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REGULATORY ASSESSMENT OF EFFECTIVENESS OF ACR-1000 EMERGENCY CORE COOLING SYSTEM

A. Tanase[1], R. Gibb[1] and A. Delja[1]
[1] Canadian Nuclear Safety Commission (CNSC) Ottawa, Ontario, Canada

Abstract

The paper presents the regulatory approach for assessment of the Advanced CANDU Reactor (ACR)-1000 Large Loss of Coolant Accident (LOCA) Emergency Core Cooling (ECC) effectiveness, describes the rationale for the selection of sensitivity cases and discusses the results of the simulations for 50% Pump Suction Break (PSB). The separate in-house simulations strengthened the CNSC staff knowledge about the ACR-1000 design and the modeling methodology. The review of representation of plant systems and plant behavior indicated no major issues. The selected accident scenarios and the limited scope sensitivity cases conducted by the CNSC staff, indicated that, overall, the ECC performance showed small sensitivity to the parameters and assumptions considered for investigation.

Introduction

CNSC staff has performed, at the request of Atomic Energy of Canada Limited (AECL), pre-project safety reviews of the ACR-1000 design and safety analysis. The first review, called Phase 1, was an overall review of design and safety issues to assess fundamental design concepts and approach. The second, the Phase 2 review, was more specific and a safety report, called the Generic Safety Case Report (GSCR), was provided. A Phase 3 was opted for in order to follow-up on specific items that required more effort. The advantage to all was to reduce the time for CNSC staff reviews at the time when an application to construct was made. One of the Phase 3 tasks was for the CNSC staff to perform simulations of ECC performance for some sample break cases to understand and assess effectiveness of the ECC design.

Key changes in the ACR-1000 design from existing Canada Deuterium-Uranium (CANDU)'s that affect ECC effectiveness are the larger steam generators, the long term cooling (LTC) injection flow is only to the headers, the header-to-header interconnect, the higher channel powers, and the direct connection of accumulators and core makeup tanks to the heat transport system (HTS).

One should note that the pre-design review is not part of the reactor licensing process. However, it is believed that this project would help the CNSC staff to be more familiar with the ACR-1000 design, thus reducing the review time during licensing.

1. Objectives

The purpose of the Phase 3 simulations was to assess the robustness of the design to changes in operating conditions such as ageing effects or differences between an as-built plant from design. This work did not include a review of code validation work and accepted the simulations as having sufficient fidelity. As well, the intent was not to study the potential benefit of making design changes, but rather investigation of sensitivity of operational or modeling parameters. The results of simulations that reflected the performance of ECC were assessed against the requirements of CNSC Regulatory Document RD-337 [1] and other relevant regulatory documents such as RD-310 [2].
Section 8.5 of RD-337 requires that:

"The ECCS meets the following criteria for all Design Basis Accidents (DBAs) involving loss of coolant:

1. All fuel in the reactor and all fuel assemblies are kept in a configuration such that continued removal of the residual heat produced by the fuel can be maintained; and
2. A continued cooling flow (recovery flow) is supplied to prevent further damage to the fuel after adequate cooling of the fuel is re-established by the ECCS."

The assessment of regulatory requirements and expectations was performed by the review of the documentation submitted by AECL, namely Generic Safety Case Report (GSCR), during the Phases 1 and 2 of the project. However, a more detailed and complete assessment was achieved by simulations of a limited number of cases, representative for ECC performance.

The question was: if the plant condition or operation were somewhat different that assumed in the GSCR, would ECC continue to still be shown to be effective?

This would be addressed by repeating some LOCA cases reported in the GSCR. Analyzing the output where we can take advantage of the ability to output more detailed information so that we can examine specific features. Then, taking advantage of this insight, running a number of sensitivity cases where we would change the plant condition or operating condition to determine the impact on ECC behaviour and performance.

The cases requested were for the Large Break 50% Pump Suction break with Class IV Power available and a 2.5% RIH break with the loss of Class IV power. These appeared to best illustrate the behaviour and the dependence on new design features as discussed in the previous section. In the present paper only 50% PSB is presented.

2. Differences in the design between ACR-1000 and CANDU-6

The ACR-1000 is a 1200MWe evolutionary pressure tube reactor, perceived as the successor of CANDU-6. It is a light water cooled, heavy water moderated reactor, with 520 pressure tubes, two primary heat transport loops, each containing two primary pumps, two steam generators, two reactor inlet headers (RIHs), two reactor outlet headers (ROHs), feeders, pressure tubes and primary pipes. A simplified schematic of one primary loop (i.e. half of the primary circuit) is presented in Figure 1. The ACR-1000 retrains main features of CANDU-6 design, such as a modular, horizontal fuel channel core, a low-temperature heavy water moderator, two independent diverse shutdown systems, as well as on-power fuelling and reactor building accessibility for on-power maintenance.

