Qinshan CANDU® 6 Main Heat Transport System High Operational Performance

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Abstract

This paper deals with the Qinshan CANDU® 6 Critical Channel Power (CCP) performance up to about 6 years of operation. Operational and aging related changes of the primary Heat Transport System (HTS) throughout its lifetime may lead to restrictions in certain safety system settings and hence some restriction in performance under certain conditions. A step in confirming safe reactor operation is the tracking of relevant data and their corresponding interpretation by the use of appropriate thermalhydraulic analytic models. Based on these analytic models up to 10 years of reactor operation are predicted and presented. These predictions, in association with an optimized parameter tracking and adjustment methodology, confirm continued safe reactor operation. This paper demonstrates that Qinshan CANDU Units 1 and 2, as compared to other CANDU 6 nuclear reactors of earlier design, exhibit significantly improved performance with much reduced plant aging effects. This high performance may, in part, be attributed to design improvements as well as improved operating practices. These performance improvements can also be expected for both new and refurbished CANDU 6 type nuclear reactors.

1. Introduction

This paper deals with the Qinshan CANDU[1] 6 operational and aging related changes of the primary Heat Transport System (HTS) throughout its first 6 years of operation, their extrapolation through analytical models up to 10 years of operation and their impact on Critical Channel Powers (CCP) performance, the power at which fuel dryout occurs. The presented analysis is based on the principles of monitoring, detecting, tracking, anticipating, understanding and then adjusting or compensating as needed. Comparisons with older CANDU 6 stations are made [1], [2] demonstrating improved CCP performance. The analysis applies and discusses improved analysis methodologies, [3], [4], [5], including recent thermal-hydraulic model developments, enabling performance optimization as well as prediction of future trends. The focus is on Qinshan CANDU data evaluation and thermal-hydraulic model development which are used for accurate diagnosis of HTS operating changes and potentially, if needed, frequent minimal operational adjustments. In addition to CCP analysis, primarily used for Regional Overpower Protection (ROP) trip setpoint analysis [4], the thermalhydraulic models generated have a multitude of application such as forming the basis for steady state

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reference models for transient safety code model development. Other applications include determining accurate coolant conditions for a number of uses such as feeder, pressure tube and fuel integrity analyses.

2. Qinshan CANDU 6 main heat transport system aging

Operational preferences as well as aging processes may cause changes to the primary HTS. These changes affect both coolant-flow and heat transfer properties of the HTS as a whole. There are several contributing effects, some acting to increase and some to decrease safety margins. The magnitudes of these effects vary over time, and thus the overall impact on the HTS is a complex integrated function of all mechanisms.

Operational changes can take place in a relatively short time frame, such as changes caused by utility operating preferences as well as changes in the reference analytical model interpretation caused by measurement-instrumentation calibration. An example would be a relatively instantaneous change in measured outlet feeder exit temperature caused by an improvement in instrumentation calibration (not a real change in temperature, only improved perception affecting the analyzed model).

Aging related changes typically take place over relatively long time periods. The following is a list of the main currently known aging processes that may occur within the HTS which would influence the CCP:

- Increase in Pressure Tube (PT) diameter due to irradiation creep (PT diametral creep): This reduces the hydraulic resistance in the channel, hence increases its coolant-flow, but causes a detailed redistribution of coolant flow within the bundle that can result in a reduction in dryout power. Because there is more creep in the higher power channels, there is a flow redistribution effect whereby some of the flow from the outer low power channels is redirected to the inner channels. Increased flow in central channels mitigates the effect of PT diametral creep on CCP for the central most important, high power channels.
- Increase in hydraulic resistance due to redistribution of iron oxides (magnetite) in the HTS: Dissolution of iron and Flow Accelerated Corrosion (FAC) has been shown to occur in CANDU 6 plants. Iron is removed from the outlet feeders and is re-deposited in the cooler parts of the circuit, including the cold leg of the steam generators, the inlet feeders, and possibly the first section of the fuel channels. The magnetite layers cause both a fouling of the inside of the steam generator tubes, leading to reduced heat transfer, and also an increase in hydraulic resistance in the steam generator tubes and inlet feeders. This affects core flow (possibly core top-to-bottom flow tilt), inlet header temperature and, consequently, CCP.
- Erosion of the edges of flow-reducing orifices: This leads to relative flow redistribution from inner to outer reactor core.

