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Supplemental Thermal-Hydraulic Transient Analyses of BR2 in Support of Conversion to LEU Fuel

Nuclear Engineering Division

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prepared by

J. Licht and B. Dionne
Nuclear Engineering Division, Argonne National Laboratory, Argonne, IL 60439

and

E. Sikik, G. Van den Branden and E. Koonen
BR2 Reactor Department, SCK•CEN

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1 Introduction

Belgian Reactor 2 (BR2) is a research and test reactor located in Mol, Belgium and is primarily used for radioisotope production and materials testing. The Materials Management and Minimization (M^3) Reactor Conversion Program of the National Nuclear Security Administration (NNSA) is supporting the conversion of the BR2 reactor from Highly Enriched Uranium (HEU) fuel to Low Enriched Uranium (LEU) fuel.

BR2 is a tank-in-pool type reactor cooled by light water and moderated by beryllium and light water (Figure 1). The reactor operates with downward flowing coolant at an inlet temperature of 35°C and an inlet pressure of 1.26 MPa. The reactor core is located inside a pressure vessel that contains 79 channels in a hyperboloid configuration. The core loading configuration is variable as each channel can contain a fuel assembly, a control or regulating rod, an experimental device, or a beryllium or aluminum plug. In case of a pressure loss accident, the primary system inside the pool is automatically opened to the pool by a pool connection valve (ABV4-1308) so that the reactor vessel can be kept full of coolant. Isolation valves on the inlet and outlet piping (ABV4-1304 and ABV4-1305, respectively) located at the pool wall automatically close to stop leakage of the water from the reactor vessel. A bypass valve (ABV4-1301) located between the inlet and outlet piping is automatically opened to complete a natural circulation loop.

The RELAP5/Mod 3.3 code [1] has been used to perform transient thermal-hydraulic safety analyses of the BR2 reactor to support reactor conversion. A RELAP5 model of BR2 has been validated against select transient BR2 reactor experiments performed in 1963 by showing agreement with measured cladding temperatures [2]. Following the validation, the RELAP5 model was then updated to represent the current use of the reactor; taking into account core configuration, neutronic parameters, trip settings, component changes, etc. Simulations of the 1963 experiments were repeated with this updated model to re-evaluate the boiling risks associated with the currently allowed maximum heat flux limit of 470 W/cm^2 and temporary heat flux limit of 600 W/cm^2 [3]. This document is an extension of Ref. [3] and provides analysis of additional transient simulations that are required as part of a modern BR2 safety analysis report (SAR). The additional simulations included in this report are:

Effect of Pool Temperature – Effect of the initial steady-state pool temperature on the Test G transient[1].

Reduced Steady-State Flow Rate – Reduction of the primary coolant flow to a value near the trip setting followed by the Test G accident scenario.

In Pool Loss of Coolant Accidents – Double ended guillotine break of the primary coolant system at various locations within the reactor pool (see Figure 1) with an initial heat flux at the maximum limit of 470 W/cm^2.

[1] The Test G transient is defined as an untimely opening of the pool connection valve (total loss of pressure) without opening of the bypass valve for an initial heat flux at the maximum limit of 470 W/cm^2. It was conservatively assumed that the pressurizer was isolated from the system and that the external cooling (vessel shroud and secondary side of the primary heat exchangers) was terminated by reducing the flow to 0 kg/s within 5 s. The value of 5 s was chosen to avoid calculation issues associated with an instantaneous stoppage of flow.

Loss of External Cooling – Loss of flow in both the reactor vessel cooling shroud and secondary side of the primary heat exchangers at nominal conditions with an initial heat flux of 470 W/cm^2 and 600 W/cm^2.

The simulations described in this document have been performed for both an HEU and LEU fueled core.

2 Simulation Results and Discussion

2.1 Elevated Pool Temperature

The nominal steady-state pool temperature for RELAP5 simulations has been set at 30°C. Additional RELAP5 simulations have been performed to investigate the impact of steady-state pool temperature on the Test G transient, which was the most limiting case simulated in Ref. [3]. Figure 2 shows the maximum clad temperature[2] during the flow reversal for initial steady-state pool temperatures of 30°C (nominal), 40°C, 50°C and 60°C. The pool does not significantly impact the cladding temperature for steady state conditions since any difference in heat loss to the pool from the primary system are compensated for by the primary heat exchangers. The flow reversal process, which occurs between ~15 s to 35 s, is also unaffected by the pool temperature. Following this, the core coolant remains in single phase until ~1350 s, when net void generation begins again. Figure 3 shows that the net void generation continues for the duration of the 24 hour simulation. The peak void is ~17% for a pool at 60°C and ~12% for a pool at 30°C. The peak cladding temperature remains near the saturation temperature while the pool temperature steadily increases. The coolability of the core can be inferred from the continual decrease in channel void. This can also be inferred by the eventual decline of the inlet coolant temperature shown in Figure 4 (represented by the value in volume 38 of the RELAP5 model, which is the coolant volume located just below the highest heat flux fuel element). The peak coolant inlet temperature is ~108°C at 3 hours for a pool temperature of 30°C and increases to ~114°C at 6 hours for a pool temperature of 60°C. The coolant temperature continually decreases following its peak value, indicating that the core is coolable for at least 24 hours without any issues related to peak clad temperature. Results are quite similar for a core fueled with LEU.

2.2 Reduced Steady-State Flow Rate

An additional Test G simulation was performed with a reduction in the initial steady-state flow rate. The reduced flow was achieved by lowering the pump speed to just above the first encountered trip setting, which was found to be the minimum allowable pressure difference across the vessel (trip occurs when DPRCA4-1301 < 1.8kg/cm^2). The nominal and reduced flow rates were 6478 m^3/hr and 4965 m^3/hr, respectively. It should be noted that the trip setting for low flow is 4500 m^3/hr. Figure 5 shows the impact of a reduced flow rate on the maximum clad temperature. For steady state conditions, the reduction in flow to 4965 m^3/hr results in the maximum cladding temperature increasing from 155°C to 180°C. This value is slightly less than observed for a nominal heat flux of 600 W/cm^2 at nominal flow, which results in a maximum cladding temperature of 183°C. The reduction in flow rate does not have a significant impact on the Test G accident scenario. Following the untimely opening of the pool

[2] The maximum clad temperature is taken from the inside face of the 6th plate of the fuel element (axial location of maximum temperature varies with time). Inside face temperature typically differs from the outside face by ~0.2^{o}C.

connection valve (ABV4-1308), there is sufficient flow following the reactor scram to bring the cladding temperature quickly down to the coolant temperature. Since the nominal power (and hence decay heat) is unchanged, the cladding temperature remains similar to the nominal case during the flow reversal process. Although the long term simulations were not performed for this case, they are expected to be nearly identical to the long term result for the nominal case shown in Figure 3 and Figure 4. The HEU and LEU results are nearly identical for this accident scenario.

2.3 In Pool Loss of Coolant Accident

Four break locations within the reactor pool have been selected for in-pool LOCA simulations; one on each of the two inlet and outlet pipes (Figure 1). For simplicity, the labeling nomenclature for the four break locations is LOCA1 through LOCA4. LOCA1+2 is the nomenclature referring to a simultaneous break at LOCA1 and LOCA2. In total, 6 LOCA simulations were performed (LOCA1, LOCA2, LOCA1+2, LOCA3, LOCA4, and LOCA3+4). Each of the LOCA simulations, as requested by BR2[3], has been treated as a double ended guillotine break. A description of the break model is given in Appendix A. For these simulations, the initial conditions are based on the maximum nominal heat flux of 470 W/cm^2 (see Ref. [3] for steady state results). As in previous work, the pressurizer and anti-syphon were isolated from the system during the transient since neither contributes significantly to the simulation. It was conservatively assumed that the external cooling in the vessel shroud and secondary side of the primary heat exchanger were turned off by reducing the flow to zero in 5 s.

Table 1 shows the event times for all of the LOCA simulations with and HEU core. In all cases, the LOCA is initiated at 0 s and the reactor scrams due to a loss in pressure (P4-1302 < 9kg/cm^2). Breaks that occur on the inlet leg within the pool cause the inlet pressure to decrease below the trip setting as fast as 3-6 ms (with reactor scram occurring following a 25 ms delay). This same trip setting is reached at 32 ms for LOCA4. The order in which the ABV4 valves open/close varies depending on break location but the timing for all ranges between 1.645 s and 4.247 s. These valves require 15 s to fully close/open. The pump trip occurs at about 3.3 s for all break locations. The average time for the pump check valves to close is 16.5 s. Table 2 contains the same information for an LEU fueled core and the results show that the event timings are quite similar to an HEU core.

Figure 6 and Figure 7 show the maximum clad temperature for each of the in-pool breaks in comparison to the results of Test F[4] from Ref [3]. Test F is included here for comparison since the transient behavior is quite similar to the LOCA scenario (e.g. maximum cladding temperature). The in-pool LOCAs on the outlet piping produce very similar maximum cladding temperatures relative to Test F (Figure 7), both in the short (<90 s) and long term (<24 hr). Breaks within the pool and on the inlet side of the reactor vessel (Figure 6) are also similar in the short term with the exception of a ~10°C spike in maximum clad temperature within the first 0.1 s of the transient. The rise in clad temperature is due to the rapid reduction in mass flow rate within the core, as shown in Figure 8. This is shortly followed by nucleate boiling, which increases the heat transfer and lowers the clad temperature. A sensitivity test was

[3] Sikik, E., Private communication, February 12th, 2016

[4] The Test F transient is defined as an untimely opening of the pool connection valve (total loss of pressure) followed by the opening of the bypass valve at an initial steady state heat flux of 470 W/cm^2.

performed to investigate the impact of the scram delay time on this spike in clad temperature. It was found that there are potentially two temperature peaks; one due to the reduction in flow, followed by one due to the scram delay time (Figure 9). For in-pool LOCAs on the inlet piping, the spike due to the scram delay remained equal to or less than the temperature spike caused by the flow reduction if the scram delay time was less than about 250 ms. For the LEU case, a delay time of 250 ms produces a slightly larger secondary peak (~169°C).

With respect to long term cooling, Figure 6 and Figure 7 show that the core is coolable for at least 24 hours in all of the in-pool LOCA cases that were investigated. For in-pool LOCA's that occur on the cold leg (inlet side), Figure 6 shows these cases produce long term maximum cladding temperatures lower than what was achieved in the Test F transient. The reason for this difference can be inferred from the natural circulation flow path at 1 day, conceptual shown in Figure 10, and the relative flow rates. Figure 11 indicates that there is 25 kg/s of coolant drawn into the primary system through the pool connection valve. In this case, some or all of the coolant can short-circuit the nominal flow path (i.e. flow path of Test F) by exiting the break at the inlet side of the vessel with coolant resupplied by the pool through the pool connection valve (Figure 10). This results in a higher (more negative) mass flow rate through the core during the transient (Figure 12).

For the hot leg LOCA's (outlet side), Figure 7 shows these cases produce long term cladding temperatures similar to the Test F transient. However, in addition to the coolant having to travel from the reactor vessel to the bypass valve and back (like in Test F), the coolant will also flow into and out of the pool before returning to the reactor vessel. This flow pattern adds additional flow resistance to the system, which reduces the in core flow rate to a value below (less negative) than what was established for the Test F transient (Figure 12). The reduced flow rate results in a bulk coolant temperatures more similar to test F, which is why the long term cladding temperature is more similar to Test F rather than inlet leg LOCA's.