Key changes in the ACR-1000 design from the existing CANDU's that may affect the ECC effectiveness are:

- larger steam generators - it was noted that the ACR-1000 has four steam generators, each steam generator being taller and larger than that of the CANDU-6 design. The height of the steam generator tubing may influence the flow rates during ECC injection and two-phase spatial distribution (i.e. vapour locking behaviour).
- LTC injection flow is only to the inlet headers - another significant difference consists in the long term cooling injecting only to the inlet headers of the broken loop, which alters the flow directions in the core passes.
- header-to-header interconnect, designed to transfer ECC coolant from one core pass to another, which is a major difference versus the "traditional" CANDU6 ECC design. (see also Figure 1).
- higher channel powers and fuel geometry (43 vs, 37 fuel pins), which requires higher ECC flow rates, to ensure adequate removal of residual heat.
- connection of accumulators and core makeup tanks to the HTS; these new features are designed to compensate for the coolant shrinkage and small coolant loss. The core make-up tanks inventory is maintained at a pressure and temperature close to that of the primary system, hence they are able to inject water into the primary system if the primary pressure or inventory decrease. These innovative features provide some passive cooling capability.

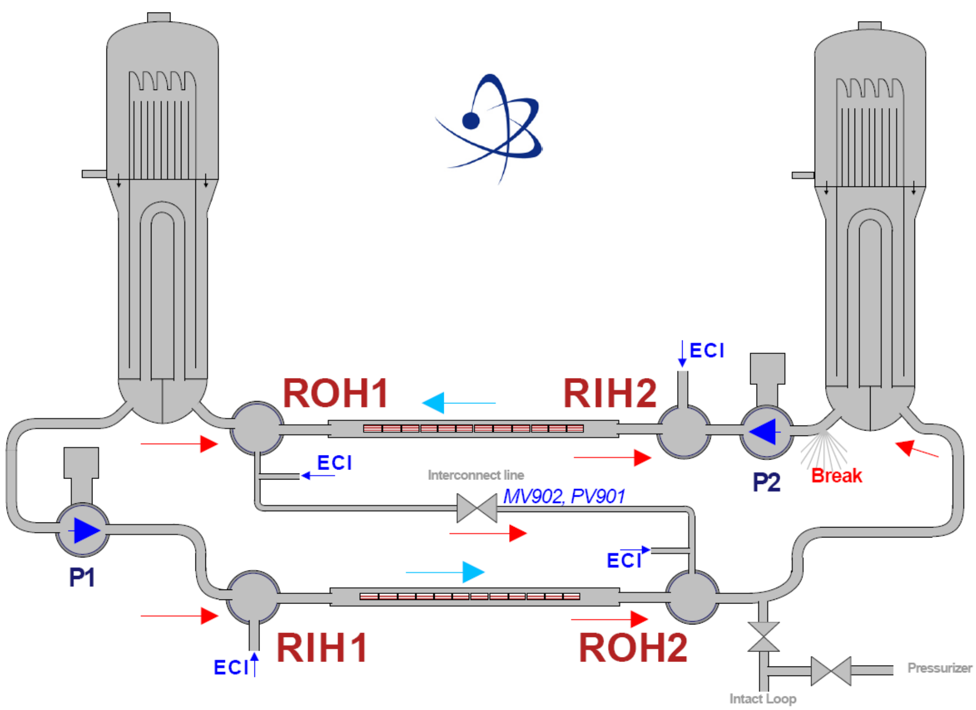

Figure 1 Schematic of broken loop PHTS and main flow directions for long term cooling 50%PSB

3. Case selection

Sensitivity analysis of Large LOCA indicated the break sizes in the range 40-50% produce the longest flow stagnation in downstream core pass; hence highest fuel temperatures occur for this break range. AECL previously performed sensitivity studies for large LOCA and in GSCR reported three limiting cases identified: 22%RIH which produces the maximum pressure tube strain, 100% ROH which produces the highest fuel sheath temperature and 45% PS pipe break which produces the highest pressure tube temperature and fuel temperature. The parameters considered for the CNSC sensitivity assessment and their rationale are presented in Table 1. Please, note that the number of sensitivity cases was commensurate with the CNSC available resources:

