Reactor Kinetics Study in Candu-6: A 3-D Versus Point Kinetics Comparison for Loss of HTS Flow Events

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Abstract

This paper describes a series of coupled neutronic-thermalhdyraulic simulations of postulated loss of primary circuit flow events occurring in CANDU 6 Nuclear Power Generating Station. The full-core time-dependent neutron kinetic calculation was performed using the *CERBRRS module in physics code, RFSP-IST. *CERBRRS solves the time-dependent neutron diffusion equation with delayed neutrons using the Improved Quasistatic Methodology. In addition, it incorporates Reactor Regulating System (RRS) capabilities, which makes it suitable for analysis of slow transients. Dynamic coupling of the thermalhydraulic code CATHENA version Mod3.5dRev.1 with RFSP was automated using RCCTS, a PERL-based script. For these coupled simulations, a full 2-energy-group WIMS-IST Simple Cell Model (SCM) approach was used with 2-group device incrementals from DRAGON.

A "point reactor" model was also used in simulation, with incorporation of reactivity coefficients and delayed-neutron data from the RFSP model. A "point reactor" version of the RRS controller, used in *CERBRRS, was incorporated in the "point reactor" model. A three dimensional (3-D) versus "point neutronics" comparison is thus established as it pertains to loss of heat transport system (HTS) flow.

1. Introduction

1.1 Background

Analysis at Canadian nuclear power stations for loss of heat transport system (HTS) flow and the corresponding trip coverage has been traditionally performed using a "point reactor" model or "point kinetics" subroutine contained in a thermalhydraulic computer code. More specifically, this methodology has been used for licensing related to trip coverage relevant to loss of HTS flow and improvements of the shutdown system process trip parameter setpoints [1**],** [2**]**.

The "point reactor" model provides physics feedback to thermalhydraulics via a bulk power or core-wide reactivity. In the same light, averaged, core-wide thermalhydraulic conditions are used to solve for the amplitude of the neutron flux while changes in the flux shape are ignored. A coupled analysis provides a more rigorous approach by allowing the physics and thermalhydraulics codes to exchange distributed 3-D properties. Moreover, the physics code would involve integration of the flux amplitude and shape as a function of time. The common example in nuclear safety analysis has been to couple the physics code, RFSP **[3]**, with the thermalhydraulics code, CATHENA [4]. The latter is the standard computer code used at Candu Energy Inc. to analyze two-phase flow dynamics and heat transfer during postulated accident scenarios. Within RFSP, neutron kinetics modules such as *CERBERUS **[5]** have been used extensively for coupled analysis **[6], [7], [8]**.

The *CERBERUS module has been the go-to physics tool for power pulse analysis for large-break Loss of Coolant Accident (LOCA). The latter is characterized by rapid core voiding and an overpower transient which is beyond the Reactor Regulating System (RRS) capability. For this reason, *CERBERUS was never equipped with an RRS controller for its Large LOCA applications. In comparison, the Reactor Regulating System is designed to control reactor power during events where void generation happens on a longer time scale. The result is that the reactor might trip first on a thermalhydraulic-related parameter such as high or low primary circuit pressure. A loss of primary circuit flow (due to pump trip) and small-break LOCA fall under this category of event.

In recent years, the nuclear safety industry has made strides to address 3-D neutronic effects on the analysis of slow transients and their potential impact on trip coverage. As such, the industry has begun to adopt a coupled analysis approach for slow transients. A main strategy has been to perform coupled analysis to support the main results obtained using point kinetics.

Coupled analysis for slow transients requires a detailed RRS response within a coupled RFSP/CATHENA framework. The addition of the RRS function to the aforementioned fast transient neutron kinetics module (*CERBERUS) has expanded RFSP-IST's capability to analyze slow transients, as will be demonstrated here. *CERBRRS is the name of the neutron kinetics module in RFSP with time-dependent RRS capability. Its validation is presented in [10].

1.2 Objective

This paper describes a series of coupled neutronic-thermalhydraulic simulations of postulated loss-of-HTS flow events occurring in a CANDU® nuclear generating station. The coupling between a thermalhydraulics computer code such as CATHENA **[4]** and an external physics code such as RFSP **[3]** provides a rigorous treatment for simulating loss of HTS flow. A comparison of the coupled simulations with the point kinetics approach will provide further confidence in the approach.

A generic CANDU 6 model, including an overall system controller, was used for the purposes of simulating the set of loss of flow transients. An RRS controller was independently added to the thermalhydraulics model. This controller is based on the *CERBRRS RRS controller's equations for the liquid zone system, and thereby provides the best possible point kinetics to 3-D comparison. It should be noted that the *CERBRRS module, and most particularly the liquid-zone portion of the RRS, has been validated **[10]**. As such, implementation of the *CERBRRS controller's equations for the liquid zone in the point kinetics model is a logical approach.

2. Numerical Models and Methodologies

2.1 Software Relationships

Figure 1 shows some of the key dynamic software relationships in the analysis. It focuses on the interaction in transient between CATHENA and the *CERBRRS module of RFSP.

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The output of each RFSP calculation includes a TAPE15 file, consisting of powers (in Watts) for each axial node (bundle location) for each thermalhydraulic channel group, which is used by CATHENA. This TAPE15 file is then converted into bundle power fractions for each channel group, which are then inserted into the CATHENA input file by coupling script. The CATHENA simulation, in turn, produces several "coreden.out" files, which contain 12 coolant densities, coolant temperatures and fuel temperatures at the beginning time and end time of the simulation interval for each thermalhydraulic channel group. Those files are merged into one TAPE2 file, in a format acceptable by RFSP, which is subsequently used its next calculation.