The LOCA results shown here suggest that the severity (e.g. maximum clad temperature) of most cases is similar to the Test F transient described in Ref. [3]. If there was a failure of the bypass valve, there would be little impact on inlet leg LOCAs (LOCA1, LOCA2 and LOCA1+2) since there is normally no significant flow resistance added by this valve. For outlet leg LOCAs, the severity due to failure of the bypass valve would be comparable to the Test G transient (described in Ref. [3]) where natural circulation is limited to within the reactor vessel.

Figure 6 through Figure 12 show that the results obtained with the LEU core are quite similar to the HEU core.

2.4 Loss of External Cooling

A loss of external cooling (LOEC) transient was performed to evaluate the impact of a stoppage in forced flow within the vessel shroud and secondary side of primary heat exchangers during normal operation. For this transient, the figure of merit is the time it takes for the coolant temperature measurements to reach alarm settings defined in Ref. [4]. Figure 13 shows the inlet temperature (T4-1301), outlet temperature (T4-1304) and temperature difference (DT4-1301) as a function of time following a LOEC

with an initial nominal heat flux of 470 W/cm^2. The high alarm for inlet temperature is set at 43°C and was reached at 1340 s for the HEU core. The high alarm for outlet temperature is set at 60°C and was reached at 2540 s for the HEU core. For the LEU core which has a lower power (77.2 MW for HEU versus 72.4 MW for LEU), the inlet and outlet temperature alarms are reached at 1440 s and 2820 s, respectively.. The temperature difference alarm (set at 11°C) was not reached for either core since the temperature difference decreases slightly within the time frame investigated (5000 s). Figure 14 shows the same plot as Figure 13 but for an initial nominal heat flux of 600 W/cm^2. In this case, the high alarm for the inlet and outlet temperatures are reached at 1040 s and 1620 s, respectively, for the HEU core (1100 s and 1820 s, respectively for the LEU core). Figure 15 illustrates the impact increasing temperature has on the thermal margins (ONBR, FIR and CHFR)[5]. It should be noted that a more thorough and accurate treatment of the margins is given in the steady state safety analysis of BR2 [5]. However, given that the magnitudes for ONBR and FIR at 0 s are not too different from the values presented in Ref. [5], some useful information can be obtained from the RELAP5 results[6]. For instance, the rate of change in thermal margins is quite small, indicating that the thermal margins will not be significantly compromised in relation to the time it takes to reach the high alarm for either the inlet or outlet temperature. The results for both 470 W/cm^2 and 600 W/cm^2 indicate there is sufficient time for operator intervention should there be a loss of external cooling. An exception to this is the ONBR for the 600 W/cm^2 case; some nucleate boiling may occur since the ONBR is initially very near 1.0.

3 Summary and Conclusions

Elevated Pool Temperature

Simulations of the Test G transient were performed to assess the impact of the initial steady-state pool temperature (30°C to 60°C) on core cooling. In the early stages of the transient (<100s) the pool temperature had little to no impact on the maximum cladding temperature. For long term cooling, the peak clad temperature remains at or near the saturation temperature. The peak coolant inlet temperature is ~108°C at 3 hours for a pool temperature of 30°C and increases to ~114°C at 6 hours for a pool temperature of 60°C. The coolant temperature continually decreases following its peak value, indicating that the core is coolable for at least 24 hours without any issues related to peak clad temperature.

Reduced Steady State Flow Rate

Simulations indicated that the steady-state flow rate could be reduced from 6478 m^3/hr to 4965 m^3/hr just before tripping the reactor on the pressure difference signal (DP4-1301). At this reduced flow rate, the steady state peak clad temperature increased from 155°C to 180°C, a value slightly less than that

[5] ONBR = onset of nucleate boiling ratio, FIR = flow instability ratio and CHFR = critical heat flux ratio; as defined in Ref. [3].

[6] In Ref. [5], the ONBR and FIR for 470 W/cm^2 was 1.3 and 2.9, respectively. The ONBR and FIR for 600 W/cm^2 was 1.0 and 2.3, respectively. It is expected that the RELAP5 results would be less conservative since it does not take into account, for example, the fuel element manufacturing uncertainties. It should be noted that the CHFR presented here differs quite a bit from the results presented in Ref. [5] since the RELAP5 post processing is limited to the less accurate direct substitution method rather than the heat balance method used in Ref. [5].

obtained for a steady state heat flux of 600 W/cm^2 (183°C). Operation at a reduced steady-state flow rate had little to no impact on the results of the Test G transient.

In Pool Loss of Coolant Accidents

In pool LOCA simulations of double ended guillotine breaks at various locations of the primary coolant system were performed. LOCA's that occur within the pool on the hot leg(s) were shown to have similar peak cladding temperatures as the Test F transient. If these LOCA cases suffered from a failure of the bypass valve the transient would be comparable to Test G. LOCA cases within the pool on the inlet leg(s) would be unaffected by a failure in the bypass valve since the natural circulation flow path is predominantly through the pipe break(s) and pool connection valve.

Loss of External Cooling

An LOEC transient was performed for an initial peak steady-state heat flux of 470 W/cm^2 and 600 W/cm^2 to evaluate the thermal hydraulic behavior of the core following a stoppage in forced flow within the vessel shroud and secondary side of primary heat exchangers. For HEU fuel, results indicate that there is sufficient time for operator intervention since the inlet coolant temperature will alarm within 1340 s and 1040 s for a heat flux of 470 W/cm^2 and 600 W/cm^2, respectively. For LEU fuel, the time at which the coolant temperature alarm is reached increases to 1100 s and 1820 s, respectively. Considering the uncertainty and small margin to ONB, some nucleate boiling may occur in the core for the 600 W/cm^2.

HEU versus LEU core

Simulations of elevated pool temperature, reduced steady state flow rate and in pool loss of coolant accidents using the BR2 representative core configuration show nearly identical peak cladding temperatures and overall system behavior for both HEU and LEU fuel. The loss of external cooling accident shows that there is additional margin for the LEU core with respect to the timing in which temperature limits are reached.

References

[1] RELAP5/MOD3.3 Code Manual, Nuclear Safety Analysis Division, July 2003, Information Systems Laboratories, Inc., Rockville, Maryland and Idaho Falls, Idaho.

[2] J. Licht, B. Dionne, G. van den Branden, E. Sikik and E. Koonen "RELAP5 model description and validation for the BR2 loss-of-flow experiments," ANL/GTRI/TM-14/10, 2015.

[3] J. Licht, A. Bergeron, B. Dionne, E. Sikik, G. van den Branden, and E. Koonen "Loss-of-Flow and Loss-of Pressure Simulations of the BR2 Research Reactor with HEU and LEU Fuel," ANL/RTR/TM-15/8, 2016.

[4] G. van den Branden, "Conversion study: design basis document LSSS", BR2/RCE/GVdB/2010/53/conversion_DBD_LSSS-ed 1Draft, August 2010.

[5] J. Licht, A. Bergeron, B. Dionne, E. Sikik, G. van den Branden, S. Kalcheva and E. Koonen "Steady-State Thermal-Hydraulics Analyses for the Conversion of the BR2 Reactor to LEU," ANL/GTRI/TM-14/8, 2016.

Tables

Table 1 Event timing for in-pool LOCA's with HEU fuel

	Cold leg / Inlet side – time in s			Hot leg / Outlet side – time in s		
Event	LOCA1	LOCA2	LOCA1+2	LOCA3	LOCA4	LOCA3+4
LOCA	0.000	0.000	0.000	0.000	0.000	0.000
Reactor scram	0.030	0.031	0.028	0.056	0.057	0.053
Opening of ABV4-1308*	1.675	1.798	1.670	1.659	1.659	1.648
Closing of ABV4-1304*	1.647	1.651	1.645	3.355	3.357	3.428
Closing of ABV4-1305*	1.647	1.651	1.645	3.355	3.357	3.428
Opening of ABV4-1301*	2.465	2.470	2.464	4.174	4.176	4.247
Pump trip	3.287	3.291	3.285	3.299	3.300	3.288
Check valve	15.705	15.708	15.703	17.304	17.306	17.373

*Time at which valve begins to open or close. Once tripped, the ABV4 valves take 15s to fully open or close.

Table 2 Event timing for in-pool LOCA's with LEU fuel

	Cold leg / Inlet side – time in s			Hot leg / Outlet side – time in s		
Event	LOCA1	LOCA2	LOCA1+2	LOCA3	LOCA4	LOCA3+4
LOCA	0.000	0.000	0.000	0.000	0.000	0.000
Reactor scram	0.030	0.030	0.027	0.056	0.057	0.052
Opening of ABV4-1308*	1.674	1.745	1.851	1.659	1.658	1.658
Closing of ABV4-1304*	1.646	1.650	1.644	3.266	3.256	3.332
Closing of ABV4-1305*	1.646	1.650	1.644	3.266	3.256	3.332
Opening of ABV4-1301*	2.465	2.469	2.463	4.085	4.075	4.151
Pump trip	3.287	3.291	3.285	3.300	3.299	3.299
Check valve	15.536	15.540	15.534	17.034	17.024	17.096

*Time at which valve begins to open or close. Once tripped, the ABV4 valves take 15s to fully open or close.

