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LOSS OF COOLANT ACCIDENT SIMULATION WITH RELAP5/SCDAPSIM/MOD4 FOR THE CANADIAN SUPERCRITICAL WATER-COOLED REACTOR

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Abstract

RELAP5/SCDAPSIM/MOD4 has been improved to accommodate supercritical water properties and transcritical fluid transitions and is used for simulation of the Canadian Super Critical Water Reactor (SCWR). The system response to several postulated Loss of Coolant Accident (LOCA) transients are studied. For a 100% (single-ended) cold-leg break located in between the feedwater pump and the inlet plenum, the system pressure immediately drops followed closely by a flow reversal with a rapid break discharge. A brief power pulse (178%FP) is observed under this non-equilibrium depressurization scenario. The transient simulations show the potential for two sheath temperature maxima, one early in the transient as a result of the power pulse and the subsequent flow-power mismatch, and the other later peak resulting from the fuel heat-up under near stagnant channel flow conditions as the heat transfer regime changes to radiation dominated. The Automatic Depressurization System (ADS) mitigates the later fuel heat-up by introducing forward channel flows. This effect is enhanced by additional coolant supplied from Low Pressure Coolant Injection (part of the Emergency Core Cooling system). Under the 100% break LOCA with coincidental failure of the Emergency Core Cooling System (ECC) transient, the core inventory is depleted even faster after the break. The highest maximum cladding surface temperature of 1331 K is achieved approximately at 141s and meets the safety criterion (1533 K). Beyond this time the sheath temperatures gradually decrease by heat transport to the Passive Moderator Cooling System (PMCS).

LOCAs initiated by a range of 5% to 100% cold-leg break sizes are simulated with coincidental loss of ECCS. In this specific design, break sizes less than 15% are defined as SBLOCAs and show an early pressurization up to the Safety Relief Valve (SRV) setpoint, while breaks larger than this threshold do not undergo a pressurization phase. . During SBLOCAs, the first MCST peak is more limiting than the LBLOCA cases because of insufficient fuel cooling caused by lower reverse flows. However, these lower discharge flows prolong the period of blowdown cooling and hence mitigate the secondary MCST peak. The worst LOCA occurs in the 15% break case with a highest MCST of 1450 K. The results showed the most sensitive parameters are delays associated with SDS action, emissivity and ADS actuation parameters.

1. INTRODUCTION

The supercritical water-cooled reactor (SCWR) is one of the six advanced reactor concepts being investigated within the Generation-IV International Forum (GIF) for deployment in the next several decades. The SCWR concept possesses several distinguished advantages [1]. Firstly, it is the only Generation IV design using water as coolant and its thermal efficiency (45%-50%) is remarkably increased compared to that of the current reactors (33%-35%). Secondly, the primary heat transport system is simplified utilizing a once-through cycle whereby the steam generator can be removed.

Thirdly, the heat capacity of the supercritical coolant has a large peak in the vicinity of pseudo-critical points thus the primary coolant pump power can be reduced. Furthermore, fluid in the supercritical region does not undergo boiling crisis and hence concerns related to CHF and PDO are reduced.

The GIF goals include a) improved economics, b) improved safety and reliability, c) improved sustainability and d) improved proliferation resistance and security. Kito et al. developed the transient safety criteria for SCWR with consideration of the unique characteristics of single phase supercritical water cooling [2]. The maximum cladding surface temperature (MCST) was regarded as the most important criterion rather than the minimum deterioration heat flux ratio (MDHFR) in LWRs.

As a design basic accident (DBA), the loss of coolant accident (LOCA) is of particular importance as historically it has been one of the most limiting accidents. There have been a large number of investigations on the response of SCWR designs to LOCA. The US-SCWR was modeled by RELAP5-3D for double-ended break LOCAs located at cold-leg, hot-leg and steam line [3]. The cold-leg break case resulted in most severe consequences yet still allowed enough time for the emergency core coolant system or ADS to come into effect. Ishiwatari performed hot- and cold-leg LOCAs with various break sizes (1-100%) [4]. They concluded that the ADS plays a significant role in mitigating fuel cladding temperature for cold-leg break large LOCA and that small break gives even higher MCST values. Shanghai Jiao Tong University (SJTU) has proposed a mixed spectrum supercritical water reactor (SCWR-M) with its safety systems derived from the passive design of AP1000, including two isolated cooling systems (ICS), two accumulator (ACC), two gravity driven cooling systems (GDCS) and four stage automatic depressurization systems (ADS) [5]. The ATHLET-SC simulation of three different hot/cold leg break LOCAs of SCWR-M proved the safety systems capability of ensuring the fuel cladding temperature below the thermal design criterion.

Some safety analyses have been performed for the Canadian SCWR. A 100% cold-leg break LOCA with loss of emergency core cooling systems of the Canadian SCWR was modeled by Wu using SCTRAN [6] and the “No-core-melt” criterion under the LOCA/LOECC events was demonstrated. Hummel performed a 3D neutronic/thermalhydraulic coupling simulation to investigate the neutronic feedbacks of the High Efficiency Re-entrant fuel Channel (HERC) [7]. It was concluded that the feedback effects from multiple coolant regions of HERC are different and a temporary positive reactivity may be induced during a non-equilibrium density decreasing transient.

The Canadian SCWR adopts the once-through steam cycle similar to the Japanese Super LWR concept. Therefore this study focuses on the same break size range (5% to 100% break). Hot-leg break initiated LOCAs are proven to be less severe by various literature and thus are not discussed in this paper. A neutron point kinetics has been created for this work in order to realistically predict the reactivity feedbacks during the transient, which was neglected in reference [6]. The acceptance criteria for fuel sheath integrity is taken to be ~1530 K, consistent with that adopted in the Japanese SCWR design as well as the acceptance criteria for early PWRs that utilized SS cladding. Such criteria are meant to ensure fast oxidization and significant exothermic oxidization heat loads do not exacerbate accident consequences [8].

2. RELAP5/SCDAPSIM/MOD4

RELAP5 is a one-dimensional thermal-hydraulic code designed to predict the reactor coolant systematic responses under normal and postulated accident conditions. It is characterized by a two-fluid, non-equilibrium, non-homogenous hydraulic model for two phase flow. Its modelling capability for thermalhydraulic response, control system behavior, reactor kinetics and various special reactor system components has been confirmed by extensive experimental data for Light Water Reactors (LWRs) conditions and accidents. In order to extend RELAP5's application in advanced reactors and to improve its simulation performance, Innovative System Software (ISS) has participated in the international SCDAP Development & Training Program (SDTP) [9]. On the basis of publicly accessible RELAP5/MOD3.3 models developed by the US Nuclear Regulatory Commission (NRC), ISS released RELAP5/ MOD4 as the first version of RELAP5 which is completely rewritten in FORTRAN 90/95 [10], resulting in increased flexibility in adding alternative working fluids and correlations.

