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Preliminary ROP Assessment for CANDU-6 with DUPIC Fuel

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Abstract

Preliminary assessments for regional overpower protection trip system of the CANDU-6 reactor with DUPIC fuel has been performed. For this study, some severe flux perturbation cases were selected, and the trip setpoint was calculated. In addition to the trip set point, some sensitivity studies for uncertainty of the channel random and common-random were performed. This study has shown that the trip setpoint of the DUPIC fuel core is slightly improved compared to the natural uranium fuel core with the same uncertainties of natural uranium core.

I. Introduction

CANDU power reactors are characterized by on-power fuelling and a relatively large reactor core. These factors result in a continuously changing burnup distribution and a potential for slow flux and power oscillations due to xenon. Reactor control thus involves a number of types of spatially-distributed reactivity control devices and spatial control in three dimensions using dual on-line computers. Combined, these factors result in a need for protection against localized fuel bundle and channel overpowers throughout the core for a wide variety of potential flux shapes.

To provide this protection, CANDU power reactor are equipped with regional overpower protection (ROP) trip systems. There are two such ROP systems in CANDU, one for each shutdown systems (SDS’ s). If either ROP system senses an overpower condition in the reactor, it immediately trips or actuate its associated shutdown system, which then rapidly shuts down the reactor.

An overpower is defined as a fuel bundle or channel power in excess of specified safety-related limits. These overpower limits are separate from and above the normal operating limits on channel and bundle powers. In CANDU reactors, the ROP trip systems in the reactor against overpower in the reactor fuel, where due to localized peaking within the core or a general increase in core power levels.

The DUPIC[1] fuel has been studied as an alternative to either the once-through or recycling fuel cycle. An important eventual consideration in the implementation of the DUPIC fuel in the CANDU-6 reactors is the determination of the setpoints required by the ROP trip system. The DUPIC fuel cycle includes several differences from the current natural uranium fuel which will impact on the ROP setpoints:

- Heterogeneous fuel composition, which may affect channel-random and common-random uncertainties
- A 43-element fuel, which provides additional margin to critical channel power
- A two-bundle-shift refuelling scheme, which alters the axial power profile, and thus the critical channel powers
- A redistribution of radial power from the inner core to the outer core.

In this study, these changes are considered to be part of a preliminary ROP calculation for a DUPIC-fuelled CANDU-6 reactor core. Several ROP calculations were performed for a number of limiting flux shapes to determine the required ROP setpoint for the DUPIC-fuelled core. Several additional calculations were performed to determine the robustness of the ROP design or to suggest improvements to it.

II. ROP System

II.1 Trip Coverage Equation

The basic ROP trip protection equation is : for any flux-shape k and ripple q, the reactor shall be tripped before any coolant channel reaches its critical channel power. That is, the ROP system shall have detector locations, channelizations and trip setpoints TSP such that for every design-basis flux shape k, there is at least one detector j_p in each safety channel i such that:

$$\mathrm{TSP}(j_p) \le \Phi^T(j_p, k, q) \tag{1}$$

where $$\Phi^T(j_p, k, q) = D_0(j)\frac{\phi(j,k,q)}{\phi_0(j,q)} * r_{CPRL}(k,q) \tag{2}$$

and TSP is the trip setpoint for detector j,

D_0 is the detector's initial calibration at 100% full power,

Φ^T is the required trip setpoint for detector j_p if it is to protect flux-shape k with ripple q,

ϕ is the flux at detector j for case k and ripple q,

ϕ_0 is the nominal flux at detector j at 100% power for ripple q,

r_{CPRL} is the minimum critical power ratio for case k and ripple q.
In practice, a number of modifications and corrections are made to the basic equation. These modifications and corrections are described precisely in reference 2.

II.2 ROP System

Each ROP system consists of 20 to 50 self-powered in-core flux detectors, contained within the core in vertical or horizontal tubes known as assemblies, together with associated amplifiers, trip comparators, display and test circuits, and other trip logic circuitry.

