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Sensitivity Study of Break Opening Characteristics for Large LOCA using CATHENA

A. Oussoren, P. Wan and A. Delja
Canadian Nuclear Safety Commission, Ottawa, Ontario, Canada
andrew.oussoren@cnsc-ccsn.gc.ca, pak.wan@cnsc-ccsn.gc.ca, aleksander.delja@cnsc-ccsn.gc.ca

Abstract

A study was performed using a CATHENA based generic model of a CANDU6 reactor to demonstrate the sensitivity of system response to single and two-stage break opening characteristics following a Reactor Inlet Header Large Break LOCA. Figures of merit for the analysis were the power pulse amplitude and the 5 second integrated normalized power. The analysis demonstrates in a qualitative manner a threshold time for 'instantaneous' break opening, and a threshold initial break size above which shutdown system trip times remain unchanged. The analysis also attempts to establish the relative impacts of initial and second stage opening times on figures of merit.

Keywords: Thermalhydraulics, LOCA, Modelling

1. Introduction

Large Break Loss of Coolant Accident (LOCA) is a design basis accident for nuclear power plants which postulates break of a large diameter pipe in the heat transport system. In CANDU reactors the early phase of this accident is characterized by a rapid drop in system pressure and slowing or reversal of flow in the fuel channels downstream of the break. Voiding of coolant in the fuel channels causes reactivity to increase at a rate for which the reactor regulating system cannot compensate. Consequently, rapid initiation of reactor shutdown system is necessary to mitigate the resulting power pulse. The rise in reactor power combined with degraded cooling results in heat up of the fuel which poses a challenge to maintaining fuel integrity.

The Canadian nuclear industry has proposed a Composite Analytical Approach (CAA) for analysis of Large Break Loss of Coolant Accidents (LBLOCA). An important aspect of the CAA methodology is to model the opening of a pipe break in two stages:

1) Break opens to 10% of the total break flow area in 5 ms (instantaneous opening)

2) Break continues to open to 100% of the total break flow area over the following 5 seconds, with a linear increase in flow area

The current practice in safety analysis is to model a break opening to 100% of the total break flow area “instantaneously”, typically assumed to be 1 to 10 ms. The proposed change will significantly impact safety analysis margins.

The purpose of this study is to qualitatively demonstrate sensitivity of the predicted safety analysis results to opening times for single-stage and two-stage break models, as well as the to the initial instantaneous opening area. The thermal hydraulics analysis code CATHENA was used for the assessment.

It should be stressed that this study is qualitative in nature. The purpose is to demonstrate the trends with changing breaking opening characteristics, as opposed to quantifying safety analysis parameters for individual break opening models.

2. Methodology

2.1 CATHENA model

This study used the safety analysis code CATHENA 3.5d revision 2. A generic nodalization of a CANDU6 reactor was used for the analysis. The nodalization of the primary heat transport system is a ‘3 and 7’ design, where the three unbroken core passes are each modelled as a single pass, each pass representing 95 fuel channels lumped together. The 4^{th} pass (the broken pass) is divided into 7 passes each of which represent a number of affected fuel channels in the core. A simplified schematic of the nodalization scheme is presented in Figure 1.

The break was modelled at the inlet header upstream of the 4^{th} core pass. The flow area of the break was controlled as a function of time via a look up table to produce a single or two-stage break. A maximum time step size of 1 millisecond was used for all analyses.

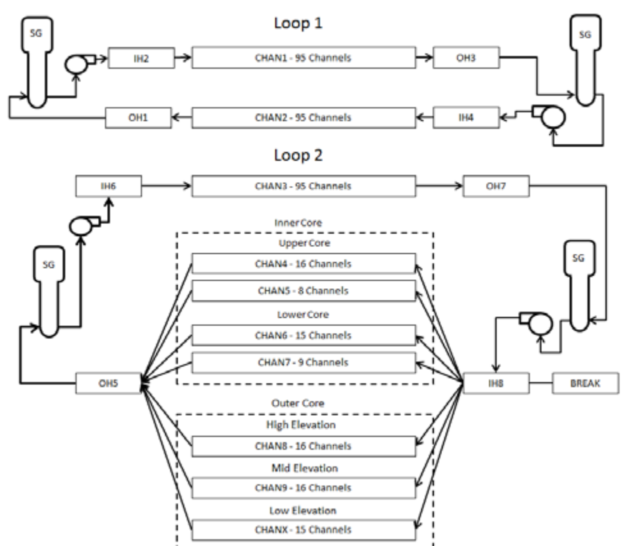


Figure 1: Simplified nodalization scheme of CATHENA CANDU6 model used in this study

2.2 Reactor Kinetics

A key phenomenon for LOCA analysis of CANDU reactors is reactivity insertion due to coolant voiding. Because CANDU reactors have a positive channel coolant void reactivity (CVR) coefficient, voiding in the core in the early stages of a transient will cause a rapid increase in power, called a power pulse. The power pulse is arrested by rapid intervention of either of the two shutdown systems.

