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LOCA POWER PULSE CHARACTERISTICS in a CANDU-6 with CANFLEX-RU FUEL

Chang Joon Jeong, Bo Wook Rhee, Jong Yeop Jung and Ho Chun Suk
Korea Atomic Energy Research Institute
P.O. Box 105, Yuseong, Daejeon Korea, 305-600

ABSTRACT

The power pulses following a large loss-of-coolant accident 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: 100% pump suction break, 40% reactor inlet header break and 100% reactor outlet header break. The coupled simulations for the reactor physics and thermal-hydraulics phenomena are done using the RFSP and CATHENA codes. The power transient is largest for the 100% reactor outlet header break. The channel power transient at 3 seconds after breaks will be used for further upstream analysis and fuel breakup margin calculations.

1. INTRODUCTION

Following a hypothetical large break in a primary circuit pipe, the heat transport system will begin to void 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. A primary consideration in the design of the two shutdown systems is the rapid detection of such a power increase and the timely insertion of a negative reactivity to prevent excessive stored energy in the fuel.

A first step in the comprehensive analysis of the consequences of large breaks is the determination of the expected reactor power transients experienced by the fuel. Such a study involves detailed modeling of both the reactor physics and thermal- hydraulics phenomena. These are coupled together since coolant voiding determines the rate of the power increase, while the power increase has some feedback (although limited) on the voiding rate as well as the fuel temperature. In the present analysis, the physics code RFSP [1] is coupled with the thermal-hydraulics code CATHENA [2].

The joint simulations were performed in a series of time steps with the coolant densities calculated by the CATHENA code used as inputs to the CERBERUS module in the RFSP which calculated the flux and power distributions in the core. Bundle powers were then fed back to the thermal-hydraulics calculation. To account for the spatial and power effects on the voiding, fuel channels in the network thermal-hydraulics model were subdivided into ten different groups (or types): seven groups in the critical core pass, and one group each for the remaining pass of the broken loop and both the passes of the intact loop.

The main objective of the present analysis is to determine the reactor trip time and the power transient during a large loss-of-coolant accident (LOCA) for a CANDU-6 core fuelled fully with a CANFLEX-RU (Recovered Uranium) fuel bundle. The fuel integrity threatened by the power pulse is assessed based on these results. Table 1 compares the channel parameters between standard 37-element fuel and CANFLEX-RU fuel bundle.

2. ANALYSIS METHODOLOGY AND ASSUMPTIONS

This analysis involves the determination of the spatial power transients following a large LOCA using a CANTHENA two-loop network model coupled to a CERBERUS module in the RFSP. The coupling is done using the power generation, coolant density, coolant temperature and fuel temperature in each CATHENA core node. The neutronic flux shape is calculated using a CERBERUS module in the RFSP at time intervals $\Delta\tau$. For a given time interval $\Delta\tau$, the CERBERUS module in the RFSP uses the core coolant densities, coolant temperatures and fuel temperatures produced by the CATHENA code and determines the 3-dimensional power generation in the core at the end of the interval. CATHENA uses this power distribution to calculate the core node coolant densities, coolant temperatures and fuel temperatures for the next time interval.

2.1 Thermalhydraulic 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 Physics Modeling

The standard RFSP W2/3/4 model is used in this study. Lattice properties are calculated with the WIMS-AECL[4] code, which generates the fuel and reflector cross sections. The model includes adjuster rods and zone-control compartments as well as all the shutoff rods (initially parked out of core).

The SIMULATE module in the RFSP simulates a power burnup history starting from a known initial condition and proceeds for a given number of time steps. This module is used for an equilibrium core model to determine the minimum allowable performance specification (MAPS) conditions which are 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.

From this configuration, the LOCA is assumed to occur and the CERBERUS module in the RFSP is used to simulate the transient. The detectors and shutoff rods are modeled to be the same as those in Ref. 3.

3. RESULTS OF LOCA THE TRANSIENT SIMULATIONS

Three breaks are simulated: a 55% pump suction break (PSB), a 35% reactor inlet header (RIH), and a 100% reactor outlet header (ROH) break. These breaks are found to be the worst case in each break location from the previous study of a 37-element standard bundle [5].

3.1 Thermal-hydraulic Behavior

The first trip was the high neutron power trip for all the cases. The backup trip was the high rate log neutron power trip. The time of actuation of the SDS1 is shown in Table 2. The shortest trip time is 0.380 s for the 35% RIH break.

The most important thermal-hydraulic parameter which affects the power pulse is the coolant density (mass) in the core during the transient. Figures 2 and 3 show the channel coolant density transient for the 35% RIH break. The most important time interval which affected the power pulse was from 0 to 1 second, since the reactor trip occurred within this time period. It can be seen that only the coolant densities in the channel groups 4 to 10 (i.e., critical core pass) were important, since the densities in the channel groups 1 to 3 were practically unchanged. Due to a greater boiling, the coolant densities in the high power channel groups (e.g. 4 to 7) were smaller than those in the low power groups (e.g., 8 to 10). Flow stagnation occurs shortly after the break. In general, the closer these flows were to zero the smaller the density which was leads to a larger power pulse.

The other thermal-hydraulic parameters which affected the power pulse were the fuel temperature and the core coolant temperature. Generally, both the fuel and coolant temperatures increased slowly. Their effect on the power pulse was much smaller than that of coolant density.