The flux detector assemblies are located within the relatively cool low-pressure moderator, between and perpendicular to the fuel channels. To ensure physical separation, the SDS1 ROP detectors and shutdown mechanisms are vertically oriented, while the SDS2 ROP detectors are located horizontally. The detectors in each ROP system are divided into three sub-sets, each associated with one of the shutdown system trip channels.

Each ROP detector has a preset trip setpoint. If the signal from any detectors in a trip channel exceeds the detector's setpoint, then that trip channel is tripped. Trip by any 2 out of 3 channels initiates a reactor shutdown.

III. Calculation Procedure

III.1 Flux Shape and Channel Powers

The RFSP[3] code was used to simulate a selection of ROP cases based on the time-average core model. These cases were selected from the limiting ROP cases for Wolsong-1 ROP analysis for a core fuelled with 37-element natural-uranium fuel[2]. A list of these cases is presented in Table 1. The required outputs for ROP analysis are:

- Channel powers (for input to ROVER-F[4])
- Bundle powers (for input to the calculation of the critical channel powers).
- The flux along the detector assemblies (for input to the calculation of detector response).

III.2 Thermalhydraulic Analysis

The thermalhydraulics code NUCIRC[5] (Version 5-05) was used to calculate the critical channel powers (CCPs) for each ROP case, for use as input to ROVER-F. As input, the bundle powers for each ROP case are required as well as feeder pipe geometry and orifice data.

In view of CCP analysis, two major differences between standard and DUPIC fuels are the axial power distribution and the fuel bundle geometry. In general, the DUPIC fuel makes the power distribution skewed toward the inlet of the fuel channel due to its large reactivity insertion of the proposed two-bundle shift refueling. This inlet-skewed cosine power distribution tends to increase the critical channel power. It should be noted that the existing feeders and orifices are set for the standard fuel bundle loaded core, the channel flow distribution in the reactor core may not be the optimized for the DUPIC loaded. Therefore, the CCP margin may not be uniformly distributed over the fuel channels.

The fuel bundle geometry may cause different CCP value for the DUPIC fuel when the same power level is used. In this analysis, the AECL's x_c-L_b correlation is used. The validity of this correlation for other fuel geometry than the standard fuel bundle string has not been fully asserted. It should be noted, however, that the 43-element fuel bundle geometry should have better heat transfer characteristics due to its larger heat transfer surface per unit volume of the fuel material. On the other hand, the radial power distribution inside the DUPIC fuel bundle may affect the heat transfer characteristics which is not understood completely. Since the NUCIRC code employs single channel model lumped over the cross section of the fuel channel, the detailed radial geometric information of the fuel bundle cannot be considered.

III.3 Detector Responses

The fluxes along the detector assemblies are converted to detector responses via a simple processing code of DETRESP[2]. This code calculates the response at the locations occupied by the detectors and adds a lead-cable response based on the lead-cable relative sensitivity and on the length of the lead cable and the flux. These are then compared to the detector responses in the nominal case. The quantity of interest is the ratio of the detector response in the case analyzed to that in the nominal case. The detector responses for each ROP case are then used as an input to ROVER-F.

III.4 Ripple calculation

Because of on-power fuelling, the core in an operating CANDU reactor will typically have channels and bundles with a mixture of widely varying irradiations and varying powers. As fuel in a channel reaches its maximum burnup it is discharged and replaced with fresh fuel. The resulting variation in individual channel powers about their time-average values is known as refueling ripple. A basic simplification in the ROP calculation is to separate the effect of refueling ripple from the other flux-shape variations – i.e., those due to reactivity devices or xenon change. The ROP detectors are calibrated to each ripple, i.e., to read 100%, then the calibration automatically divides out the rippled nominal flux distribution.

The ripples used for the ROP trip setpoint calculation were obtained from 600 full power day (FPD) refueling simulation. A total 61 ripples were supplied, at approximately 10-day intervals.