RELAP5/SCDAPSIM/MOD4 has incorporated the National Institute of Standards and Technology (NIST) database Version 10 based on the International Association for the Properties of Water and Steam (IAPWS) 1995 formulation for water thermodynamic properties [11]. Compared to the original database, NIST 10 provides improved accuracy of water properties in the supercritical region, especially in vicinity of the pseudo-critical line. Two-dimensional property look-up tables at discrete pressure and internal energy entries are created for property interpolation at given state points. Above the critical pressure, the fluid properties change continuously without the presence of phase change phenomenon. However, at a constant pressure and over a very narrow range of temperatures near the pseudo-critical temperature threshold, fluid properties vary significantly. Below this threshold the fluid property magnitudes and sensitivities closely resemble those of liquid water at high pressure. Above the threshold, the magnitudes and sensitivities resemble those of gaseous water at high pressure. In RELAP5/SCDAPSIM/MOD4 the fluid properties in the supercritical region are determined from the NIST database and the vapor void fraction is set to zero for the purposes of computation of the two-fluid equations (i.e. the equations are solved as if there were only a liquid within a volume, and using the fluid properties of the supercritical fluid). The heat transfer and frictional models above the supercritical pressure are determined using the standard single-phase correlations in RELAP, and while these correlations have some inaccuracy near the pseudo-critical point, overall very few nodes pass through these conditions in a given time step, and the point of MCST exclusively occurs well away from the pseudo-critical transition.

Special treatments have been implemented in MOD4 to cope with the water property transitions through the threshold temperature range as well as when pressure transitions through the critical pressure in order to prevent numerical errors [12]. Firstly, the fluid property tables are made denser near the critical point such that width of the interpolation region is smaller in regions where fluid properties change the most. In addition the following codes change are implemented:

1. three possible state change paths from supercritical to subcritical pressure (to subcooled liquid, to superheat steam and to a two phase mixture) as illustrated in Figure 1 are considered;
2. simple linear interpolation is used instead of the cubic interpolation when the critical point or its adjacent points in the fluid property table are used. Given the highly non-linear relationship of fluid property relations in this region, using smaller intervals with linear interpolation provides improved stability;
3. the junction donor in between a supercritical and a subcritical volume inherits the density and internal energy from the upstream supercritical volume and the void fraction from the downstream subcritical volume, rather than inheriting them from the upstream volume at normal conditions;
4. the latest thermal conductivity is incorporated in the source code to provide more accurate conductivity predictions.

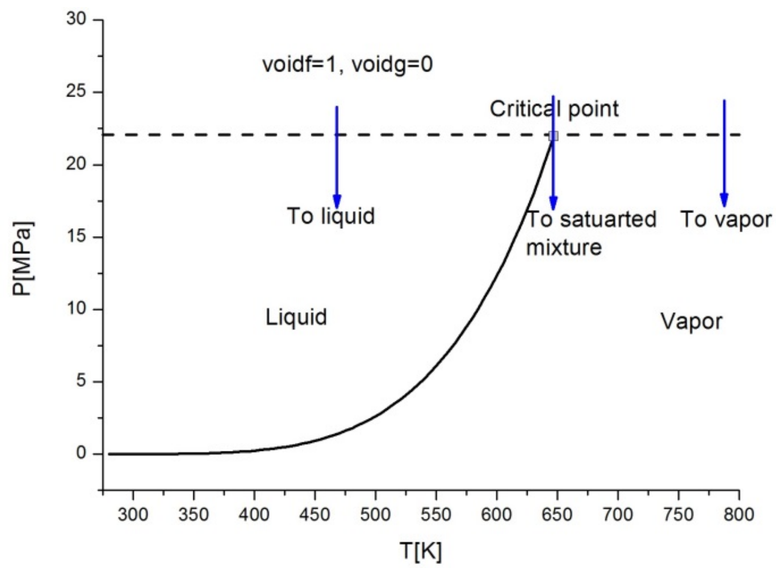

Figure 1 P-T diagram for H_2O

3. SCWR DESIGN AND MODELLING

3.1 Canadian SCWR systematic design

The conceptual design of Canadian SCWR proposed by Canadian National Laboratory (CNL) consists of 336 vertical fuel channels using Th-Pu mixture fuel, cooled with supercritical light water and moderated with a separate low pressure heavy water system. This concept uses a High Efficient Re-entrant Channel (HERC), consisting of a central downward coolant flow path, and an outer upward flow path through the fuel region. Surrounding the outer flow path is an insulator with inner and outer liners which serve to provide thermal isolation between the pressure tube and the high temperature coolant. The pressure tube is in direct contact with the low pressure and low temperature heavy water moderator. Figure 2 shows the schematic of the reactor core. Cold coolant from four identical feedwater pumps enters the inlet plenum via the cold-leg pipes at 25.8MPa and 350°C. Flow passes from the inlet plenum to each individual fuel channel and flows downward through the central flow tube. Once the coolant reaches the bottom of each channel, it is redirected upwards in the outer portion of the channel and through the fuel containing region taking away the thermal energy. The hot coolant from the fuel channels gets uniformly mixed in the outlet plenum with a temperature of 625℃ at 25.0MPa and subsequently flows into the steam turbine through the main steamlines. The mass flow rate in each channel is matched to the reference channel power via an inlet orifice similar to that in a Boiling Water Reactor (BWR).

The fuel assemblies are surrounded by the low-pressure, low-temperature moderator system which is driven by both active (i.e. pumped) and passive (i.e. natural circulation) mechanisms during normal operation. In case of loss of class IV power, while the active components of moderator cooling are lost, the passive component continues operation and is capable of removing the decay heat [1]. The passive moderator heat removal system is designed to remove approximately 3.6 % of the full power (FP) under normal operating conditions, which guarantees that the decay heat can be removed effectively in the event of a coincidental loss of Class IV and III power and Emergency Core Cooling systems.

The active shutdown systems for the design have not yet been finalized, however consistent with previous CANDU designs, it is assumed the SDS1 (rod based) and SDS2 (poison based) systems will be retained. For this analysis only the less effective SDS1 (in terms of depth and speed) is credited, while the single failure criterion is applied to SDS2. However, given that the analysis below demonstrates small LOCAs to be limiting, additional failures of the ECC are also analyzed.

