Power Pulse Characteristics Following a LOCA in a DUPIC Fuel CANDU-6 Reactor

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1. Introduction

The core characteristics following a large loss of coolant accident (LOCA) in a Canada deuterium uranium (CANDU) reactor loaded with the direct use of spent pressurized water reactor (PWR) fuel in CANDU reactors (DUPIC) [1] fuel has been was assessed as a part of the DUPIC fuel safety analysis. As the DUPIC fuel bundle is designed and fabricated based on the same mechanical and structural design criteria as the CANDU flexible (CANFLEX) [2] bundle, its geometric characteristics are the same as those of the CANFLEX fuel bundle. However due to the differences in the fuel material composition and the fuel management method in the core, the steady-state full power distribution of the DUPIC fuel core is significantly different from that of the standard fuel core. Therefore the LOCA analysis of the DUPIC fuel CANDU-6 reactor was performed to confirm the compatibility of the DUPIC fuel with the existing CANDU-6 reactor from the viewpoint of the fuel safety.

In this study, the LOCA characteristics have been analyzed for typical cases such as a 55% pump suction break (PSB), 35% reactor inlet header (RIH) break and 100% reactor outlet header (ROH) break. The safety analysis will determine the reactor trip time and the transient power during the LOCA. The fuel integrity threatened by a power pulse is assessed based on these results. The LOCA analysis was performed by the system code CATHENA [3] and the physics code RFSP [4].

2. Analysis Procedure

A two-loop network model [5] of the heat transport system is used in the analysis. As shown in Fig. 1, the core pass downstream of the break (critical pass) was modeled as 7 average channels with different powers, channel elevations and header/feeder connections. The passes in the intact loop are represented as channel group 1 and 2, The return pass of the broken loop (95 channels) is represented as channel group 3. The core region of an individual average channel is represented as 12 nodes, which is to ensure an enough accuracy in predicting the coolant density in the core region.

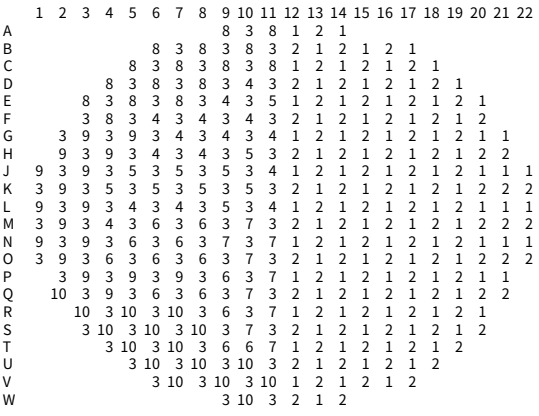

Fig. 1 Channel nodalization of the 10-channel model

The standard RFSP model in Ref. 5 is used in this study. Lattice properties are calculated by the WIMS-AECL [6] code, which generates the fuel and reflector cross sections. The SIMULATE module of the RFSP is used for the equilibrium core calculation under the minimum allowable performance specification (MAPS) conditions as follows:

- Crept pressure tube (2.5% creep in diameter),
- Startup after a long shutdown for 12 hours, in which 2.4 ppm of boron is needed to maintain the criticality,
- The 8% side-to-side power tilt.

3. Results and Discussion

The first trip signal was detected by the high neutron power for all the cases. The backup trip was activated by the high log rate neutron power. The trip times of shut down system 1 (SDS1)

were 0.360, 0.318 and 0.672 sec for the PSB55, RIH35 and ROH100, respectively. The 35% RIH resulted in the highest peak reactivity of 3.826 mk at 0.845 sec. The peak reactivity of the 55% PSB and 100% ROH is 3.798 mk (at 0.887 s) and 3.449 mk (at 1.199), respectively. Figure 2 compares the relative total reactor power transients for the three breaks. The peak value of the relative power is 2.993 at 1.042 sec for the 35% RIH. The peak relative powers for the 55% PSB and 100% ROH are 2.916 (at 1.084 sec) and 2.457 (at 1.396 sec), respectively.

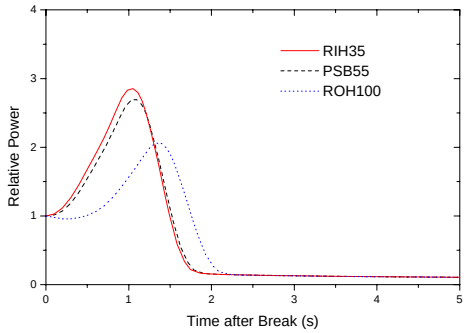

Fig. 2 Comparison of the Power Pulse

The criterion used for the assessment of the fuel integrity was based on the energy content in the fuel at 3 sec after the break, which was compared to the conservative lower limit of the fuel breakup. The minimum energy content of the CANDU-6 fuel breakup is 840 J/g, which is 200 cal/g. As summarized in Table I, the total energy added to the bundle was 556 J/g for the PSB55, 570 J/g for the RIH40, and 550 J/g for the ROH100, respectively. These values correspond to a 32%, 31% and 35% margin for the fuel breakup.

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

ACKNOWLEDGEMENT

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Table I Margin for the Fuel Breakup Threshold

Break	Initial-Power Second (MW.s)	Pulse Energy [1] (J/g)	Total Energy [2] Deposition (J/g)	Margin to Breakup (%)
PSB55	4.29	226.4	556.4	34
RIH35	4.55	240.1	570.1	32
ROH100	4.17	220.1	550.1	35

[1] Pulse Energy = Initial Power Seconds x 23.8/451 J/g

[2] Total Energy Deposition = Pulse Energy + 330 J/g