III.5 Trip Setpoint and Confidence Calculation

The probabilistic assessment is performed to determine the ROP trip confidence and setpoint The first step in the probabilistic assessment is the classification of the expected errors and uncertainties into four categories:

- Detector-random errors: these affect the ratio of detector setpoint to detector reading, and vary from detector to detector. These errors include detector calibration errors, flux-shape simulation errors, and errors in setting the trip setpoint.
- Channel-random errors: these affect the ratio of critical channel power (CCP) channel power, and vary from channel to channel. An example is the uncertainty in the CCPs arising out of uncertainties in channel hydraulic resistance or axial power distribution.
- Common-random errors: These are random in their probability of occurring, but apply in the same way to all detectors or all fuel channels. Examples include uncertainties in CCP correction, or fluctuations in the coolant inlet temperature.
- Systematic errors: these are non-random errors, present either at all times, or only when specific conditions arise. Theses include the average value of measured random errors.

Table 2 shows the uncertainties used in this study.

The individual errors are summed together in each of the four categories. All cases in the simulation-set must meet the 98% confidence requirement. Thus, error allowance for which the limiting cases or cases just meet the 98% limit defines the maximum allowable level of ROP setpoint.

IV. Calculation Results

Several ROP calculations were performed. The basic result is the determination of the required ROP setpoint for the DUPIC-fuelled core. In addition to this calculation, additional results to determine the robustness of the ROP design were obtained.

IV.1 ROP Trip Confidence

The primary result is the calculation of the trip confidence and trip setpoint in the DUPIC core, as compared to the result for the natural-uranium fuelled core. This was calculated for a number of limiting cases. The results are presented in Table 3 and the results for natural uranium fuel cases are shown in Table 4. As can be seen, an increase in the setpoint appears to be attainable with DUPIC fuel, with current values of the various current uncertainties.

IV.2 Uncertainty Sensitivity

The non-homogeneous nature of DUPIC fuel, as well as the changes in bundle geometry and fuelling scheme will result in changes to the uncertainties that ROP uses to calculate trip probability and trip setpoint. In anticipation of this, a response table of the required setpoint for a range of channel-random and common-random uncertainties has been produced. This is presented in Figures 1-3. From these the sensitivity to channel-random uncertainty can be seen to be greater than the sensitivity to the common-random uncertainty.

IV.3 Single-Detector Failures

The calculation of the effect on trip probability in the event that a single detector has failed was performed. This task is performed for each detector and calculates the trip probability and trip setpoint in the event that that detector has failed. This is outlined in Table 4. Given these setpoints, the values to be used upon the failure of a single detector is determined. It may be noted that the setpoint change for some detectors is negligible. Either the detector in question is well backed up by a second detector, or the cases protected by these detectors are not near-limiting.

V. Summary and Future Work

In the CANDU-6 reactor with DUPIC fuel, a preliminary ROP assessment has been performed for selected design basis cases. In this study, the calculation procedures were investigated for DUPIC fuel core, and assessment results can be summarized as:

- The trip setpoint for DUPIC fuel core is slightly increased compared to natural uranium fuel core, with current values of the various uncertainties.
- The sensitivity of the channel-random uncertainty can be seen to be greater than the sensitivity to the common-random uncertainty.
- In case of single detector failure, the setpoint change for some detectors is negligible, but the setpoints are decreased for SDS2 detector failures.

This study determined ROP characteristics for a preliminary study of the DUPIC core. Several tasks may be added as the study continues and more data becomes available. In future, the uncertainties introduced by the non-homogeneous nature of DUPIC fuel should be evaluated as they affect the channel powers and critical channel powers. Also, it is likely that the analysis for a larger selection of the design basis cases will be required. Furthermore, the changes of the ROP setpoint due to changing ripples, and the optimum radial core-power for a DUPIC fuel core should be performed, which will increase the ROP margins by maximizing channel powers in regions of the core with ample margin, and decreasing the channel powers in limiting areas.

