Proceedings of Korean Nuclear Society Autumn Meeting
Seoul, Korea, October 2001

POWER PULSE FOLLOWING A LARGE LOCA IN CANDU-6 WITH DUPIC FUEL

Chang Joon Jeong and Bo Wook Rhee
Korea Atomic Energy Research Institute
P.O. Box 105, Yusong, Taejon Korea, 305-600

ABSTRACT

The power pulses following a large loss-of-coolant accident have been analysed for CANDU-6 core fuelled fully with DUPIC 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 simulation for reactor physics and thermalhydraulics phenomena are done using RFSP and CATHENA codes. The power transient is largest for the 40% reactor inlet header break. The channel power transient at 3 seconds after breaks will be used for further upstream analysis and fuel breakup margin calculation.

1. INTRODUCTION

Following a hypothetical large break in a primary circuit pipe, the heat transport system would begin to void rapidly. This is due to the loss of inventory, depressurization and increased boiling in fuel channels due to degraded fuel cooling. The decrease in coolant density would be most pronounced in fuel channels of the broken loop downstream of the break. Coolant voiding in the core would introduce positive reactivity at a rate and depth for which the reactor regulating system could not compensate. This would lead to an increase in 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 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 modelling of both the reactor physics and thermal-hydraulics phenomena. These are coupled since coolant voiding determines the rate of power increase, while the power increase has some feedback (although limited) on the voiding rate as well as feedback and 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 input to the CERBERUS module in 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 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 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 CANDU-6 core fuelled fully with DUPIC fuel bundle. The fuel integrity threatened by power pulse is assessed based on these results. Table 1 compares the channel parameters between standard 37-element fuel and DUPIC fuel bundle.

2. ANALYSIS METHODOLOGY AND ASSUMPTIONS

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

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 form Wolsong 2/3/4 to consider the real geometry of the DUPIC 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 modelled 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 modelled 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 coolant density in the core region. Flow resistance in the fuel channel is increased from 10.33 for a standard 37-rod bundle to 14.3 for CANFLEX fuel bundle. The fuel dryout is prohibited during the transient to be conservative.

2.2 Reactor Physics Modelling

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

The SIMULATE module in RFSP simulates a power burnup history starting from a known initial condition and proceeding 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 8 hours, in which 2.4 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 RFSP is used to simulate the transient. The detectors and shutoff rods are modelled the same as in Ref. 3.

3. RESULTS OF LOCA TRANSIENT SIMULATIONS

Three breaks are simulated: a 100% pump suction break (PSB), 40% reactor inlet header (RIH), and 100% reactor outlet header (ROH) break. These breaks are found to be the worst case in each break location from the previous study for a 37-element standard bundle [5].

3.1 Thermalhydraulic Behaviour

The first trip was the high neutron power trip for all cases. The backup trip was the high rate log neutron power trip. The time of actuation of SDS1 is shown in Table 2. The shortest trip time is 0.318 s for 40% RIH break.

The most important thermalhydraulic 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 40% RIH break. The most important time interval which affected the power pulse was from 0 to 1 seconds, since the reactor trip occurred within this time period. It can be seen that only the coolant densities in channel groups 4 to 10 (i.e., critical core pass) were important, since the densities in channel groups 1 to 3 were practically unchanged. Due to 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, leading to a larger power pulse.

The other thermalhydraulic 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 Behaviour

Table 3 lists the peak relative powers for three transients. The maximum value of the peak relative bundle and channel powers occurred in 40% RIH. The peak reactivity for the transients and the time of reaching the peak reactivity are shown in Table 4. The 40% RIH gave the highest peak reactivity (3.826 mk at 0.845 s), also the 40% RIH break gave the highest peak relative neutronic power (2.993 at 1.08 s).

Figure 4 compares the total reactor power transients for three breaks. The peak value of the relative power is 2.993 at 1.08 s. For 100% PSB and 100% ROH, the peak relative powers are 2.916 (at 1.08 s) and 2.503 (at 1.245 s), respectively.

3.3 Fuel Integrity

The energy deposited in the fuel by the power pulse was computed by integrating the bundle power to 3 seconds after the break. This was done for each of the 4560 bundles in the core. Table 5 shows the maximum time integrated bundle power up to 3 s. Among the power pulses of the present study, the highest hypothetical value of energy that could be deposited at 3 seconds in the hot pin of a 935 kW bundle was found in the case of a 40% RIH break. The maximum integrated bundle power to 3 s in that case was 3.942 MJ, corresponding to 4.959 initial-power seconds.

