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Regional Overpower Protection System Analysis for a CANDU-6 Reactor with the DUPIC Fuel

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ABSTRACT

The regional overpower protection (ROP) system of a Canada deuterium uranium (CANDU) reactor was assessed for direct use of spent pressurized water reactor (PWR) fuel in CANDU reactors (DUPIC), including the validation of WIMS/RFSP/ROVER system used for the estimation of the ROP trip set-point (TSP). The comparison has shown that the WIMS/ RFSP/ROVER system produces the results consistent with the design code system for the estimation of the ROP TSP for a CANDU reactor. For the DUPIC fuel CANDU core, the ROP TSP was estimated to be 123%, which is higher than that of the standard natural uranium core by ~1%. This study has shown that the DUPIC fuel does not deteriorate the current ROP TSP designed for the natural uranium CANDU reactor.

I. INTRODUCTION

In a Canada deuterium uranium (CANDU) reactor, the regional overpower protection (ROP) system protects the reactor against overpowers in the core. The design requirement of the ROP system is that the integrity of the primary heat transport system (PHTS) is maintained if an overpower is to occur.[1] Though the CANDU reactor can be operated in dryout conditions without damages to pressure tubes, prevention of fuel centerline melting is nevertheless set as a convenient and conservative safety design criterion for the CANDU ROP system.[2]

The ROP system is activated by signals from two detector systems which are distributed in the core to ensure the coverage of the local flux and power peaks that could arise due to normal maneuvering or abnormal combination of reactivity devices. The shutdown system 1 (SDS1) ROP detectors are located in 26 vertical assemblies, which are shared with other flux detectors used for reactivity control and flux mapping. The shutdown system 2 (SDS2) ROP detectors are located in seven horizontal assemblies. Each ROP detector has a preset trip set-point (TSP) and each shutdown system is connected to three logic channels. If two detectors in three different trip channels (but the same ROP system) detect a high local flux in excess of their preset TSP, the reactor is shutdown. The SDS1 and SDS2 detector locations are shown in Figs. 1 and 2.

The basic ROP system design requirement is that the reactor is tripped for any flux-shape and ripple before any coolant channel reaches its critical channel power (CCP). Therefore the ROP system shall have detector locations, channelizations and TSP such that for every design-basis flux-shape there is at least one detector j_p (the "protecting" detector) in each safety channel, that satisfies following condition:[3]

$$\mathrm{TSP}(j_p) \leq \Phi^T(j,k,q) = D_0(j)\frac{\varphi(j,k,q)}{\varphi_0(j,q)} r_{CPRL}(k,q)$$

where D_0 is the detector's initial calibration at 100% power, Φ^T is the required TSP for detector j if it is to protect flux-shape k with ripple q, φ is the flux at detector j for flux shape k and ripple q, and φ_0 is the nominal flux at detector j at 100% power and ripple q. The r_{CPRL} is the minimum critical power ratio (CPR) for flux shape k and ripple q, that is the minimum value (over 380 fuel channels) of the ratio of the CCP to the actual channel power (CP). A number of modifications and corrections are needed to consider the uncertainties and some operating conditions. The detailed modification and correction methods are described in Ref. 4.

In this study, the ROP TSP of the CANDU reactor is assessed for the DUPIC fuel. In order to ensure the safe and economic operation of the DUPIC fuel CANDU core, an appropriate reactor trip margin should be reserved so that unnecessary reactor trips are avoided and the fuel integrity is maintained, which has motivated this work. For the ROP analysis, the computer code system used for the DUPIC core analysis is validated against the current design code system, and the ROP analysis for the DUPIC core will be performed.

II. ASSESSMENT OF WIMS/RFSP/ROVER-F SYSTEM FOR ROP ANALYSIS

In the ROP analysis of the DUPIC core, a lattice code WIMS-AECL [Ref. 5], a core simulation code RFSP [Ref. 6] and an ROP TSP calculation code ROVER-F [Ref. 7] are used, while the current ROP analysis method is based on POWDERPUFS-V (PPV) [Ref. 8] for the lattice parameter generation. This section describes the assessment of the WIMS-based code system for ROP analysis by comparing the results with the PPV-based calculation. The calculations are performed for the standard 37-element natural uranium core.