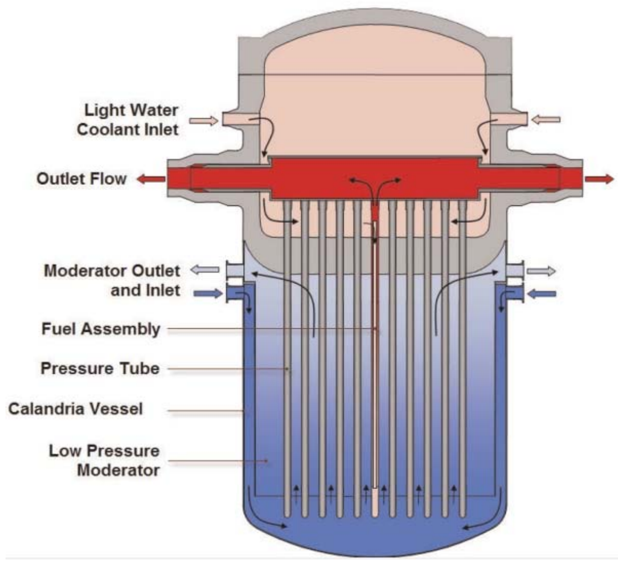

Figure 2 Schematic of the Canadian SCWR reactor core

3.2 RELAP5 modelling

A RELAP5 idealization of the Canadian SCWR primary heat transport system model has been created for this study, including a series of hydraulic volumes, heat structures, control variables, trips, and a reactor point kinetics model. While validation of the tools is limited by sparsity of experimental data, code to code comparisons to CATHENA which uses an entirely different set of numerical approaches and empiricism was performed by Lou [12] and showed excellent agreement during the blowdown phases of a LOCA. The coolant flow path was modeled volume by volume, starting from the feedwater storage tank, the main feedwater pumps through the reactor core to the main streamline pipe. There are four identical cold-leg pipes linking the feedwater tank and the inlet plenum, only one of which is assumed to break. Thus the break-path in Figure 3 was to model the broken path and the intact pipes were lumped as an averaged flow path in RELAP5. In the absence of a detailed pump specification for this design, the RELAP5 default Westinghouse pump model was used to simulate the feedwater pumps for each loop. The 336 fuel channels were divided into two representative groups, the high-power (HP) group with the highest channel power and the average-power (AP) group with the remaining 335 channels, for simplification. After the coolant from the broken and intact flow paths (volume 102 and 202) merges in the inlet plenum (volume 105), it flows through the 5-metre long center flow tube inside the fuel channel (volume 111 for the AP group and 211 for the HP group). The 180-degree bend at the channel bottom was modeled by two identical volumes (volume 113&115 and 213&215) with opposite directions, connecting the center flow tube and the outer fuel channel flow region (volume 117 and 217). The orifice sizes have been optimized by Hummel and Novog [13] to accommodate the channel powers variation as a function of burnup along the reactor life cycle. In this paper the channel power distribution in middle of cycle (MOC) from Hummel's quarter-core neutronic calculation has been used as the axial power profile [14].

The 64 fuel elements in each channel are configured in two concentric rings equidistant between the outer liner/insulator and the central flow tube. They were modeled as two vertical heat structures and are coupled to the reactor point kinetics model. The pressure tube wall with an insulation layer and two liner layers, as well as the center tube wall were modeled as heat structures with no heat sources and radiation heat transfer two and from all heat structures is considered (e.g., inner ring to outer ring, inner ring to central flow tube, etc.).

The separated coolant flow pathways are significantly different from the unidirectional flow configuration in a CANDU fuel channel. While the reactivity induced from a decrease in coolant density is negative under equilibrium conditions (i.e., when both the coolant in the center flow tube and outer fuel channel tube void uniformly at the same rate), there may be transients wherein there may be a significant, albeit short duration, difference of void between these regions. This differential voiding effect was examined extensively by Hummel and Novog [7] and showed that the reactivity feedback effects of the inner and outer region coolant densities need to be considered separately. This is in contrast to the work of Shen [15] which applied a uniform reactivity feedback assuming a constant coolant density across the assembly.

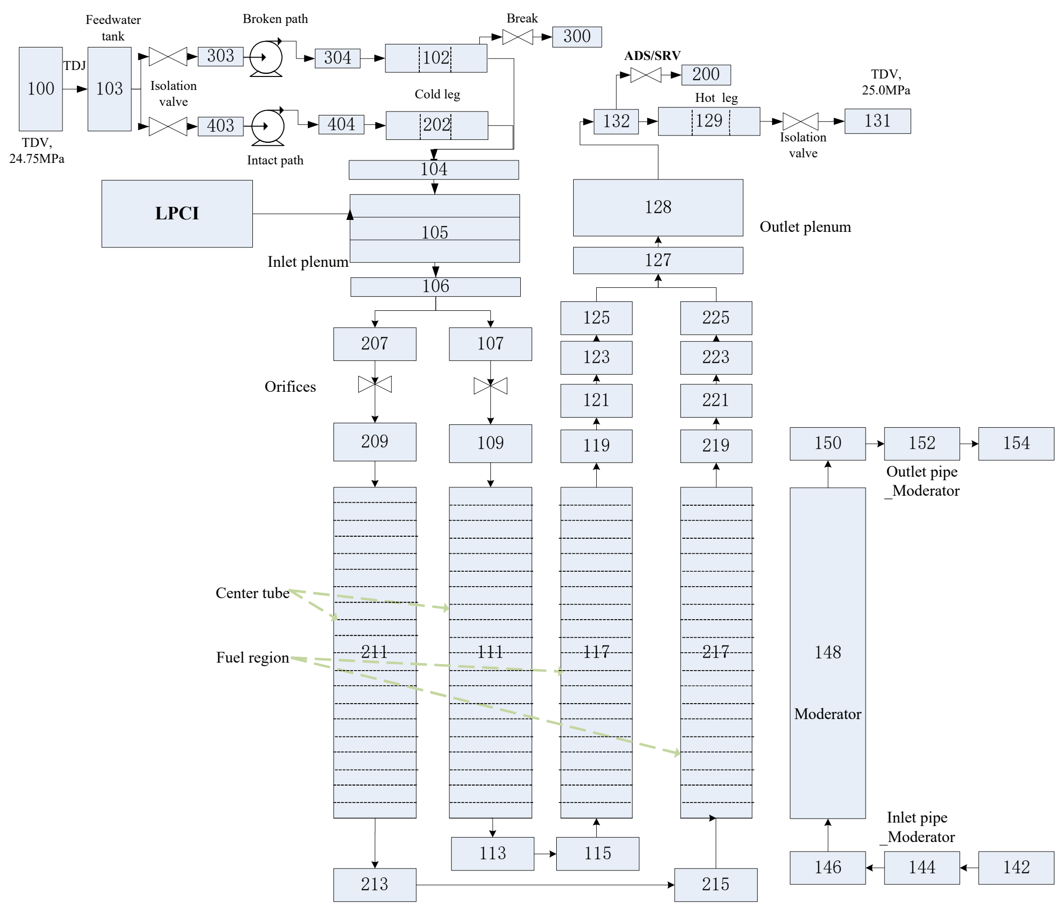

Figure 3 The RELAP5/SCDAPSIM/MOD4 model for Canadian SCWR primary heat transfer system

A neutron point kinetics model with separated reactivity feedbacks was adopted here to update the thermal power as a function of various feedback terms during the transient. Figure 4 shows the reactivity feedback as change of coolant density in the two regions and fuel temperature. The overall sensitivity to a coolant decrease in the central flow tube is negative whereas for the outer flow tube is positive. Under uniform voiding conditions the negative feedback dominates and a net negative reactivity would be introduced. During a cold-leg break induced LOCA transient, the coolant along the channel will not void uniformly, and in particular the early flow reversal will move low density fluid at the outlet towards the bottom of the channel, thereby decreasing the density in the outer flow tube while the inner flow tube may be less affected. The net result is a small increase in reactivity until such time as the low density fluid travels into the central flow tube. While the inner and outer coolant reactivities are assumed independent in this work (i.e., with no covariance), analysis by Hummel [14] has shown that the net error in this assumption is small. Another negative reactivity feedback during this transient is from the fuel temperature as the blue line shown in Figure 4. Previous LOCA studies [6] of the Canadian SCWR did not consider this non-equilibrium voiding effect. The RELAP5 point kinetics model updates the thermal power in each time step as a result of the combined effects from the inner and outer coolant densities as well as the change in fuel temperature. A decay heat curve provided by CNL is incorporated in this model for power prediction after the reactor shutdown [16]. The validity of the point-kinetics approximation for similar depressurization transients was tested against a full 3-dimensional kinetics model with the agreement in power being better than 2% [12].