In CATHENA, core reactivity changes and their impact on fission power is tracked using a point kinetics model. In this model, the initial, steady-state core conditions are assigned a reactivity value of 1.0. Core-wide average values for coolant void, coolant temperature and fuel temperature are tracked as time progresses and compared against reactivity feedback tables for these same parameters. At each time step, core reactivity changes are calculated and its impact on core fission power is quantified.

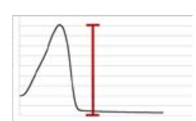
For this analysis, the reactivity effects of coolant temperature and fuel temperature were ignored, as they are small compared to the coolant void reactivity feedback. Void reactivity is calculated as a fraction of the full core CVR based on the average void fraction over all channels at a given time step.

Neutronic trips are the first to activate during Large Break LOCA. The Reactor Over Power (ROP) and High LOG rate (HLOG) trips for SDS1 were modeled using trip times and time constants typical for the CANDU6 design. As well, a typical SDS1 reactivity insertion table is used, representing reactivity insertion due to rod drop in core. SDS1 activation was assumed to occur when the second (backup) neutronic trip reached its set point.

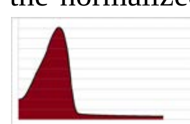
2.3 Figure of Merit

The two-stage break model will impact the characteristics of the power pulse, therefore the two figures of merit (FOM) chosen for this analysis were:

1) Max normalized reactor power, or the amplitude of the power pulse:



2) Integrated normalized reactor power, or adiabatic energy deposition in the fuel over the first 5 seconds of the transient. This is calculated by integrating the area under the normalized power curve:



The FOM are strongly influenced by voiding rate in the core and reactor trip time, therefore these values are also presented for each study.

3. Benchmarking

In order to assess the accuracy of the model, the point kinetics model was first benchmarked against a CANDU6 Large Break LOCA assessment conducted using the WIMS/CATHENA /RFSP code suit. This benchmark analysis is detailed in [1]. The transient modeled in [1] was a 100% Reactor Inlet Header (RIH) LOCA. The same transient was simulated using the present CATHENA model for a direct comparison of the results.

Full core CVR was initially implemented in the generic model based on the most recent values reported in [2]. As shown in Figure 2, the resulting power pulse amplitude was significantly smaller than the reported value in the benchmark case. Therefore a scoping study was performed by adjusting the full core CVR to match the power pulse amplitude in the benchmarking case. The result was a full core CVR of 22.40mk which was implemented for the rest of the analyses in this report.

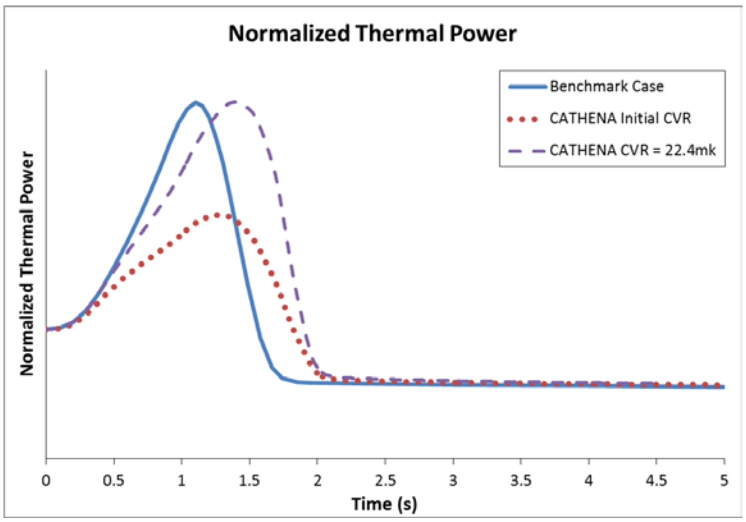


Figure 2: Comparison of CATHENA predicted power pulse for initial and 22.40mk CVR, and the benchmark case

As seen in Figure 2 the power pulse produced using the CATHENA model with a CVR of 22.40mk produces a delayed peak and is wider than the benchmark case. This results in an integrated 5 second power which is approximately 30% higher than the benchmark case. The discrepancy is a result of using a point kinetics approximation for reactivity feedback as opposed to the 3D model used in the benchmark case. The point kinetics model does not discern localized changes in reactor power, which could activate neutronic trips earlier than is seen in the current model, which applies power changes to the bulk reactor power. Again, the primary interest of this study is the relative change in values between different break opening models.

4. Test Matrix

The test matrix consisted of a reference case (comparison of the instantaneous break model and the CAA two-stage break model) followed by a series of sensitivity studies for single and two-stage break opening models. Break opening sizes are reported as a percentage of a full double ended guillotine break.