Figures

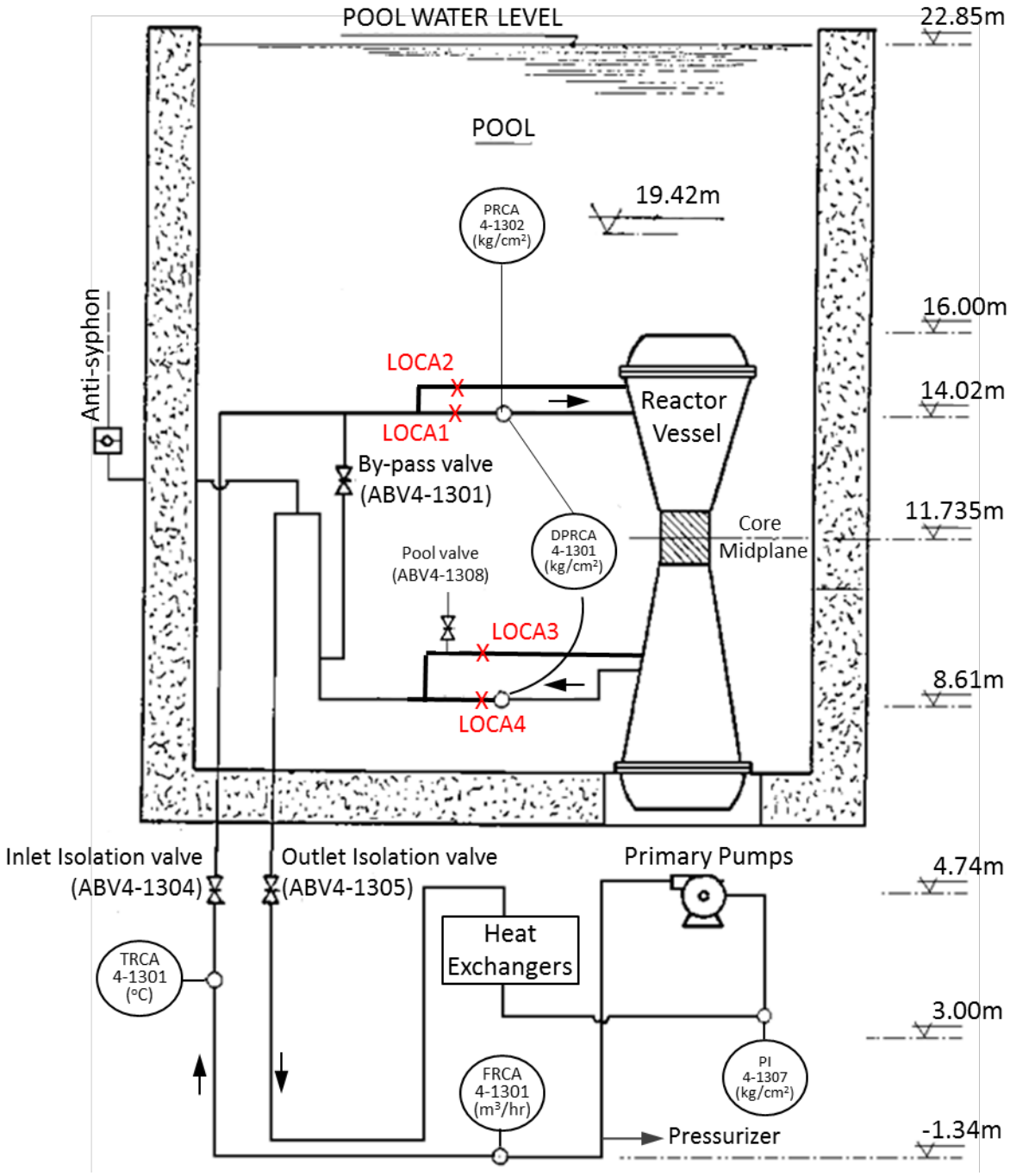
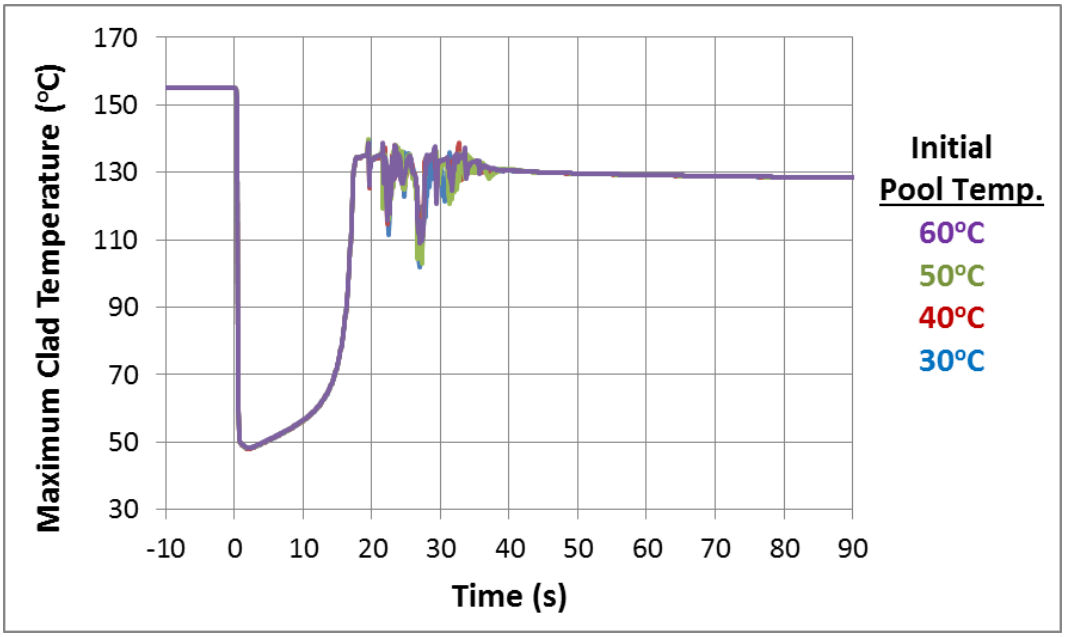
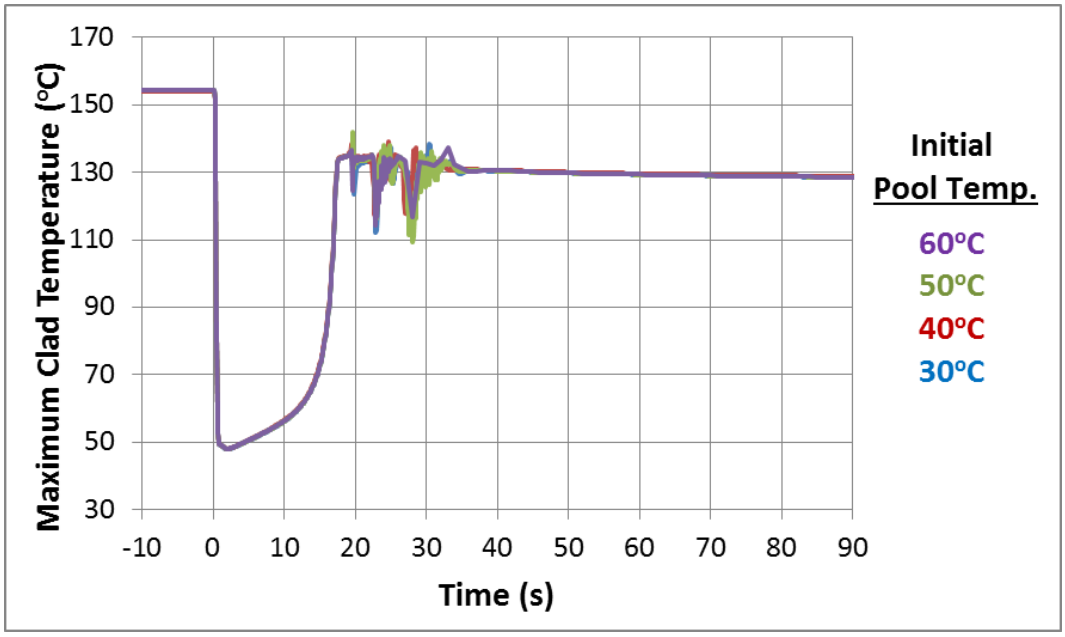


Figure 1 Conceptual drawing of BR2.

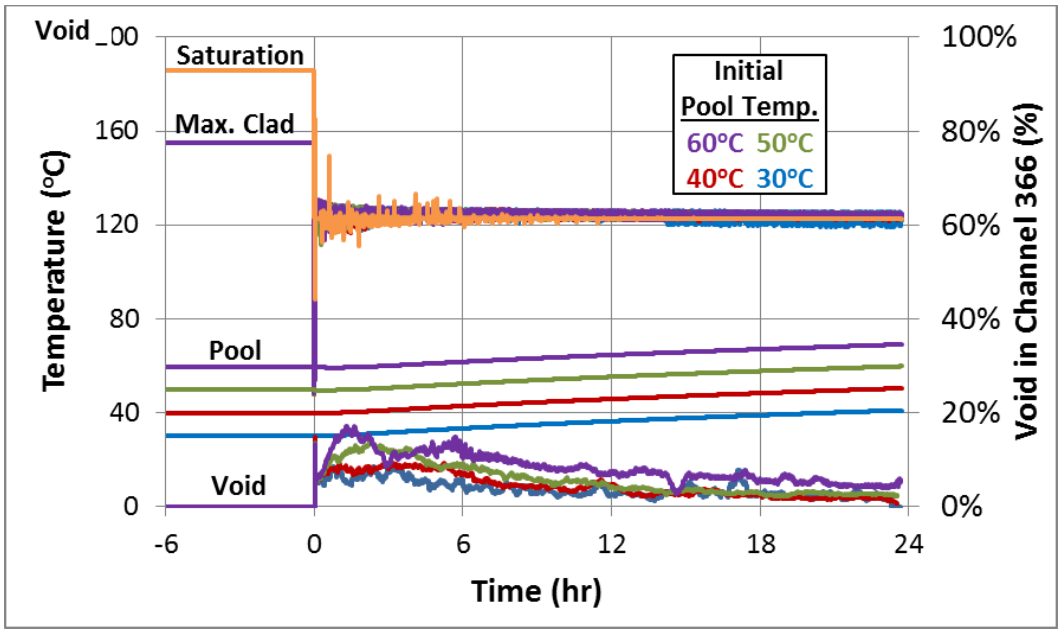


a) HEU

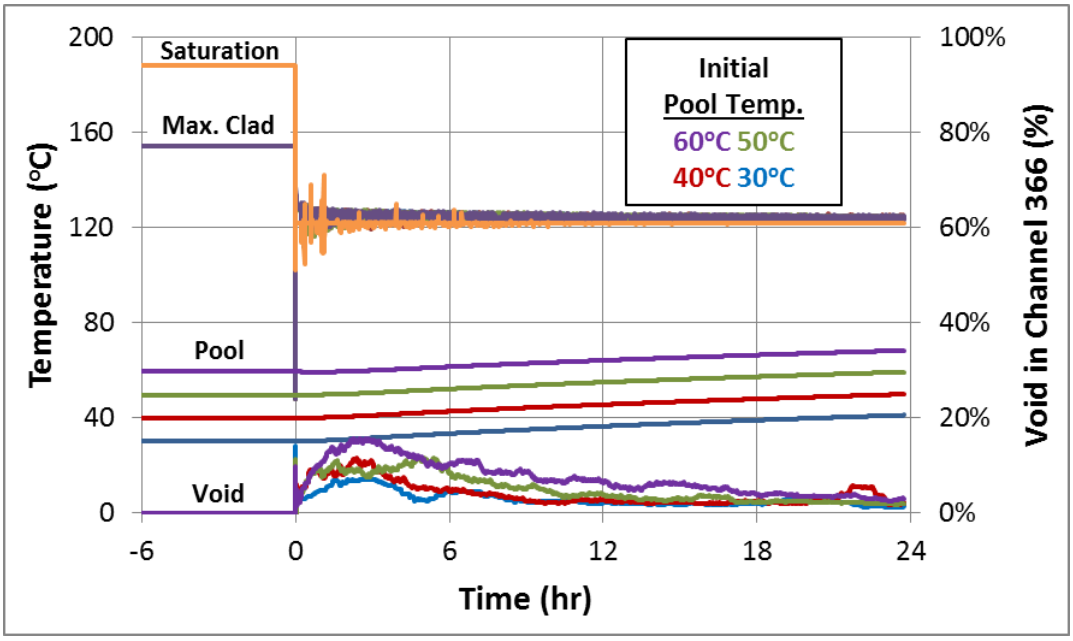


b) LEU

Figure 2 Impact of pool temperature on maximum clad temperature for Test G.

