Best Estimate Analysis Of Loss Of Flow Events At CANDU Nuclear Power Plants

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Abstract

During plant operation there is a potential for a loss of forced circulation in the primary heat transport system due to a Loss of Class IV power to an single electrical bus, which leads to a loss of a main heat transport pump, or through a failure of power to multiple busses or to the plant which will lead to the loss of 2 or all 4 pumps. Historically, nuclear safety analysis for these events have adopted bounding assumptions for key parameters to ensure that the outcome of the analysis would envelope those expected during an event, and did not take credit for possible process system mitigation. While this provided conservative estimates of the consequences of these events, and met the analysis requirements for licensing at the time, the existing analyses do not provide any knowledge on true response of the plant. The objective of this work is to perform best estimate and deterministic analyses, including the impact of anticipated Reactor Regulating System actions for Loss of Flow (LOF) events in a CANDU station, and to provide the sensitivities to process system component availability. This initial work will feed into downstream Best Estimate and Uncertainty (BEAU) which will explicitly account for the uncertainties in the key parameters, and will eventually provide a measure of the true safety margins for these events.

1. Introduction

1.1 Background

Loss of Forced circulation events occur as a result of complete and partial failures of the Class IV electrical power system in CANDU reactors. Historically, nuclear safety analysis for these events have adopted bounding assumptions for key parameters to ensure that the outcome of the analysis would envelope those expected during an event, and did not take credit for possible process system mitigation. While this provided conservative estimates of the consequences of these events, and met the analysis requirements for licensing at the time, the existing analyses do not provide any knowledge on true response of the plant. Depending on the frequency and nature of Loss of Forced (LOF) circulation events draft regulatory guidelines such as S310 [1] make provisions for both crediting automatic control functions as well as probabilistic analysis of less frequent, accident scenarios. Specifically, draft regulatory standard S310 defines accident classes as either Abnormal Operating Occurrences (AOOs) or Design Bases Accidents (DBA) dependent on the frequency of occurrence of these events. AOO events are classified by event frequencies of 10^{-2} or greater while DBA events occur in the range 10^{-2} to 10^{-5}.[i]

[i] Beyond Design Basis events are not considered in the present study.

Historically, Nuclear Safety Analysis has been performed using a large number of conservative simplifying assumptions with respect to operating conditions and modelling methodologies. As a result of these simplifications, it is impossible to determine the exact margins to safety limits. Due to the importance of such predictions in ensuring public safety it is necessary to have an accurate quantification of these margins. Furthermore, Risk Informed Decision (RID) making practices and maintenance optimization [2] at each plant rely on accurate quantification of the impact of upgrades/refurbishment on safety margins.

With the advent of more realistic computational tools and detailed plant modelling, best estimate predictions of plant response are now possible. However, the accuracy of the tools and models becomes increasingly important since operating safety limits will be defined on these new methods. The Canadian Nuclear Safety Commission (CNSC) has recognised that best estimate predictions of plant response, along with accurate assessments of uncertainties, is an acceptable alternative to more limiting and bounding analyses for demonstrating safety system response [3]. The United States Nuclear Regulatory Commission (USNRC) has also revised its acceptance criterion for Emergency Core Cooling Systems (ECCS) response to allow for use of best estimate methods [4]. The calculation of the integrated uncertainty is usually performed for a limiting component within the CANDU industry using a combination of Monte-Carlo methods, multi–dimensional Functional Response Surfaces (FRS) [5], Physical Interdependency Functional Relationships [6] or Wilk's method [7].

This work presents the initial results and comparisons between a pure best estimate Loss of Flow event, including expected Reactor Regulating System response, and one based on limiting deterministic assumptions and will be used as a basis for determining the integrated uncertainty in reactor response to this scenario.