Considerable advances have been made to mitigate these aging characteristics, by making design as well as operational changes. This paper presents the high performance results of such mitigating actions for the Qinshan CANDU reactors. It is expected that Qinshan

CANDU performance will also be achieved by new CANDU 6 type reactors and refurbished CANDU 6 reactors.

3. CANDU 6 main heat transport system description

Figure 1 gives a simplified presentation of the HTS components and coolant flow of a typical CANDU 6 reactor. The HTS consists of two “figure-of-8 loops” with four main HTS pumps (P1, P2, P3, P4), four steam generators (B1, B2, B3, B4), and associated headers (HD) servicing the 380 reactor-core channels (ranging from channels A09 to W14). Each HTS loop consists of two HTS passes. Outlet header, purification and pressuriser interfaces are also shown.

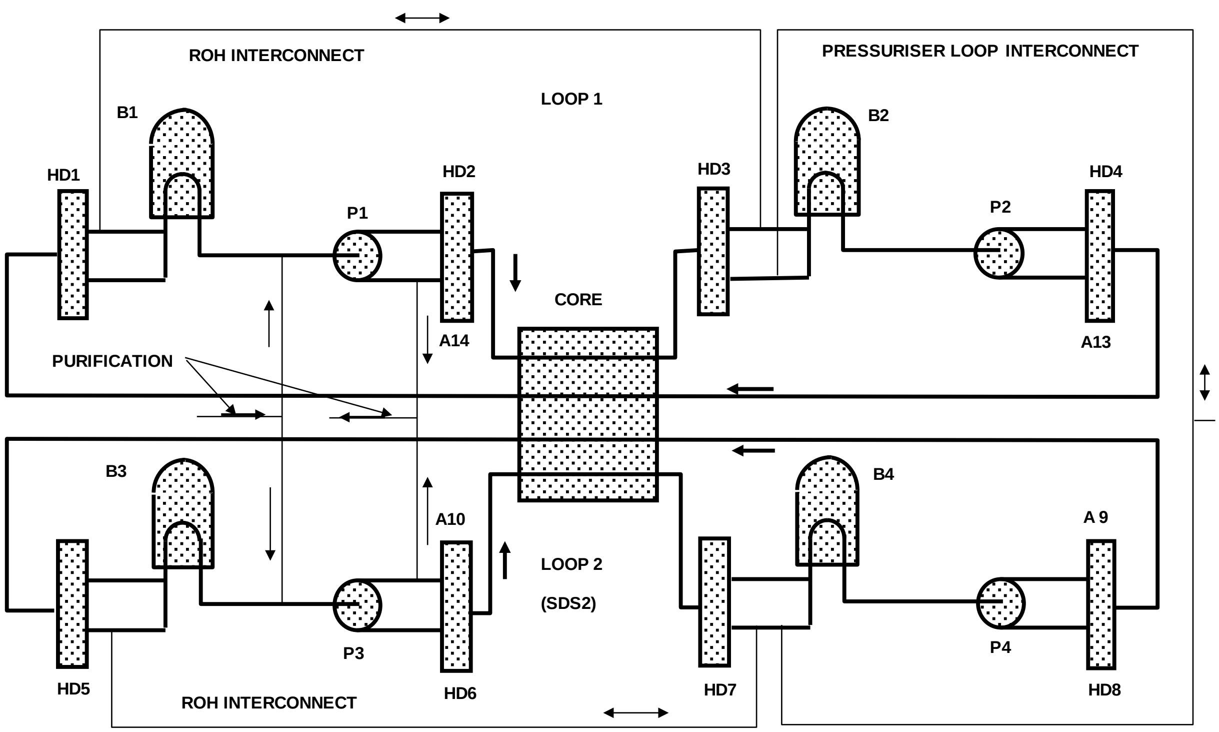

Figure 1 CANDU 6 simplified HTS flow diagram

4. CANDU 6 reactor-core thermalhydraulic model development

HTS temperature, pressure and flow as well as pressure tube diametral creep have long been established as the key parameters in establishing Critical Heat Flux (CHF) and CCP performance. These parameters, therefore, define the test matrix for associated laboratory tests at fuel dryout conditions. To simulate in-reactor conditions, CHF tests are performed in well-controlled test rigs in laboratory settings. In test set-ups radioactive heat sources (fuels) can be avoided and one can accurately measure local pressure, temperature, power, fuel-channel and fuel geometry as well as channel coolant-flow at fuel dryout conditions. From the collected data, CHF correlations can be developed. CHF is a function of temperature, pressure, channel-flow and pressure tube diameter. For the models used in this case, specific correlations are needed to use more easily predicted bundle average conditions for specific bundle geometries. Conditions upstream in the channel as well as appropriate, complementary models for local quality prediction also need to be accounted for. In fact, the entire hydraulic model for pressure-drop, two-phase flow condition prediction, and the CHF model need to work together in an integrated manner to yield a reliable prediction of the CCPs. Also, the system hydraulic model needs to properly predict the boundary conditions for the individual channels. Thermal-hydraulic codes (such as NUCIRC [3]), accurately predicting CCPs once the thermally-hydraulic code, by arbitrarily increasing channel power, has established local temperature, local pressure and flow associated with fuel dryout. A plant-specific thermal-hydraulic model is required for this analysis. This plant-specific thermal-hydraulic model is generally referred to as a reactor core model (see Figure 2) with plant specific geometry, together with plant specific boundary conditions, the measured HTS header conditions. For the CCP/CHF analysis to remain effective, one has to ensure that this plant-specific thermal-hydraulic model remains appropriate regardless of the operational changes, due to aging or otherwise. This assurance is given by tracking measured HTS header conditions and reactor-core geometry followed by ROP analysis adjustments whenever the plant conditions drift from the reference conditions of the previous ROP analysis.

Therefore, the ROP/CCP thermal-hydraulic model (see Figure 2) is a reactor core geometric model typically consisting of feeders, end fittings, feeder orifices, and pressure tubes with fuel bundles. The 4 parameters defining the site-specific characteristics of the ROP/CCP thermal-hydraulic model, therefore, can be identified as the following model boundary conditions:

1. The inlet header temperature, (T_{rih}),
2. The outlet header pressure, (P_{roh}),
3. The header to header differential pressure, (ΔP_{hh}), and
4. The HTS reactor-core geometry.

These 4 parameters must be tracked and adjustments made whenever the operational characteristics differ from the design reference, used in a corresponding full ROP trip setpoint reference analysis. The associated methodology is called the 4-parameter methodology and forms the basis for the original-design ROP tracking and adjustment methodology.