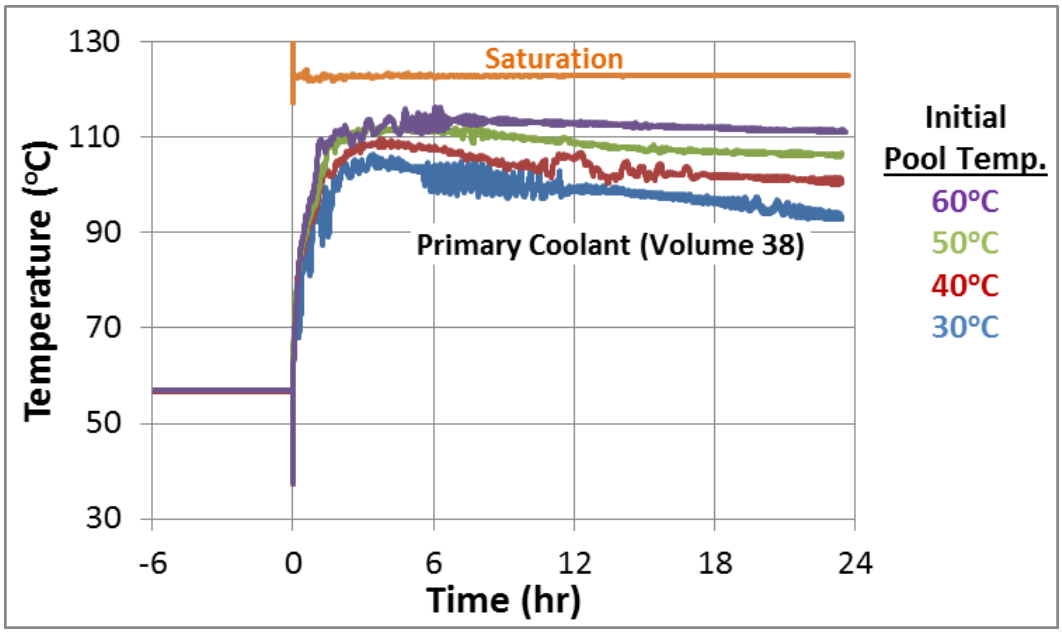


(a) HEU



(b) LEU

Figure 3 Impact of pool temperature on maximum clad temperature and channel void for Test G.



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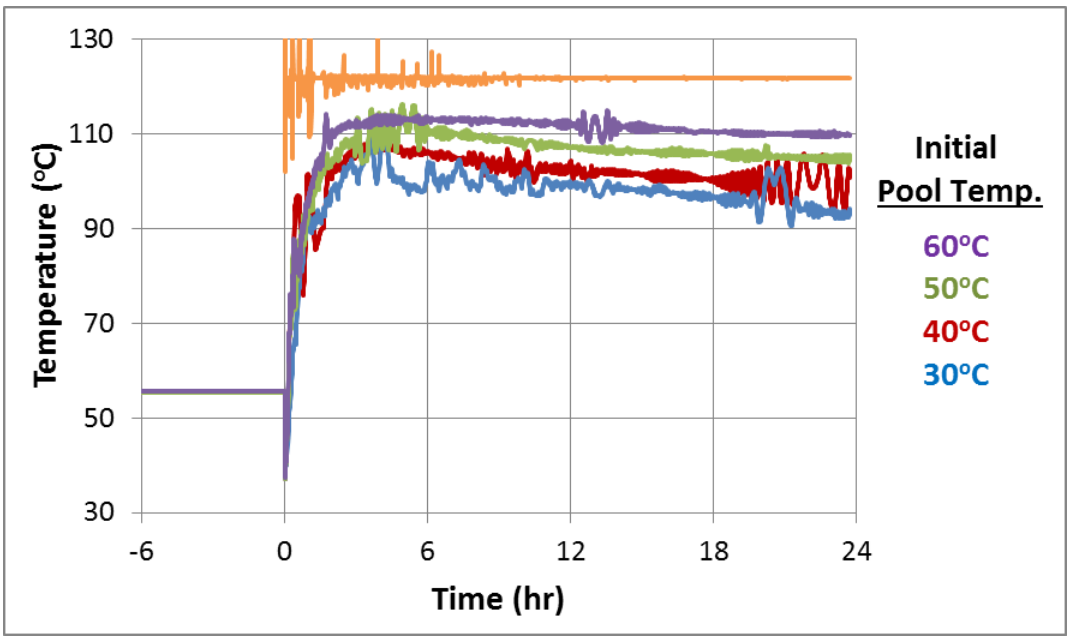
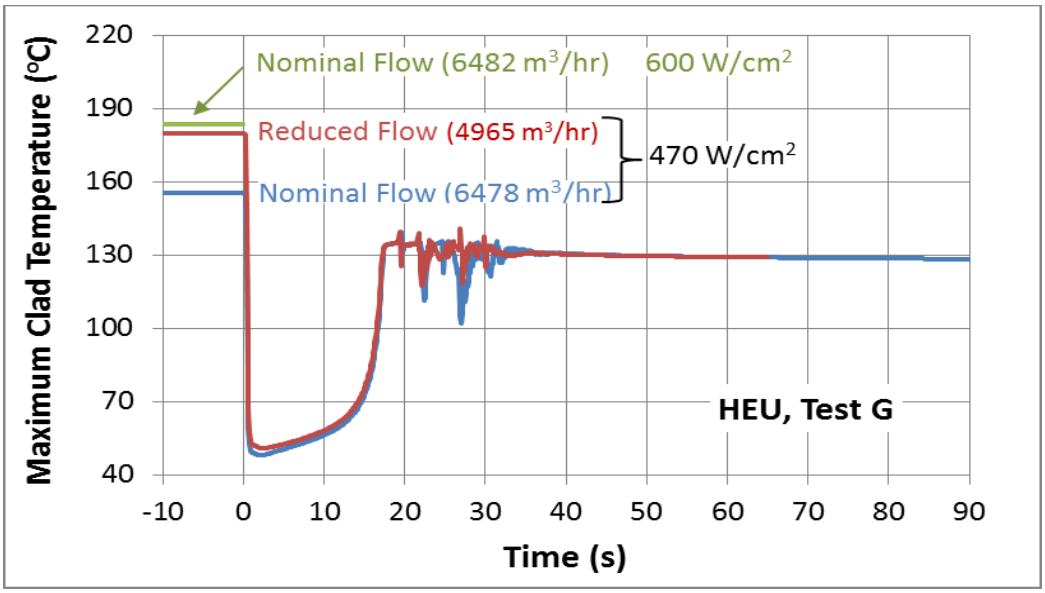
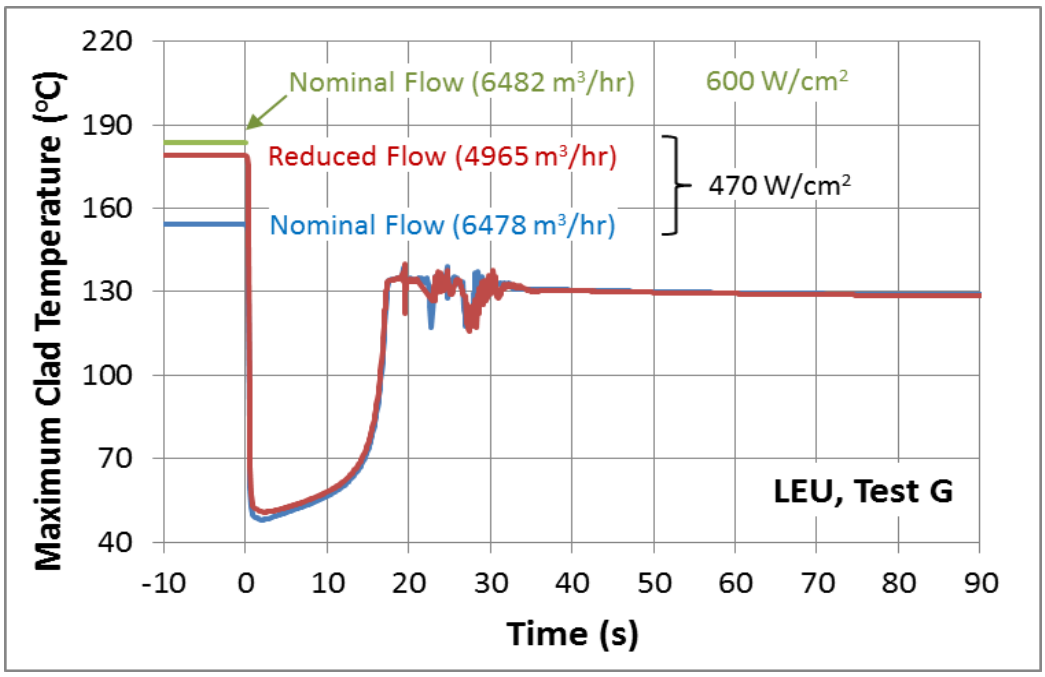


b) LEU

Figure 4 Impact of pool temperature on coolant temperature for Test G.

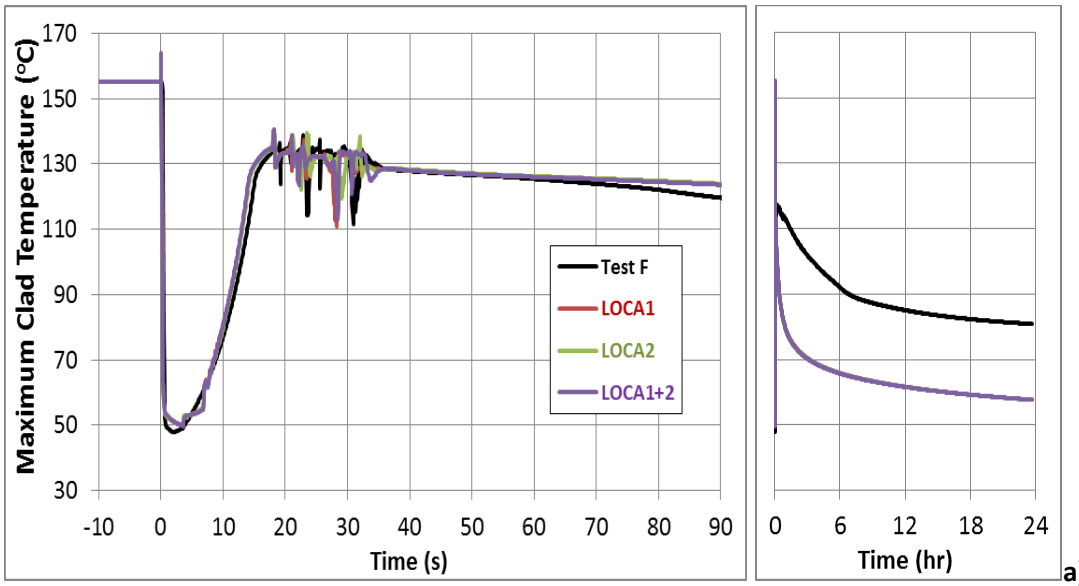


a) HEU

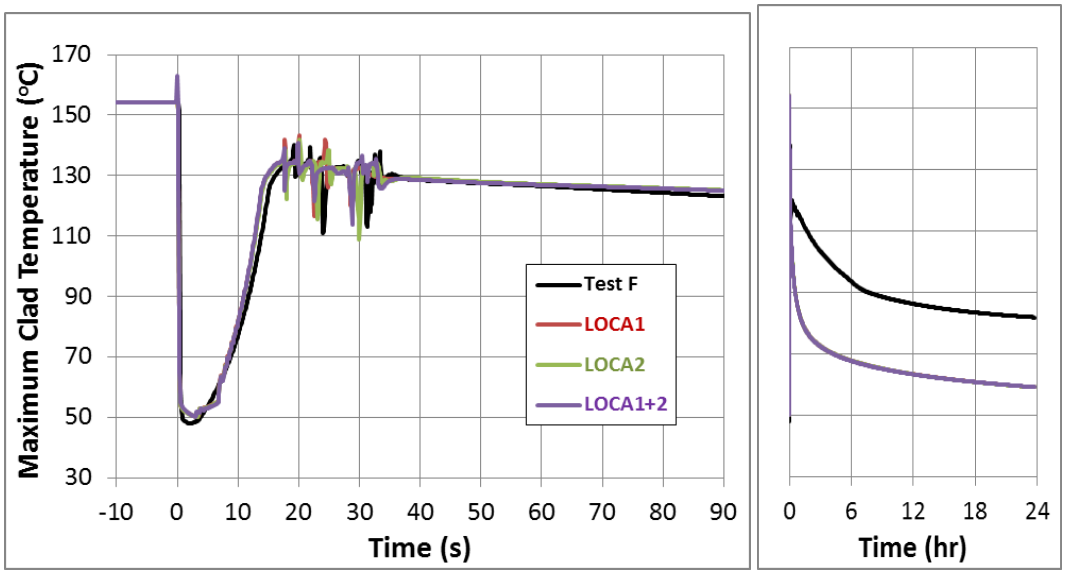


b) LEU

Figure 5 Impact of reduced flow on maximum clad temperature for Test G.

