	5	3	4	1	2	1	2	1	2	1	2	1	1	1
K	3	9	3	5	3	5	3	5	3	5	3	2	1	2	1	2	1	2	1	2	2	2
L	9	3	9	3	4	3	4	3	5	3	4	1	2	1	2	1	2	1	2	1	1	1
M	3	9	3	4	3	6	3	6	3	7	3	2	1	2	1	2	1	2	1	2	2	2
N	9	3	9	3	6	3	6	3	7	3	7	1	2	1	2	1	2	1	2	1	1	1
O	3	9	3	6	3	6	3	6	3	7	3	2	1	2	1	2	1	2	1	2	2	2
P		3	9	3	9	3	9	3	6	3	7	1	2	1	2	1	2	1	2	1	1	
Q		10	3	9	3	6	3	6	3	7	3	2	1	2	1	2	1	2	1	2	2	
R			10	3	10	3	10	3	6	3	7	1	2	1	2	1	2	1	2	1		
S			3	10	3	10	3	10	3	7	3	2	1	2	1	2	1	2	1	2		
T				3	10	3	10	3	6	6	7	1	2	1	2	1	2	1	2			
U					3	10	3	10	3	10	3	2	1	2	1	2	1	2				
V						3	10	3	10	3	10	1	2	1	2	1	2					
W									3	10	3	2	1	2								

Group 1 : Core pass 1 (loop 1)

Group 1 : Core pass 2 (loop 1)

Group 1 : Core pass 3 (loop 2)

Group 4 to 10 : Core pass 4 (loop 2)

Fig. 1 Channel Grouping for Whole Core

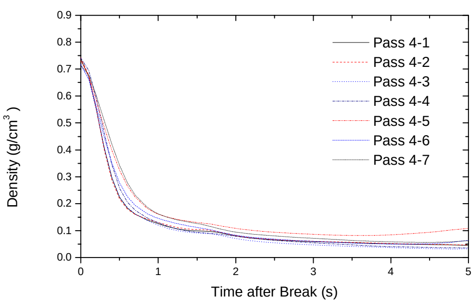

Fig.2 Channel Coolant Density in Channel Groups 4 to 10 (55% PSB)

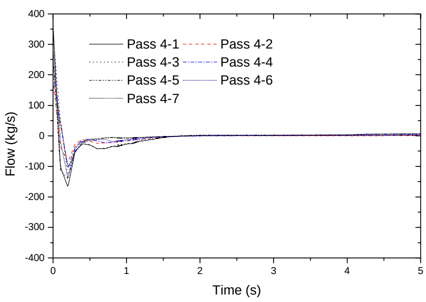

Fig.3 Channel Coolant Flow in Channel Groups 4 to 10 (55% PSB)

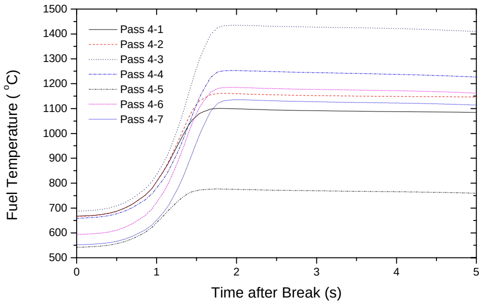

Fig.4 Channel Average Fuel Temperature in Channel Groups 4 to 10 (55% PSB)

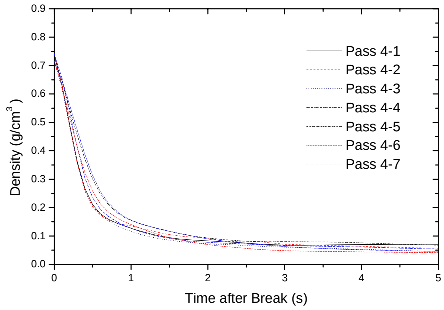

Fig.5 Channel Coolant Density in Channel Groups 4 to 10 (35% RIH)

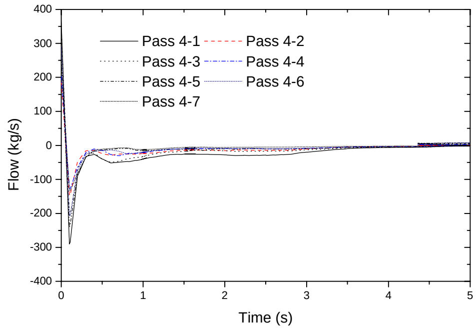

Fig.6 Channel Coolant Flow in Channel Groups 4 to 10 (35% RIH)

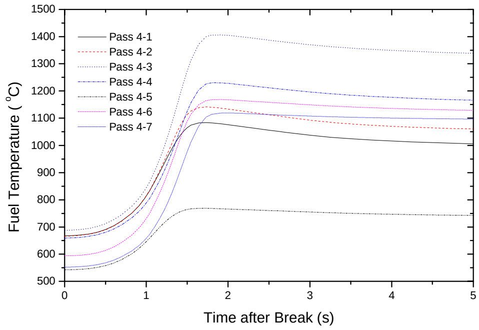

Fig.7 Channel Average Fuel Temperature in Channel Groups 4 to 10 (35% RIH)

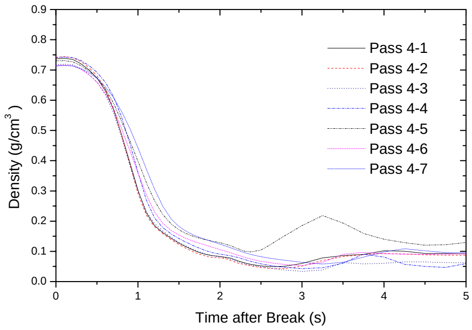

Fig.8 Channel Coolant Density in Channel Groups 4 to 10 (100% ROH)

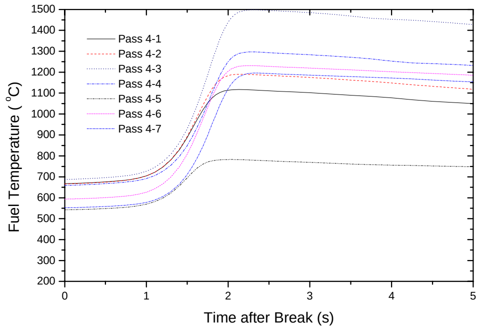

Fig.9 Channel Average Fuel Temperature in Channel Groups 4 to 10 (100% ROH)