Reference Case

This study compared the results for the CAA two-stage break model to the 'instantaneous' break model.

Instantaneous: 100% break opening in 5ms

Two-stage: 10% break opening in 5ms, opening to 100% over 5s.

Sensitivity Studies - Single Stage Breaks

Study A: In order to demonstrate the impact of single stage opening time, a series of 100% RIH breaks were modeled with the opening time varied between 1ms and 10s.

Study B: In order to demonstrate the impact of break opening size, a series of single stage RIH breaks with a 5ms opening time were modeled, with break size varied between 5% and 100% of total flow area.

Sensitivity Studies - Two-Stage Breaks

Sensitivity to the timings of the two-stage break model was demonstrated in studies C through E. A first stage opening time of 5ms was used in all cases, while the size of the initial break was varied between studies. For each study the second stage time for 100% break opening time was varied between 0.5s and 10s.

Study C: 5% RIH break opening in 5ms, 100% opening time varied between 0.5 and 10s

Study D: 10% RIH break opening in 5ms, 100% opening time varied between 0.5 and 10s

Study E: 40% RIH break opening in 5ms, 100% opening time varied between 0.5 and 10s

5. Results and Discussion

5.1 Reference Case

The objective of this study was to compare the CAA break model to the instantaneous break model. The normalized thermal power and the integrated void transients for the inner core are presented in Figure 3, with trip timings shown in Table 1.

As seen in the integrated void fraction plot in Figure 3, the void fraction for the two-stage break model rises at a much slower pace compared to the instantaneous single stage break. This results in a low void fraction during the early stage of the transient, which limits the positive reactivity insertion due to CVR. However the amount of initial voiding is sufficiently large to initiate reactor trip in almost the same time as the reference case, thus reactor trip is not significantly delayed (the trip time in this case is mostly dependent on trip time constants). The overall result is a smaller power pulse and the adiabatic energy deposition in the fuel (the integrated 5 second power) is reduced by 43% in the two-stage model compared to the instantaneous break model.

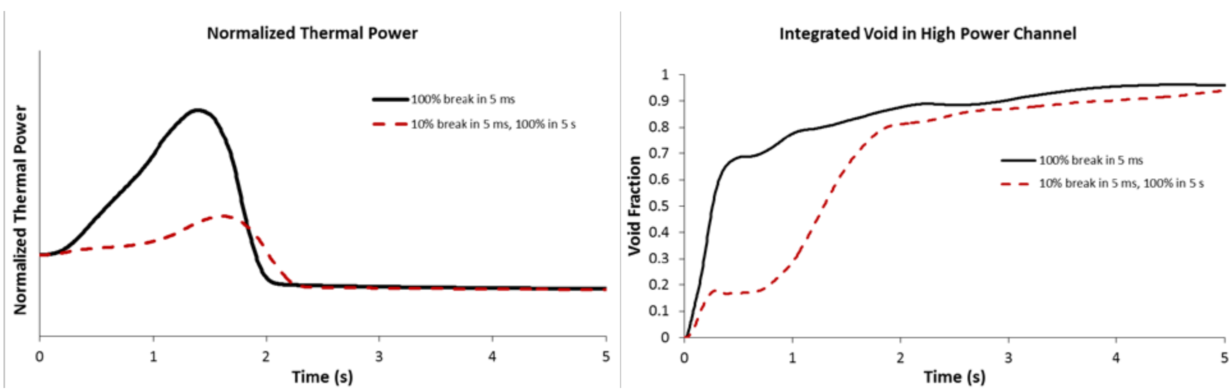


Figure 3: Normalized thermal power and channel void for the reference two-stage break model

Break Model	Backup Trip	Trip Time
Instant	HLOG	0.57
Two-Stage	HLOG	0.61

Table 1: Figures of merit and trip times for the instantaneous and reference two-stage break model

5.2 Study A: Single Stage Break Opening Time Study

The objective of this study was to investigate sensitivity to single stage break opening time. The FOM trends, as a function of break opening time, are presented graphically in Figure 4. Trip times and the backup trip identifier are listed in Table 2. Integrated void in the inner core is shown in Figure 5.

This study demonstrates that the FOM values are insensitive to break opening times below 100ms. This demonstrates a threshold break time below which a break may be considered "instantaneous". The exact value of this threshold time is expected to be dependent on the geometry of the primary heat transport system.

The threshold break time arises from the inertia of the coolant: there is a finite amount of time required for the coolant, initially flowing from the inlet header to the outlet header, to slow down, stop, and reverse direction towards the break. At the threshold break time, the coolant is turning around and leaving the core at its maximum rate, therefore a further reduction in break opening time has no impact. Figure 5 shows that voiding behavior remains unchanged up to a 100 millisecond opening time.