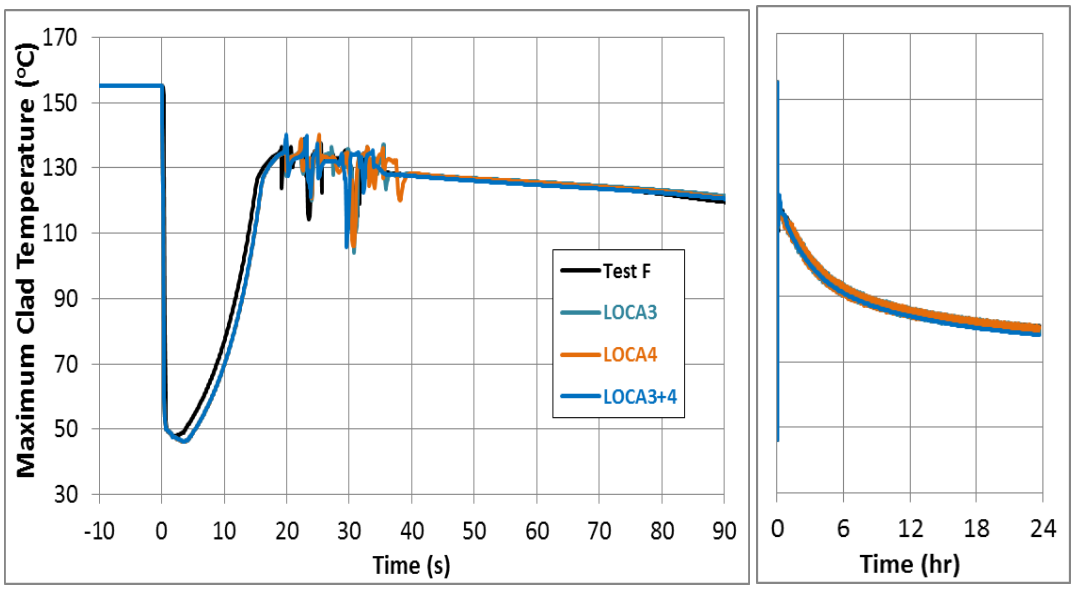


a) HEU

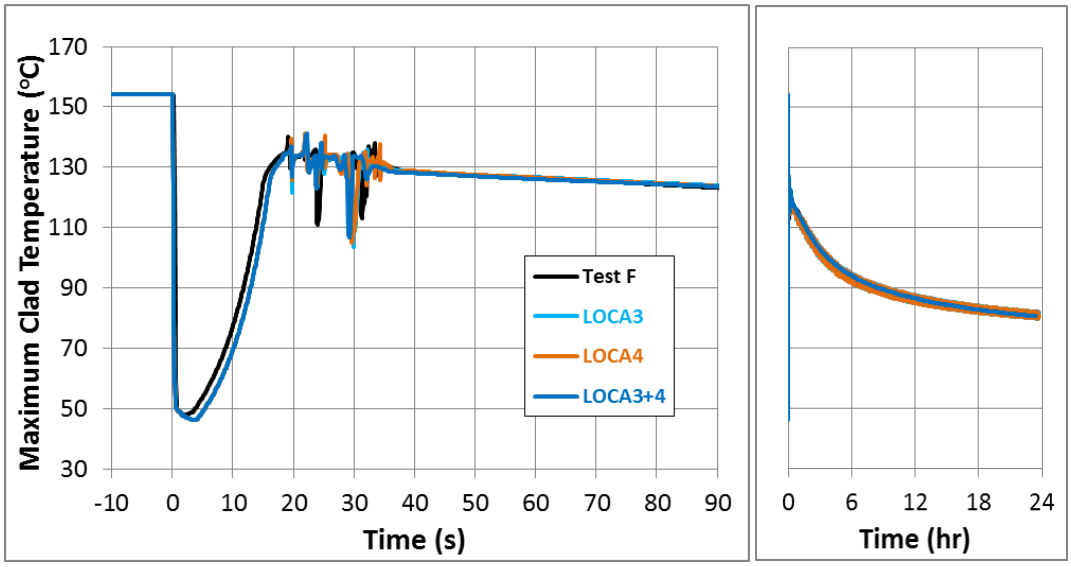


b) LEU

Figure 6 Maximum clad temperature for inlet pipe LOCA.

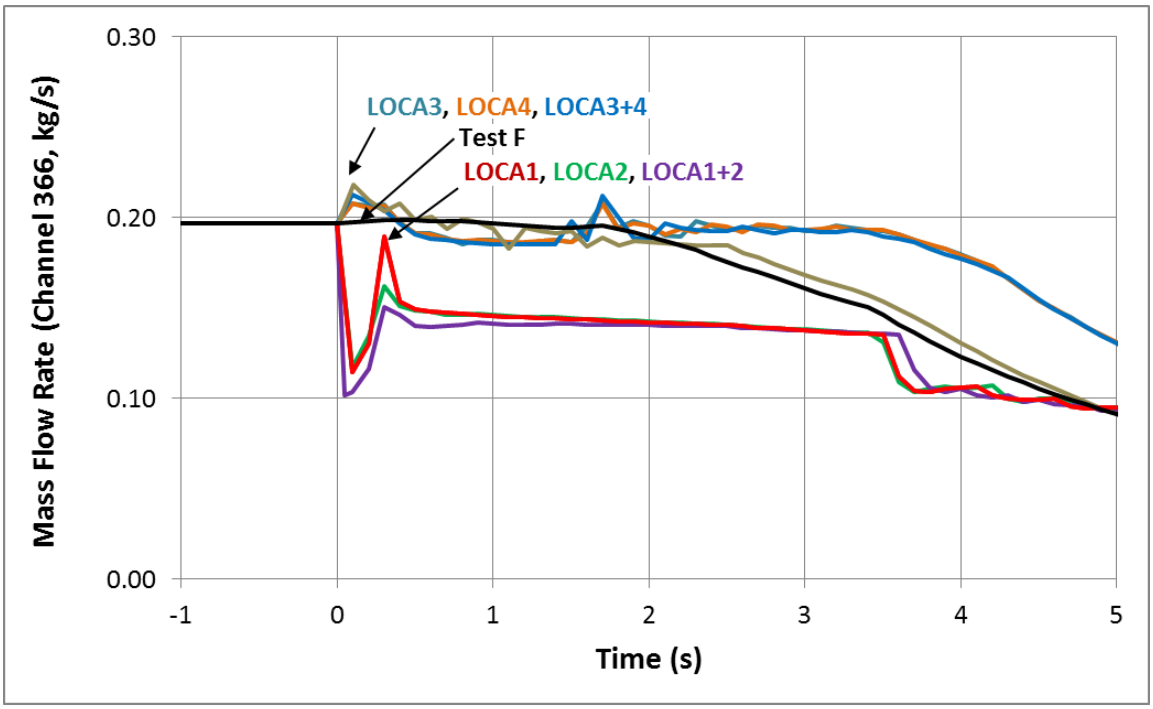


a) HEU

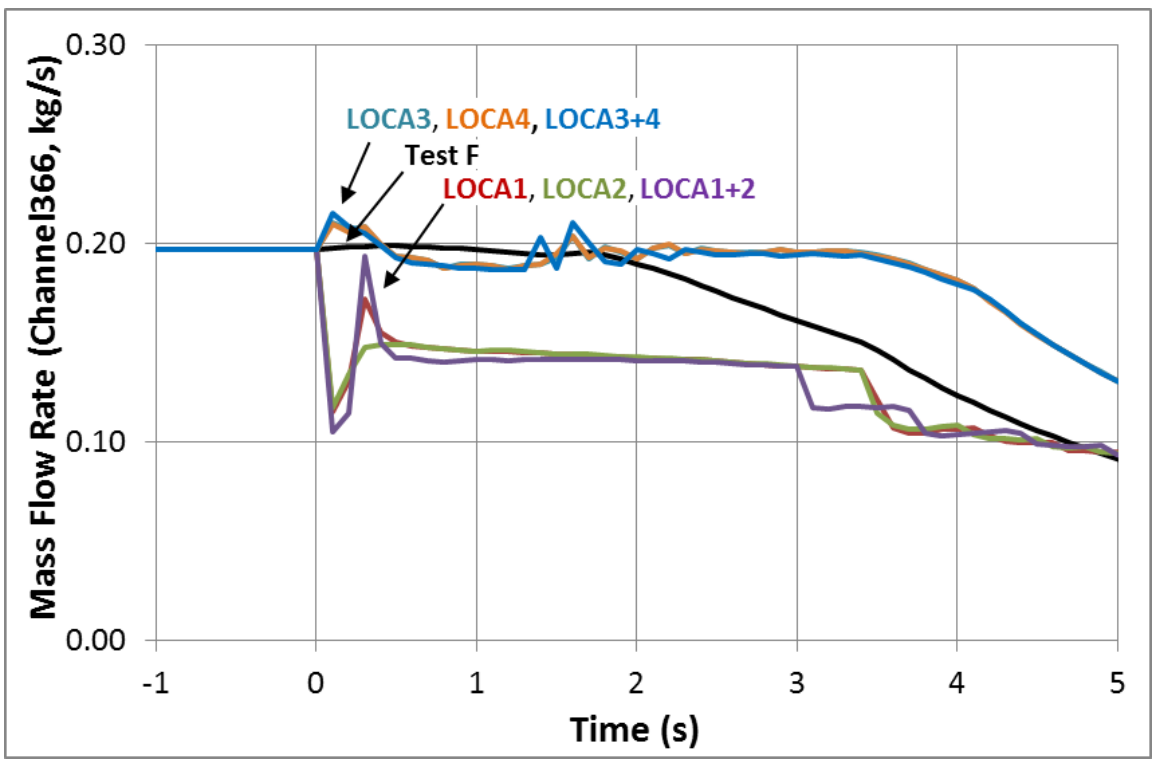


b) LEU

Figure 7 Maximum clad temperature for outlet pipe LOCA.