Table 1 Parameters selected for the sensitivity analysis and their rationale

Parameter	Base case value	Modified values	Rationale
Stopped pump resistance (K)	22.05	20 (-10%) 18 (-20%) 26 (+20%)	Due to the potential of diverting the LTC ECC flow through the break, this parameter was investigated first. It has been judged that a decreased value would allow more ECC flow to be diverted through the break, thus leaving less cooling water to go through critical pass
Change of condensation index	Q=2 (conduction turbulent heat transfer)	Q=1 (conduction)	Heat transfer inside steam generator tubes during large LOCA, when they became vapour filled, is governed by condensation. It is believed that change in condensation index alters the heat transfer to reflect the possible flow regime of laminar versus turbulent condensate flow, hence the behavior of SG during the accident progression. The default option in CATHENA is Q=2 MAX - conduction, turbulent heat transfer. The sensitivity considered Q=1 - conduction through the liquid film.
Change of vapour convective heat transfer index	R=3	R=1 McAdams	The rationale for choosing this parameter is that for an important part of the transient the steam generators are vapour locked and the heat transfer between vapors and secondary side of boilers is influenced by the correlation(s) that govern the convective heat transfer from vapour to tubes. The default option CATHENA uses for this parameter is R=3. The option explored was R=1 (McAdams)
Decreased heat transfer (primary-secondary) coefficient (aging, fouling)	The correction factors are applied on the inside of the boiler tubes	-10% -20%	This sensitivity case is intended to simulate thedegradation of heat transfer processes that occur in time, such as aging and fouling. It is expected that the accident progression and ECC effectiveness are not significantly changed by decrease of heat transfer coefficient by 10 or 20%. A lower heat transfer coefficient for the inside surface of boiler tubes implies lower convective heat transfer from liquid and vapour phases and, consequently smaller oscillations but also higher temperatures of the coolant in the primary system.
Gap conductance	20,000 W/m^2K	18,000 W/m^2K	In the base case CATHENA model, the gap conductance is assumed constant: 20,000 W/m^2K

Parameter	Base case value	Modified values	Rationale
	20,000 W/m^{2}K	(-10%)	It is expected that in real accident scenarios the gap conductance changes mainly due to variations in primary system pressure, fuel pellet temperature, and gap gaseous composition
		16,000 W/m^{2}K (-20%)	
		22,000 W/m^{2}K (+10%)	
Emergency feedwater (EFW) temperature	50 °C	40 °C	Emergency feedwater (EFW) injects in the secondary side of steam generators when main feedwater system is unavailable due to loss of class IV power and provides residual heat removal. CNSC staff considered that the temperature of EFW may influence the condensation process in boilers, thus it has the potential to change the progression of the accident. In the reference case is assumed that EFW temperature is constant, 50°C. However, in realty the temperature of EFW may vary, hence two sensitivity cases were conducted: EFW temperature 40°C and EFW temperature 60 °C.
		60 °C	
Feedwater(FW) temperature	155 °C	100°C	Feedwater temperature has been included in the analysis because of the influence on the steam generators. Class IV is lost at about 16 s and normal feedwater is supplied to the boilers up to that time in the transient. The base case assumed a constant reservoir boundary condition for SG feedwater at 0.54E6 Pa at saturation, which corresponds to a constant temperature of 154.75 °C. The sensitivity case assumed a constant temperature of feedwater of 100 °C. A lower temperature was considered because in reality the temperature of feedwater is expected to decrease after initiation of the break.
Heat loss from feeders	Adiabatic	Assumed constant and uniform heat transfer coefficient (HTC) 12 W/m^{2}K and 100°C air temperature	The claim that insulated feeders model appears conservative is somehow straightforward since a source a heat sink is neglected. However, on the other hand condensation induced in feeders may have the potential to create water plugs, especially in small diameter feeders, that may disturb the flow rates calculated in the adiabatic feeders case
Headers flow stratification	FIX-MIXED	FIX-SEPARATED	This option was investigated to determine the sensitivity of accident progression to a different option in header mixing flow regime. Reference [3] recommends FIX-MIXED numeric option rather than FIX-SEPARATED, which is susceptible to numerical instabilities. However, the complexity of the flow in headers and low flow conditions can justify the flow separated option.

Parameter	Base case value	Modified values	Rationale
Interconnect flow	Motorized Valve (MV) 902 open Pressure Valve (PV) 901 open	MV902 open PV901 closed	In the accident progression, two large interconnect valves are assumed to open after receiving the ECI signal at 37.5s. For the reference case it was assumed that both valves open normally, as per design expectations. However, for the sensitivity case it was assumed that pressure valve PV 901 is failed close.
Time step size	Min=1ms Max=250 ms	Min=0.5 ms Max=100ms	In order to assess the temporal convergence of numerical solutions provided by CATHENA, a sensitivity case with half minimum and maximum time steps (relative to the base case) has been performed.

4. Results and discussion

4.1 Base case

The base case (or the reference case) was performed with the unaltered input files provided by AECL. It has been noted that AECL used UNIX operating system, while the CNSC simulations were performed on WindowsXP workstations. A comparison for several parameters of interest (such as break discharge rate, reactor power, channel flows, fuel temperatures, coolant void) revealed some minor differences between AECL and CNSC simulations results; however, an overall a satisfactory match was noticed. It was concluded that the simulations results have a relatively weak sensitivity to the operating system, thus giving confidence that the sensitivity cases to be analyzed are meaningful.