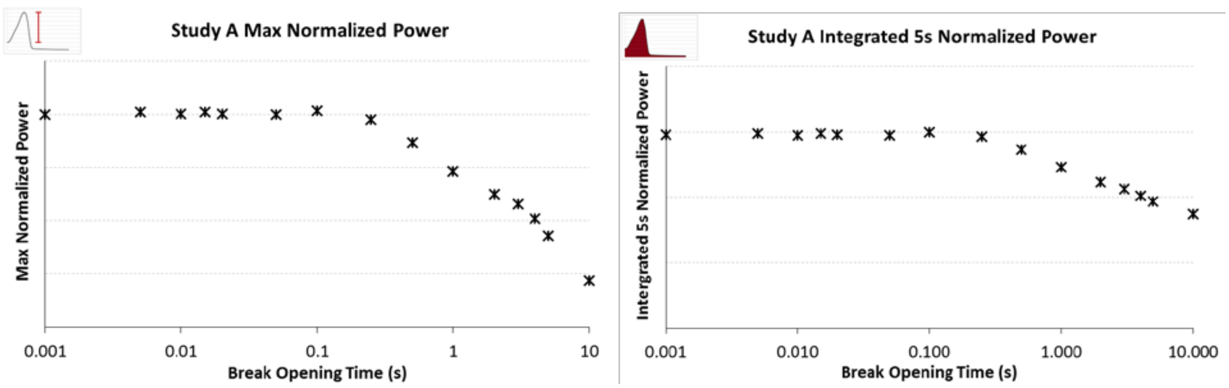


Figure 4: Max normalized power vs. break opening time, study A

Break Opening Time (s)	Backup Trip	Trip Time (s)
0.001	HLOG	0.57
0.050	HLOG	0.57
0.100	HLOG	0.59
0.250	HLOG	0.61
0.500	HLOG	0.66
1.000	HLOG	0.72
2.000	HLOG	0.86
3.000	HLOG	0.95
4.000	HLOG	1.04
5.000	ROP	1.15
10.000	ROP	1.53

Table 2: Study A Trip Timings

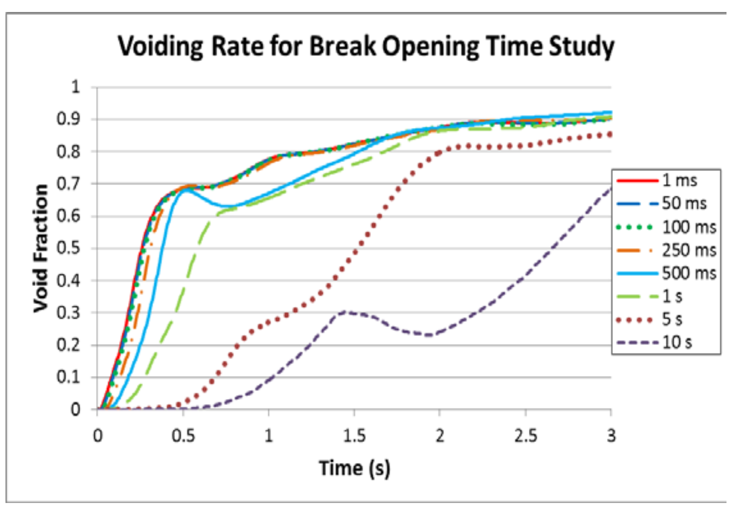


Figure 5: Voiding rates for different break opening times

5.3 Study B: Single Stage Break Opening Size Study

The objective of this study was to investigate sensitivity to single stage break size. The FOM trends, as a function of break opening time, are presented graphically in Figure 6. Trip times and the backup trip identifier are listed in Table 3. Integrated void in the inner core is shown in Figure 7.

This study demonstrates that there is a threshold break size between 10% and 20% flow area above which reactor trip time does not change. This minimum trip time is a characteristic of the shutdown system. As break size is reduced from 100% coolant voiding rate decreases, which reduces the reactivity insertion and the FOM. However at very small break sizes, the reactivity insertion is low enough to delay neutronic trip activation. This causes integrated 5 second power to increase as the break gets smaller, as the reactor remains at power longer. At very small breaks the maximum power may also increase.