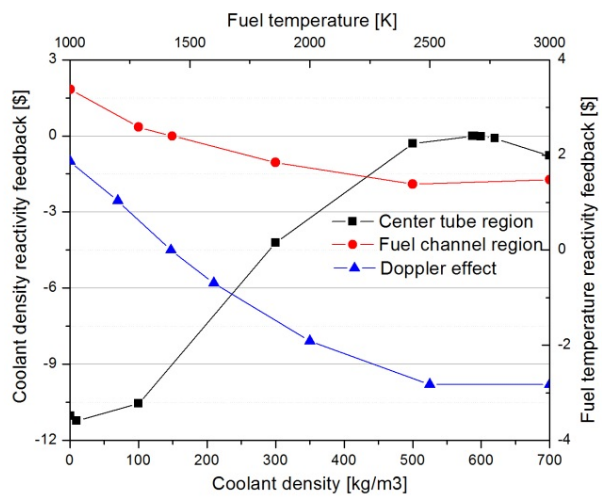

Figure 4 Reactivity feedback data, generated by SCALE at MOC with burnup of 25 MWd/kg U [17]

The Main Steam Isolation Valves (MSIVs) will close on the outlet side of the core as a consequence of LOCA, thereby reducing the magnitude of the reverse flow and eventually approaching near stagnation conditions. As a result of the reduction in cooling, the fuel will heat up and radiate the decay heat energy to the liner/insulator/pressure tube components. The radiation heat transfer enclosure consists of the center tube wall outside surface, two fuel elements ring outside surfaces and the pressure tube inner liner surface. The view factors were calculated by an external code GEOFAC [18] and the emissivity was assumed to be 0.8. The radiation heat transfer model plays an essential role in cooling the fuel elements as the coolant is completely lost and the cladding temperature gets rather high.

The cold-leg break was modeled by a valve component connecting the affected cold-leg pipe and a standard atmospheric control volume. The reactor system isolation valves were modeled as motor valves and are assumed to close at certain rate by the trip signal, thereby isolating the reactor core from the feedwater tank and the turbine. The coolant flow from the feedwater tank was controlled by a Time Dependent Junction (TDJ) while the inlet and outlet boundary conditions were modeled by two Time Dependent Volumes (TDVs).

The moderator system is simply modeled with an upward flow path including an inlet pipe, a main moderator tank and an outlet pipe. Two TDVs were used to model the inlet and outlet boundary conditions in the moderator system. After the loss of Class IV power, a TDJ was utilized to maintain a constant mass flow rate so that 5% FP can be taken by convection from the pressure tube wall to the passive components of the moderator system.

4. LOCA TRANSIENT RESULTS

After the break occurrence the main heat transport system would be depressurized with most of the coolant discharged rapidly from the break, resulting in reverse flow in the fuel channels. A low steam flow (90% of nominal value) signal and/or a low pressure (24.0MPa) signal were detected to shut down the reactor by quickly inserting a worth of 11 dollars negative reactivity with a instrumental delay of 0.55 s [4]. Meanwhile, the feedwater pumps were tripped and the feedwater tank was assumed to linearly drain out within 6s after the trip. With another delay of 0.05s, the MSIV was closed. The Automatic Depressurization System (ADS) were actuated at the low pressure signal and the Low Pressure Coolant Injection (LPCI) were actuated as the pressure drops to 3MPa.

4.1 100% cold-leg break LOCA with ECCS

A 100% cold-leg break (i.e., the cross-sectional area of a main feedwater pipe, $0.07548m^2$) LOCA transient simulation is discussed below. As shown in Table 1, the break occurred at 5.0s and the low coolant flow signal was registered 0.09s later, initiating the reactor trip and the feedwater pump trip at 5.645s (with a 0.55s delay). The low pressure signal occurred slightly later at 6.45s and initiated the ADS with a 2s delay. After trip initiation, negative reactivity is added to the core at rate identical to the typical CANDU SDS1 insertion capability. Then the steam isolation valve closed at 5.695s while the coolant flow from the feedwater storage tank decreased linearly and flow stopped at 11.645s. The LPCI actuated once the system pressure drops below 3MPa and continues until a total amount of 10,000 kg inventory is ejected. The simulation continued for 500s after the break occurrence.

Table 1 Transient sequences for the 100% break LOCA cases

Event sequence	Case 1	Case 2	Case 3
Break occurrence	5.00	5.00	5.00
Reactor trip signal/low flow	5.09	5.09	5.09
Reactor trip	5.645	5.645	5.645
Main coolant pump trip	5.645	5.645	5.645
Main steam isolation	5.695	5.695	5.695
ADS activation	8.45	8.45	-
LPCI initiation	18.53	-	-
LPCI drain out	106.94	-	-
Feedwater tank drain out	11.645	11.645	11.645

During the LOCA transient, the mass flow rates in the fuel containing region of the fuel channels are important as they provide convective cooling of the fuel especially in the early phases when the power levels are high. Figure 5 indicates that the combined effects of ADS and LPCI keep the channel flow lasting for approximately 120s before the core coolant is depleted. The transient can be divided into the following phases based on the important phenomena and MCST behaviors: 1) initial MCST excursion phase; 2) blowdown cooling phase; 3) interim heat-up phase; 4) sustained cooling phase (Figure 6).

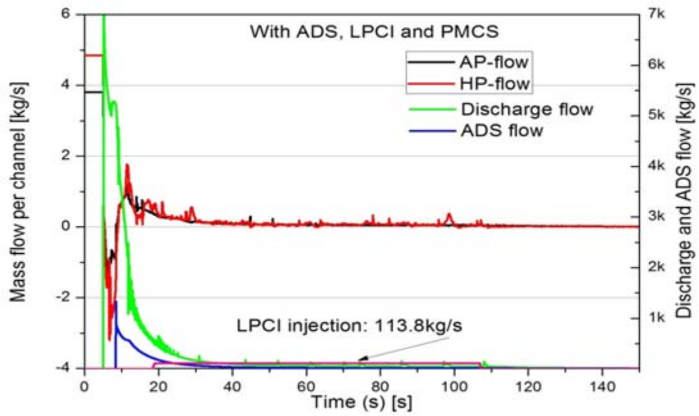

Figure 5 Channel flow and discharge flow in early phases of the 100% LOCA transient (reference case)

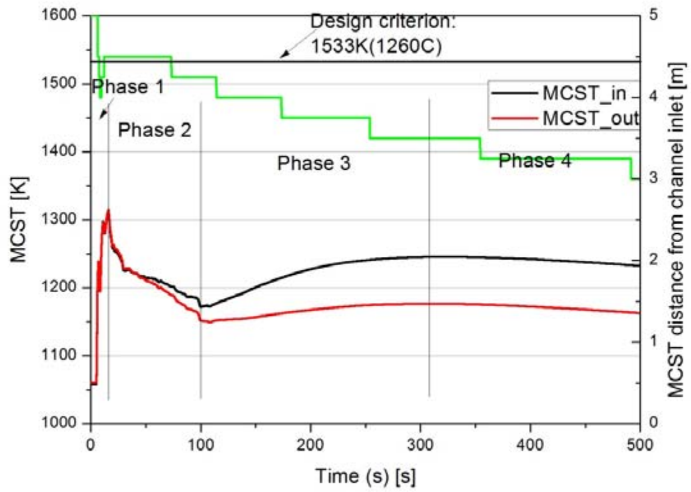

Figure 6 MCST values and positions variation during the 100% LOCA transient (reference case)

Figure 7 and Figure 8 show the system behaviors in Phase 1 and Phase 2. The cold-leg break caused a sudden pressure drop in the inlet plenum, which spread to the outlet plenum with a small time delay. The large outflow from the break quickly initiated reverse channel flows, resulting in hot coolant from the outlet plenum to the fuel region. The ADS acted to accelerate the depressurization and re-establish forward channel flows. The ADS action while eventually favorable, temporarily impedes heat transfer as the flow undergoes another change in direction from reverse to forward flow. In addition, a temporary positive reactivity feedback was generated due to the differential voiding effect in Phase 1, resulting in a brief power excursion (178%FP) as shown in Figure 8. The power pulse is terminated by the shutdown system action and negative reactivity feedback as the density further decreases. The decay power combined with the energy deposition during the power pulse exceeds the heat removal capability of the channel flows and degraded fuel cooling causes the temperature to increase in Phase 1. The MCST approached the first peak value of 1315 K at 16.0s.