References

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Table 1 ROP Cases Simulated for DUPIC Fuelled Core

Case	File Name	Description
1	SSSC50	STEADY-STATE
38	D01C80	ZONE DRAIN 01 FROM 80%
39	D02C80	ZONE DRAIN 02 FROM 80
46	D09C80	ZONE DRAIN 09 FROM 80%
51	D14C80	ZONE DRAIN 14 FROM 80%
130	ZT2A01	2ND AZIMUTHAL 135,315 HI
150	SSSD01	ALL BANKS OUT/Xe at 44.5 MIN
151	SSSD02	ALL BANKS OUT/Xe at 67.4 MIN
152	SSSD03	BANK 7 FULL-IN/NO TIMESTEP
153	SSSD04	BANK 7 FULL-IN/Xe at 91.4 MIN
154	SSSD05	BANK 6 FULL-IN/NO TIMESTEP
170	SA4402	ALL BANKS OUT/Xe at 5.3 MIN
172	SA4404	BANK 7 FULL-IN/Xe at 5.2 MIN
222	ABHO01	STARTUP BANK 7 HALF-IN

Table 2 Estimated ROP Errors and Uncertainties for Wolsong-1

Source of Errors	Estimated Magnitude (%)			
	Detector Random	Channel Random	Common Random	Bias Error
1. Detector-Related Errors				
Trip Setpoint	±0.18%		±0.14%	~0
Buffer Amplifier	±0.10		±0.10	
Dynamic Compensation	±0.60			
2. Flux-Shape Errors				
Simulation Error	±1.88	±1.06	±1.07	0.14
Change due to Boiling				-0.20
Lead-Cable Contributions	±0.20		±0.10	-0.20
Off-Nominal Core			±0.80	
3. CCP Errors				
CHF Correlation Errors			±1.70	0.66
Incomplete Instrumentation				-0.19
NUCIRC Pressure Loss Term		±0.89	±0.83	-2.20
Uncert. In HTS Bndy Cond's			±2.32	
Chge in Ref. HTS Bndy Cond's				4.18
HTS variations			±0.15	
Channel Age Correction		±0.15		
CCP Change		±0.10		
Normal Operating Flux Tilt				-0.20
Different Fuel Type		±0.44		
Allowance for PT Creep				-1.00
4. Calibration Errors				
CP/CPPF Calculation			±1.50	0.20
Thermal Power Calculation			±1.70	
CPPF Drift Erro			±0.80	
Calibration Drift Error	±1.60			
TOTALS	±2.55%	±1.46%	±4.07%	+0.19%

Table 3 Trip Confidence for DUPIC Fuelled Core with ROP Setpoint of 125%

Case		CPRL	Avg. Cons.	SDS1	SDS2	LimDets	
1	SSSC50	1.423748	1.028413	.999592	.999477	6D	3G
38	D01C80	1.338534	1.053011	.998309	.997819	8F	3H
39	D02C80	1.310183	1.042594	.999597	.979876	2E	6G
44	D07C80	1.308966	1.037323	.998173	.980055	2F	6H
46	D09C80	1.279503	1.045893	.991170	.984605	5D	4H
51	D14C80	1.296593	1.039573	.991182	.981716	10E	8G
130	ZT2A01	1.320480	1.056017	.998591	.993387	2D	6G
150	SSSD01	1.096475	1.044001	1.000000	1.000000	4E	1H
151	SSSD02	1.058506	1.034462	1.000000	1.000000	4E	1H
152	SSSD03	1.159355	1.032921	1.000000	1.000000	6E	1H
153	SSSD04	1.128970	1.031984	1.000000	1.000000	6E	1J
154	SSSD05	1.180705	1.038996	.999999	1.000000	6E	4H
170	SA4402	1.153746	1.036850	1.000000	1.000000	4E	1H
172	SA4404	1.284571	1.036787	.999999	.999998	6E	4H
222	ABHO01	1.098705	1.042798	1.000000	.999991	6E	4H

Required ROP Setpoint for 98% Trip Confidence = 1.2764

Table 4 Trip Confidence for Natural Uranium Fuelled Core with ROP Setpoints of 125%

