Best Estimate Prediction of NFR with Uncertainty following a LBLOCA of CANDU plants

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1. Introduction

Limit of radioactive dose to the public following an accident has been the important safety criteria for CANDU nuclear power plants [1]. For a large break loss-of-coolant accident (LBLOCA), the number of rods of fission product release (NFR) which determines the amount of the released fission product is the most important contributor to the dose. Prediction of the NFR means to identify the fuel rods whose properties such as sheath temperature caused by the accident exceeded the fuel failure criteria. The present paper aims to apply the methodology of best estimate (BE) calculation plus uncertainty [2] to predict the NFR for LBLOCA of CANDU plants. To this purpose, the MARS code [3] which has some CANDU specific models was used as a BE code. As a preliminary approach, uncertainty related to the critical break flow model is considered.

2. Code, Modeling, and Calculation Scheme

2.1 MARS Code

MARS-KS-2.0 code is used in the present study. The code, as a BE code, has been applied to calculate system thermal-hydraulic response during the LOCA and transient of NPP. Also, the code has some CANDU specific models as follows [3]:

- Horizontal fuel bundle heat transfer correlations
- CANDU 37 fuel bundle D_2O CHF model
- CANDU ANS94-4 decay heat model
- Henry-Fauske/Moody critical flow model for D_2O
- Digital sampling model for control function
- MOV different open/close time rates model
- Spray flow regime model
- Horizontal flow regime model
- Fuel heatup model for horizontal stratification
- Header component model

The MARS code has been applied to the simulation of the CANDU specific experiments, actual incidents and the postulated accidents of Wolsong NPP.

2.2 Modeling of CANDU NPP

The modeling scheme of Wolsong NPP which was developed in the previous KINS study is used in the present study. The major components of primary heat transport (PHT) system including pressure tubes, feeders, headers, PHT pumps, steam generators (SG), pressurizer, loop isolation, etc were simulated. The total number of volumes, junctions, and heat structures were 547, 580 and 508, respectively. The break is assumed to occur at reactor inlet header (RIH) and the size is 35 % of the RIH cross-sectional area according to the FSAR [4]. The core pass 4, a downstream of the break, may experience the significant stagnation by the break. Thus, detailed modeling containing seven groups of pressure tube is used for the pass 4 (Fig.1). The channels are grouped based on channel power and elevation of pressure tubes [3]. Separate fuel rod modeling within a pressure tube for horizontal stratification is not adopted in the present study. Therefore, 37 fuel rods per pressure tube times number of pressure tubes belonging to each group ($37\times G_i$ rods) are modeled as one heat structure.

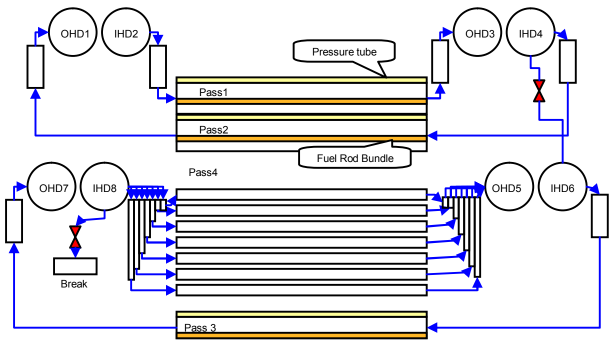

Fig. 1. Detailed modeling of core pass 4.

2.3 Calculation Scheme

From single calculation, thermal-hydraulic response can be obtained for 84 rods (12 bundles × 7 groups). Generally, the fuel failure criterion involves the maximum fuel centerline temperature, the maximum sheath temperature, the maximum sheath circumferential strain, etc. In the present study, only the criterion on maximum sheath temperature is used. There are lots of parameters and models affecting the sheath temperature. Among them, only the parameters related to the critical flow at the break are considered for purpose of the study. Currently, the default critical flow model in MARS code, (Henry-Fauske) is as follows:

$$G_c^2 = \left\{\frac{x_0 v_v}{\eta P} + (v_v - v_{l,0})\left[\frac{(1-x_0)N}{(s_{v,eq}-s_{l,eq})}\frac{ds_{l,eq}}{dP} - \frac{x_0 C_D (1/\eta - 1/\gamma)}{P_t (s_{v,0}-s_{l,0})}\right]\right\}^{-1}$$

$$N = \min(1, x_{eq,t}/C_{ne})$$

In this equation, two parameters, discharge coefficient (C_D) and non-equilibrium constant (C_{ne}) are considered as uncertainty variables. Means and standard deviations are as follows:

Table I: Uncertainty variables

Parameter	Mean	Distribution	Standard Deviation
C_D	1.0	Normal	0.05
C_{ne}	0.14	Normal	0.02

With the sampled sets of two uncertainty variables, 124 code runs are conducted based on the 2nd order Wilk's formula. From the result of calculation, sheath temperatures for 124 x 84 rods are traced to find the 3rd highest one.