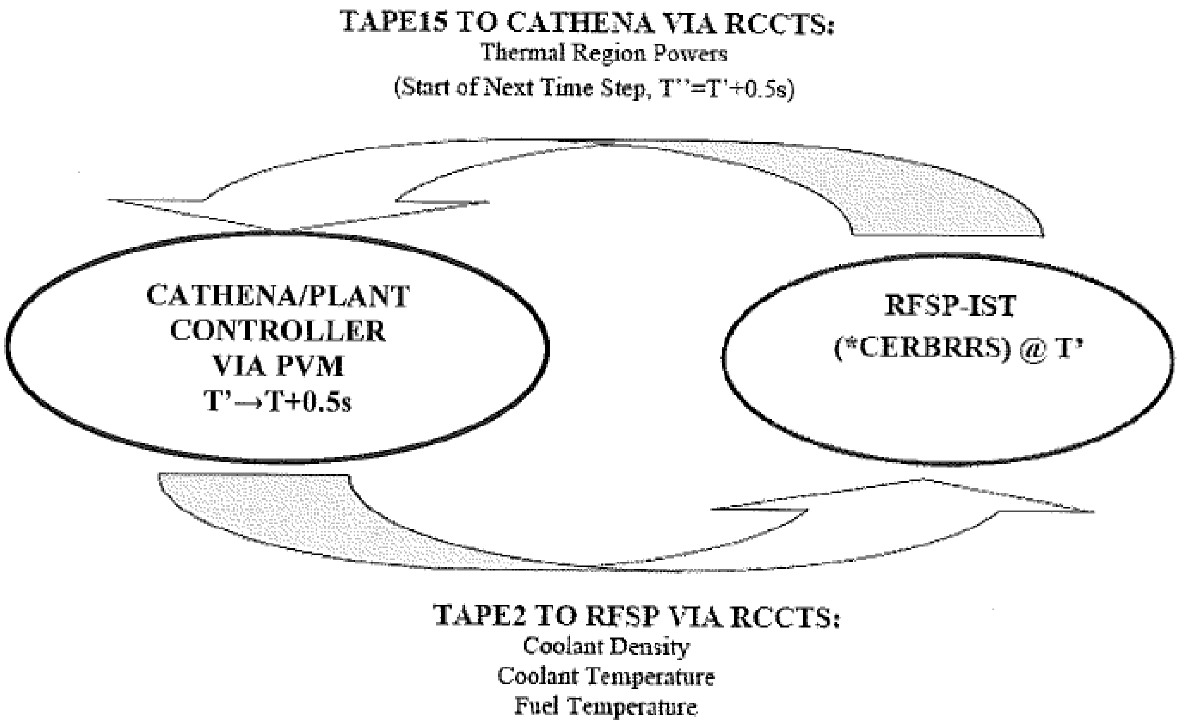

Figure 1 Software Relationship (CATHENA – RFSP)

The flow of data between the thermalhydraulic code and RFSP can be automated via a computer script or utility program written in the PERL computer language [11] or can be exchanged via the Parallel Virtual Machine (PVM) software package [9]. In steady state application (or *CERBBRS case 1), the time is not advanced. RFSP simply is performed in static mode and iterations between CATHENA and RFSP are performed until the desired power shape and thermalhydraulics are converged.

2.2 Computer Codes and Models

2.2.1 Full-Core Physics

The 3-D neutron dynamic calculations of core response and reactivity for the loss of HTS flow cases were performed with the neutron kinetics module, *CERBRRS, in RFSP-IST version 3-04-04PC **[3], [9]**, in conjunction with the WIMS- IST-based Simple-Cell Method (SCM) **[13]**. The RFSP models for this analysis are the same as those used in the loss of flow analysis of **[4]**. Table 1 summarizes the physics-related parameters used in this analysis.

The starting point for the equilibrium fuel model (Direct Access File or DAF) was a time-average model with uncrept pressure tubes. WIMS-IST (WIMS-AECL version 2-5d (**[14])** along with the nuclear data library ENDF/B-VI version 1a, is used to generate the SCM fuel and reflector tables of this analysis [13**]**. This methodology was successfully tested and compared against running WIMS-AECL for every bundle in a sample LLOCA analysis for a baby CANDU case in [7**]**. The three-dimensional neutron- transport code DRAGON-IST [15**]** was used to calculate the incremental nuclear cross sections of the shutoff rods of Shutdown System One (SDS1) and other devices (such as liquid zone compartments) and structural materials.

Given the aforementioned time average model, the next step was to compute the moderator boron corresponding to the equilibrium core snapshot. The amount of moderator boron is an important parameter for coolant voiding transients. This is because when coolant voids the flux shape in the cell changes: the flux increases in the fuel cluster and decreases in the moderator. This effect gets amplified if more boron (a strong neutron absorber) is present in the moderator because it will result in less absorption of neutrons in the moderator as the flux decreases there. For similar reasons as that for maximizing the moderator boron, a low D_2O isotopic for the coolant and moderator result in increased positive reactivity on primary circuit void generation. The values in Table 1 of 99.72 a% and 98.40 a% for the moderator and coolant D_2O isotopic are typical operational low limits for CANDU 6.

A moderator boron concentration of 3.15 ppm was computed for the equilibrium fuelled core at 100% FP. Given this value of moderator boron concentration and the core states at full power, the reference detector data was read in to the RFSP model for equilibrium fuelled core to be used to determine trip times. The resulting RFSP models are the starting point for *CERBRRS analysis, used in *CERBRRS case 1. In this stage of the analysis, the remaining parameters of Table 1 (i.e., loop-loop tilt, reactor power, neutron velocities, thermalhydraulic conditions) are incorporated into the model. The loop-loop tilt is defined in terms of a percent of the steady state power or present power:

$$TILT = \frac{P_{LOOP\,1} - P_{LOOP\,2}}{P_{LOOP\,1} + P_{LOOP\,2}} * 100\text{, [units, \% PP].}$$

A 4% loop-loop tilt was incorporated in Core_State 2. The loop-loop tilts were modelled via adjustment of the LZC System level configuration and biased such that Loop 1 has a higher power than Loop 2. The final two steps in the steady state are modelling of a particular coolant void reactivity (CVR) and computation of the reference zone fluxes (PHINOMS), used in spatial flux control.

2.2.2 Full-Circuit Thermalhydraulics

The conservation equations for mass, momentum and energy related to motion of a two-phase fluid are performed via the thermalhydraulic code CATHENA **[4]**. A generic CANDU 6 CATHENA model was used for the purpose of this analysis. The model also incorporates an overall plant controller. An independent RRS controller was added to mimic *CERBRRS equations of the liquid zone controller and mechanical absorbers. This is discussed in Section 2.5.1.

CATHENA is a two-fluid thermalhydraulic computer code, developed by AECL for analysis of flow transients in reactors and piping networks. CATHENA uses a one-dimensional, non-equilibrium, two-fluid thermalhydraulic representation of two-phase flow. Conservation equations for mass, momentum and energy, together with flow-regime dependent constitutive relationships describing the interfacial transfer of mass, momentum and energy are solved for each phase.

The CATHENA code includes component models such as pumps, valves, pipes, generalized tank models and point-kinetics models, and has extensive control modelling capability. It also contains a GENeralized Heat Transfer Package (GENHTP) which allows the modelling of solid components such as pipes and fuel elements, heat generation within these components and heat transfer between these components and the surrounding fluid.

The CATHENA computer model is based on the CANDU 6 Heat Transport System design and depicted in Figure 2; each core pass is represented by a multiple average channel approach, which permits a more realistic prediction of void generation following a loss of flow event. Each channel group is modelled with twelve axial nodes corresponding to 12 bundle locations. The model also contains detailed representations of other related systems including the Pressure and Inventory Control System and the Steam and Feedwater system. A plant controller is linked to the main portion of CATHENA via PVM [12]. The controller simulates the different control systems such as Boiler Level and Pressure Control.