Phase 2 is defined as the interval of time with decreasing temperatures after the first peak (Figure 8). The cooling provided by small channel flows sustained by the ADS and LPCI actuation is sufficient to cool the fuel at decreased power level. The convection in conjunction with radiation heat transfer exceeds the decay power generation in the fuel and the temperatures begin to decrease. These combined effects cause the MCST to approach a local minimum of 1172 K at the end of Phase 2 (100s).

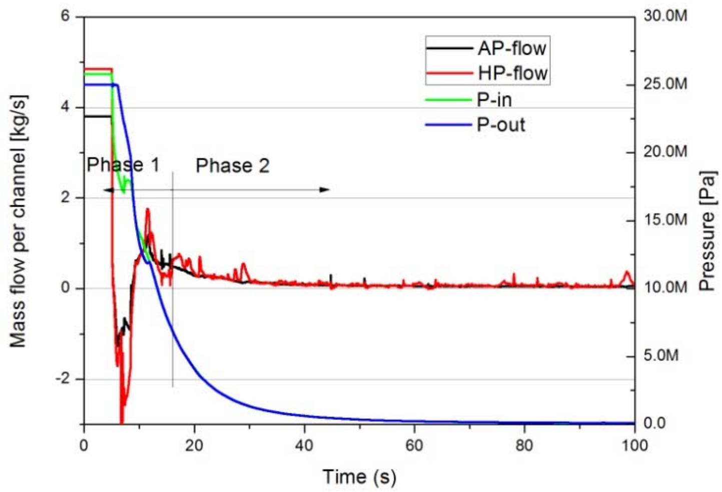

Figure 7 Pressure and channel flow in Phase 1 and Phase 2 of the 100% break LOCA(reference case)

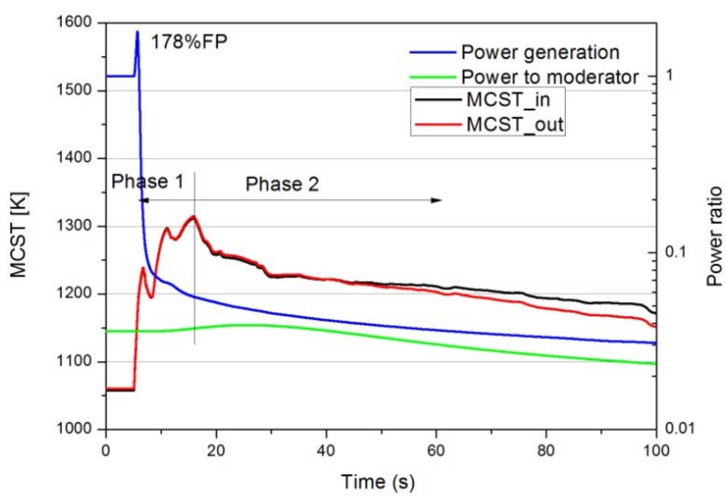

Figure 8 MCST, power generation and removal in Phase 1 and Phase 2 of the 100% break LOCA(reference case)

Figure 9 presents the fuel cladding temperature variation as the transient further developed. During Phase 3 the contribution of convection drops to insignificant values due to near stagnant channel flows (Figure 5) after core coolant depletion. Heat rejection during this phase is primarily via radiation heat transfer from the fuel to the PT assembly and ultimately to the PMCS. The decay power generation in the fuel exceeds the available radiative cooling and therefore the fuel temperature begins to increase. The maximal value of MCST, i.e. 1245.7 K, occurred at the end of Phase 3. This peak MCST is mitigated by

ADS and LPCI actuation and is less severe compared to the first peak. Figure 10 shows the fraction of heat removed by radiation increased to approximately 80% in this phase and remained at this level for the remainder of the simulation time.

At the beginning of Phase 4, the cladding surface temperature has risen to such extent that the heat radiated from the fuel cladding surfaces balances the further reduced decay heat generation, resulting in gradual decrease in the fuel temperatures. Figure 9 shows that the heat removal remains higher than the decay heat generation for the rest of the simulation, indicating effective long term cooling by combined ECC flow and PMCS.

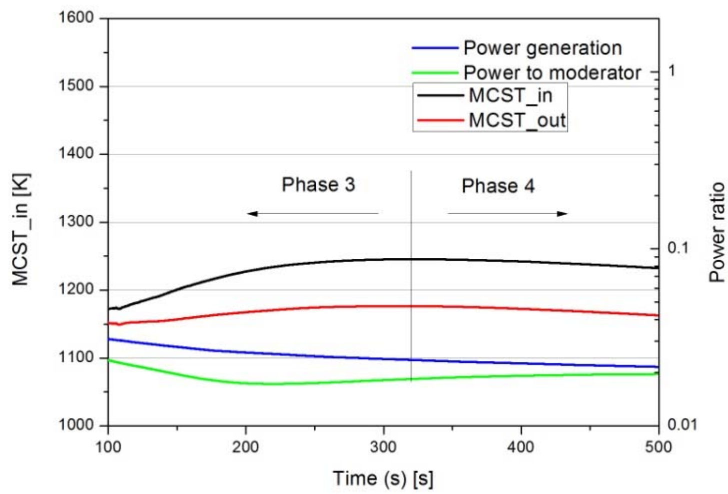

Figure 9 MCST, power generation and removal in Phase 3 and Phase 4 of the 100% break LOCA(reference case)

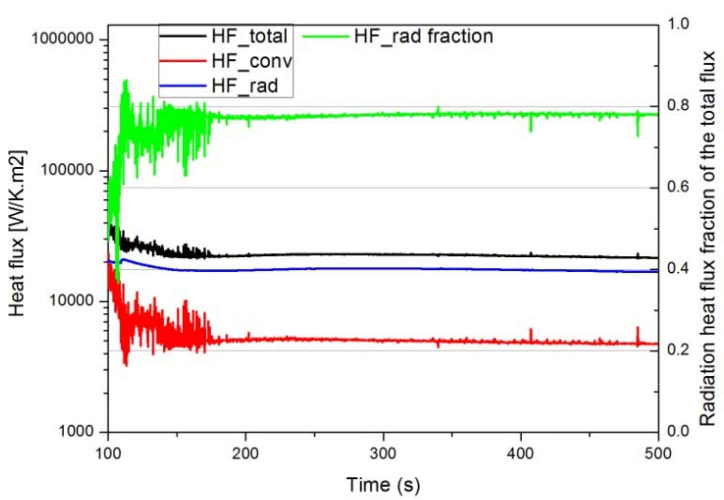

Figure 10 Radiation and convection heat flux on the fuel cladding surface of the 100% break LOCA(reference case)

4.2 100% cold-leg break LOCA with partial or total failure of ECCS

In contrast with the above discussed 100% LOCA with actuation of ADS and LPCI (reference case, i.e. Case 1), a single failure of LPCI for the 100% break (Case 2) and a dual failure of ADS and LPCI for the 100% break (Case 3) were also studied. The transient sequences are listed in Table 1.