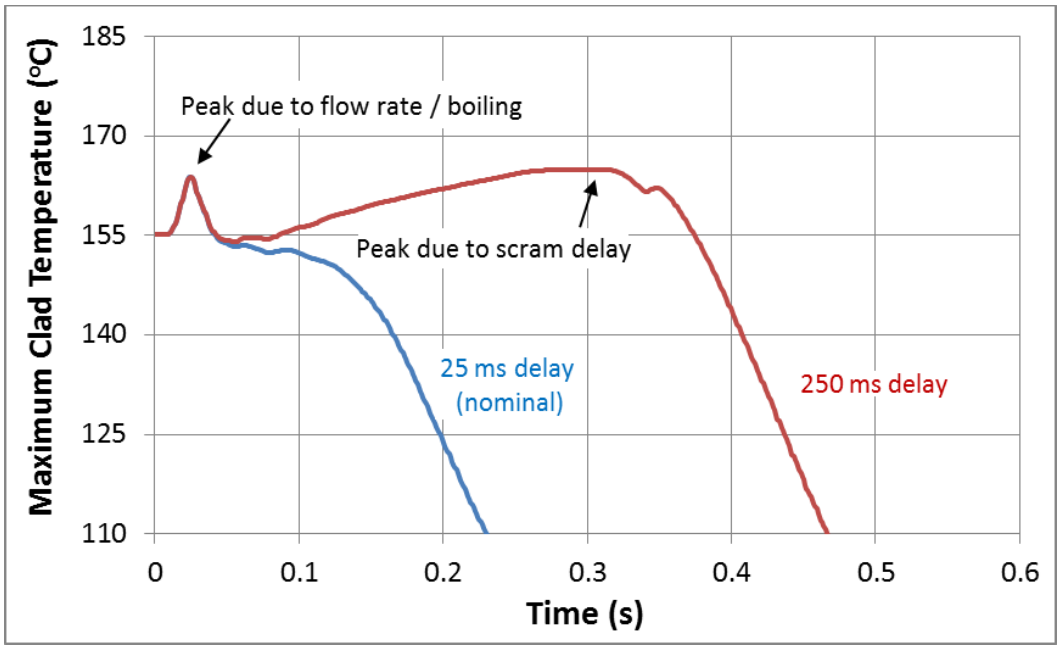


a) HEU

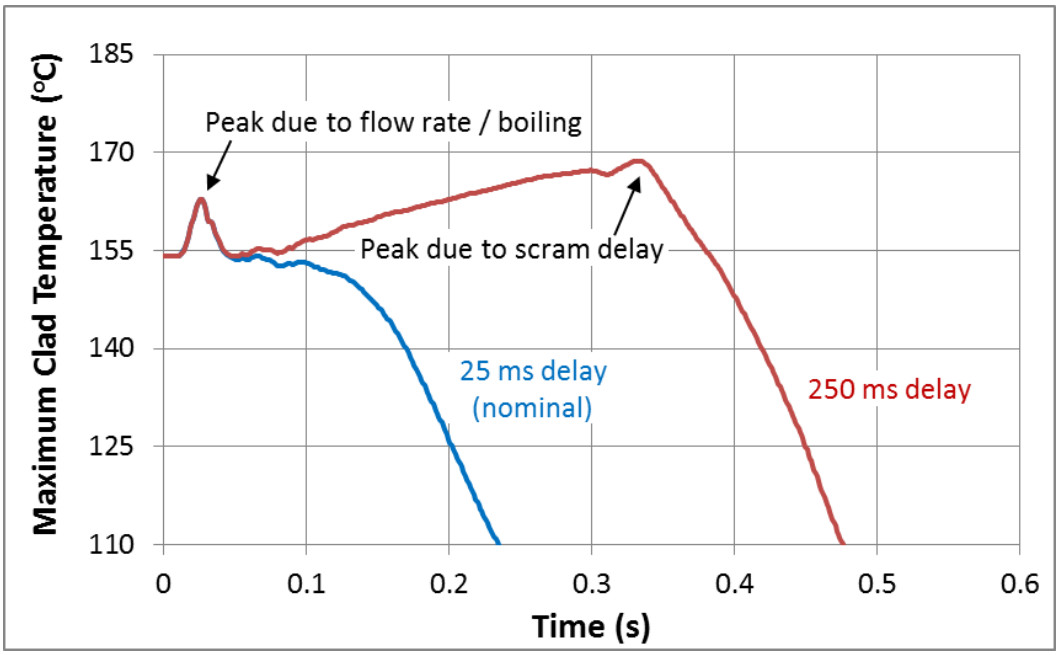


b) LEU

Figure 8 Mass flow rate in channel 366 of the high heat flux fuel element for LOCA's at various locations (positive flow is downward).



a) HEU



b) LEU

Figure 9 Peak clad temperature as function of scram delay time for LOCA1+2.

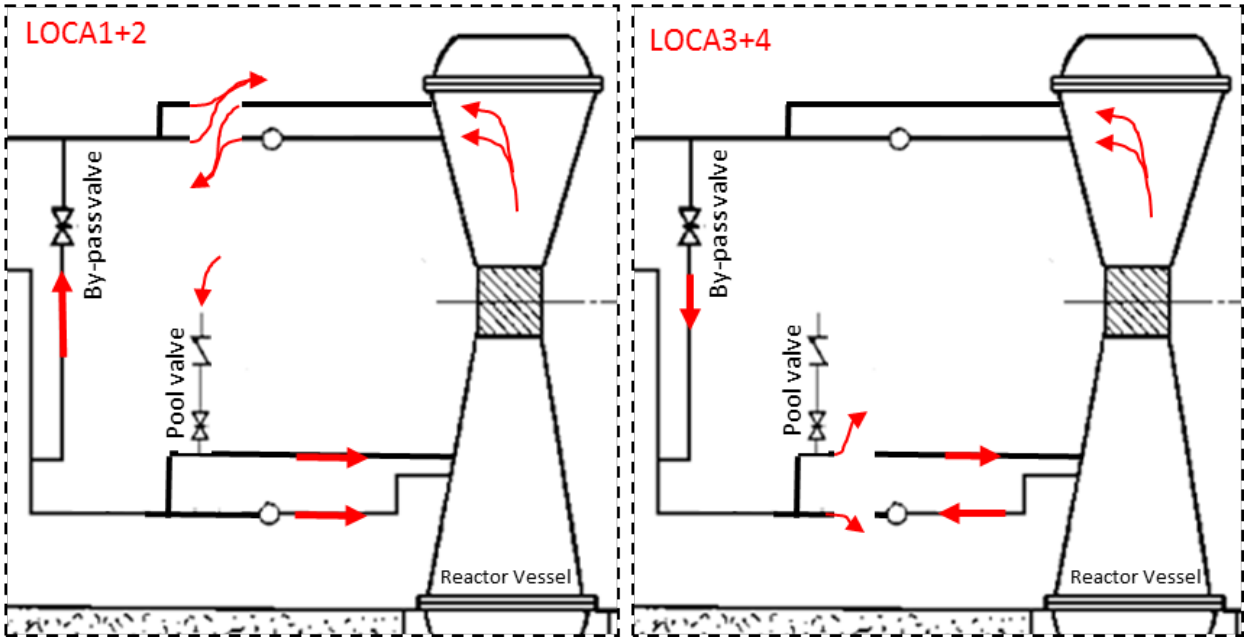
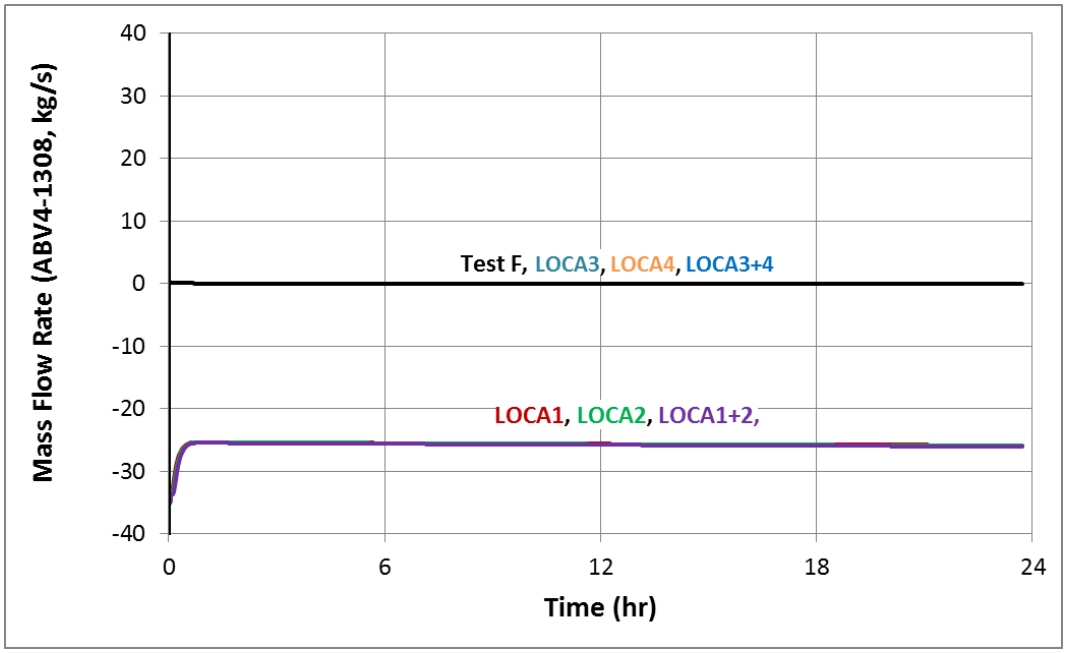
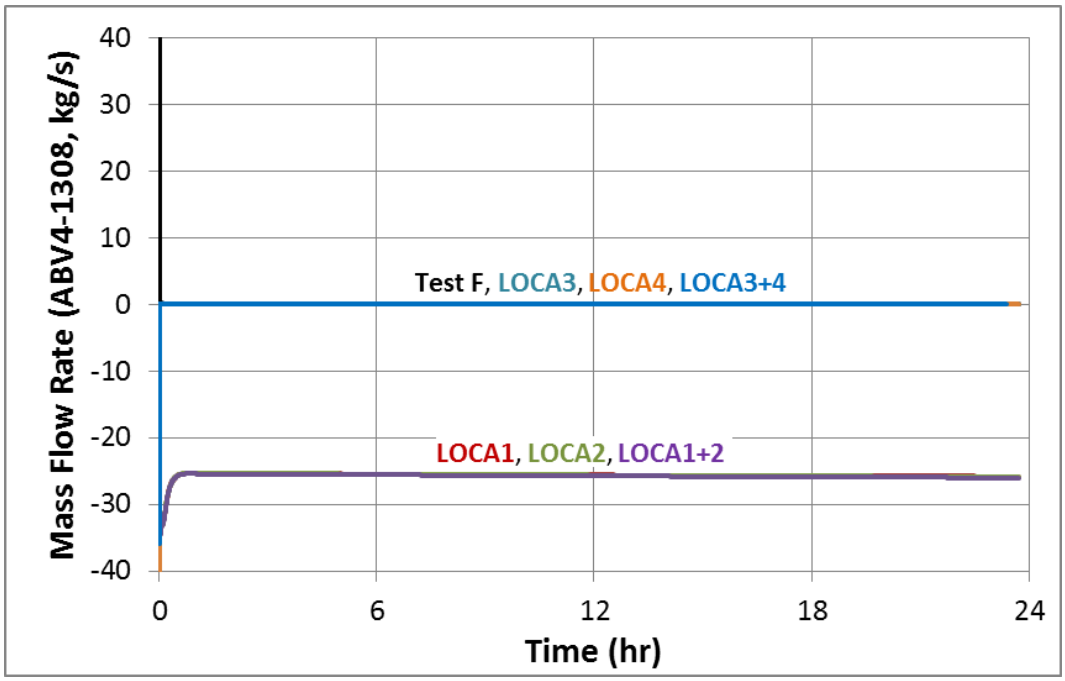


Figure 10 Conceptual drawing of the long term natural circulation flow paths from LOCA simulations of the HEU and LEU fueled cores.