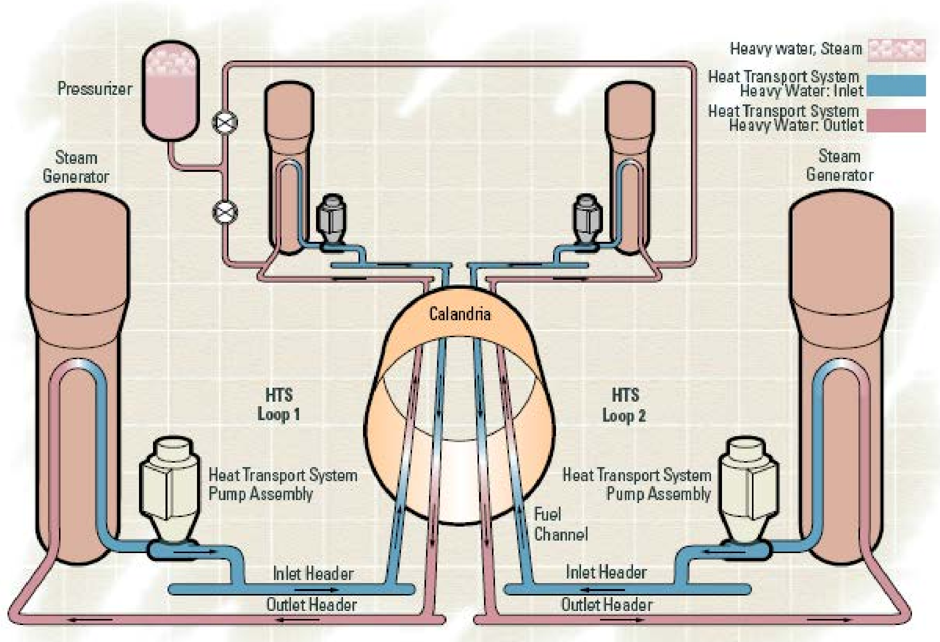

Figure 2 Two-Loop Reactor Coolant System in CANDU 6

Best-estimate heat transfer and two-phase parameters were applied to quantities such as nucleate boiling and critical heat flux (CHF). The only exception is the use of the Leung Onset of Significant Void (OSV) model for sub-cooled boiling (option 9 of the wall-interface-heat transfer model). The CHF model used is the default Groeneveld-Leung 37-element-bundle CHF tables without correction. The Boiling Length Average CHF model is included in the circuit model.

Important operational thermalhydraulic parameters include the Reactor Inlet Header (RIH) temperature and Reactor Outlet Header (ROH) pressure. The RIH temperature is an important operational condition because an initially high RIH temperature is favourable to an early onset of intermittent dryout. For this analysis, an initial RIH temperature of 266.7ºC was modeled. The initial ROH pressure is also an important operational parameter from a fuel sheath dryout point of view. An initially low ROH pressure increases the void fraction in the HTS, which favours the early onset of dryout. It also provides a larger margin between the initial ROH pressure and the HP trip setpoint. An initial ROH pressure of 9.93 MPa(a) is assumed for this analysis.

2.2.3 Single Channel Analysis

The onset of fuel sheath dryout - breakdown of nucleate boiling due to the local heat flux exceeding the critical heat flux – was estimated using a CATHENA model of a high power channel O6. Similar to the full-circuit model, best-estimate heat transfer and two-phase parameters were applied to quantities such as nucleate boiling and critical heat flux (CHF). The fuel model is based on a 37-ROD-1 representation with associated HOTPIN. A developing post-dryout (PDO) model is incorporated to predict fuel sheath temperatures after dryout has occurred. A best-estimate correlation is implemented for the average pin, whereas a minimum PDO model is implemented for the HOTPIN. The predicted hydraulic conditions in core pass 1 (header 4 to header 1) were applied to the single-channel in conjunction with the bundle power transient from a channel with similar geometry to O6 but located in core pass 1. Channel O16 was used for this purpose.

2.3 Neutron Kinetics

Physics-related parameters important for time-dependent neutronics are the neutron velocities and delayed-neutron fraction, and reactivity coefficients. Although these parameters are specified on the N and V card, respectively, of the RFSP input for *CERBRRS case 1, the actual values are only used in solution of the time-dependent diffusion equations (*CERBRRS case 3 and higher). Six direct delayed-neutron groups, corresponding to Blachot [16**]** and 11 photoneutron groups corresponding to the Baumann set [19**]** were used for the transient RFSP-IST calculation. The *CERBRRS calculation in RFSP-IST derives bundle specific values for the delayed neutron fraction and prompt neutron lifetime.

An exchange interval (between CATHENA and RFSP) of 100 ms is performed during these transients. Within CATHENA a maximum time step of 10 ms is specified. The solution of the time-dependent neutron diffusion equations in RFSP is based on factorisation of the flux into an amplitude and shape contribution (the flux shape). This factorisation together with a constraint equation on the flux shape ([17**],** [18**]**) yields the following equation for the amplitude:

$$\dot{A}(t) = \left(\frac{\rho(t) - \beta_{eff}(t)}{\Lambda(t)} \right) A(t) + \sum_{g=1}^{G} \lambda_g c_g(t) \tag{1}$$

The above equation for the amplitude is in the general "point kinetics" form. It is coupled to an equation for the six delayed neutron precursor groups and another for the flux shape. The flux shape equation is integrated on macro-intervals which coincide with the CATHENA-RFSP exchange interval (i.e., 100 ms). The amplitude equation is integrated on finer micro-intervals. Solution of the spatial flux on a macro-interval with integration of the amplitude on finer micro-intervals is the basis of the Improved Quasistatic Method ([17], [18]) and encoded in *CERBRRS and *CERBERUS.

The "point kinetic" equation for the amplitude (1) is defined in terms of four basic parameters: (1) a dynamic reactivity, $\rho(t)$ (2) a mean neutron generation time, $\Lambda(t)$ (3) an effective delayed neutron fraction, $\beta_{eff}(t)$ and (4) delayed precursor decay constants, λ_g. To obtain the best comparison between the 3-D model and point kinetics, one would use the *CERBRRS computed values for $\Lambda(t)$ and $\beta_{eff}(t)$ at time 0 as input to the point kinetics equations as is discussed in Section 3.2. Moreover the dynamic reactivity $\rho(t)$ is function of coolant density, coolant temperature and fuel temperature. These reactivity coefficients are computed by RFSP at the analysis conditions prior to the point-kinetics calculations and input in CATHENA point kinetics model.

2.4 RRS Modelling

RRS modelling for this analysis pertains to the LZC System and Mechanical Control Absorbers (MCAs). The RRS response is under *CERBRRS control for the 3-D coupled simulations and would generally be driven by the general plant controller for the point kinetics cases. Stepback and Setback are not credited in both 3-D and point kinetics analysis. A description of the LZC system is given here and how it is implemented in the point kinetics cases.

2.4.1 Modelling of the LZC System

The initial average normalized level of the LZC System compartments is assumed to be 0.5. For the 3-D cases where no loop-loop tilt is modelled, the initial level in all compartments is modelled as 0.5. However, adjustments to these values are required for the core state where a 4% tilt is modelled.