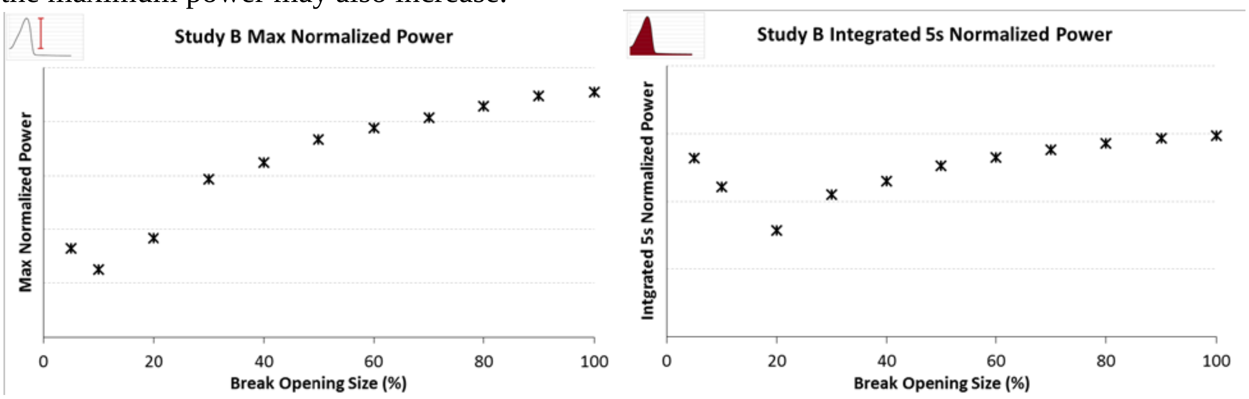


Figure 6: Max normalized power vs. break size, study B

Break Size (%)	Backup Trip	Trip Time (s)
5	HLOG	12.64
10	ROP	2.56
20	HLOG	0.57
30	HLOG	0.57
40	HLOG	0.57
50	HLOG	0.57
60	HLOG	0.57
70	HLOG	0.57
80	HLOG	0.57
90	HLOG	0.57
100	HLOG	0.57

Table 3: Study B Trip Timings

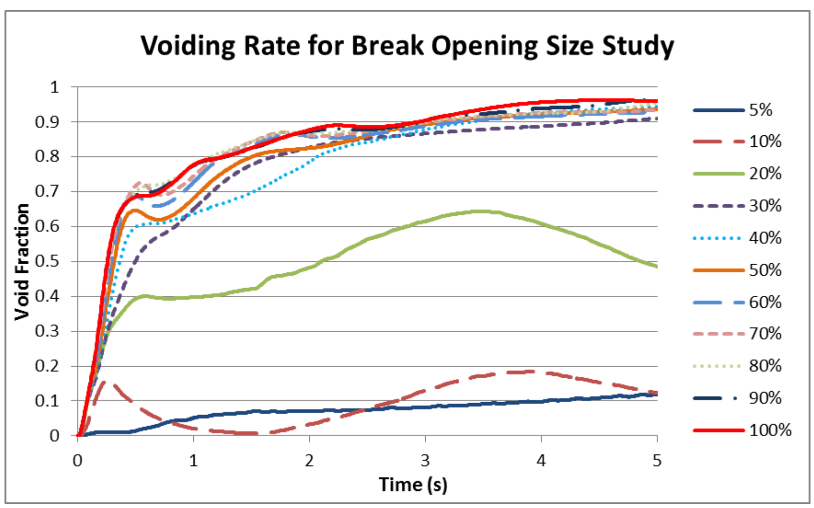


Figure 7: Voiding rates for different break sizes

5.4 Study C: Two-Stage Break with 5% Initial Break

The objective of this study was to investigate sensitivity of predicted FOM to second stage opening time in combination with an instantaneous initial opening size of 5% flow area. The FOM trends as a function of break opening time are presented graphically in Figure 8. Trip times and the backup trip identifier are listed in Table 4. Integrated void in the inner core is shown in Figure 9.

This study demonstrates that for a small initial break, the voiding rate, trip timing and FOM values depend strongly on second stage opening time. FOM values drop most rapidly as second stage opening time increases from 0.5 to 2 seconds.

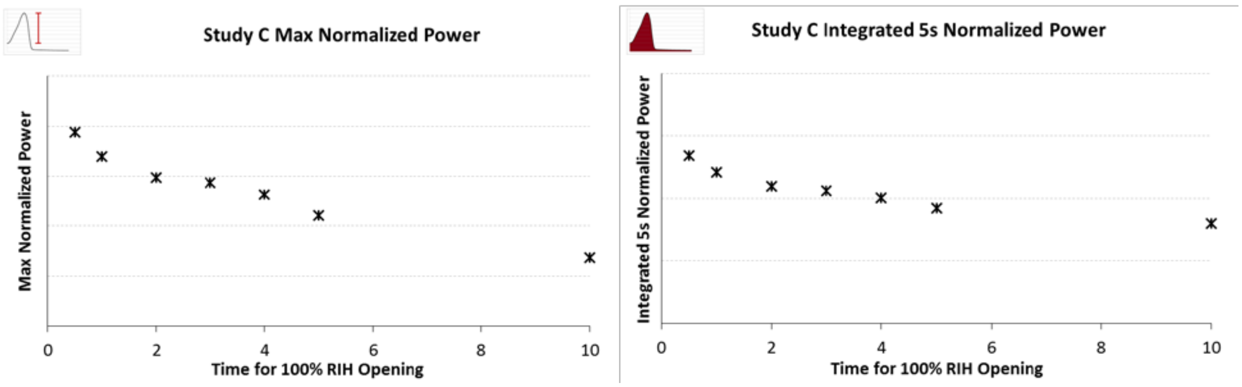


Figure 8: Max normalized power vs. second stage break opening time, study C

Time for 100% Opening	Backup Trip	Trip Time (s)
0.5	HLOG	0.63
1	HLOG	0.68
2	HLOG	0.78
3	HLOG	0.85
4	HLOG	0.90
5	HLOG	0.93
10	HLOG	1.17