3.2 Neutronic Behavior

Table 3 shows the peak relative powers for the three transients. The maximum value is of the peak relative bundle and channel powers occurred in the 100% ROH. Also, the 100% ROH break gave the highest peak relative total core power (4.896 at 1.897 s).The peak reactivity for the transients and the time to reach the peak reactivity are shown in Table 4. The 100% ROH gave the highest peak reactivity (4.571 mk at 1.489 s).

Figure 4 compares the total reactor power transients for the three breaks. The maximum value of the peak relative core power, which is occurred in the 100% ROH, is 4.896 at 1.897 s. For the 55% PSB and 35% RIH, the peak relative powers are 3.661 (at

1.199 s) and 3.663 (at 1.165 s), respectively.

3.3 Fuel Integrity

Table 5 shows the total energy deposition up to 3 sec. Among the power pulses as shown in the present study, the highest hypothetical value of energy that could be deposited at 3 seconds from a hot pin of a 935 kW bundle was found in the case of the 100% ROH break. The total energy deposition of fuel up to 3 sec in that case was 698.4 J/g, which is lower than the fuel break up threshold of 840 J/g.

4. SUMMARY

Coupled reactor physics/thermal-hydraulics simulations of hypothetical large-loss-of-coolant accidents, and terminated by shutdown system 1 using the backup trip signal, have been carried out. Various LOCA transients were simulated, assuming different locations and sizes of the breaks in the primary heat-transport system.

The main results and conclusions of the analysis are summarized as follows:

- The 100% ROH break gave the largest power pulse among those studied with different break sizes and locations.
- The highest peak reactivity of 4.571 mk (at 1.489 s) was reached in the 100% ROH break.
- 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 a 16.9% margin for the fuel break up threshold of 840 J/g.

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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TABLE 1 CHANNEL PARAMETER COMPARISON BETWEEN STANDARD 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	43.31	43.84
Intermediate Elements	27.53	30.75
Inner Elements	14.88	17.34

TABLE 2 TIMES OF ACTUATION OF SHUTDOWN SYSTEM 1
FOR VARIOUS TRANSIENTS

Break	Nature of SDS-1 Backup Trip	Time of Actuation of SDS-1 (s after break)	Time Used as Origin of Shutoff-Rod-Drop Curve* (s after break)
55% PSB	Rate-of-log-power	0.415	0.431
35% RIH	Rate-of-log-power	0.380	0.396
100% ROH	Rate-of-log-power	0.962	0.978

* 16 ms is added to consider the current to clutch cutoff.

TABLE 3 COMPARISON OF PEAK RELATIVE POWERS FOR VARIOUS TRANSIENTS

Break	Peak Relative Bundle Power	Peak Relative Channel Power	Peak Relative Full-Core Power	Peak Relative Broken-Loop Power
55% PSB	6.049 (M4/5)	5.523 (M4)	3.660	4.378
35% RIH	6.116 (M4/5)	5.577 (H7)	3.663	4.386
100% ROH	8.935 (M4/5)	8.065 (H7)	4.896	6.081

TABLE 4 PEAK REACTIVITY AND TIME AT WHICH PEAK REACTIVITY IS REACHED FOR VARIOUS TRANSIENTS

Break	Time to Reach Peak Reactivity(s)	Peak Reactivity (mk)
55% PS	0.942	4.146
35% RIH	0.908	4.138
100% ROH	1.489	4.571

TABLE 5 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

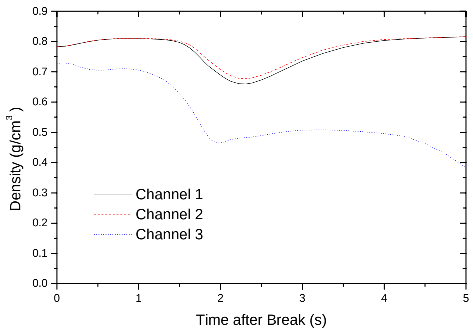

Fig.2 Channel Coolant Density in Channel Groups 1 to 3 (100% ROH)

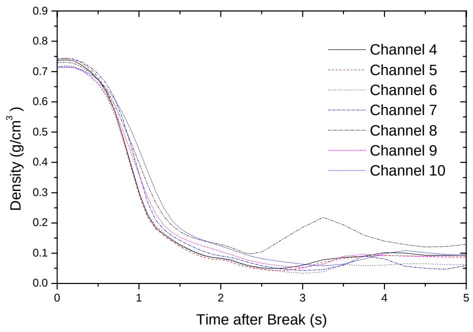

Fig.3 Channel Coolant Density in Channel Groups 4 to 10 (100% ROH)

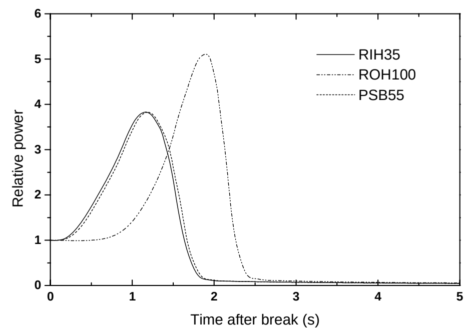

Fig.4 Total Reactor Power Transients after Break