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# The Evaluation of ROP Trip Set Point for the Different CANDU-6 Fuel Type

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## 1. Introduction

The regional overpower protection (ROP) system in CANDU6 reactor is to prevent OID (Onset of Intermittant Dryout) caused by the various thermal flux shape or the Slow Loss of Regulation (SLOR). This system has the 58 detectors divided into three safety channels and actuates according to the two out of two, trip logic. The each safety channel must provide the trip signal with 98% trip probability. The ROP trip set point is decreasing due to core aging effect and bad thermal condition. Therefore, the operating power is decreased to maintain the trip probability. Recently the new designed fuel is being developed to enhance the trip set point. The fuel bundle used in domestic CANDU reactor has the 37 elements to be the same diameter. The diameter of central element in the new designed fuel is reduced to increase the coolant flow area. In this paper the ROP trip set point is evaluated to estimate the effect of fuel design change.

## 2. Methods and Results

*2.1Model of ROP Physics Cases*

The various flux shapes are considered for ROP analysis and are categorized according to the conditions of reactor operation which are the normal operation, startup and shim operation, liquid zone and moderator level, poison concentration, power tilt, device configuration and so on. There are 926 ROP physics cases. The flux shapes and ROP detector response at each detector location are generated using WIMS-AECL 3.1, DRAGON 3.06 and RFSP 3.5 with ENFD/B-VII [3,4,5]. Case 001 is the reference case and the steady state core with 40% average liquid zone level and all adjuster inserted. In this study case 222 and 252 is selected because of the worst cases in the previous analysis. Both cases are the startup operation case with the abnormal adjuster configuration. Normally the 7 adjuster banks to startup the reactor are inserted step by step based on the power and liquid zone level. The 7$^{th}$ adjuster bank (10th and 12th adjuster) in case 222 is inserted with half-length. The 10th adjuster in case 252 is inserted with half-length and 12$^{th}$ adjuster is not inserted. For the generating case 222 and 252, case 151 is also modeled because of the initial case to startup after short shutdown. The channel power of each case is provided to NUCIRC code to calculate the critical channel power. The detector signals are used in ROVER-F code to evaluate the trip set point.

*2.2 Calculation of Critical Channel Power*

NUCIRC is a steady-state one-dimensional thermal-hydraulic code used for the design and performance analysis of the HTS (Heat Transport System) and components of a CNADU reactor for a variety of operating conditions. In this study, the conditions at 8000 EFPD (Effective Full Power Day) is assumed and the pressure tube creep is not applied. NUCIRC code calculates the critical channel power using the different critical heat flux correlations obtained from experiments of each fuel type. When some flux shape without reactor operator's recognition is increased, the critical channel power to occur the dryout is calculated. The critical channel power is provided to the ROVER-F for the ROP analysis.

*2.3 ROP Analysis*

ROP trip set point is evaluated using ROVER-F code. ROVER-F code is to calculate the trip probability of ROP system [1]. The input files of ROVER-F are the channel power, the critical channel power, detector information, uncertainty, ripples and so on. The ripples are generated from the snapshot of the actual operating reactor. The uncertainty is divided into the channel random error, common random error and detector random error. Channel and common random error is changed according to the thermal hydraulic model. Calculations for two fuel type use the same input files and calculation option except of the channel power and critical channel power.

*2.4 Results*

1) Channel Power and Critical Channel Power

The maximum power and location of the channel and bundle for each case is similar to each other fuel type as shown as table 1. Figure 1 shows the relative error of the channel power for the ROP case 001. Figure 2 shows the relative error of the critical channel power for the ROP case 001. The channel power is similar below 1% error between two fuel types. But the maximum relative error of critical channel power is 11.26% because the critical heat flux of the new designed fuel is improved due to increasing the coolant region. For the case 222 and 252 the relative channel power error is below 0.5% and 0.65% and the relative critical channel power error is below 13.95% and 13.10%. The fuel design change is not effect to the thermal neutron flux and power distribution and only improves the thermal operation margin.

There is no or little effect of the flux shape and power distribution due to the partial design change of fuel bundle. To reduce the diameter of center fuel element is expected only to improve the performance of cooling because of increasing the quantity of coolant.