Table 4: Study C Trip Timings

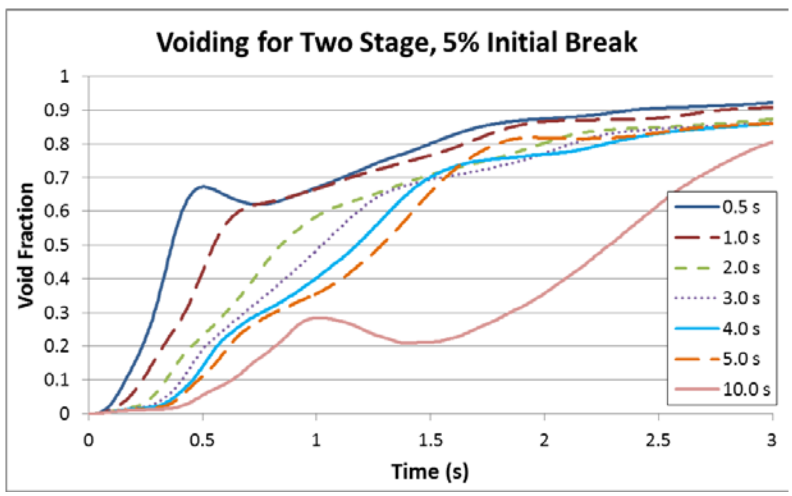


Figure 9: Voiding rates for study C

5.5 Study D: Two-Stage Break with 10% Initial Break

The objective of this study was to investigate sensitivity of predicted FOM to second stage opening time in combination with an instantaneous initial opening size of 10% flow area. The FOM trends as a function of break opening time are presented graphically in Figure 10. Trip times and the backup trip identifier are listed in Table 5. Integrated void in the inner core is shown in Figure 11.

This study demonstrates that for this size of initial break the second stage opening time has little impact on trip timing, however it still has a significant influence on the FOM values. Figure 11 shows that there is an initial rise in void for all second stage opening times, which is characteristic of the initial break size. This initial voiding is sufficient to initiate reactor trip, as trip times are relatively unchanged for all cases. Subsequent voiding behavior, which is still strongly dependent on second stage opening time, determines the characteristics of the power pulse, with longer opening times resulting in a more gradual increase in void.

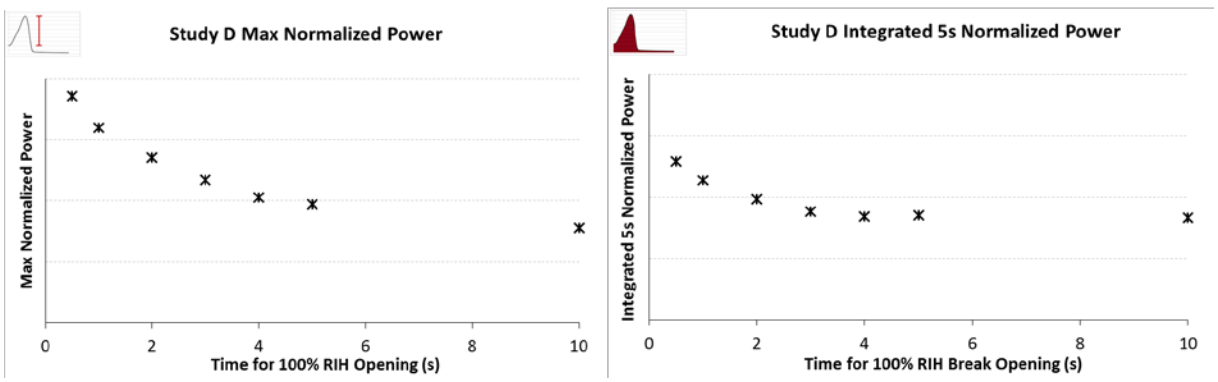


Figure 10: Max normalized power vs. second stage break opening time, study D

Time for 100% Opening (s)	Backup Trip	Trip Time (s)
0.5	HLOG	0.60
1	HLOG	0.61
2	HLOG	0.61
3	HLOG	0.61
4	HLOG	0.61
5	HLOG	0.61
10	HLOG	0.62