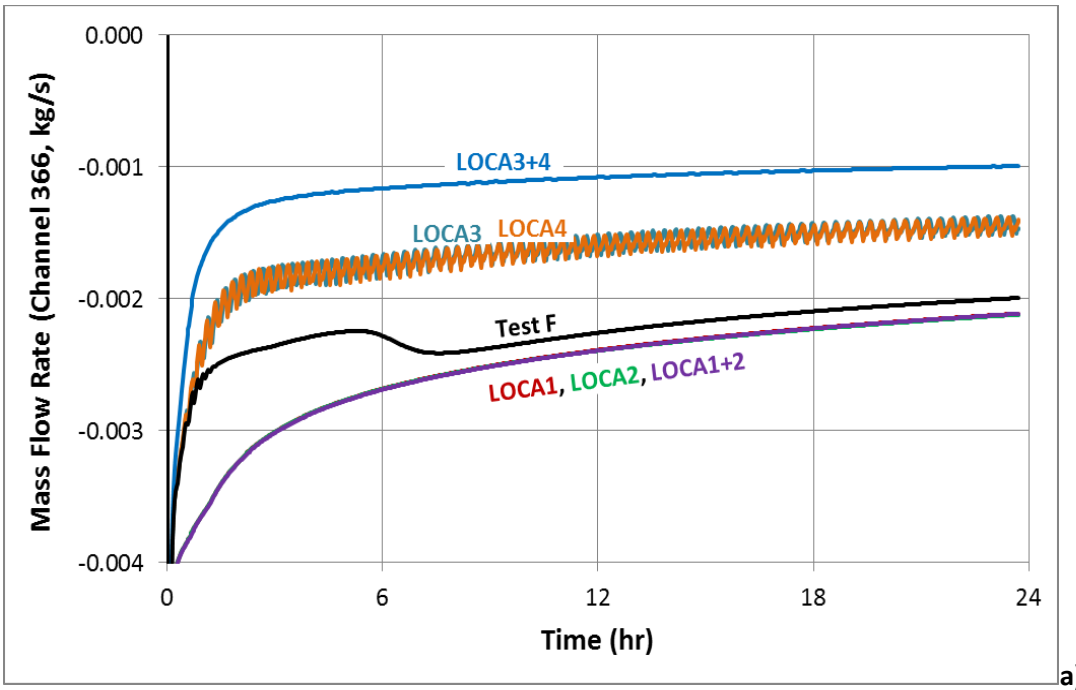


a) HEU

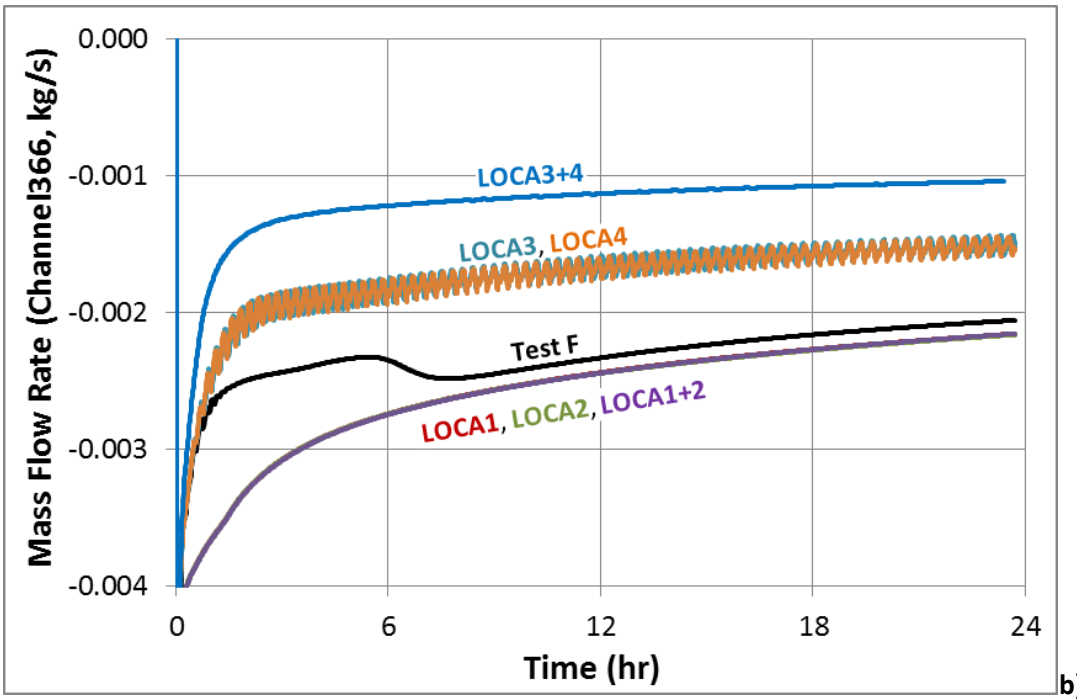


b) LEU

Figure 11 Mass flow rate through pool connection valve (ABV4-1308), negative values represent flow into the primary system.

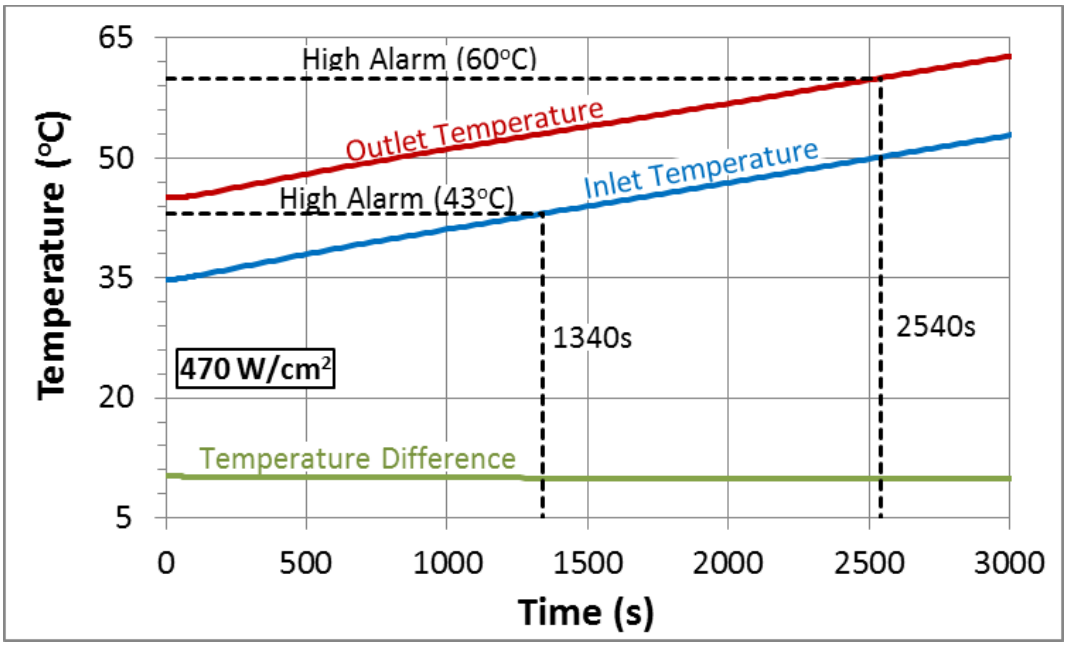


a) HEU

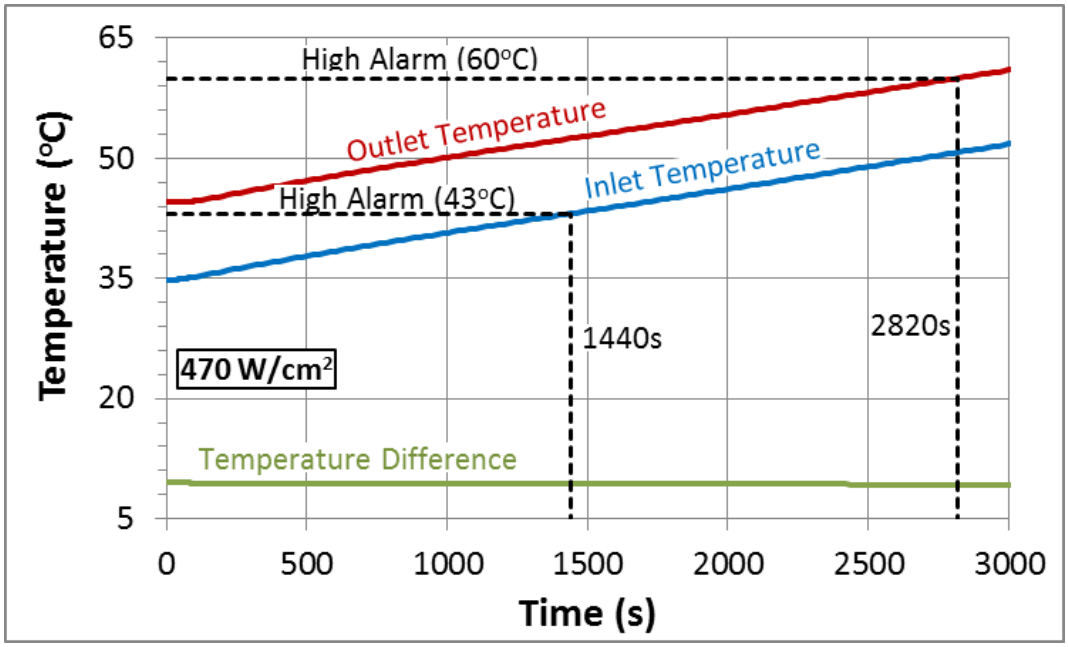


b) LEU

Figure 12 Mass flow rate in channel 366 of the high heat flux fuel element for LOCA's at various locations (positive flow is downward).