Canadian Algorithm for Thermalhydraulic Network Analysis (CATHENA) MOD-3.5d rev 2 and input file provided by AECL have been used for simulating the ACR -1000 plant response to 50% PSB initiating event. For the base case, initial and boundary conditions as presented in the GSCR were applied. They are summarized as follows:

- ACR-1000 Preliminary Safety Analysis Report (PSAR) model
- Initiator: 50%PSB with consequential loss of class IV power
- 16 (7+7+2) channel groups
- Reactor power: 102% Full Power (FP)
- Flux tilt: +/- 4% side-to-side tilt (high power on broken loop) for coupled Reactor Fuelling Simulation Program (RFSP)-CATHENA analysis and for CATHENA ECC effectiveness
- Bleed&Feed not credited, spray is blocked
- Pressure tube diametrical creep: end-of-life with 4% allowance plus 5% uncertainty analysis
- Decay power ANS 5.1 plus 20%

Conservative assumptions in the initial and boundary conditions, trip setpoints and some of the CATHENA models were applied.

The break occurred at t=0; break opening time was 10ms. Initial break discharge rate is 15,200 kg/s. The primary system depressurizes rapidly and headers and channels voiding occurs almost immediately following the break opening. Pressurizer and loop isolation valves start to close at 7.5s and ROH large interconnection valves start to open after 38s from the beginning of the initiating event. The highest and longest void fraction was observed in the RIH2, the closest to the break. High pressure ECC starts injection as early as 10s (ROH2 and ROH3) and starts collapsing the void in the headers and reactor channels; however, significant void persists in all headers up to 120s. After 150s all headers are full of water. The void fractions in broken loop core passes are presented in Figure 3 and Figure 4. In the core pass far from the break (CP1) the flow increases for a short time, during loop depressurization. Development and spreading of vapor phase in the primary system degrades the pump head rapidly and after loss of class IV power (~16s) the primary circulation pumps trip, hence the flow rate decreases continuously and shows relatively large amplitude oscillations. However, a positive flow rate is maintained for the whole duration of simulation (5000s) and no subsequent voiding developed after header and channels refilling (150s). A different behavior is observed for the core pass closest to the break (critical pass), CP2. Immediately after break opening, the flow in the critical pass is reversed, due to extremely high negative pressure gradient induced by the break. The flow reversal takes place rapidly, in less than 0.5s, therefore the fuel cooling is not disrupted. Very high flow rate in reverse direction is maintained for about five seconds. Reactor trips at 0.43s, on low core differential pressure. After the pressure in the primary system became low enough that the channel pressure gradient is balanced by the pumps head and momentum of the rest of the fluid in the system the flow in the critical pass almost stagnates between 5 and 30s. As mentioned, high pressure ECC injection starts after 10s and refilling of headers creates large flow oscillations in the headers that end the stagnation period, at 30s after break initiation

Both steam generators in the broken loop (SG1 and SG2) are voided, therefore they became vapour locked and remain in this state up to 1100s from the break initiation. The ECI injection in the interconnection line tends to create and maintain a negative flow rate (opposite to normal flow direction) through the critical core pass (CP2). Reversed flow direction through CP2 is maintained even after the depletion of accumulators (~220-250s). An important aspect is that the long term cooling start to inject light water into the broken loop before the accumulators inventory is depleted, ensuring the continuity of fuel cooling and smooth transition to long term cooling; furthermore, at the end of high pressure phase and beginning of low pressure injection phase, the flow direction in the broken loop core passes are the same.

Long term cooling system injects only in inlet headers RIH1 and RIH2. Injection to RIH 2 goes through P2 in reverse direction and is lost to the break. The only useful injection came from RIH1, cools CP1 in positive direction, goes to interconnect line, via ROH2, cools CP2 in negative direction, enters RIH2 and exits through the break via P2. Unlike CP1, where subsequent voiding after refilling was not predicted (see Figure 2), in the critical pass void starts to develop between 550-700s in the upper core channel group. As shown by Figure 3, no void is developed in the upper channel groups after 1000s. A different situation is observed in the lower channel groups, particularly group 6 and 7. The periodical behavior of void fraction in these channel groups suggest that due to almost stagnant flow rates, a growing vapor bubble is formed and as soon as it reaches the vertical portion of feeders, vents through them to the reactor headers and the void is replaced by coolant in the channels.