II.1. Calculation Procedure

The RFSP physics calculations were performed to obtain flux shapes and channel powers, using lattice parameters generated by the WIMS and PPV. The physics calculations were performed for 26 limiting design-basis cases, which are given in Table 1. The thermal neutron fluxes calculated for each case are processed to obtain the detector response at each detector location, which was performed by INTREP module of the RFSP code. The CCPs were then calculated for all flux shape cases. For both the PPV- and WIMS-based calculations, the same operating conditions were used. The detailed methodology and calculation procedure are described in Ref. 9. For the ripple data, the operation data of Wolsong-1 reactor were used, which consist of 150 sets obtained from the operation from 1997 to 1999. The average CPPF of this data is 1.0706. The uncertainties used for the calculations are shown in Table 2.

II.2. Calculation Results

The results are shown in Tables 3 and 4 for the PPV- and WIMS-based calculations, respectively. In these tables, minimum CPR, average contribution of ripple conservatism, trip probabilities of the SDS1 and SDS2 and limiting detectors are presented. The TSP based on the WIMS is 122.9%, while that based on PPV is 121.8%. The difference is ~1%, which is small enough to accept that the WIMS-based code system is consistent with the current ROP design code system based on the PPV code.

In principle, the ROP analysis includes the errors and uncertainties in order to consider the accuracy of the physics calculations used to produce the power distribution. Because the power distribution is directly dependent on the lattice parameters, it is possible that the sequences of limiting cases are different for the PPV- and WIMS-based analyses. The results here indicate that the trip set point is reasonably estimated by the WIMS-based calculation, when considering all cases with a 98% trip probability.

III. ROP TRIP SETPOINT ASSESSMENT FOR DUPIC CORE

This section describes the ROP TSP estimation for the CANDU reactor with the DUPIC fuel. For the ROP TSP assessment, 232 design-basis cases were considered with some exceptional cases of harmonic tilts and restart after a long shutdown. The cross-sections of the DUPIC fuel were produced by WIMS-AECL, and the rest of the procedure to estimate the ROP TSP of the DUPIC core is the same as that of the natural uranium core.

III.1. Calculation Procedure

The physics calculations were performed for 232 design-basis cases with exception of some cases: four cases of startup after long shutdown and ten cases of harmonic top-to-bottom and side-to-side tilts. The ROP detector responses at each detector location were obtained from the thermal neutron fluxes calculated for each flux shape, which was performed by the INTREP module of the RFSP code. The ripples used in this assessment were obtained from a 600 full power days (FPDs) refueling simulation and a total of 121 ripples were generated for every 5-FPD.

III.2.Calculation Results

The TSP calculations were performed by the ROVER-F to assess the ROP system characteristics of the DUPIC core. The uncertainties used for calculations are given in Table 2 and the calculation results are presented in Table 5 for 15 worst cases. In this result, the TSP of the DUPIC core is 123%, which is almost the same as that of the current ROP TSP of the natural uranium core (Wolsong-1), 122%. Therefore, it is expected that the loading of the DUPIC fuel in the CANDU-6 reactor does not deteriorate the ROP TSP. In fact, an increase of the TSP seems to be attainable with the DUPIC fuel with current values of the various uncertainties.

The effect on trip probability was calculated in the event that a single detector has failed. This task is performed for each detector to obtain the trip probability and TSP in the event that a detector has failed, which is outlined in Table 6. Given these set-points, the value to be used on the failure of a single detector is determined. It can be seen that the TSP changes are negligible for some detectors, which indicates that either the detector in question is well backed up by a second detector or the cases protected by these detectors are not near-limiting. For example, the TSP does not change in case of the SDS1 detector failure, while the maximum decrease of the TSP is ~11% in case of some SDS2 detector failures.

REFORM is a process that attempts to improve the ROP margin by changing the reference power shape of the core. In the REFORM process, an excess margin (the amount by which the margin to dryout exceeds the margin to trip) is determined for each channel in the core. The channel power in each fuel channel is then adjusted, in small increments, to minimize this excess margin. Since the overall reactor power is to be conserved, the revised power shape is normalized. This has an effect of adding powers to the channels with excess margins and removing powers from channels with small excess margins. As a result, in the most limiting channels, the channel power is reduced, which leads to a larger margin to dryout and an increased permissible ROP TSP.

The possibility of increasing TSP in the DUPIC core was investigated by performing the REFORM calculation. The calculation result showed that the TSP increases to 125.7%, which is 3% higher than that of the nominal TSP. This is, however, a theoretical improvement and it would be necessary to be checked against operational considerations.

IV. CONCLUSION AND FUTURE WORK

The ROP assessment was performed for a CANDU 6 reactor with the DUPIC fuel. The validation calculation showed that the difference between the WIMS/RFSP/ROVER-F and the design code system is less than 1%, which indicates that the ROP system analysis based on the WIMS is acceptable.