For point kinetics cases, an independent LZC controller was added to the CATHENA model, separate from the general plant controller. The equations of this controller in terms of the bulk power error, demand lift of the zone's valve, actual (or filtered lift, "ALIF") and speed of filling follow those given in reference [10]. For example, the speed of filling in "%Filling/second", in terms of the filtered lift, is given by:

$$V = \frac{1}{T_{fill}} \frac{ALIF - BIAS}{1 - BIAS} \text{ [\%Filling/s]} \qquad (2)$$

In point kinetics, the value of T_{fill} is taken as 0.48. This is based on an average of 0.36 for T_{fill} of zones 5 and 12 in *CERBRRS and 0.60 for the remaining zones; these values are hard-coded in the *CERBRRS module. Also consistent with [10] and *CERBRRS logic, the filter - demand lift to "ALIF", which is used to drive the valve - is based on a first-order differential equation with a 100 ms time constant. It should be noted that the calculation of "ALIF" and speed of filling are characteristics of the LZC valve whereas the demand lift (as a function of the reactor power error) is part of RRS program specification.

The addition of the separate LZC controller, which mimics the equations in the *CERBRRS module, results in a better 3-D/point kinetics comparison. In CATHENA, CALCULATE models were implemented for ALIF as a function of bulk power and LZC speed filling from Eq. (2). The overall LZC impact on reactor reactivity is used directly in CATHENA. It should be noted that the *CERBRRS module, and most particularly the liquid zone portion of the RRS with associated valve characteristics, has been validated **[10]**. As such, implementation of the *CERBRRS controller's equations for the liquid zone in the point kinetics model is a logical approach.

2.4.2 MCA Modelling

To simplify this analysis and isolate the impact of RRS via the LZC, the MCAs are not credited to enter the core. This is because in the time frame of interest (0 – 6 seconds) for single pump trip and 0-3 seconds for Loss Class IV Power, the MCAs do not have sufficient time to enter the core under RRS control. While loss of flow transients beyond this time frame would require a careful treatment of the MCAs and any top-bottom tilt which may result, the MCAs are assumed disabled in the transient cases of this analysis. The objective is to isolate the effectiveness of the LZC System alone in maintaining the reactor power.

3. Results

3.1 Analyzed Conditions

The main plant conditions analyzed pertain to those in existence at the beginning of its life (BOL) and when the fuel has reached its equilibrium concentration. The initial core states considered in this analysis are presented in Table 1. Two core states were constructed using the 3-D physics methodology, one with point kinetics. The core states are as follows:

- Two core states (Core_State_1 and Core_State_2) with the fuel in an equilibrium core configuration, reactor power at 100% FP and poisoned moderator. No tilt is modelled in Core_State_1, whereas a loop-loop tilt of 4% is modelled in Core_State_2.

- One core state (Core_State_1PK) with the fuel at equilibrium, reactor power at 100% FP and poisoned moderator. This is the point kinetics equivalent of Core_State_1.

Thermalhydraulic-related assumptions corresponding to these initial core states are presented in Table 2. Two postulated scenarios are considered for this purpose. They are: (1) loss of a single HTS pump (2) total loss of Class IV Power.

Table 1 Physics-Related Assumptions for the Initial Core States

Initial Core State Description	Core_State_1	Core_State_2
Fuel Configuration	EQ	EQ
Initial Reactor Power	100	100
% Loop-Loop Tilt	0	4
Moderator Temp (°C)	68	68
Initial Moderator Poison Load (PPM)	3.15	3.15
Coolant D_2O Purity (a %)	98.40	98.40
Moderator D_2O Purity (a %)	99.72	99.72
Zone Level	50% Fill (Average of 14 Zones)	50% Fill (Average of 14 Zones)

CVR Bias	Nominal (WIMS-1.6 mk)	Nominal (WIMS-1.6 mk)
Delayed Neutron Setting	Nominal	Nominal

Table 2
Important Thermalhydraulic-Related Assumptions (applies to all core states)

Parameter	Value
Reactor Inlet Header Temperature	266.7°C
P&IC Pressure Setpoint	9.93 MPa(a)
ASDVs, CSDVs	Operating normally
Feedwater Temperature at 100% FP	187°C
Gap conductance	35 kW/m^2°C
HTS Pump Model	USER-P
CHF Setting	Nominal
CVR bias (for reactivity tables for point kinetics and in coupled analysis)	Nominal(WIMS-1.6 mk)
Delayed Neutron Data (coupled runs)	6 groups

Loss of power to a single pump in CANDU-6 can be initiated by a loss of power to the heat-transport pump. Loss of power to a HTS pump results in coastdown of the pump's rotational speed according to the characteristics of the pump rundown model in the CATHENA model. Such a coastdown results in a fairly rapid rundown of coolant flow in the channels downstream of the tripped pump. This has the potential to lead to transiently high heat-transport pressure and degraded fuel-cooling conditions. The event is asymmetric in that the affected pump is located in one loop of the core. As such, the neutronic response would be most pronounced in that half of the core. Moreover, since it is typically the outlet header upstream of the tripped pump where the primary circuit pressure increases most rapidly, HTS pressurization is minimized by tripping the HTS pump directly downstream of the pressurizer.

The pump-trip accident scenario is simulated for the two equilibrium fuelled core states with poisoned moderator (Core_State_1 and Core_State_2) to ascertain the impact of a pre-existing loop-loop flux tilt on the primary circuit response. The pump-trip accident scenario is also simulated for Core_State_1PK to ascertain the adequacy of point kinetics to simulate the single-pump-trip event.

A total loss of Class IV power in CANDU 6 can be initiated due to the failure of the main transformer along with subsequent failure of the stand-by transformer. As such, power to all of the main Class IV buses is lost. The result is a loss of power to all four main HTS pumps, in addition to other key plant systems as summarized in Table 3. It is a symmetric event which complements the core behaviour of the single pump trip event. The void generation and subsequent overpower transient is generally much faster than that for the loss of a single pump. In the simulation, the loss of LZC pump is not considered.

Table 3 Affected Equipment due to Total Loss of Class IV Power

HTS Pump	All HTS pumps are lost
Boiler Feedwater Pump	All feedwater pumps are lost
Condenser	Condenser Cooling Water (CCW) Pump and Condenser Extraction (CEP) Pump lost resulting in loss of Condenser

Condenser Steam Discharge Valves (CSDVs)	Opening is limited due to loss of CCW and CEP pumps
Pressurizer Heaters	All pressurizer heaters are lost
LZC Pump	Generally lost but this effect is not credited so that LZC remains available to act

3.2 Steady State Core

Thermalhydraulic conditions for the two initial core states following *CERBRRS case 1 are presented in Table 2. The initial power for all core states is 100%. A 4% loop-loop tilt was incorporated in Core_State 2. The loop-loop tilts were modelled via adjustment of the LZC System level configuration and biased such that Loop 1 has a higher power than Loop 2, as shown on Figure 3.