Case		CPRL	Avg. Cons.	SDS1	SDS2	LimDets	
1	SSSC50	1.388429	1.040686	.999760	.999695	6D	3G
38	D01C80	1.302188	1.060765	.998493	.998502	1F	3G
39	D02C80	1.281467	1.049015	.999568	.988055	2E	6G
44	D07C80	1.272290	1.054976	.998263	.988792	5F	6H
46	D09C80	1.270645	1.055635	.996725	.993799	5D	4H
51	D14C80	1.283386	1.045286	.996089	.991087	10E	8G
130	ZT2A01	1.300184	1.054576	.998707	.994789	2D	6G
150	SSSD01	1.013091	1.058394	.999975	.999977	4E	1J
151	SSSD02	.958042	1.054548	.999998	.999999	4E	1J
152	SSSD03	1.039796	1.059936	.999738	.997505	6F	1J
153	SSSD04	.992353	1.055020	.999973	.999860	4E	1J
154	SSSD05	1.112696	1.050421	.999931	.999967	10E	1H
170	SA4402	1.067599	1.053050	.999997	.999998	4E	1J
172	SA4404	1.130085	1.052836	.999912	.999287	1F	1J
222	ABHO01	.982455	1.053918	.999990	.997609	5D	4H

Required ROP Setpoint for 98% Trip Confidence = 1.2764

Table 5: Setpoints for Single Detector Failure

SDS1 Detector			SDS2 Detector		
Failed Detector	**Limiting Case**	**Required Setpoint**	**Failed Detector**	**Limiting Case**	**Required Setpoint**
1D	44	127.64	1G	39	127.35
2D	44	127.64	2G	51	127.55
3D	44	127.11	3G	44	127.40
4D	44	127.64	4G	51	127.05
5D	44	127.64	5G	44	126.75
6D	44	127.64	6G	46	123.69
7D	44	127.64	7G	44	127.06
8D	44	127.64	8G	51	126.53
9D	44	127.64	1H	44	127.56
10D	44	127.64	2H	44	127.59
11D	44	127.64	3H	46	127.48
12D	44	127.64	4H	46	124.18
1E	44	127.64	5H	44	127.60
2E	44	127.64	6H	44	123.84
3E	44	127.64	7H	39	126.84
4E	44	127.64	8H	44	127.46
5E	44	127.64	1J	39	127.55
6E	44	127.64	2J	44	125.88
7E	44	127.64	3J	39	127.54
8E	44	127.64	4J	51	127.49
9E	44	127.64	5J	39	123.64
10E	44	127.64	6J	51	124.33
11E	44	127.64	7J	39	127.61
1F	44	127.64	8J	46	124.69
2F	44	127.64			
3F	44	127.64			
4F	44	127.64			
5F	44	127.64			
6F	44	127.64			
7F	44	127.6			
8F	44	127.64			
9F	44	127.64			
10F	44	127.64			
11F	44	127.64			