Table 1. Maximum Channel and Bundle Power and Location

| Case No. | Standard 37-element Fuel | | New 37-element Fuel | |
|---|---|---|---|---|
| | CP_max [kW] | BP_max [kW] | CP_max [kW] | BP_max [kW] |
| 001 | 6614.0 O17 | 796.68 P-17/6 | 6603.81 O17 | 795.96 P-17/6 |
| 151 | 9383.0 M13 | 1420.59 M-11/6 | 9282.3 M13 | 1420.34 M-11/6 |
| 222 | 9168.9 N13 | 1327.62 N-11/6 | 9166.89 N13 | 1327.06 N-11/6 |
| 252 | 9079.2 M13 | 1310.97 M-13/7 | 9080.4 M13 | 1310.88 M-13/7 |

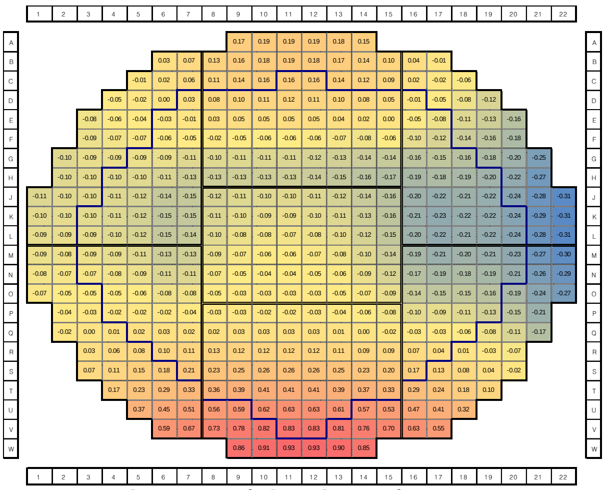

Fig. 1. Relative Error of Channel Power for Case 001, (CP_new-CP_standard)/CP_standard*100

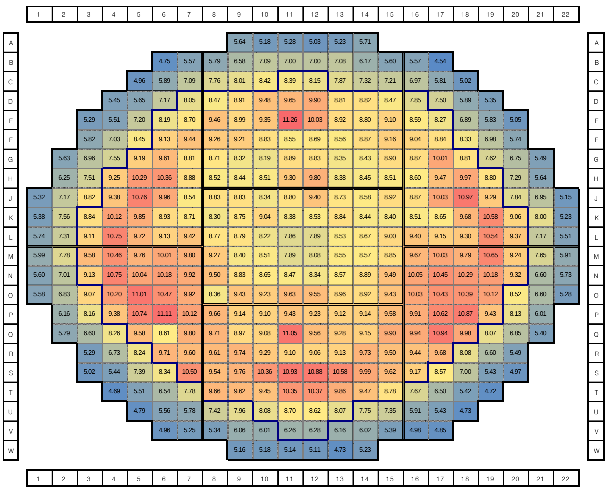

Fig. 2. Relative Error of Critical Channel Power for Case001, (CCP_new fuel – CCP_standard)/CCP_standard *100

2) ROP Trip Set Point

Table 1 shows the trip set point for each ROP cases. The trip set point for the new 37 element fuel is increased above 10%. Because the thermal hydraulic model at this analysis is assumed the no pressure creep the trip set point is estimated over than realistic condition. But the improvement of the trip set point for the new designed fuel can be expected. Consequently the loading of the new designed 37 element fuel into the core increase the trip set point.

Table2. ROP Trip Set Point at 8000 EFPD without Pressure Tube Creep

| ROP Case Number | Trip Set Point (%) | | Difference (B-A) |
|---|---|---|---|
| | Standard 37-element (A) | New 37-element (B) | |
| 252 | 94.65 | 105.74 | 11.09 |
| 222 | 95.35 | 105.95 | 10.60 |
| 151 | 103.02 | 113.79 | 10.77 |
| 001 | 108.99 | 119.61 | 10.62 |

## 3. Conclusions

At 8000EFPD with the assumption of no pressure tube creep, the trip set point of new fuel design is respectively increased above 10%. This study has shown that the trip set point is enhanced by means of the fuel design change. Because the thermal hydraulic model in this paper is assumed without the pressure creep, the trip set point to be applied to the plant operating must be reevaluated with the proper plant's data. In the future the trip set point of Wolsong plant loading new designed fuel is evaluated with each proper thermal hydraulic condition, pressure tube creep and ripples.

## REFERENCES

[1] D. Kastanya, ROVER-F version 3.0.1 Application Manual, AECL Report CW-117390-MAN-004, 2008.

[2] D.Kastanya, V. Caxaj, Regional overpower protection analysis for CANDU6 reactors using ROVER-F code, Annals of Nuclear Energy 37(2010) 28-33, 2010.

[3] G.Jonkmans, WIMS-AECL Version 3.1 User's Manual, ISTP-05-5115, COG, 2006.

[4] G. Marleau, A.Herbert and R.Roy, A User Guide for DRAGON 3.06, IGE-174 Rev.7, Ecole Polytechnique de Montreal, 2008.

[5] P.Schwanke and A. Ho, RFSP 3.5 User's Manual, SQAD-12-5022, COG, 2013.