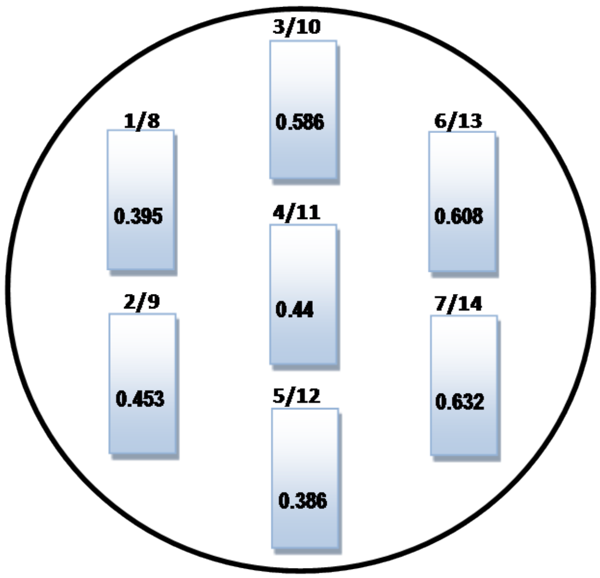

Figure 3 Zone Controller Fractional Fills to Produce the 4% Flux Tilt (Credits: E-L Pelletier)

Point kinetics related parameters are provided in Table 4. These values were taken from the steady state results of Core_State_1, which corresponds to the untilted core at equilibrium. The reactivity coefficients for coolant density, coolant temperature and fuel temperature are computed as well from the steady state results of Core_State_1.

Table 4
Dynamic Parameters Used in "Point Kinetics" Analysis

Delayed Neutron Precursor Group	Dynamic Parameter (β and Λ)	Dynamic Parameter λ (seconds)
1	0.0001993	0.000600
2	0.0010109	0.031552
3	0.0008421	0.119574

4	0.0017875	0.313276
5	0.0009335	0.938468
6	0.0004168	2.931496
β_{eff}	0.000519	
Λ (seconds)	0.00815	

3.3 Pump-2 Trip Results

Loss of HTS flow is a pressurisation event during which the pressure in the affected core pass of the primary circuit is rapidly increasing, whereas the flow and core differential pressure is decreasing. Main HTS Pump-2 is tripped for all single pump trip events. As such, the greatest pressurisation will be observed in the upstream outlet header, that being ROH-3. The transient cases were simulated as un-terminated, meaning that a reactor trip was not simulated on any process trip or neutronic trip parameter. The unterminated transient was carried out until 6.0 seconds. Figure 4 shows the ROH-1 (Reactor Outlet Header) and ROH-3 pressure transients for the Pump-2 trip of Core_States_1 and Core_State_2. Included in Figure 4 is the point kinetics result (Core_State_1PK). The ROH pressure trends are similar for each of the different transients. However, the ROH-1 and ROH-3 pressures increase progressively from the point kinetics case to the 3-D case with untilted core to the 3-D case with 4% tilt.

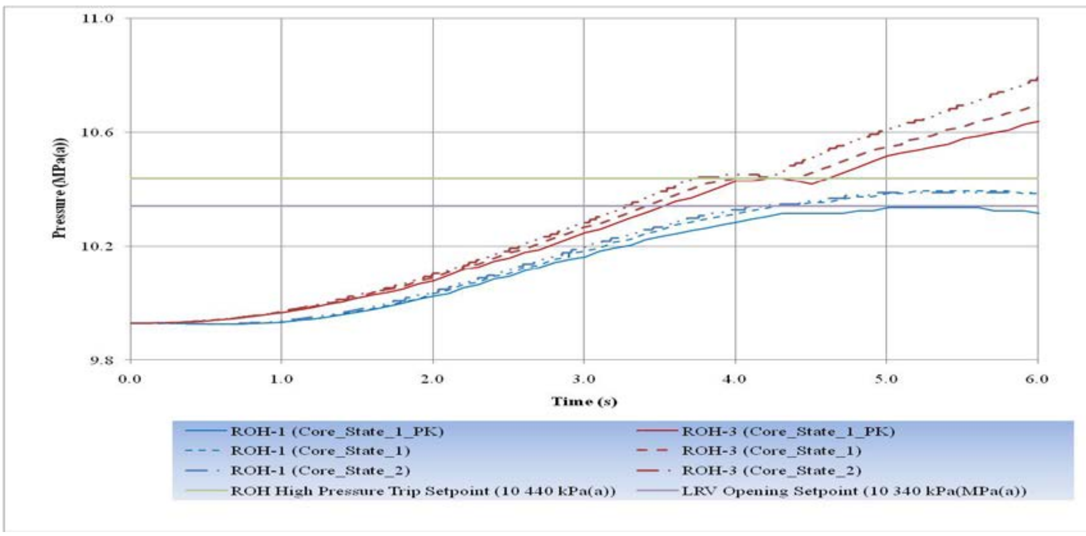

Figure 4 ROH Pressure Transient for Pump-2 Trip

The core differential pressure transient of the core passes in the affected loop is presented in Figure 5. Again, similar to the ROH pressures in Figure 4, the trend is similar for each case. This figure shows that a reactor trip on low coolant flow or low core differential pressure would occur at approximately the same time regardless of the initial loop-loop tilt or the methodology used to perform the transient analysis (i.e., 3-D versus point kinetics).

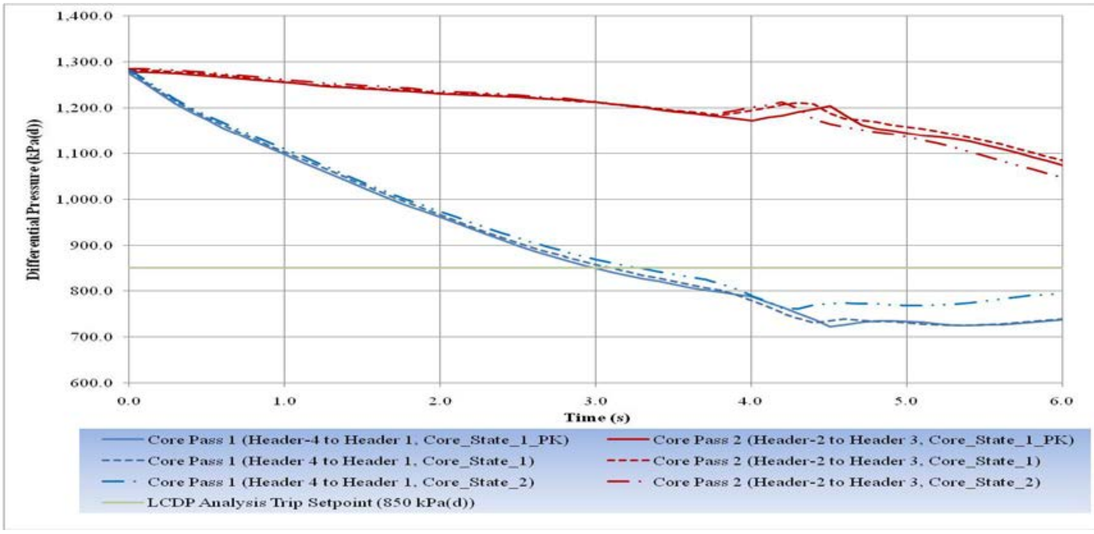

Figure 5 Core Differential Pressure Transient for Pump-2 Trip

Figure 4 and Figure 5 show that a reactor trip on high outlet header pressure, low core pass flow or low core differential pressure would occur at approximately the same time regardless of the initial loop-loop tilt. It also shows that 3-D neutronic affects, which are not captured in the one case generated using point kinetics, do not have a large impact on the transient thermalhydraulics. In other words, 3-D neutronic affect do not appear to play a very large role in the transient thermalhydraulics, which are used at site to actuate the shutdown systems on process parameters such as high outlet header pressure.