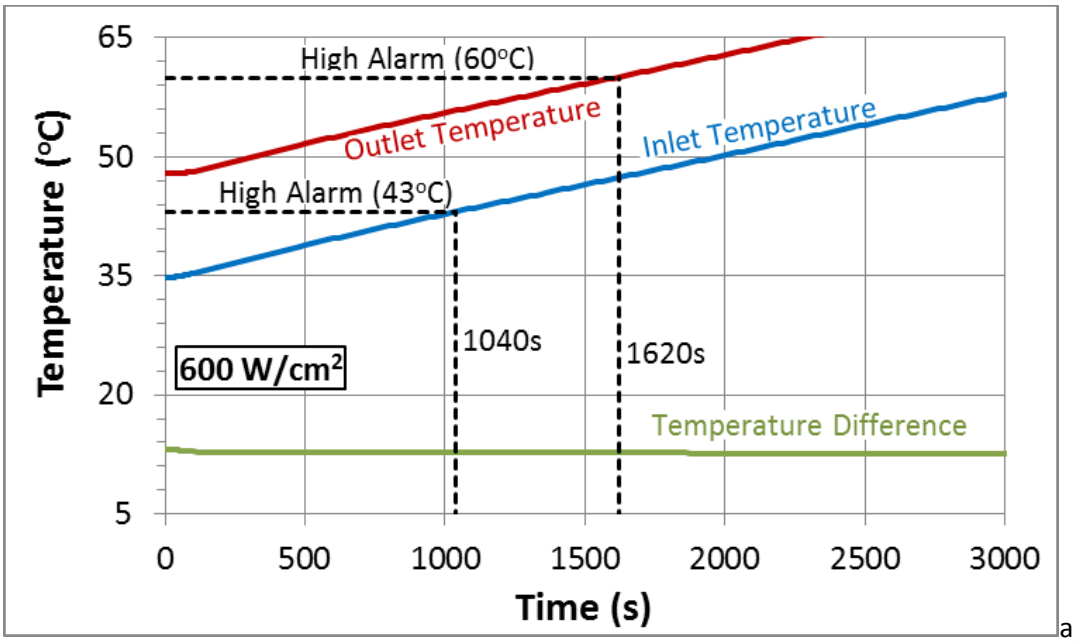


a) HEU

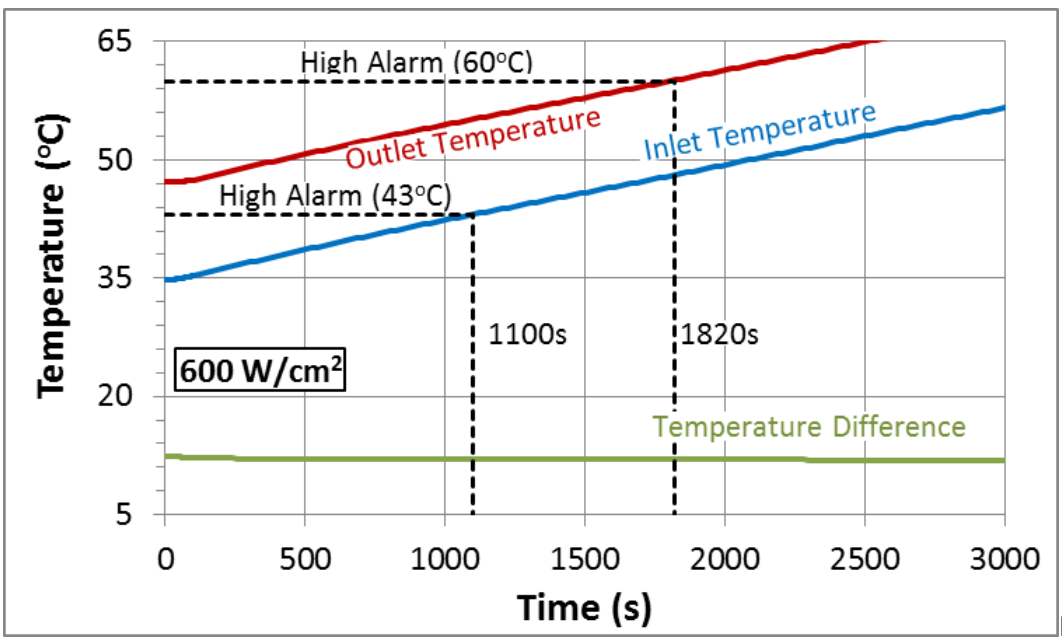


b) LEU

Figure 13 Coolant temperatures in relation to high alarm settings for nominal heat flux = 470 W/cm^2.

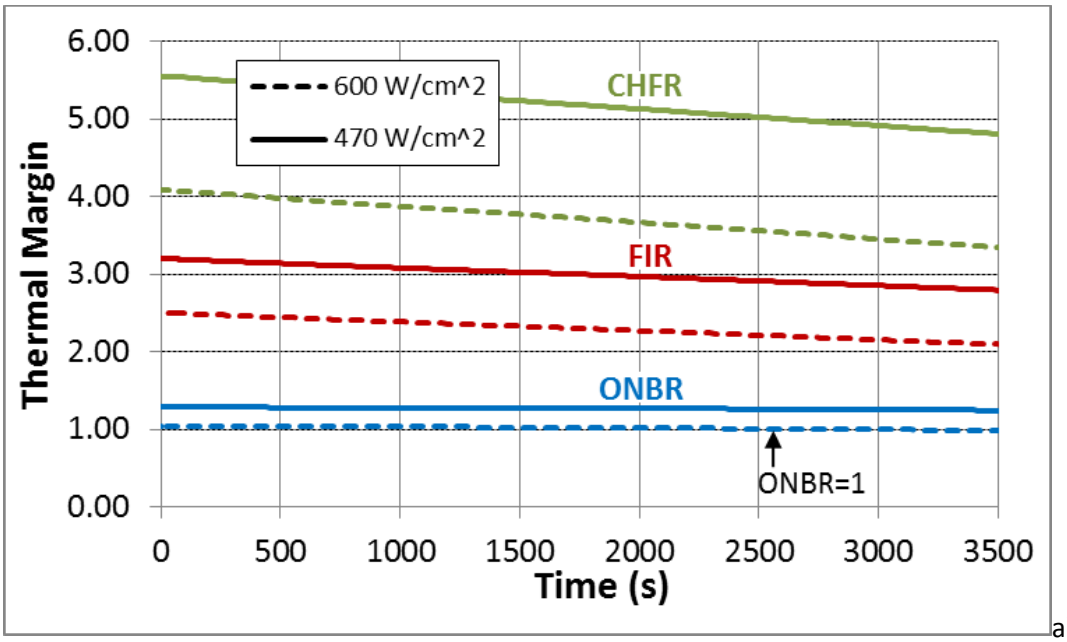


a) HEU

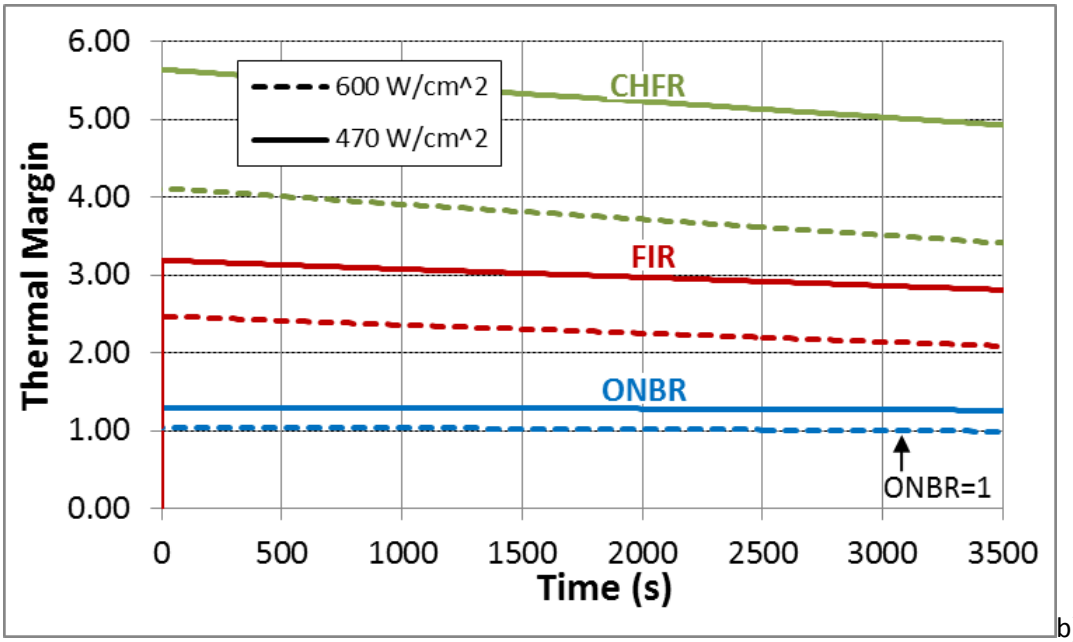


b) LEU

Figure 14 Coolant temperatures in relation to high alarm settings for nominal heat flux = 600 W/cm^2.



a) HEU



b) LEU

Figure 15 Thermal margins for a LOEC transient at 470 W/cm^2 and 600 W/cm^2.

Appendix A LOCA model

Figure A-1 shows a schematic for a double ended guillotine break model in RELAP5. It consists of 3 valves and 2 time dependent volumes. A motor valve (disconnect valve) is placed between two sections of piping to facilitate a pipe disconnect. Motor valves (break valves) are connected to the primary piping on either side of the pipe disconnect to divert the pipe connection to the pool. Motor valves have been used to allow control of the valve opening /closing rate (default value is 10 s^{-1}). As recommended in the RELAP5 manual for break valves, the modified PV term, choking model and abrupt area change model were all applied (efvcahs : 1000100). To initiate a LOCA, all three valves are tripped at the same time.

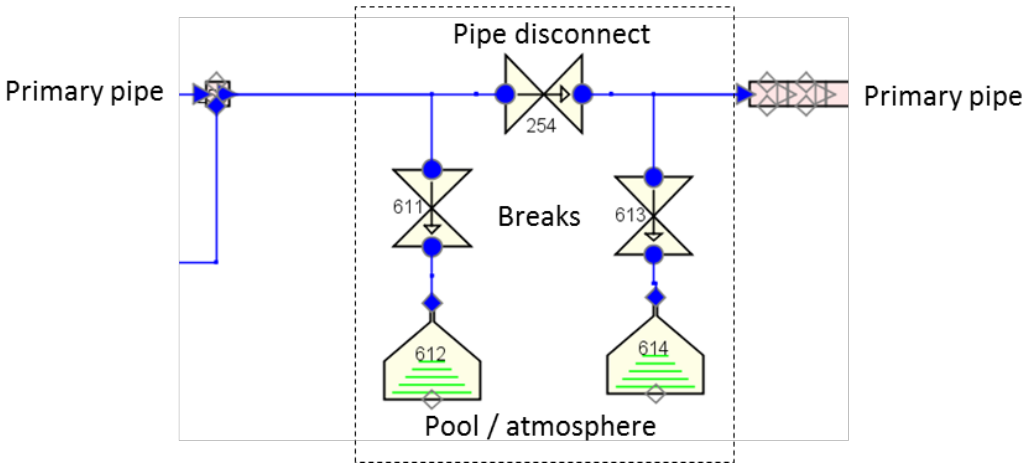


Figure A-1. Schematic for a double guillotine break model in RELAP5.

The time dependent volumes connected to breaks within the pool have been given a pressure based on the elevation in the RELAP5 model, as shown in Table A-1. The temperature of these time dependent volumes was maintained at the same value as the pool volume.

Table A-1. Pressure specified at break locations within pool.

Location	Distance from pool surface	Specified pressure[7]
Hot leg	14.24 m	240693 Pa
Pool connection valve	13.61 m	234514 Pa
Cold leg	8.83 m	187631 Pa

[7] Pool pressure at break elevation = (pool density) x (gravity) x (elevation) + (atmospheric pressure) ≈ 1000 (kg/m^3) x 9.81 (m/s^2) *elevation (m) + 101000 (Pa)

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Nuclear Engineering Division
Argonne National laboratory
9700 South Cass Avenue, Bldg. 208
Argonne, IL 60439

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