1.2 Objective

The Canadian CANDU industry is currently pursuing the use of Best Estimate and Uncertainty (BEAU) methodologies to resolve various issues related to Loss of Power Regulation, Loss of Coolant and Loss of Station Power Accidents [8]. The objective of this work is to perform analysis and document the results of two separate LOF analyses using both a pure best estimate and more limiting deterministic assumptions in order to assess the effectiveness of process and safety system actions. This work will provide the initial estimates of plant response which will form the basis of downstream BEAU work.

1.3 Scope

This work will analyse the single-pump trip transients for a typical 900 MW CANDU reactor operating at 100%FP for both beat estimate and deterministic assumptions. Specifically, the following analyses are presented:

- Analysis of a single-pump trip flow rundown from limiting initial conditions along with the anticipated process system response (RRS control and STEPBACK), Shutdown System Response (assuming process action does not initiate) and sheath temperature transients assuming no shutdown system or process action occurs.

- Analysis of similar parameters as above using best estimate methodologies.
- Sensitivity of the best estimate results to key operating variables.

These results provide an interesting comparison on the plant response from deterministic and best estimate conditions.

2. Accident Description and Analysis Methodology

An electrical failure leading to a single heat transfer pump trip is characterized by a reduction in flow due to the decrease in driving pump head and will subsequently reach a new steady-state condition[ii]. Immediately following the trip, the affected pump will continue to spin based on the inertia of the flywheel attached to the pump shaft and hence there will be a period of time between the initial and final heat transport system conditions. During the transient portion (prior to the new steady-state 3-pump operation being reached) the heat transport system pressure may increase due to coolant swell and the core exit temperature may rise prior to a reactor trip or automatic protective process system action. The Reactor Regulating System (RRS) will detect the changes in power and will initiate light water liquid zone filling to compensate for any reactivity insertion caused by void produced in the core. Within the Digital Control Computers, a separate STEPBACK routine may also detect the fault and initiate power reduction via control absorber (CAs) drop. Finally, in the event the fault occurs and both Setback and STEPBACK fail to initiate, Shutdown System (SDS) 1 or 2 will activate via detection of low flow in the SDS instrumented channels or the high transport high pressure (HTHP) trip.

The heat transport system simulated in this analysis is a 900 MW CANDU station utilizing 2 separate figure-of-eight. One-half of the heat transport system is shown in Figure 1 (i.e., on figure-of-eight loop). Each figure-of-eight loop contains two main heat transport system pumps which flow into Reactor Inlet Headers (RIH) and then into fuel channels which are fuelled against the direction of flow. Flow from the core is collected in Reactor Outlet Headers (ROH) and then transferred to separate steam generators to remove the heat and transfer it to the secondary side for electricity generation. Balance headers are included in each figure-of-eight which allow the loop pressures and flows to equalize in the event of abnormal transients. A feed system is used for coolant make-up and feeds into one of the pump suction headers immediately upstream of a heat transport system pump. Finally, a pressurizer is attached to one of the outlet headers and allows both in-flow and out-flow into the heat transport system dependent on the nature of the transient.

[ii] Provided that the Reactor Regulating System (RRS) is capable of maintaining reactor power through the transient. In the event that conditions arise during the fault where reactivity increases are sufficiently high such that power cannot be maintained, additional Setback, STEPBACK and SDS trips on high neutron or zone powers will activate. These events are not considered as part of the present work.