The ROP assessment of the CANDU DUPIC core was carried out for design-basis cases. The results of the assessment can be summarized as follows:

- The TSP of the DUPIC fuel core is increased by 1% compared to the natural uranium fuel core, with current values of various uncertainties. The improvement of the ROP TSP is due to the 43-element fuel bundle geometry and the flattened axial channel power shape.
- In case of a single detector failure, the TSP changes for some detectors are negligible, but the TSP could be decreased by ~11% for the SDS2 detector failures.
- It was found that the reforming process of the CP distribution can increase the TSP by 3%.

Consequently, it is expected that the loading of the DUPIC fuel in a CANDU 6 reactor does not deteriorate the current ROP trip TSP designed for the natural uranium fuel. In the future, the uncertainties introduced by the variation of the DUPIC fuel composition should be evaluated as they affect the CP and CCP.

ACKNOWLEDGEMENT

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REFERENCES

1. C.M. BAILEY and F.A.R. LARATTA, "Design and Assessment of the Neutron Overpower Trips in the 600 MWe CANDU-PHW Reactors," TDAI-315, Atomic Energy of Canada Limited, 1982.
2. C.M. BAILEY, R.D. FOURNIER and F.A.R. LARATTA, "Regional Overpower Protection in CANDU Power Reactors," AECL-7549, Atomic Energy of Canada Limited, 1982.
3. C.M. BAILEY and F.A.R. LARATTA, "Final Assessment of the Point Lepreau Neutron Overpower Trips," TDAI-322, Atomic Energy of Canada Limited, 1982.
4. C.M. BAILEY and G.K. GOMES, "Design ROPT Error Analysis for Wolsong-1," TTR-289 Part 3 (W1), Atomic Energy of Canada Limited, 1995.
5. J.V. DONNELLY, "WIMS-CRNL: A Users Manual for the Chalk River Version of WIMS," AECL-8955, Atomic Energy of Canada Limited, 1986.
6. D.A. JENKINS and B. ROUBEN, "Reactor Fuelling Simulation Program - RFSP: User’s Manual for Microcomputer Version," TTR-321, Atomic Energy of Canada Limited, 1991.
7. J. PITRE, "ROVER-F Manual," TTR-605 (Rev. 1), Atomic Energy of Canada Limited, 1999.
8. D.B. MILLER and E.S.Y. TIN, "POWDERPUFS-V Users Manual," TDAI-31 Part 2, Atomic Energy of Canada Limited, 1976

Table 1. Case Set for Natural Uranium Core Analysis

Case	Name	Description
1	SSSC50	Steady-state with spatial control
37	D14C50	Zone drain 14 from 50%
39	D02C80	Zone drain 02 from 80%
42	D05C80	Zone drain 05 from 80%
44	D07C80	Zone drain 07 from 80%
46	D09C80	Zone drain 09 from 80%
49	D12C80	Zone drain 12 from 80%
51	D14C80	Zone drain 14 from 80%
53	D02N50	Zone drain 02 from 50%
58	D07N50	Zone drain 07 from 50%
60	D09N50	Zone drain 09 from 50%
65	D14N50	Zone drain 14 from 50%
114	MCAN2H	MCA 1st bank full-in and 2nd bank half-in
122	ZTSFSE	1st azimuthal side/side
123	ZTSESF	1st azimuthal side/side
130	ZT2A01	2nd azimuthal 135,315 HI
131	T2A02	2nd azimuthal 45,225 HI
152	SSSD03	Adjuster bank 7 full-in/no time step
171	SA4403	Adjuster bank 7 full-in
173	SA4405	Adjuster bank 6 full-in
174	SA4406	Adjuster bank 6 full-in/Xe @ 4.3 min
177	SA4409	Adjuster bank 4 full-in
178	SA4410	Adjuster bank 4 full-in/Xe @ 3.9 min
195	SBCK05	Adjuster bank 2 out/Xe @ 18.3 min
196	SBCK06	Adjuster bank 3 out
197	SBCK07	Adjuster bank 3 out/Xe @ 28.5 min
222	ABHO01	Startup bank 7 half-in

Table 2. Uncertainty Data for DUPIC Core Analysis

Uncertainty	Value (%)
Detector Random	2.60
Channel Random	1.97
Common Random	4.18
Bias	+0.14

Table 3. Trip Confidence for Natural Uranium Core with ROP TSP of 125% (PPV-based)