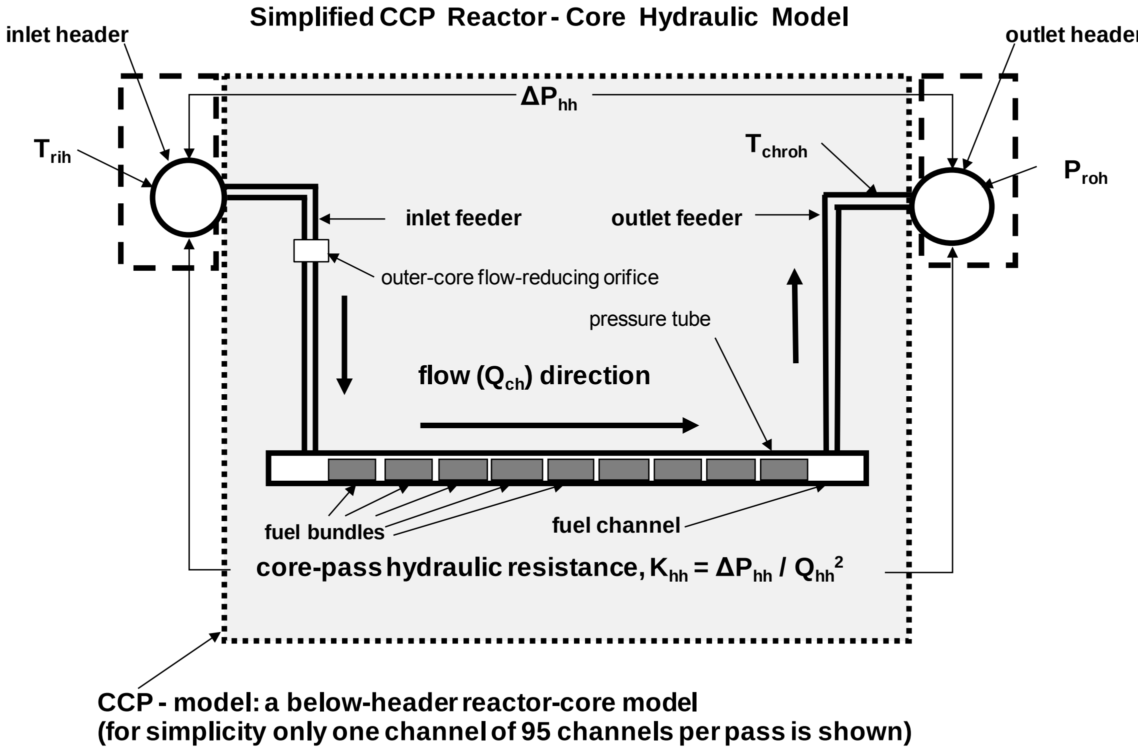

Figure 2 Simplified CCP reactor-core hydraulic model

In addition to the boundary conditions associated with the headers and the bundle powers, usually tracked at 100%Full Power (FP), the CCP model needs to consider the HTS reactor-core geometric model. Here, historic, present and future reactor operating points are based on a best-estimate thermalhydraulic model for the following components:

1. Pressure tube diametral creep [5] (based on CANDU-6 validated code predictions, it is one of the most important aging parameters),
2. Feeder-orifice degradation (obtained from core radial flow distribution consideration, [1] [4]), and
3. Feeder-pipe inner roughness changes (caused by magnetite transfer/deposition).

Pressure tube diameter may be based on direct measurements, if available, for some channels and predictions using an associated predictive code, such as an RC1980-based code [4] [5] for the remaining channels. Inlet feeder orifice degradation can be deduced from the core radial flow distribution once PT diameter distribution is accurately modeled. Generally direct performance measurements of the feeder-roughness reactor-core geometry components are not available. Therefore, one has to find an appropriate surrogate for thermalhydraulic model development as well as subsequent tracking and adjustments methodologies as required [4]. In the search for an appropriate surrogate, one has to make sure that the surrogate only changes when there is a change in the specific reactor-core geometry components. Observing that changes in geometry, such as changes in feeder roughness, will change the core flow, flow may be considered a potential surrogate candidate. However, it is noted that a change in flow may have many causes and as many different consequences on CCP. An HTS tracking and adjustment methodology, based on flow, may necessarily need to divide flow according to its source components, derive appropriate adjustments for each flow component, and then integrate the adjustments for an overall CCP adjustment. To illustrate further one may consider the following examples. Flow generally increases with increased PT diameter (PT creep) due to reduced hydraulic resistance. However, the flow through the fuel bundle, cooling the fuel, is reduced since a less resistive flow path is established above the fuel string diverting the flow through the bundle. As a consequence, for this scenario, an increase in channel flow is associated with a decrease in CCP. If, however, an increase in channel flow is the direct consequence of better main HTS pump performance, leading to