Table 5: Study D Trip Timings

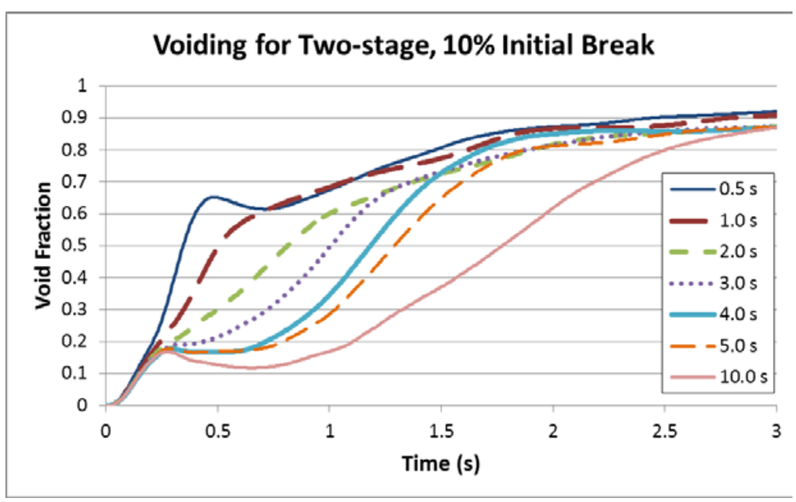


Figure 11: Voiding rates for study D

5.6 Study E: Two-Stage Break with 40% Initial Break

The objective of this study was to investigate sensitivity of predicted FOM to second stage opening time in combination with an instantaneous initial opening size of 40% flow area. The FOM trends as a function of break opening time are presented graphically in Figure 12. Trip times and the backup trip identifier are listed in Table 6. Integrated void in the inner core is shown in Figure 13.

This study demonstrates that if the initial break is large the FOM are relatively insensitive to second stage opening time. As seen in Figure 13, the second stage opening time does not significantly impact voiding rate for this study. The impact of the large initial break dominates the voiding characteristics, and subsequent changes to the break flow area have relatively little impact on the FOM values.

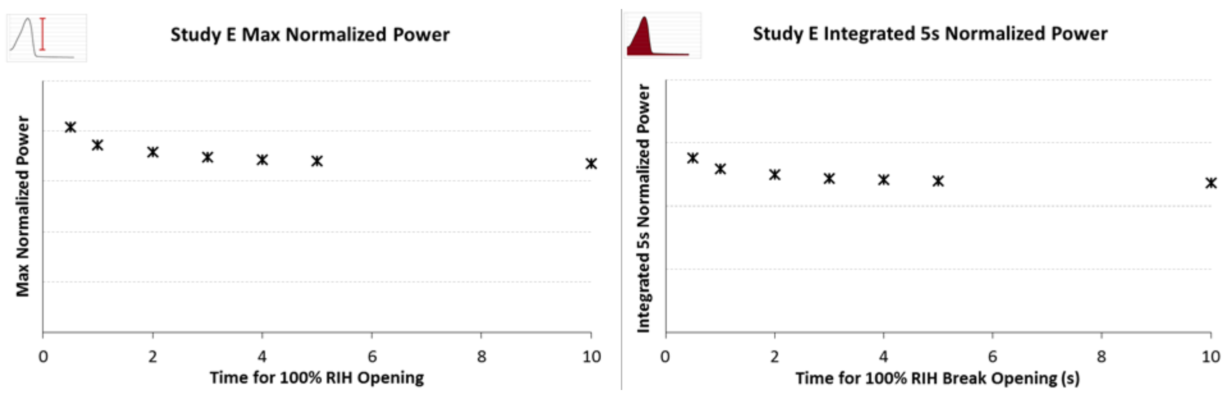


Figure 12: Max normalized power vs. second stage break opening time, study E

Time to 100% Opening (s)	Backup Trip	Trip Time (s)
0.5	HLOG	0.57
1	HLOG	0.57
2	HLOG	0.57
3	HLOG	0.57
4	HLOG	0.57
5	HLOG	0.57
10	HLOG	0.57