As vapor condenses in SG2 tubes, the pressure in ROH2 decreases further and becomes lower than pressure in RIH1, therefore after 1000s, the flow starts to flow in positive direction in CP2, (see Figure 4). Flow reversal in CP2, induced by condensation in tubes of SG2, is a relatively slow process and creates very low flows and relatively poor cooling in CP2.The maximum sheath temperature is attained during first stagnation period, after the first flow reversal, for channel group 3, outer fuel ring. This behavior is anticipated because at that moment the residual heat is relatively high. At the second flow flip (~1000s) another fuel sheath overheat is observed. However, due to lower residual heat, the amplitude of temperature excursion is much lower, around 200 °C.

Early loop isolation and inventory from the pressurizer and core make-up tanks helped in maintenance of coolant inventory in the intact loop, hence no concerns about fuel and fuel channel overheating exists for this loop.

4.2 Sensitivity cases

Simulations results have shown that some sensitivity cases bounded in consequences the base case (lower EFW temperature, different mixing header option or smaller time step size) or were bounded or comparable with the base case. Typical examples of simulations results and comparison with the reference case results are shown in Figures 5 and 6. Figure 5 plots the integrated void fraction for critical core pass for the sensitivity case heat transfer from feeders. Comparison with Figure 4 (base case) indicates more significant voiding for the base case, therefore the consequences more severe than the sensitivity case. Figure 6 shows the comparison of flow rates in the interconnect line valves, between the reference case (both valves operates as designed) and the sensitivity case – one valve failed close. The simulation indicated that the operational interconnect valve allowed a flow rate comparable to the reference case.

It was concluded that for all cases analyzed, ECC succeeded to ensure sufficient fuel cooling, thus no fuel or fuel channel overheat occurred. The fuel pellets and fuel sheath temperatures were limited, hence it appears that the robustness and effectiveness of ECC design was maintained for all cases analyzed for these parameters, albeit with less margin for some cases. The maximum fuel sheath temperatures for the upper (low EFW temperature) and the lower (high FW temperature) bounds, as well as the comparison with the reference case are plotted in Figure 7.
If ECC effectiveness is judged by the amount of void that is formed in the channels and reactor headers, the base case and sensitivity cases show channel voiding after high pressure injection ends, caused by flow stagnation when flow reverses in critical pass. Flow complexity and oscillatory behavior of void fraction makes it difficult to objectively quantify and compare cases. It appears that void fraction and void distribution varies between different cases analyzed, thus selected parameters may be of interest for validation.