The liquid zone level transient (average normalized fill) is presented in Figure 6. Core-wide reactivity and reactor-power transients are presented in Figure 7 and Figure 8, respectively. These figures demonstrate good agreement between the un-tilted core state and the one point kinetics simulation. The incorporation of a loop-loop tilt in the pre-existing core state, as in Core_State_2, results in greater voiding (particularly in the affected loop) and feedback to the reactor core physics. The result is a significantly greater power transient for Core_State_2. This is likely due to the initially higher void fraction in loop 1 and the fact that there is greater void generation occurring in that loop.

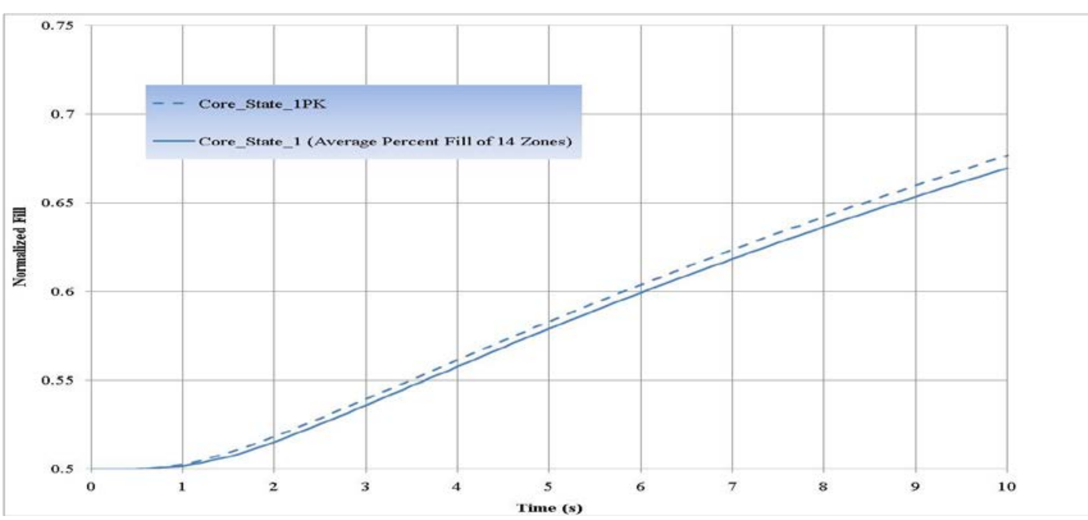

Figure 6 Zone Fill Transient for Core State 1 (Single Pump Trip)

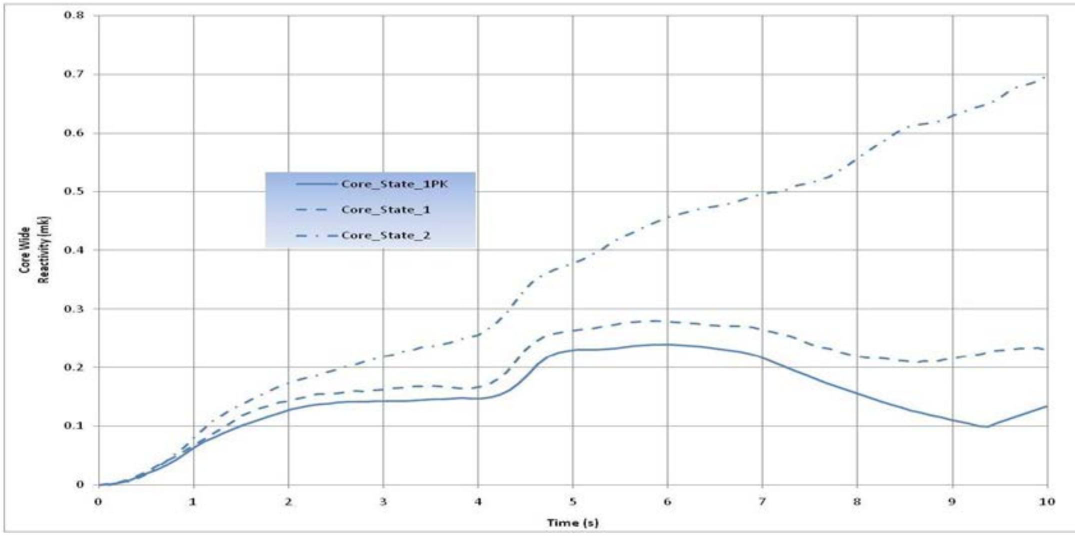

Figure 7 Core-Wide Reactivity Transient for Pump-2 Trip

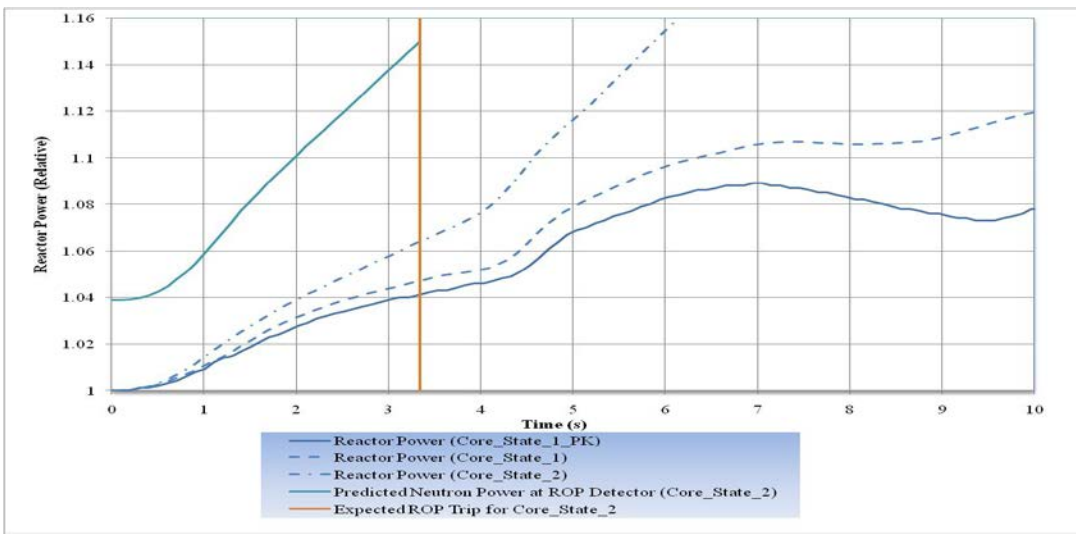

Figure 8 Reactor Power Transient for Pump-2 Trip

The greater overpower transient of the tilted core state (Core_State_2) is also an indication that reactor trip on regional overpower would occur quite early. This is especially true because the detectors for regional overpower are typically calibrated to an untilted core state whose power shape would resemble that of Core_State_1. The power transient for the highest reading flux detector of Core_State_2 is also presented in Figure 8. The initial reading of this flux detector is approximately 1.10 relative to the neutron power at that detector in the untilted core state. Due to this high initial reading, the regional overpower trip would occur within about 3.40 seconds, which is more than one second earlier than any process-related trip.