a higher header to header differential pressure, then an increase in CCP will result. A third example: inlet header temperature is decreased at 100%FP operation with significant two-phase (liquid and gaseous phases) flow present. In this case there will be a reduction in two-phase flow resistance and an increase in coolant flow. This increase in flow is not associated with changes in the reactor-core geometry but would be accounted for in the tracking and adjustment associated with inlet header temperature. An additional adjustment based on the observed flow change would necessarily double-account for this effect. Therefore, in this case, any change in flow due to a temperature change in two-phase conditions must not be considered. Generally we find that for all major flow affecting parameters, with the exception of feeder roughness, flow changes, can be accounted by direct prediction of the various component ages. Therefore, one has to develop a methodology that isolates the flow component due to feeder roughness changes. This is most directly achieved by the concept of reactor-core hydraulic resistance defined by:

$$K_{hh} = \Delta P_{hh}/Q_{hh}^2 \tag{1}$$

Where ΔP_{hh} is the reactor-core-pass single-phase header to header differential pressure and Q_{hh} is the single-phase reactor-core-pass flow. Q_{hh} is the sum of the 95 core-pass channels flow rates, Q_{ch} (see Figure 2). The single-phase requirement ensures that two-phase resistance effects are eliminated avoiding interference with inlet header temperature tracking and adjustment. Further hydraulic resistance changes due to PT creep or orifice degradation are eliminated by independent model development for these two parameters as outlined above. This necessarily results in a reference reactor-core hydraulic resistance (Kref) that changes with time (generally decreases with increasing PT diameter). Therefore, reactor-core hydraulic resistance is the choice surrogate for feeder roughness tracking and corresponding CCP adjustments [4]. The reactor-core hydraulic resistance surrogate eliminates the first order (high impact) differential pressure effect, as compared to the flow surrogate. However, without a change in feeder-roughness, the reactor-core hydraulic resistance surrogate may vary, for example, according to the choice of reactor power chosen for flow-verification (generally second order effects in magnitude that may be addressed by second order (low impact), mostly CCP-insignificant, adjustments [4]). Reactor power sensitivity can be totally avoided by performing consistent flow verifications at the same single-phase power level.