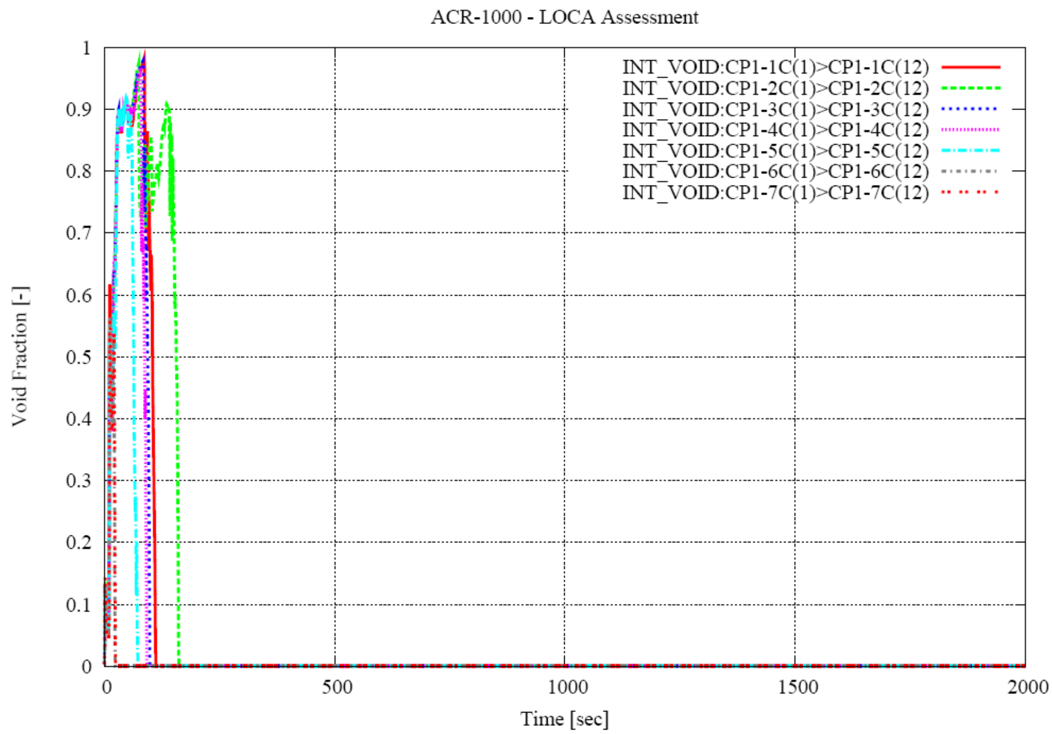

Figure 2 Integrated void fraction channel groups CP1, base case 50% PSB

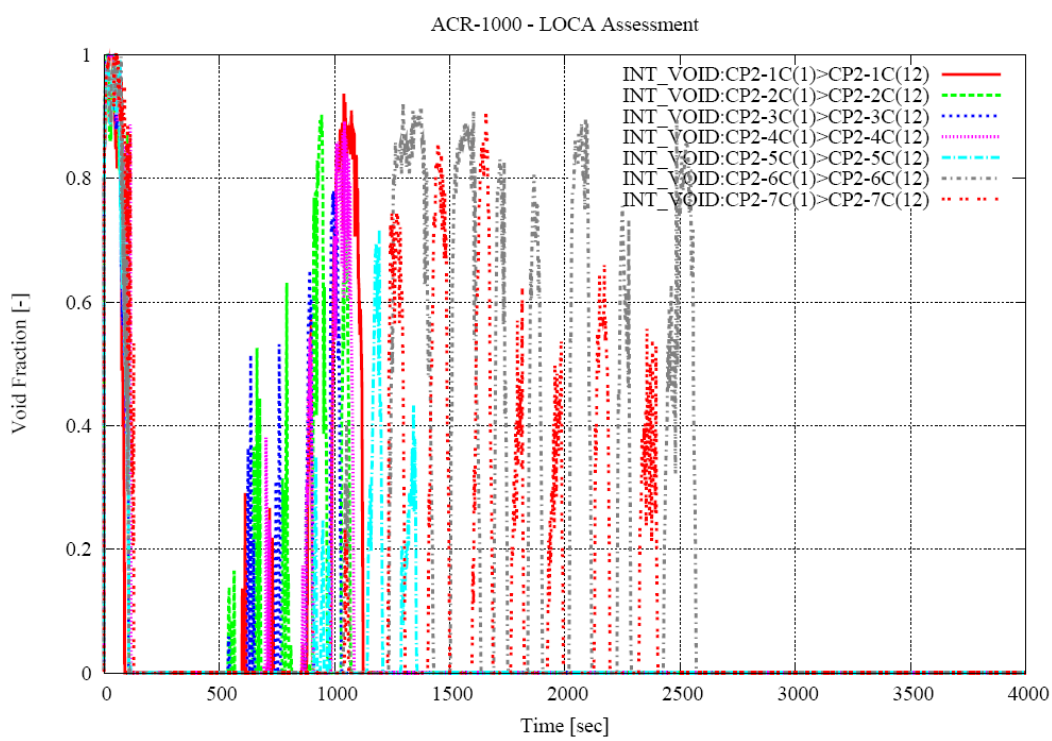

Figure 3 Integrated void fraction channel groups critical pass (CP2), base case 50% PSB

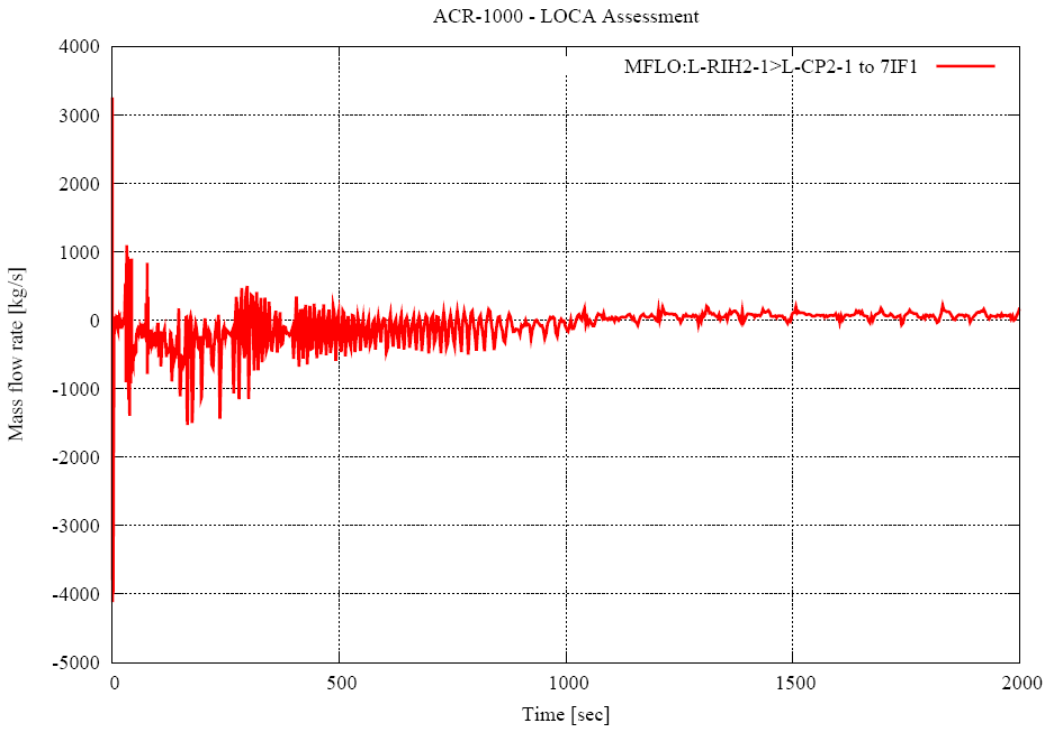

Figure 4 Mass flow rate for critical pass (CP2)