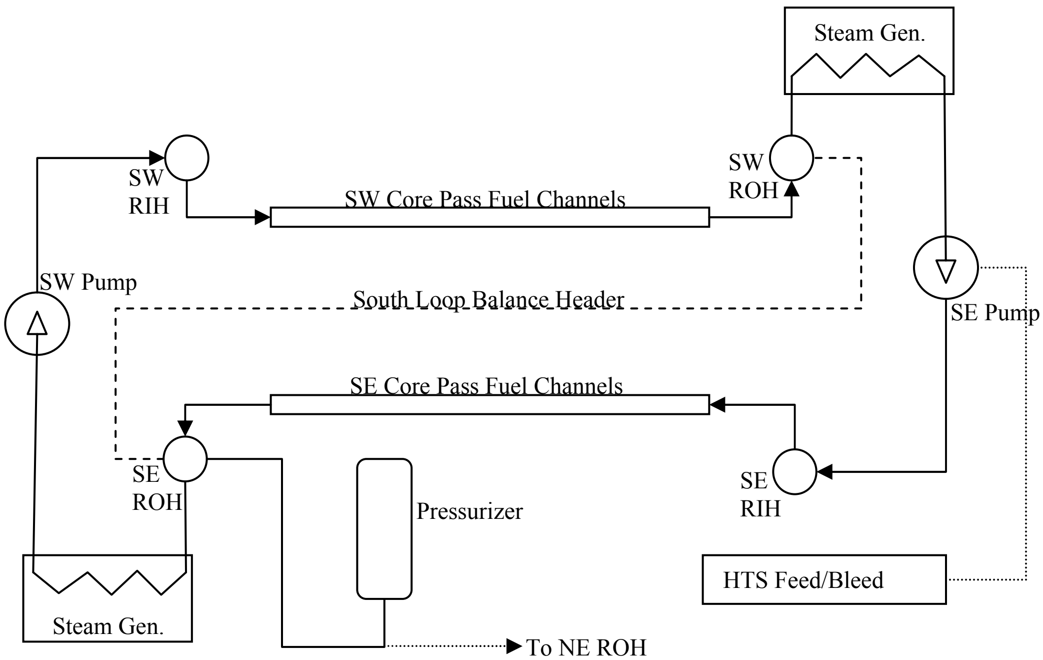

Figure 1: South figure-of-eight loop for a 900 MW CANDU heat transport system used in the Loss of Flow analysis

The Loss of Flow analysis presented in this paper was performed using the SOPHT thermalhydraulic computer code which has been used in CANDU safety and licensing for over 20 years. It is a 1-dimensional two-phase mixture formulation capable of simulating a wide rage of accident scenarios including single pump trip events. For events such as a single pump trip, the flow rates in the core remain high, with limited voiding expected. Due to the relatively low flow quality involved and the high pressures and flows during this postulated event the two-phase mixture model implemented in SOPHT is well suited for this application [iii]. The simulations are performed with 13 bundles in each fuel channel with a heat flux profile as shown in Figure 2 and assuming a South-West pump trip. South-East trips were included as sensitivity studies but were in general less limiting. Since the objective of this work is to compare the limiting and best estimate cases, no specific plant aging is included in the heat transport system as it has been demonstrated that HTS aging is highly station dependent and generic analysis of plant transients is not possible. Therefore the specific intent of this work is to demonstrate the differences between the Best Estimate plant response and the response of the plant starting from limiting operating conditions.

[iii] Activities to develop a 900 MW CATHENA model are ongoing so that results from a two-phase separated flow simulation can be compared to the mixture model results.