As shown in Figure 11, the channel flows in Case 2 started deviating from the reference case beyond 18.53s (coolant injection initiation time in Table 1). In absence of coolant injection the channel flow stagnation occurred earlier. The discharge flow from the break decreased slightly compared to the reference case which implies that part of the injected coolant in the reference case left the reactor core via the break.

For Case 3 the reverse channel flows initiated by the cold-leg break remained reversed for the duration of the transient until the coolant was depleted at around 50s, after which the convection heat transfer became minor and radiation heat exchange dominated the decay heat removal. In absence of the ADS, the system was depressurized at a slightly lower rate and the discharge flow rate was greater in the early phase (due to a higher system pressure).

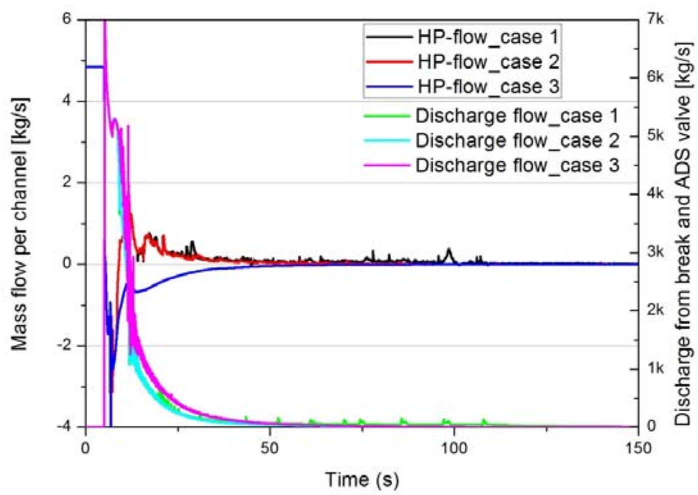

Figure 11 Channel mass flow and discharge flow in Case 2 and Case 3 of 100% break LOCA compared to the reference case

Figure 12 shows the MCST variations in Case 2 and Case 3 compared to the reference case. The faster coolant depletion in Case 2 shortened the effective blowdown cooling period and the MCST increase in Phase 3 occurred earlier compared to that of Case 1. Thus the subsequent heat-up in Phase 3 appeared at higher decay heat levels which leads to higher secondary MCST peak.

The higher magnitude reverse channel flows in early phase of Case 3 lead to a reduction in power pulse as compared to Case 1 and Case 2. Furthermore, since the flow remains reversed, the temporary heat transfer degradation caused by the channel flow direction transition in Case 1 are avoided. Therefore the first peak MCST in Case 3 is less severe than the other cases. However, since LPCI is unavailable in Case 3, the subsequent phases take place earlier at higher decay heat levels. As such the limiting MCST occurred at the secondary peak. The highest MCST (1331.5 K) among three cases was observed as the secondary peak, though it is well below the safety criterion for the stainless steel cladding.

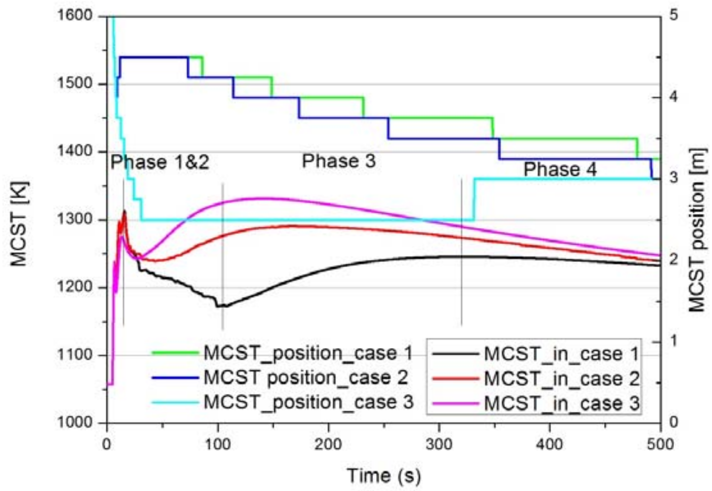

Figure 12 MCST in the Case 2 and Case 3 of 100% break LOCA compared to the reference case

4.3 5% cold-leg break LOCA with loss of ECCS

A 5% cold-leg break was studied here as a representative of small break LOCAs (SBLOCAs). The LOCA with loss of ECC was demonstrated to be more severe for the 100% break case and therefore ADS and LPCI are not credited in the SBLOCA as well. The transient sequences were listed in Table 2, which are in the same order yet with a time delay for each event occurrence than the 100% break LOCA (Case 3) due to a smaller break size.

Table 2 Transient sequences for the 5% break LOCA

Event sequence	Time[s]
Break occur	5.00
Reactor trip signal/low flow	5.41
Reactor trip	5.97
Main coolant pump trip	5.97
Main steam isolation	6.02
Feedwater tank drain out	11.97

With the small break size, the coolant discharge rate is much smaller compared to the 100% case. Therefore smaller reverse channel flows are expected to last longer for the transient. The combination of the smaller break size, reduced coolant flow, delayed coolant pump trip and closure of isolation valves leads to a temporary pressurization of the heat transport system up to the Safety Relief Valve (SRV) setpoint. This low-flow and high pressure scenario is similar in nature to the initial stages of a loss of forced flow accident [19]. The frequent SRV actions cause a periodic pressure oscillation, resulting in a fluctuating channel flows in early stage of the transient. Similar to the 100% break case, the transient is split into several phases depending on the MCST behavior (Figure 13), with a typical early pressurization phase where flow oscillations driven by SRV action dominate.

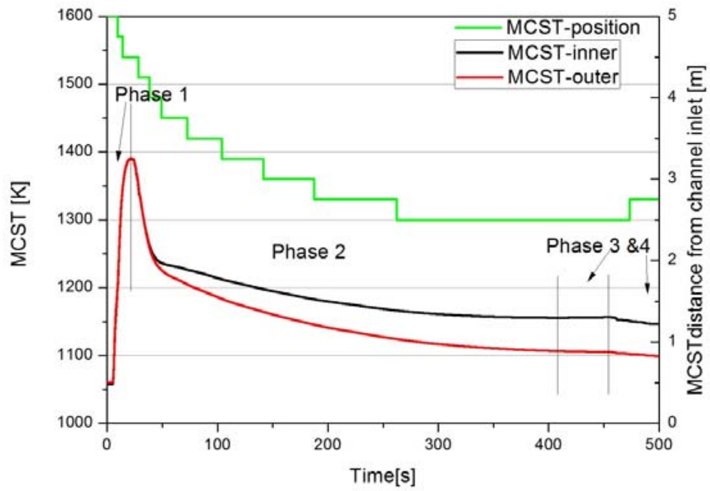

Figure 13 MCST value and position variation during the 5%break LOCA

In Phase 1 the combination of loop isolation, coolant pump trip and small break outflow causes early flow stagnation in the core and the imbalance between heat generation and removal, resulting in the system pressurization. A smaller power pulse (108.5% FP) was observed at 6.09s since the difference in inner flow tube and outer flow tube coolant density was smaller than the large LOCA case. The SRV was activated when the system pressure increased to its opening setpoint and was closed as the pressure decreased to its closing point. Subsequent repressurization occurred and the SRV opened again to relieve pressure. During this initial oscillatory phase SRV actions oppose the reverse flow while some improved cooling is observed (Figure 14(a)). At about 11s the inventory is depleted to the extent that the bulk pressure drops and the SRV action is terminated. Beyond this time continuous small reverse flow occurs in all channels.