As noted in Section 2.2.3, single channel analysis was performed using the predicted hydraulic conditions in core pass 1 (header 4 to header 1) in conjunction with the bundle power transients from a channel with similar geometry to O6 but located in core pass 1. Channel O16 was used for this purpose. The reactor power transient is simply applied for the point kinetics case. The overpower transients become increasingly greater from Core_State_1PK to Core_State_1 to Core_State_2. Despite this increasing power transient the impact on the onset of dryout at bundle 8 is small. Dryout first occurs at bundle 8, at 4.64, 4.41 and 4.39 seconds, for Core_State_1PK, Core_State_1 and Core_State_2, respectively. As a result, the difference in post-dryout fuel sheath temperatures at bundle 8 (Figure 9) is on the order of 20˚C or less. The similar times of dryout, despite the trend in overpower transient, demonstrates that the CHF correlation which determines onset of dryout is more sensitive to the local thermalhydraulic conditions (i.e., pressure and mass flux) than the local power or heat flux.

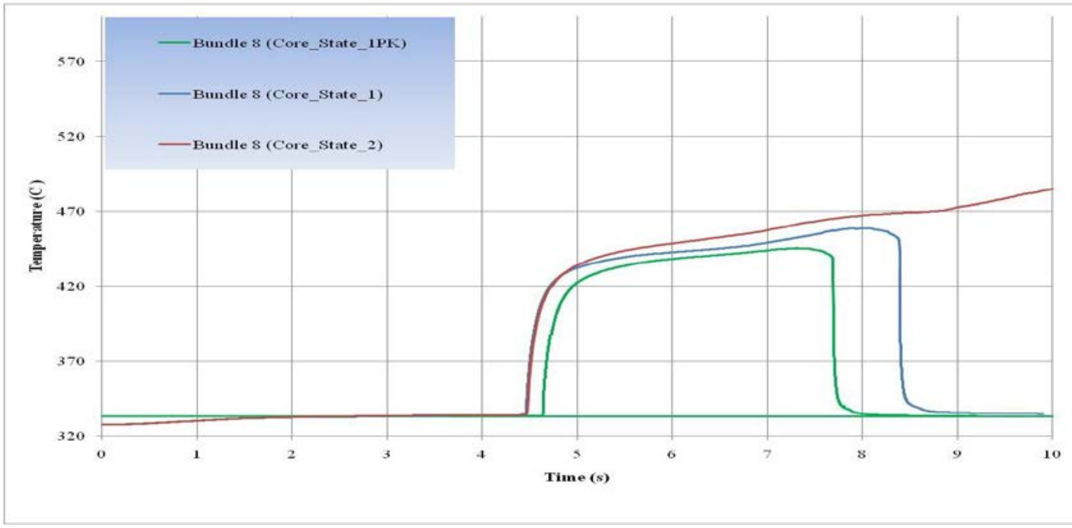

Figure 9 Bundle 8 Fuel Sheath Temperature Transients (Single Channel Analysis, Pump-2 Trip)

3.4 Total Loss of Class IV Power

A total loss of Class IV Power, resulting in rundown of all four main HTS pumps, is a symmetric pressurisation event, in the sense that the core passes of loops 1 and 2 would show similar behaviour. It compliments the asymmetric single pump trip event. Loss of feedwater to all four steam generators and closing of the turbine governor valves accentuate the system pressurisation and overpower transient. The location of ROH-3 and ROH-7 to the pressurizer would limit the pressure increase in these headers. Conversely, the pressure in ROH-1 and ROH-5 would be greater because they are not in closer proximity to the pressurizer. As a consequence, the flow reduction in core pass 1 and 3 (headers 4 to 1 and 8 to 5, respectively) will result in limiting thermalhydraulic conditions for the onset of dryout.

Figure 11 shows the ROH-1 pressure transients for Core_State_1. Included in Figure 11, is the point kinetics result (Core_States_1PK). The core differential pressure transient of the core passes in the affected loop is presented in Figure 12. Again, similar to the ROH pressures in Figure 11, the trend is similar for each case. These figures show that a reactor trip on high pressure, low coolant flow or low core differential pressure would occur at approximately the same time regardless of the methodology used to perform the transient analysis (i.e., 3-D versus point kinetics). In other words, 3-D neutronic affect do not appear to play a very large role in the transient thermalhydraulics, which are used at site to actuate the shutdown systems on process parameters.

The normalized average zone fill (0-1) and reactor power transients are presented in Figure 10 and Figure 13, respectively. In the simulation, the loss of LZC pump is not considered as shown on table 3.

These figures demonstrate reasonable agreement between the two equilibrium core cases with poisoned moderator (i.e., Core_State_1 and the one point kinetics simulation, Core_State_1PK).

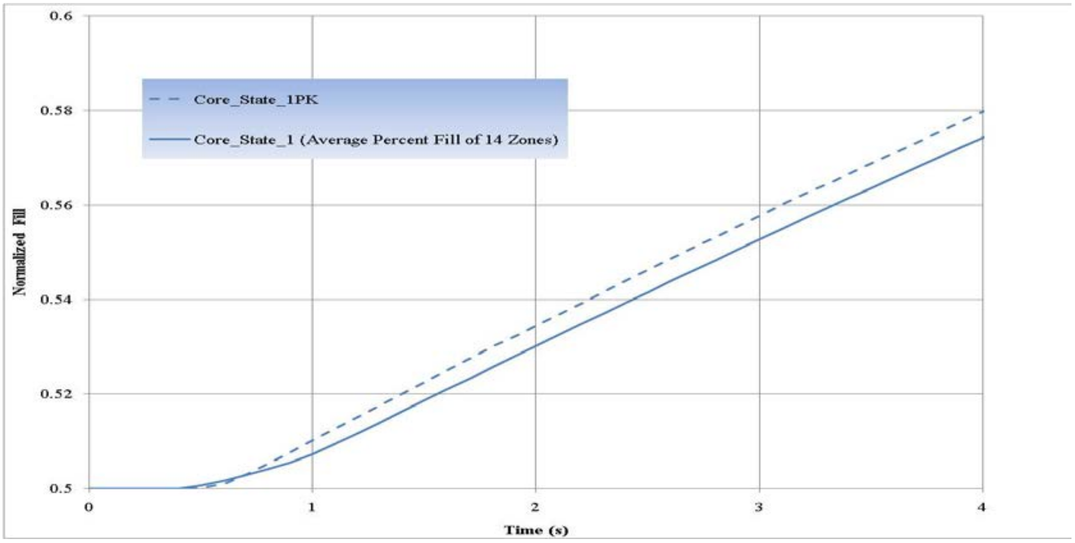

Figure 10 Average Zone Fill Transient for Core State 1 (Total Loss of Class IV Power)