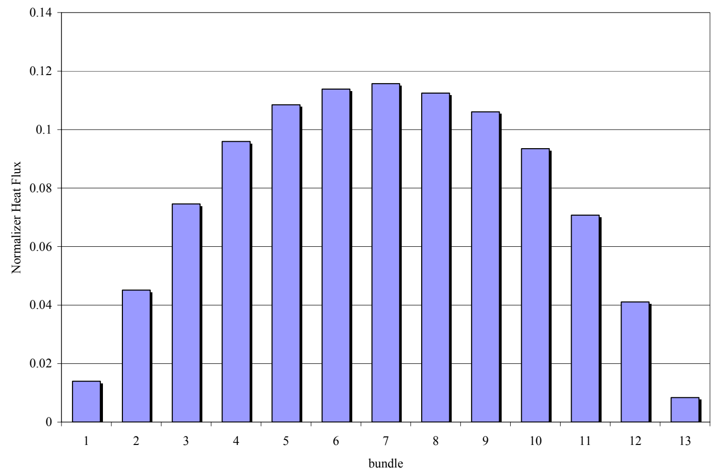

Figure 2: Fuel axial heat flux profile used in single-channel LOF simulations

The operating conditions considered for the Limiting Condition and Best Estimate cases are summarized in Table 1. System simulations are performed using SOPHT for the entire heat transport system, the pressure and inventory control system, liquid pressure relief system, reactor regulating system, and secondary side boiler control systems. The transient pressure, temperature and flow are used, along with the relevant setpoints and instrument timing characteristics, to determine the STEPBACK and SDS1 and SDS2 activation times. Furthermore, the header to header pressure drop, reactor inlet temperature, and outlet pressure transients are then used as boundary conditions for single-channel analysis. Specifically, these boundary conditions are used for simulation of the maximum power channel (e.g. simulation of the hot channel response to assess maximum sheath temperature transients in the core) and shutdown system flow instrumented channels to assess the Heat Transport Low Flow (HTLF) trip. For these simulations the single and two-phase pressure drop, Critical Heat Flux (CHF), and post dryout sheath temperatures are predicted using correlations developed from full-scale single-channel experiments. Modelling uncertainties in these parameters are treated consistently for both approaches so that comparisons on the sensitivity to operating assumptions can be clearly established. In addition to the Limiting and Best Estimate cases, and additional case assessing the sensitivity to CHF models is also included.

As previously discussed, SOPHT uses the heat transport system and single channel results to determine the activation times for the STEPBACK, SDS1 and SDS2 system, and simulates the time response of the trip signals using a combination of first-order time response and pure time delays. The total time response (including delays) is then used to activate the shutdown systems which are modeled in SOPHT using reactivity characteristics as a function of time. The setpoints and instrument loop timing information are provided in Table 2 for each of the cases analyzed.

Table 1: Operating conditions assumed in the LOF accident simulations

Operating Parameter	Limiting Condition	Best Estimate
Initial Core Power	97%[iv]	100%
Hot Channel Power	7100 kW	6800 kW
Heat Transport System Inlet Temperature	270 °C	266 °C
Initial Heat Transport System Pressure	9900 kPa	9950 kPa
Initial Hot Channel Flow[v]	26 kg/s	27 kg/s
Low Flow Trip Channel Power	5570 kW	5800 kW

Table 2: Reactor STEPBACK and shutdown system setpoints and timing information

Safety Parameter	Limiting	Best Estimate
SDS1 and SDS2 HTLF Trip Setpoint	80 %	88%
SDS1 & SDS2 HTLF Time Constant/Time Delay	1300/300	1000/300 ms
SDS1/SDS2 HTHP Trip Setpoint	10.7[vi]	10.4 MPa
SDS1/SDS2 HTHP Time Constant/Time Delay	100/300[iv]	50/300 ms
STEPBACK Pump Trip Time Delay	1.2s	0.6s[vii]
STEPBACK High Outlet Pressure	10.5 MPa	10.2 MPa
STEPBACK High Outlet Pressure Time Delay	1.0s	0.5s[viii]

[iv] The reactor power error due to control and measurement uncertainties is taken in the downward direction in order to delay the high pressure actuation of STEPBACK and shutdown system trip.

[v] The hot channel flows and fuel temperatures are simulated using header-to-header boundary conditions from the system based simulations. The initial hot channel flows are adjusted by modifying the inlet end fitting form loss factor such that the initial flow just prior to the pump trip is at the indicated value in Table 1.

[vi] Under the draft S-310 requirements for high frequency events considering deterministic assumptions, process system action is credited and hence the secondary trip on each shutdown system is not required.

[vii] The best estimate of the STEPBACK routine is based on a clock cycle of 0.25s. If a STEPBACK signal is received partway through a clock cycle it may take to the next clock cycle to activate the STEPBACK. Therefore, a conservative time of two clock cycles is used plus 0.1s for the detection relays to open on the pump trip, i.e., a total fixed delay time of 0.6s. Conservatively a delay of 1.2s is used for the limiting case.