Case		CPRL	Average conservatism factor	Trip confidence		Limiting detectors	
Number	Name			SDS1	SDS2	SDS1	SDS2
42	D05C80	1.2603	1.0347	0.9929	0.9559	4D	7J
173	SA4405	1.1863	1.0294	0.9615	0.9605	5F	4J
49	D12C80	1.2614	1.0391	0.9967	0.9642	9E	8H
114	MCAN2H	1.0417	1.0243	0.9918	0.9647	11D	7H
152	SSSD03	1.0826	1.0305	0.9976	0.9734	6F	4J
178	SA4410	1.2724	1.0274	0.9959	0.9748	7D	4J
44	D07C80	1.2802	1.0339	0.9953	0.9782	4E	4H
51	D14C80	1.2886	1.0306	0.9929	0.9859	10E	4G
122	ZTSFSE	1.2471	1.0372	0.9937	0.9909	11E	1G
39	D02C80	1.2881	1.0427	0.9994	0.9916	2E	2G
171	SA4403	1.2085	1.0303	0.9992	0.9917	3F	4J
196	SBCK06	1.2501	1.0398	0.9991	0.9918	6F	4J
123	ZTSESF	1.2483	1.0444	0.9968	0.9933	1D	2G
46	D09C80	1.2811	1.0512	0.9973	0.9936	6D	3H
197	SBCK07	1.2630	1.0416	0.9990	0.9940	4F	4J
Required ROP Set-point for 98% Trip Confidence = 1.2177							

Table 4. Trip Confidence for Natural Uranium Core with ROP TSP of 125% (WIMS-based)

Case		CPRL	Average conservatism factor	Trip confidence		Limiting detectors	
Number	Name			SDS1	SDS2	SDS1	SDS2
44	D07C80	1.2710	1.0298	0.9942	0.9701	9F	4H
51	D14C80	1.2636	1.0333	0.9893	0.9800	10E	1G
39	D02C80	1.2541	1.0479	0.9987	0.9802	2E	2G
152	SSSD03	1.1018	1.0262	0.9998	0.9806	6F	4J
123	ZTSESF	1.2072	1.0462	0.9897	0.9812	1D	2G
171	SA4403	1.1919	1.0239	0.9997	0.9832	3F	4J
122	ZTSFSE	1.2143	1.0418	0.9893	0.9841	11E	1G
46	D09C80	1.2615	1.0408	0.9925	0.9874	5D	3H
130	ZT2A01	1.2752	1.0525	0.9962	0.9890	1D	2G
114	MCAN2H	1.0318	1.0303	0.9988	0.9897	11D	7H
53	D02N50	1.2959	1.0492	0.9983	0.9915	2E	7G
131	ZT2A02	1.2833	1.0422	0.9916	0.9917	9F	7G
58	D07N50	1.2950	1.0450	0.9967	0.9919	3F	4H
60	D09N50	1.2867	1.0558	0.9964	0.9923	6D	3H
65	D14N50	1.3050	1.0367	0.9955	0.9942	9F	2J
Required ROP Set-point for 98% Trip Confidence = 1.2290							

Table 5. Trip Confidence for DUPIC Fuel Core with ROP TSP of 125%

Case Number	Case Title	CPRL	Average conservatism factor	Trip confidence SDS1	Trip confidence SDS2	Limiting detectors SDS1	Limiting detectors SDS2
49	D12C80	1.2873	1.0281	0.9984	0.9749	9E	8H
42	D05C80	1.2873	1.0288	0.9966	0.9751	4D	7J
112	MCAN1H	1.1730	1.0279	0.9993	0.9785	11D	7H
44	D07C80	1.2877	1.0431	0.9982	0.9859	9F	4H
39	D02C80	1.2938	1.0404	0.9996	0.9900	2E	2G
108	MCAC1H	1.2908	1.0260	0.9993	0.9908	2E	8J
46	D09C80	1.2856	1.0438	0.9968	0.9918	6D	3H
51	D14C80	1.2966	1.0399	0.9965	0.9919	10E	4G
114	MCAN2H	1.1020	1.0285	0.9951	0.9944	11D	7H
123	ZTSESF	1.2555	1.0440	0.9976	0.9952	1D	2G
110	MCAC2H	1.1682	1.0241	0.9972	0.9955	2E	7H
126	ZTT045	1.2659	1.0499	0.9992	0.9957	1D	8G
122	ZTSFSE	1.2576	1.0455	0.9974	0.9962	11E	1G
120	ZT1ATB	1.2837	1.0335	0.9997	0.9965	10F	8H
38	D01C80	1.3159	1.0524	0.9974	0.9967	5F	2H
Required ROP Set-point for 98% Trip Confidence = 1.2336							