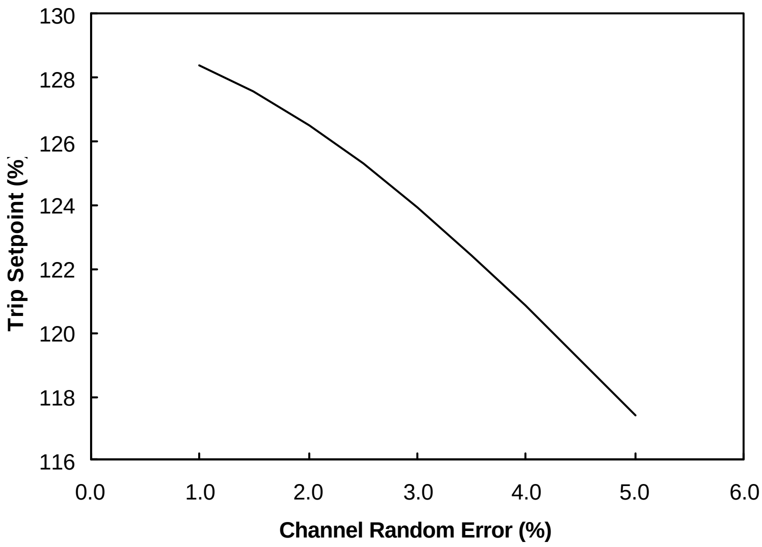

Fig. 1 Response of Trip Setpoint to Varying Channel Random Error

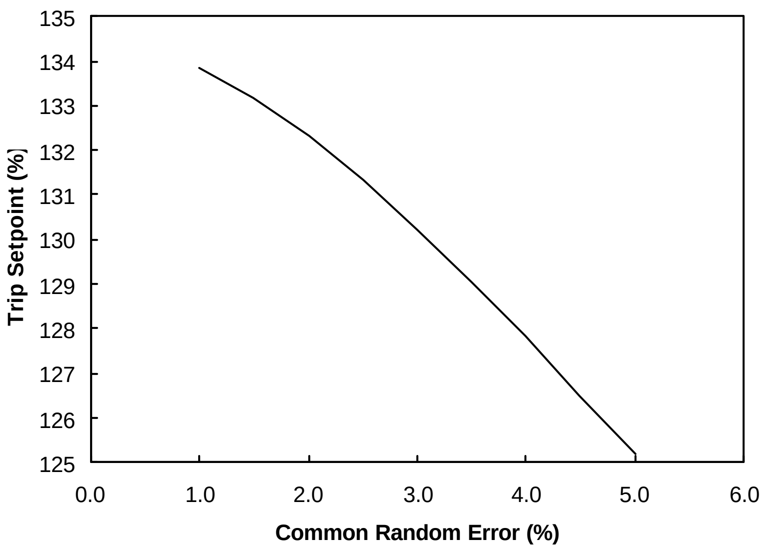

Fig. 2 Response of Trip Setpoint to Varying Common Random Error

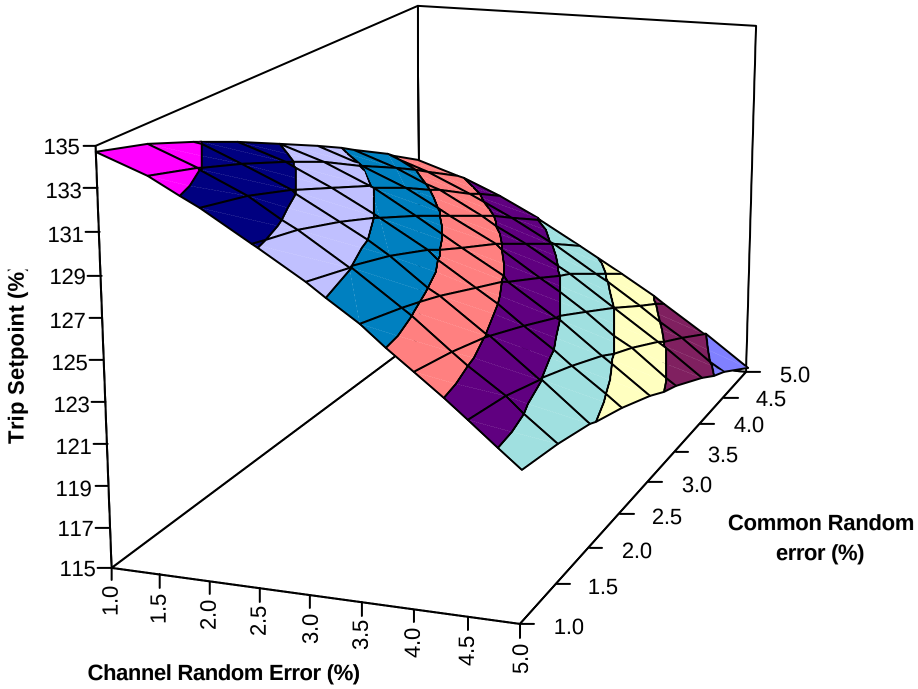

Fig. 3 Response of Setpoint to Variations in Channel- and Common-Random Uncertainties