[viii] The best estimate of the STEPBACK routine is based on a clock cycle of 0.25s. If a STEPBACK signal is received partway through a clock cycle it may take to the next clock cycle to activate the STEPBACK. Therefore, a conservative time of two clock cycles is used, i.e., 0.5s for the best estimate case.

4. Results

4.1 Reference best-estimate case results

The best estimate case for a single heat transport system pump trip is characterized by an early STEPBACK on detection of a loss of heat transport system pump electrical power. The STEPBACK signal is registered at 1.4s and initiates Control Absorber insertion which causes a fast reactor power reduction within 2 seconds from the start of the pump rundown. In the event that the pump trip STEPBACK fails to operate, shutdown system action on the heat transport low flow (HTLF) parameter will be initiated in 2 out of 3 logic channels in less than 2 seconds on both shutdown systems. In the event that both the initial STEPBACK and HTLF trips are inoperable the High Heat Transport Pressure SDS1 and SDS2 trips or the STEPBACK on High Heat Transport Pressure will activate within 4.5 seconds. A summary of the STEPBACK and SDS1 and SDS2 initiated signals is provided in Table 3 and the power and outlet header exit quality transients are provided in Figure 3.

Table 3: Summary of best estimate results

Corrective Action	Action Time [s]	Max. Fuel Sheath Temperature [C]
STEPBACK on HTS Pump Trip	1.4	No Dryout[ix]
SDS1 / SDS2 HTLF Trip	1.9	No Dryout
STEPBACK on HTS High Pressure	3.6	No Dryout
SDS1 & SDS2 HTHP Trip	4.3	No Dryout

The core flow transients in the affected core passes are shown in Figure 4 for the cases involving the STEPBACK on loss of pump. The simulations show the flow reductions in the affected core pass are quite fast and drop to 50% levels within approximately 4 seconds and then recover as the balance header redistributes flow. For the unaffected-pass the flow is initially unaffected as due to the operable pump until void is collapsed in the outlet header which subsequently causes significant pressurization. After void collapse and pressurization, flow reductions are then predicted in this pass due to redistribution to the affected pass through the balance headers. The Reactor Outlet Header pressure transients are shown in Figure 5 and clearly show the point of void collapse and subsequent pressurization in the unaffected pass.

[ix] Simulations were performed assuming that STEPBACK, SDS1 and SDS2 fail to activate, for these simulations dryout was predicted to occur at approximately 6s. For all best estimate cases the analyzed trips and STEPBACKs activated before dryout.

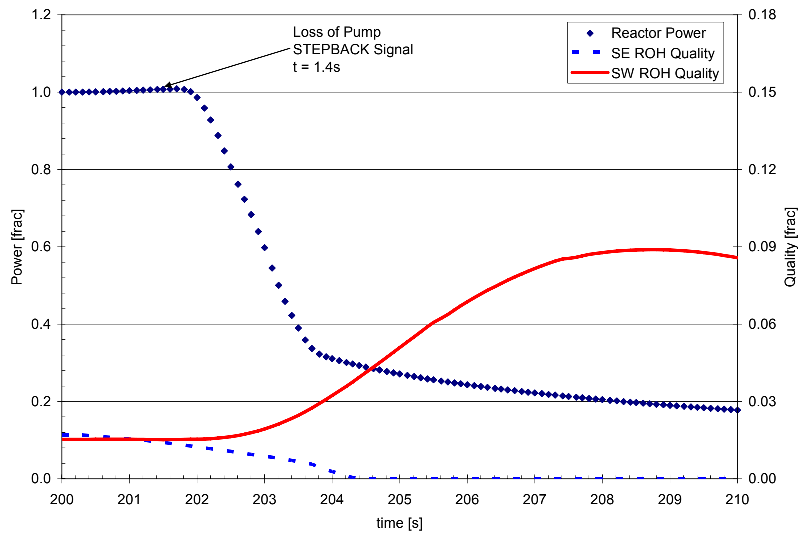

Figure 3: Power transients for the STEPBACK (initiated on high heat transport system pressure) and SDS2 HTHP trips.