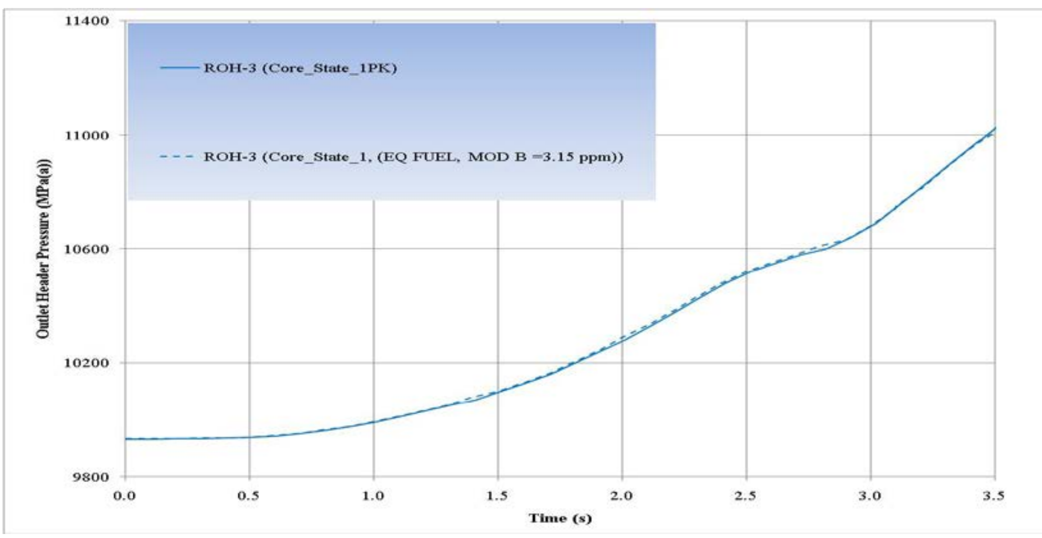

Figure 11 ROH-1 Pressure Transient for Total Loss of Class IV Power

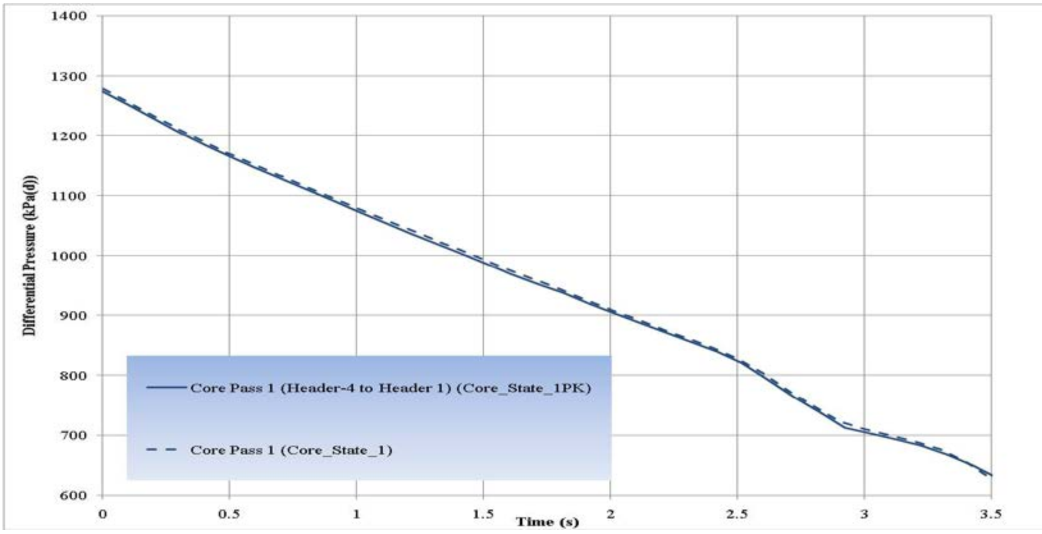

Figure 12 Core Differential Pressure Transient for Total Loss of Class IV Power

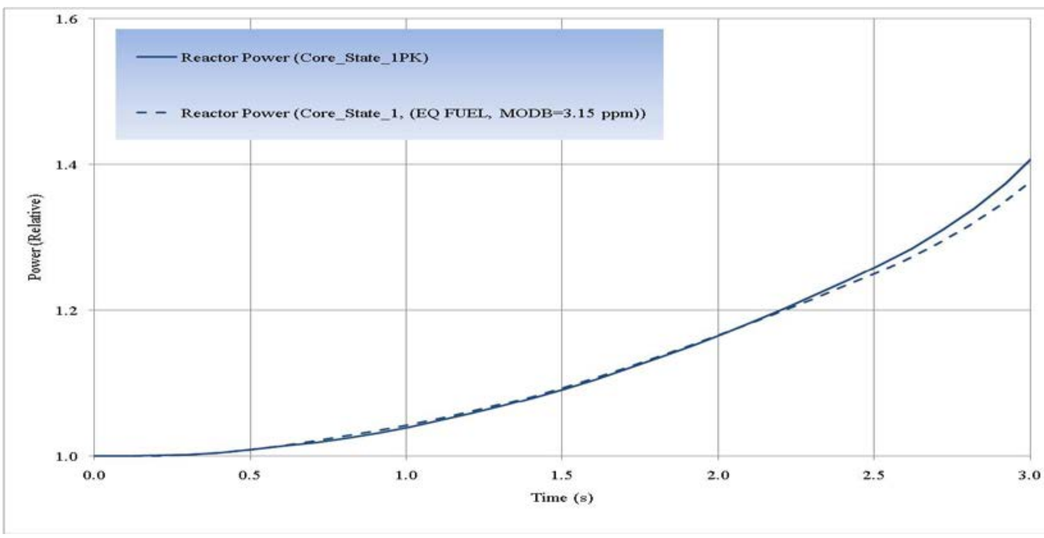

Figure 13 Reactor Power Transient for Total Loss of Class IV Power

4. Conclusions

Loss-of-flow analysis was performed for two types of transients: an asymmetric event corresponding to loss of One Heat Transport System Pump and a symmetric event corresponding to a total Loss of Class IV Power. Both events were simulated using a coupled CATHENA/RFSP methodology. The latter approach computes a 3-D representation of the neutron flux as a function of time.

The loss-of-flow events were also simulated using point kinetics. These simulations were performed using reactivity coefficients and delayed neutron data from the corresponding 3-D core state. In addition, the point kinetics simulations used a "point reactor" version of *CERBRRS's equations for the LZC System. The result is a very good overall core comparison between the 3-D and point kinetics simulations. It should be noted that future safety analysis using point kinetics should employ the same LZC model as that used in this comparison. For example, an independent RRS controller could be developed, which includes this LZC model, and linked to CATHENA via PVM.

Since the *CERBRRS LZC model is validated against station data for any CANDU 6 station, the integration of the same LZC model (or a “point reactor” version thereof) in CATHENA point kinetics, will result in a more realistic system response for safety analysis application.

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