Table 6: Study E Trip Timings

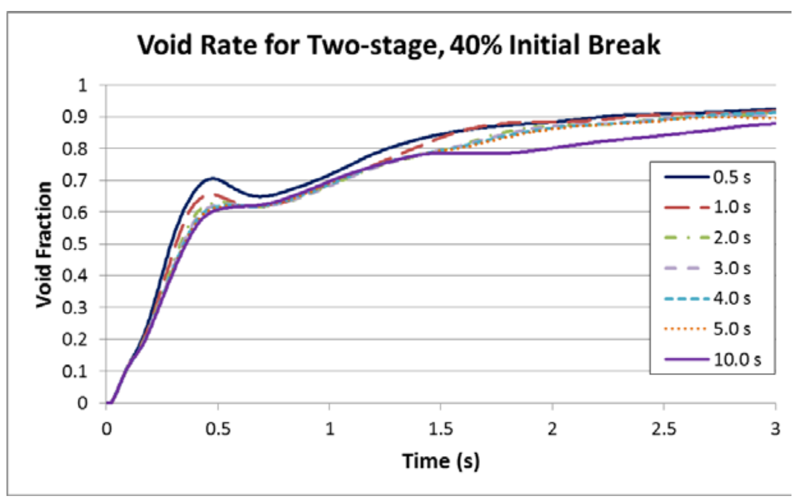


Figure 13: Voiding rates for study E

6. Summary of Two-Stage Model Results

The results of the two-stage break opening models are summarized in Figure 14, where FOM values are plotted against second stage opening time for various initial break sizes.

FOM are minimized for a 10% initial break. For larger initial breaks the more rapid voiding causes an increase in FOM due to a larger power pulse, while for smaller initial breaks the slower voiding results in a delayed reactor trip time. It is also evident that as initial break size increases the impact of second stage opening time decreases.

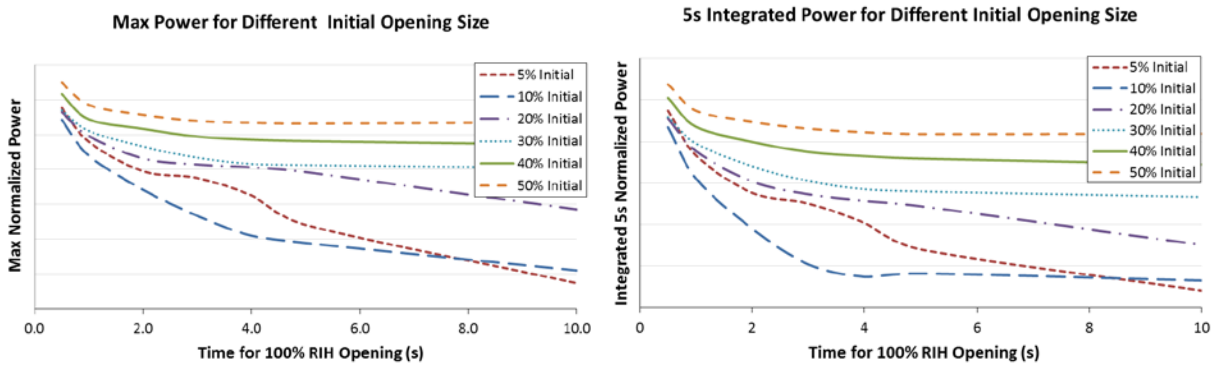


Figure 14: FOM values vs. 100% break opening time for different first stage break sizes

7. Conclusion

The two FOM parameters of maximum power and integrated 5 second power are affected by the voiding rate in the core and the reactor trip time. Increasing the voiding rate or delaying the reactor trip time will result in a larger FOM parameter.

Study A demonstrated that there is a threshold break opening time (in this case on the order 100ms) below which the initial voiding rate becomes insensitive (almost constant) due to inertia effects of the reactor coolant. Break opening times less than the threshold break opening time can be considered to be “instantaneous” breaks. For non-instantaneous breaks, a slower break opening time will result in a smaller initial voiding rate, which in turn will results in a smaller FOM parameters.

Study B demonstrated a threshold break size (in our case on the order of 10% to 20% RIH break) above which the reactor trip time remains essentially unchanged. This threshold break size will yield the minimum reactor trip time, which is a characteristic of the reactor shutdown system design. For break sizes below the reactor trip threshold break size (*i.e.*, break size <10%), a smaller break size will result in a longer reactor trip time, which in turn will result in a larger FOM. On the other hand, the break size also affects the voiding rate in the longer term. A smaller break size will result in a slower voiding rate in the longer term, which in turn will result in a smaller FOM.

The two-stage break models are defined by the initial opening size (the size of break which opens “instantaneously”) and the second stage time for the break to reach full flow area. For initial break sizes larger than the threshold break size of 10% trip timing is at a minimum, and longer second stage opening times results in lower FOM values. The influence of second stage time is more pronounced when the initial break is small; for larger initial breaks the high initial rate of voiding dominates system behavior.

For two-stage breaks with initial break size smaller than the threshold break size of 10% the slower voiding rate results in a smaller rise in power but also delayed reactor trip. As the initial break size decreases the impact of trip timing is the dominant effect and FOM values increase.

8. References

[1] W. Ross, "WIMS-IST Based CANDU6 LLOCA Power Transients Terminated by SDS1 - Break Survey Assessment," AECL Report TTR-771 Volume 3 Rev0, 2004.

[2] Fred P. Adams, “Coolant Void Reactivity Bias Closure Report, Rev1”, COG Joint Report COG-JP-4367-42-R1, April 2013.