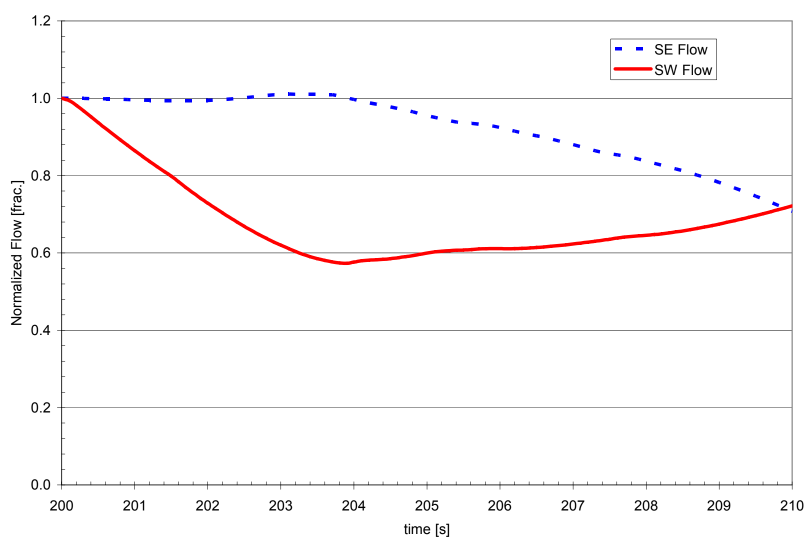

Figure 4: Sample flow transient for a South-West heat transport system pump trip

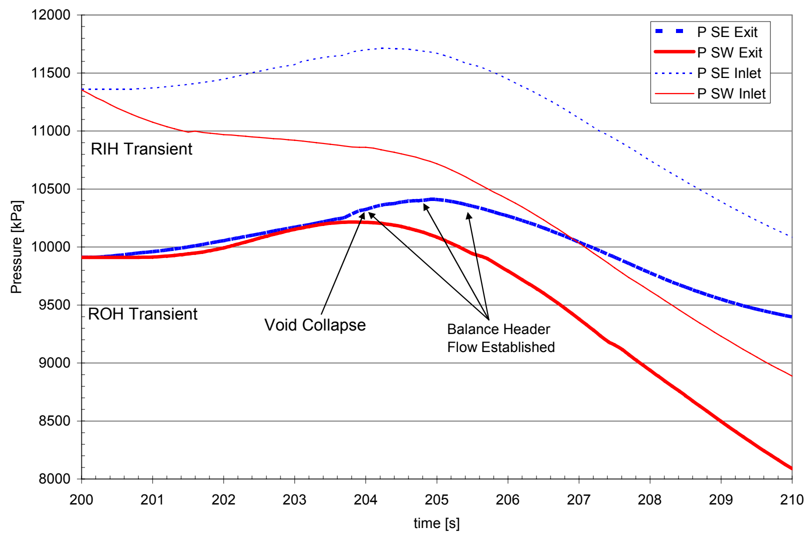

Figure 5: Reactor outlet header pressure transients for a South-West heat transport system pump trip

For each of the cases discussed in Table 3, single channel simulations were performed to determine the maximum fuel sheath temperature during the transients. For all best estimate cases fuel sheath dryout was precluded and hence fuel temperatures remain low.