Table 6. TSP for Single Detector Failure

SDS1 Detector				SDS2 Detector			
Detector	TSP	Detector	TSP	Detector	TSP	Detector	TSP
1D	1.2336	6E	1.2336	1G	1.2336		
2D	1.2336	7E	1.2336	2G	1.2226	1J	1.2333
3D	1.2336	8E	1.2336	3G	1.2334	2J	1.2326
4D	1.2336	9E	1.2336	4G	1.2332	3J	1.2215
5D	1.2336	10E	1.2336	5G	1.2336	4J	1.1689
6D	1.2336	11E	1.2336	6G	1.2336	5J	1.2336
7D	1.2336	1F	1.2336	7G	1.2100	6J	1.2336
8D	1.2336	2F	1.2336	8G	1.2202	7J	1.1947
9D	1.2336	3F	1.2336			8J	1.1960
10D	1.2336	4F	1.2336	1H	1.2332		
11D	1.2254	5F	1.2336	2H	1.2312		
12D	1.2336	6F	1.2336	3H	1.2165		
1E	1.2159	7F	1.2336	4H	1.2111		
2E	1.2336	8F	1.2336	5H	1.2323		
3E	1.2336	9F	1.2336	6H	1.2336		
4E	1.2336	10F	1.2336	7H	1.1204		
5E	1.2336	11F	1.2336	8H	1.2002		

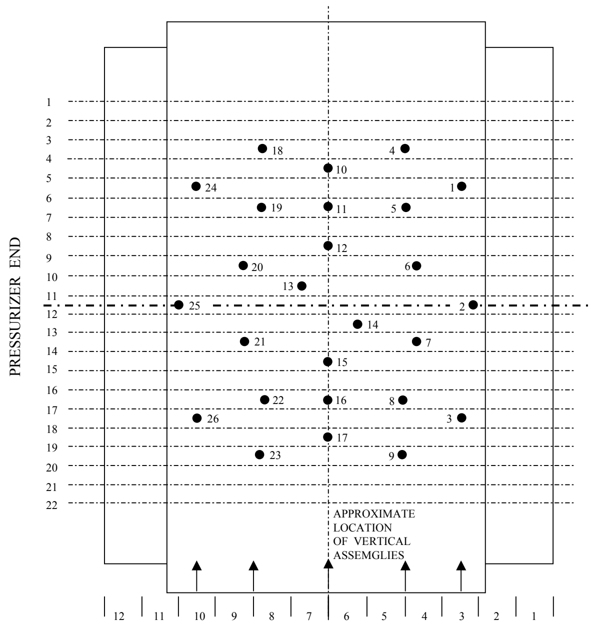

Fig. 1 Top View Showing Vertical Flux Detector Assemblies

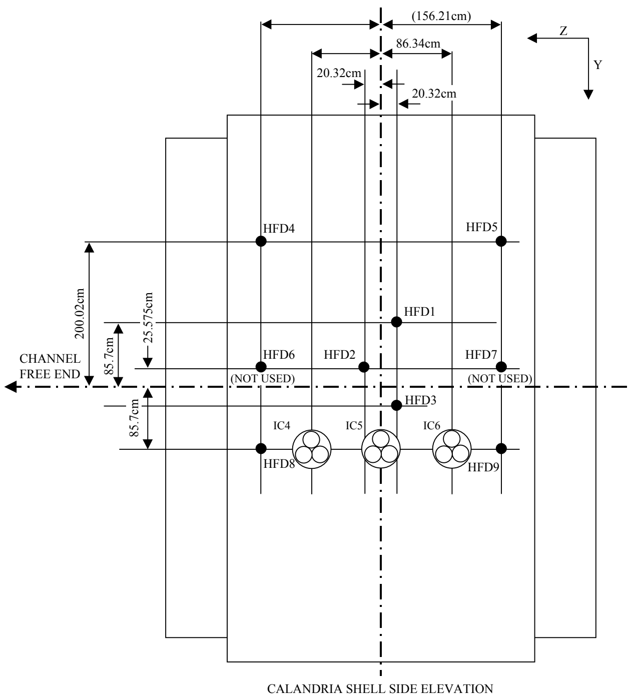

Fig. 2 Side Elevation View of SDS2 Horizontal Flux Detector Assemblies