4. Result and Discussion

Fig. 2 shows the responses of sheath temperature at 6th bundle of the group 5 of the pass 4 for 124 cases. The maximum sheath temperature is 1573 K.

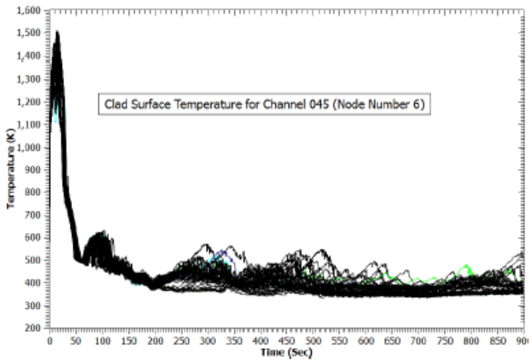

Fig. 2. Sheath temperature at 6th bundle of the group 5

The node for which the temperature exceeds 1477.15 K is assumed to be failed. The number of fuel rods for each node equals to 37 multiplied by the number of clubbed channels. The NFR for all the cases is presented in Figure 3.

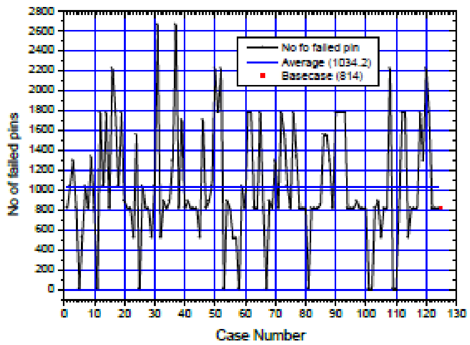

Fig. 3. Number of failed rods for all cases

The distribution of the NFR can be plotted as in Fig.4. The 3rd highest NFR was found 2,220. Additionally, the correlation between the sheath temperature and NFR was investigated as in Fig. 5.

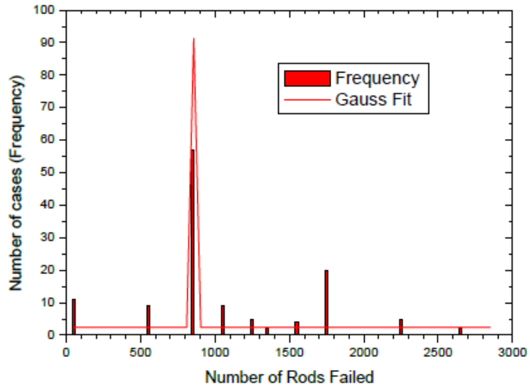

Fig. 4. Distribution of NFR

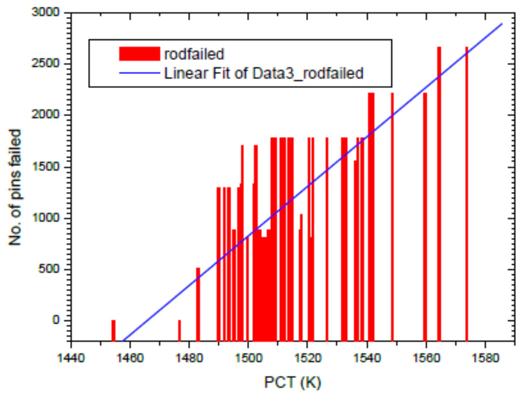

Fig. 5. Fraction of counts lost with voltage and charge

A liner relation between two parameters can be obtained, which needs more specific study.

5. Conclusions

The number of rods of fission product release (NFR) following a LBLOCA of CANDU plants was predicted by the best-estimate calculation using MARS code with uncertainties related to the critical flow model. The NFR in 95 percentile was 2,220. This scheme can be easily extended to the full-scope calculation of NFR.

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