Figure 14(b) shows that the first peak MCST is 1390 K at the end of Phase 1, which is higher than that of the 100% break case. Even though the net energy deposition (NED) due to power excursion is much larger in the 100% break case, the subsequent large magnitude reverse flow limit the initial temperature excursion. In the SBLOCA case, the much smaller reverse flows at this early period is insufficient to remove the NED and thus the first peak MCST is more limiting than in large LOCAs. Beyond this time the long-lasting small reverse flow during the blowdown cooling phase provided sufficient convection heat transfer to remove the decay heat generation.

A series of unexpected channel flow oscillations are observed in Figure 14(a) at about 28s as the system undergoes transition from supercritical to subcritical pressure. This is likely caused by the code limitation in dealing with the dramatic coolant property variations near the critical point. Such variation occurs rapidly in the 100% break LOCA and thus are less noticeable. However, the duration of time in the trans-critical region is longer for smaller break LOCAs, and therefore additional code issue are observed. Further improvement of water property interpolation scheme near the critical point is necessary in future work.

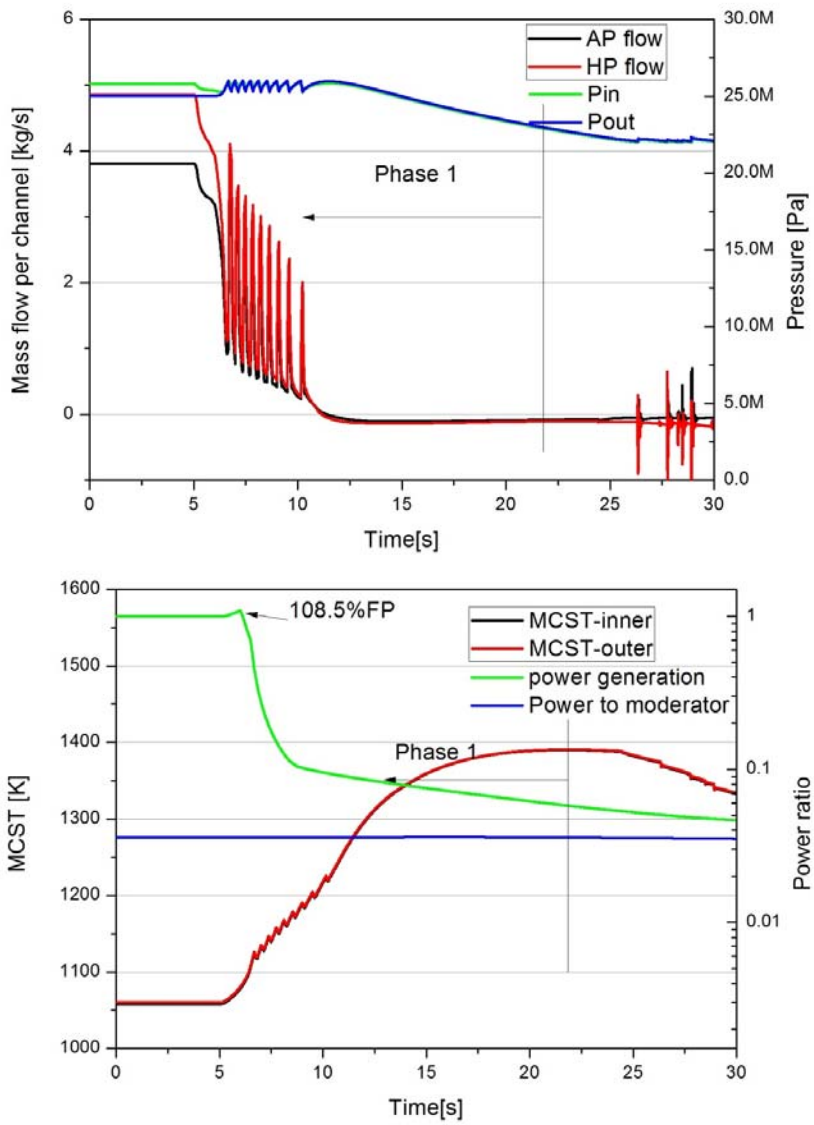

Figure 14(a) Pressure and channel flow behavior in Phase 1&2 for 5% break LOCA ;(b) Reactor power and MCST variation in phase 1&2 for 5% break LOCA

In Phase 2 the radiation heat transfer became increasingly important in removing decay heat deposition apart from convection. The dark green line in Figure 15 represents the radiation heat flux as a fraction of the total heat flux. It increases gradually from 30% to 60% at around 500s, indicating the radiation heat transfer became dominant in the latter phases. The green line and blue line in Figure 16 show the competition between decay heat generation and heat removal by radiation which is eventually deposited in the moderator system.

The 5% break LOCA has a slower depressurization rate with a smaller discharge flow rate. The distinctive initial pressure fluctuation phase caused by frequent SRV actions induces some forward flow pulses. The early peak temperature dominates since the reverse flows in this phase are much lower than that in the 100% break case. The secondary MCST peak in SBLOCA is much less significant than the LBLOCA case since the lower discharge rate and subsequent longer inventory depletion process ensures that effective convective fuel cooling lasts for a longer period of time, whereas in the LBLOCA the mass inventory is rapidly depleted and the convection heat transfer is no longer significant after core depletion.

In this 5% break case, the MCST remains around 1150 K after 500s and the highest value of 1390 K occurred at the end of Phase 1.

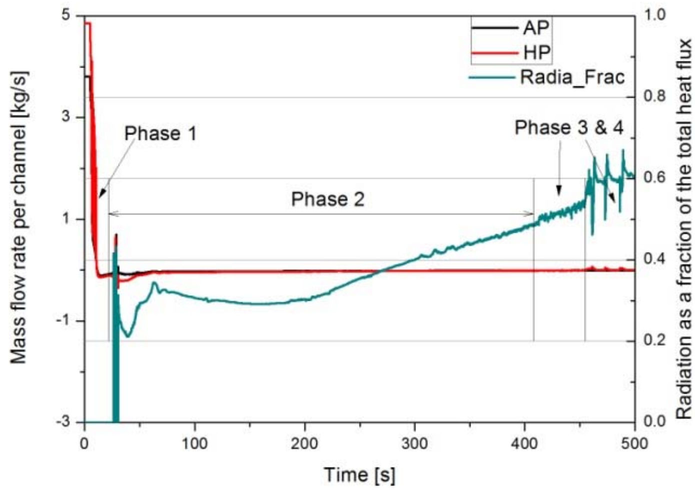

Figure 15 Channel flow rate and radiation heat transfer fraction for 5% break LOCA