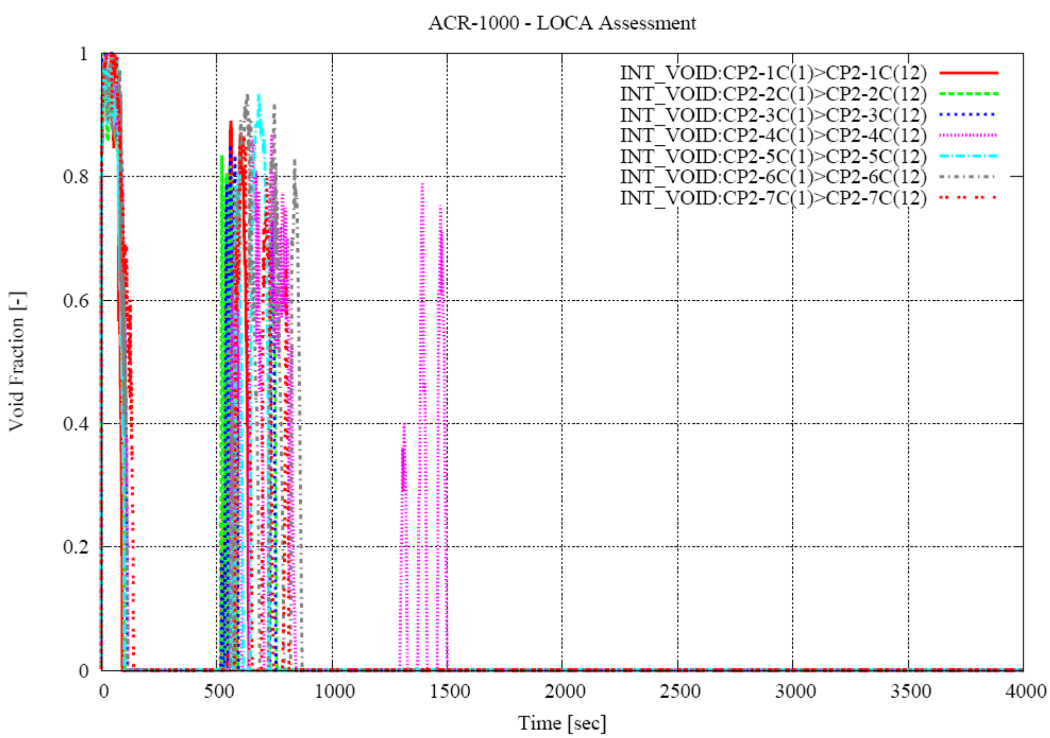

Figure 5 Integrated void fraction for sensitivity case, critical pass (CP2) 50% PSB sensitivity case heat transfer from feeders