4. SUMMARY

Coupled reactor physics/thermalhydraulics 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 breaks in the primary heat-transport system.

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

- The 40% reactor inlet header break gave the largest power pulse among those studied with different break sizes and locations.
- A highest peak reactivity of 3.83 mk was reached in the 40% reactor inlet header break.
- For the 4th bundle in the H-7 channel, the peak bundle power and the time integrated bundle power (over 3 seconds) were 3.0283 times the initial power and 4.959 Initial-Power Seconds (3.942 MW.s), respectively.

In the near future, the fuel breakup margin will be calculated when the steady-state fuel temperature is provided.

ACKNOWLEDGEMENT

This work has been carried out under the research and development program of the Korea Ministry of Science and Technology.

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Table 1 CHANNEL PARAMETER COMPARISON BETWEEN STANDARD 37-ELEMENT AND DUPIC BUNDLE

Parameters	Standard 37-element Bundle	DUPIC 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
Bundle Radial Power Distribution (Burnup 0)		
Outer Element	1.1310*	1.19679
Intermediate Element	0.9206	0.81696
Inner Element	0.8051	0.84955
Center Element	0.7613	0.48329

* at the burnup of 60 Mwh/kgU

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)
100% PS	Rate-of-log-power	0.360	0.376
40% RIH	Rate-of-log-power	0.318	0.334
100% ROH	Rate-of-log-power	0.671	0.687

* 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
100% PS	2.9723 (H7/4)	3.2845 (H7)	2.916	3.866
40% RIH	3.0283 (H7/4)	3.3652 (H7)	2.993	4.041
100% ROH	2.5408 (H7/4)	2.7995 (H7)	2.503	3.313

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

Break	Time to Reach Peak Reactivity(s)	Peak Reactivity (milli-k)
100% PS	0.887	3.798
40% RIH	0.840	3.826
100% ROH	1.198	3.490

Table 5 MAXIMUM TIME-INTEGRATED BUNDLE POWER TO 3 S FOR EACH TRANSIENT STUDIED

Break	Bundle Position	Initial Bundle Power(kW)	Maximum Time-Integrated Bundle Power to 3s	
			MJ	Initial-Power Seconds
100% PS	H7/4	795.0	3.816	4.800
40% RIH	H7/4	795.0	3.942	4.959
100% ROH	H7/4	795.0	3.466	4.360

	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

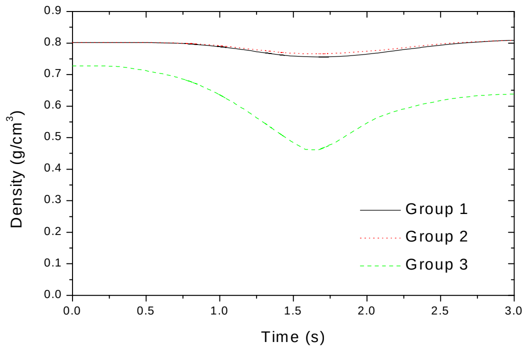

Fig.2 Channel Coolant Density in Groups 1 to 3 (40% RIH)

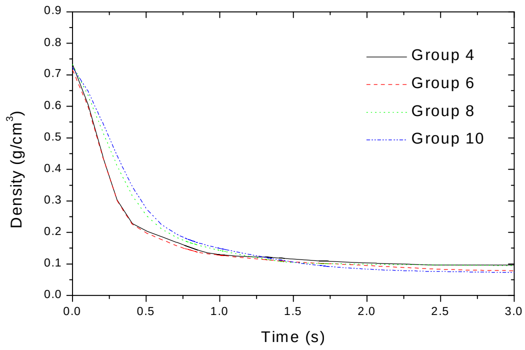

Fig.3 Channel Coolant Density in Groups 4 to 10 (40% RIH)

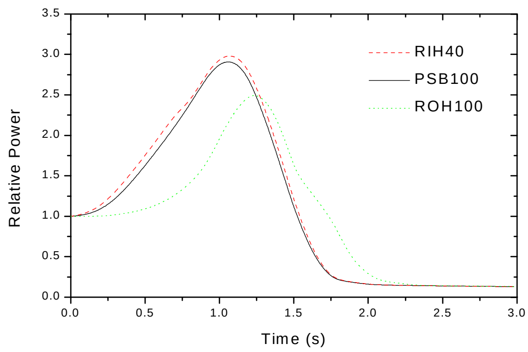

Fig.4 Total Reactor Power Transients after Break