4.2 Limiting condition results

Detailed simulations were performed for a South-West pump trip using the limiting conditions defined in Table 1, assuming a lower bound reactor power to delay the onset of the high pressure STEPBACK and SDS trip, and the results are summarized in Table 4. After the pump trip, a STEPBACK will be initiated at approximately 2.1 seconds causing a rapid decrease in reactor power and avoiding fuel sheath dryout. If the STEPBACK on pump trip is unavailable then SDS1 or SDS2 trips on low flow will occur on 3-out-of-3 logic channels well before fuel sheath dryout. In the event that the pump trip STEPBACK and SDS1 and SDS2 HTLF signals fail to actuate, then dryout will occur at approximately 3.6 seconds.

Table 4: Summary of results for an LOF using limiting conditions

Corrective Action	Action Time [s]	Max. Fuel Sheath Temperature
STEPBACK on HTS Pump Trip	2.1	No Dryout
SDS1 / SDS2 HTLF Trip	2.4	No Dryout
STEPBACK on HTS High Pressure	Not Reached	
SDS1 & SDS2 HTHP Trip	Not Reached	
Heat Transport System Liquid Relief	Not Reached	

Figure 6 shows the reactor power transients for the limiting operating conditions when the reactor STEPBACK on pump trip activates to reduce power. Furthermore, Figure 6 also shows that the void remains in both headers throughout the transient. Figure 7 shows the core pass flow transients are similar to those shown in Figure 4, however due to the lower initial flow and power in the flow instrumented channel power and the lower setpoint, the trip is slightly delayed relative to the best estimate case. The pressure transients are shown in Figure 8 and demonstrate that due to the milder pressurization transient and higher activation setpoints, both the STEPBACK and SDS1/SDS2 high pressure setpoints are not reached. In order to establish the dryout time for these limiting conditions, single channel simulations were performed simultaneously assuming the limiting channel power, inlet temperature and initial flows shown in Table 1. Furthermore, the Critical Heat Flux was reduced by a factor of 0.85 to include uncertainties and possible degradation due to aging. Even under these low probability conditions, activation times, and CHF penalties the STEPBACK on loss of pump and SDS1 and SDS2 heat transport low flow trip activated prior to dryout.