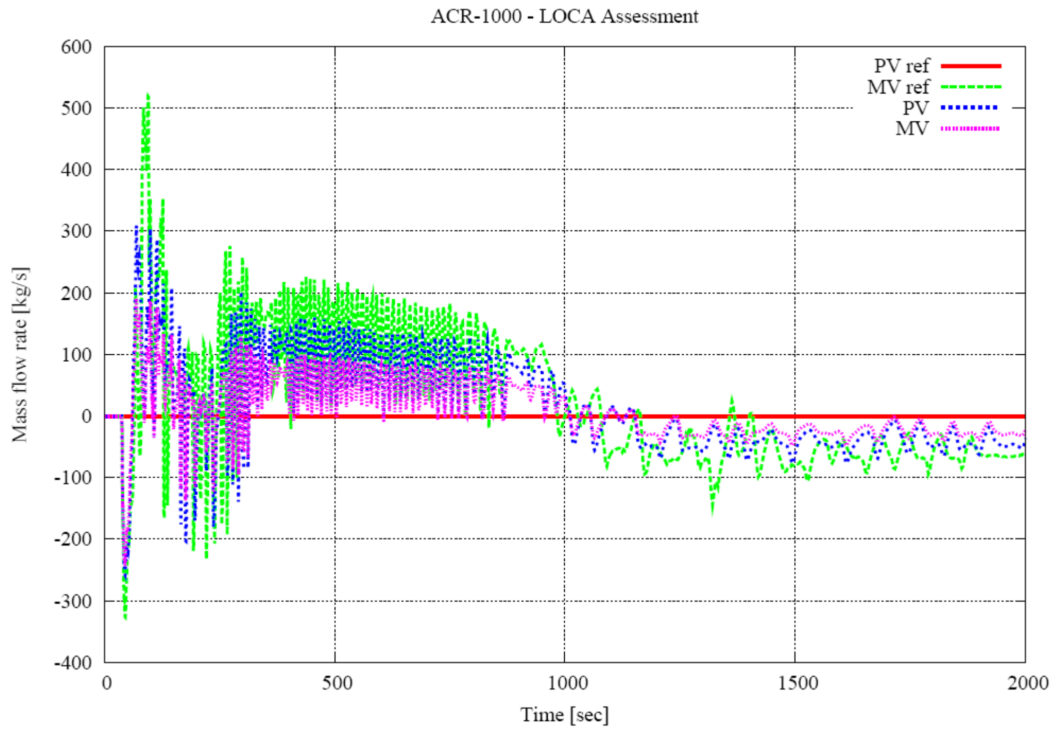

Figure 6 Comparison of flow rates in large ROH1-ROH2 interconnect valves, 50% PSB, base case vs sensitivity case one interconnect valve failed close

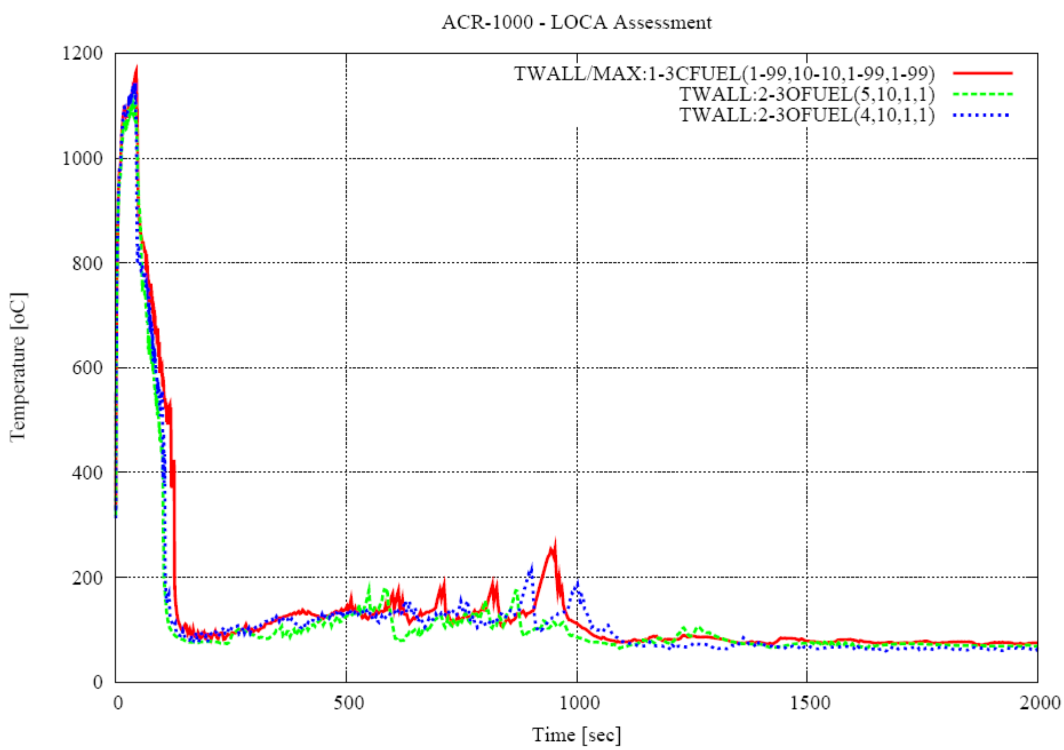

Figure 7 Maximum fuel sheath temperatures for the upper (red) and lower (green) bounds and the base case (blue)

5. Conclusion

In-house CATHENA simulations performed by the CNSC staff contributed to the achievement of the Phase 3 objectives, as follows:

- the separate simulations strengthened the CNSC staff knowledge about the ACR-1000 design, the thermalhydraulics simulation codes (CATHENA and Element Loss of Coolant Accident (ELOCA)) and modelling methodology.
- the review of representation of plant systems and plant behavior indicated no major issues. However, CNSC staff advised on a partial sensitivity study of boiler tube nodalization and time step convergence for 50% PSB.
- the selected accident scenarios and the limited scope sensitivity cases conducted by CNSC staff, indicated that, overall, the ECC performance showed small sensitivity to the parameters and assumptions considered for investigation. It should be noted that the above conclusions are valid only for the cases and assumptions investigated and cannot be extrapolated outside of this range.

6. References

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