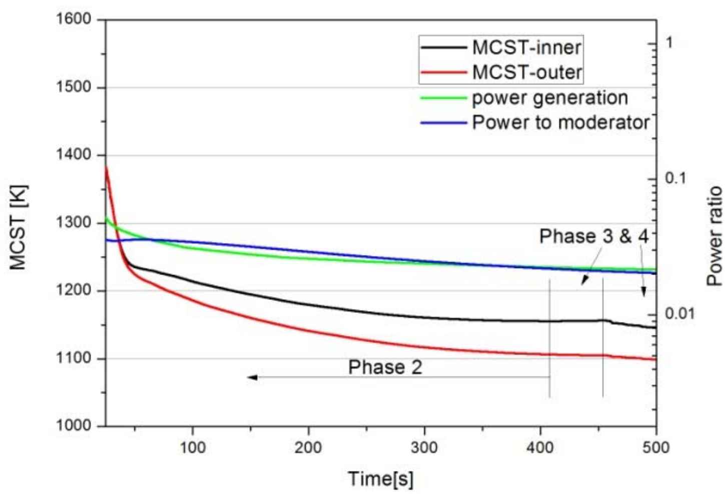

Figure 16 MCST variation, power generation and removal in later phases for 5% break LOCA

4.4 Sensitivity analyses

Various break size LOCAs, from 5% to 100% of the cold-leg cross-section area were simulated to investigate the corresponding behaviors. In this specific design, break sizes less than 15% are defined as SBLOCAs characterized by some early SRV driven flow phenomena while the other break sizes are LBLOCAs. 15% break is defined as the boundary break since it is the largest size that causes early pressurization to SRV setpoint. Figure 17 shows the depressurization process as a function of break size from 100% to 5%. The slower depressurization rate in SBLOCAs provide longer periods of flow reversal (longer flow discharge), and hence the secondary peak MCST is expected to be reduced in magnitude with decreasing of the cold-leg break size.

In terms of the maximum cladding temperature in Figure 18, generally two peak MCSTs appeared in each case. Consistent with the previous discussion, the first peak is related to the energy deposition during the power pulse and reduced cooling, while the secondary peak is related to the transition from convection to radiation dominated heat transfer. For LBLOCA, the secondary peak MCST is more limiting since the first peak is mitigated by the relatively large reverse flow in the early phase. However, the first peak MCST becomes more restrictive for SBLOCAs due to a reduction in the reverse channel flows and SRV action in an opposing direction to the reverse flow. The SBLOCA secondary peak is reduced due to longer sustained reverse channel flows and a much later transition to radiation heat transfer. Compared to the 5% break, the first peak MCST in the 15% break is much higher because of the larger energy deposition caused by a higher power pulse. Table 3 summarizes the power pulses and MCSTs for all the LOCA transient cases conducted in this study. The highest MCST (1450.4 K) occurs at 12.86s in the 15% break LOCA.

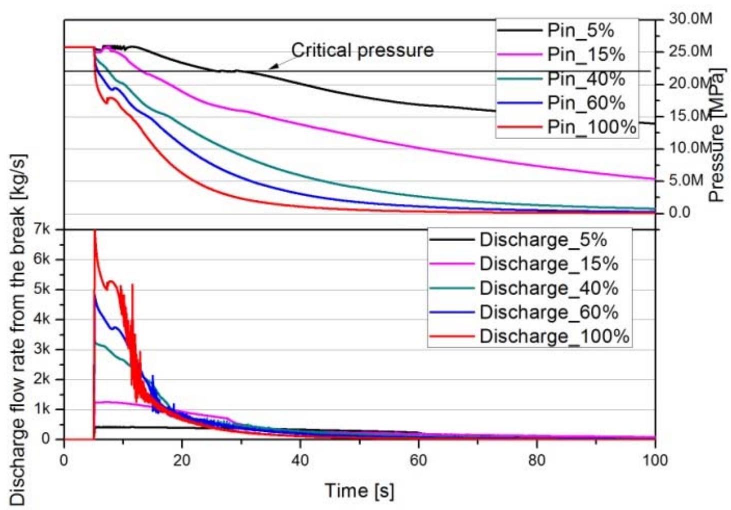

Figure 17 Depressurization and discharge flow for numerous LOCA cases

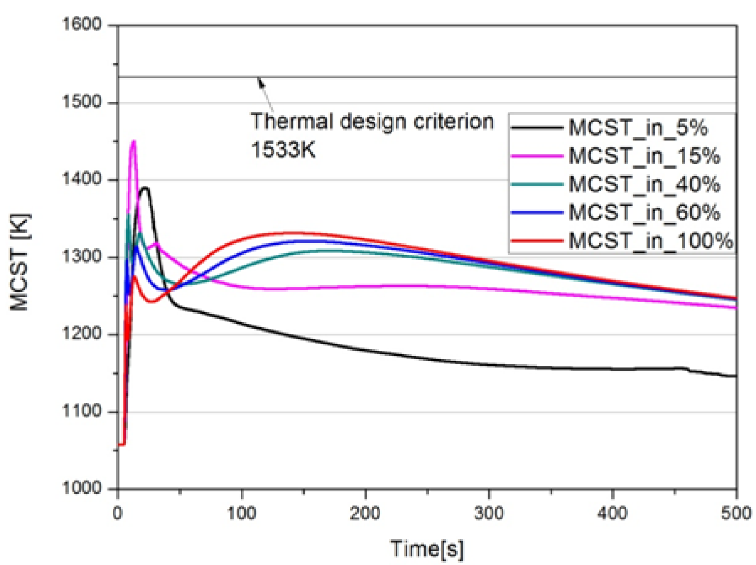

Figure 18 MCST variations for different LOCA cases

Table 3 Peak MCST and occurrence time for different break size LOCAs

Break size	Power pulse [%FP]	First Peak [K]	First peak time [s]	Second peak [K]	Second peak time [s]
5%	108.5	1389.9	21.8	1156.5	454.0
15%	118.6	1450.4	12.86	1263.2	228.0
40%	143.4	1331.7	17.73	1308.6	171.0
60%	156.8	1313.7	14.96	1320.9	155.0
100%	178.3	1275.5	13.1	1331.5	141.0

5. CONCLUDING REMARKS

A RELAP5/SCDAPSIM/MOD4 model has been established for the cold-leg break LOCA analyses for the Canadian SCWR. A two-group scheme was adopted for the LOCA transients and the simulations continued for 500s after the break occurred. The radiation and convection heat transfer effectively removed the decay heat with the moderator system being the ultimate heat sink. The inherent safety feature of the SCWR under LOCA accident with/without emergency core cooling systems has been demonstrated through this study.

- Two MCST peaks were observed for each LOCA case. For LBLOCAs, the secondary peak occurred after core depletion when the heat transfer regime transitions from convection to radiation dominated is limiting. However, for SBLOCAs the first peak MCST tends to be more restrictive due to the insufficient fuel cooling caused by lower reverse flow to remove the energy deposition because of early power pulse.

- The worst LOCA case occurred in the 15% break with the peak MCST being 1450 K. This is lower than the design limitation for stainless steel cladding 1533 K by a 83 K safety margin. Fast acting shutdown system and early ADS intervention were proved to mitigate the peak MCST and provide additional safety margin.

- Further optimization of the reactor shutdown system and the automatic depressurization system is recommended for mitigating the accident consequence especially for SBLOCAs.

- The RELAP5/SCDAPSIM/MOD4 code is recommended to improve the water property interpolation scheme near critical point.

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