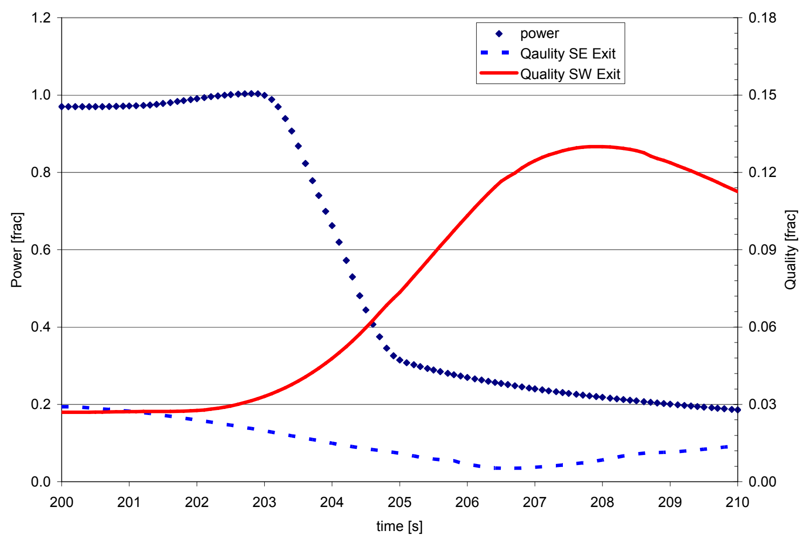

Figure 6: Reactor power transients for the limiting operating conditions

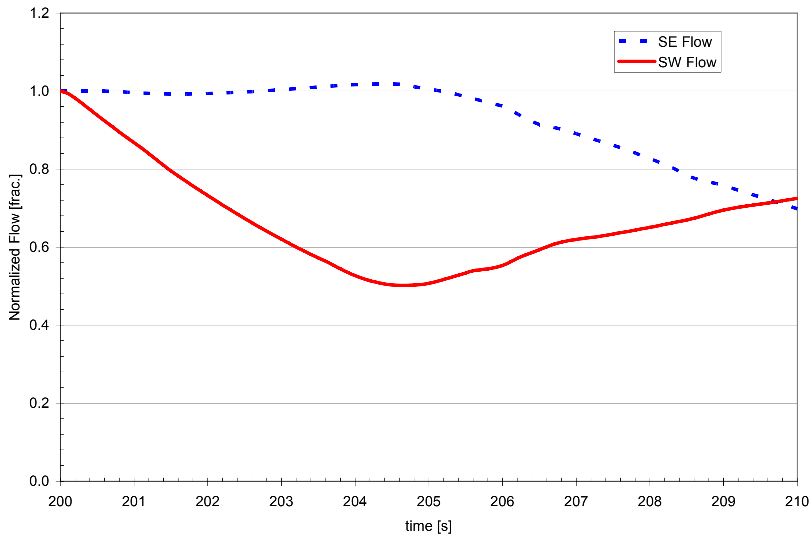

Figure 7: Core pass flow transients from limiting initial conditions [credited STEPBACK on pump trip]

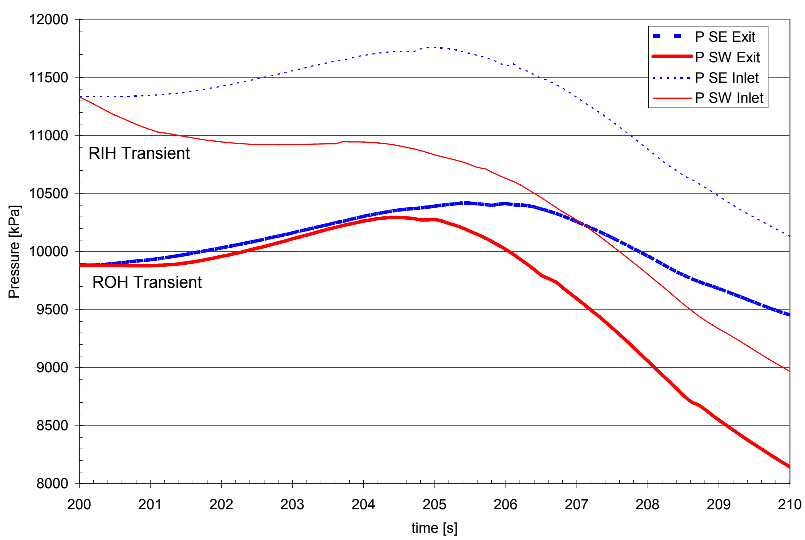

Figure 8: Reactor outlet header pressure transients from limiting conditions [credited STEPBACK on pump trip]

6. Conclusions

This paper has presented to the results for a single heat transport system pump trip event using best estimate and deterministic analysis of plant operating assumptions. The best-estimate cases demonstrate that there is significant margin between fuel heat-up and power reductions based on

STEPBACK on pump trip, STEPBACK on high pressure, as well as HTLF and HTHP trips on both SDS1 and SDS2. Analysis using more limiting deterministic assumptions for the initial plant operating state has shown that the STEPBACK parameter on heat transport pump trip is sufficient to prevent dryout and hence will prevent fuel overheating. For single-pump trip transients from these limiting initial conditions, the Heat Transport Low Flow trips on SDS1 and SDS2 are also effective at preventing dryout. Irrespective of the analysis methodology selected for the event, both the best estimate results and the more limiting operating assumptions demonstrate reactor shutdown on at least one parameter from both SDS and STEPBACK prior to fuel heat-up. Ongoing research is being directed to:

- assess the impact of plant aging on these conclusions.
- quantify the simulation and operational uncertainty in these results and establish the probability density function for the sheath temperature during these events.
- assess the impacts of additional trip parameters which may be effective.

7. References

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