5. Qinshan CANDU 6 reactor operational data and model application

5.1 Qinshan CANDU 6 HTS steam generator hydraulic performance data

For CCP analysis using the reactor-core model as shown in Figure 2 a steam generator model is not required. However, steam generator data provides valuable information about HTS aging, specifically as it relates to magnetite transport. A steam generator hydraulic resistance can be defined by the following equation:

$$K_{SG} = (P_{roh} - P_{(pump\text{-}suction)})/(Q_{SG}^2) \qquad (2)$$

where P_{roh} is the outlet header pressure at the Steam Generator (SG) inlet, $P_{(pump\text{-}suction)}$ is the pump suction pressure at the SG and Q_{SG} is the primary side coolant flow through the SG. As summarized in Figure 3 the hydraulic resistance is relatively constant for the first 6 years of reactor operation. This is significantly different from the SG hydraulic resistance trend for CANDUs commissioned in the 1980s [1] where SG hydraulic resistance is shown to increase at a rate of about 5% per kEFPD mainly due to magnetite deposition.

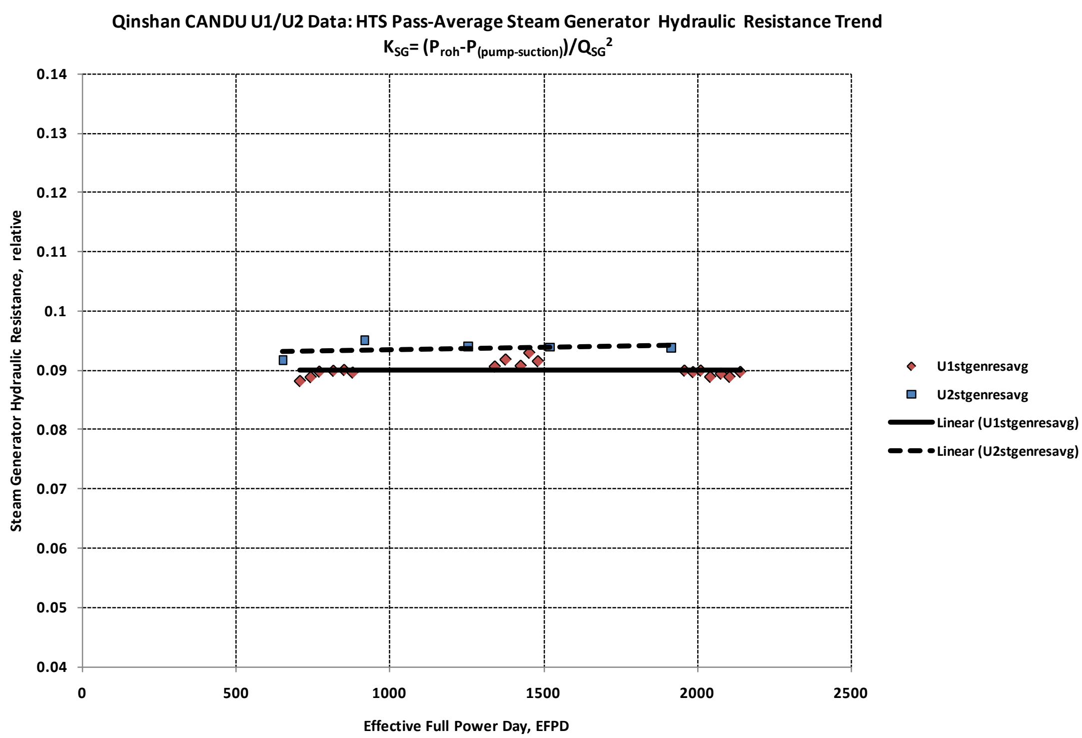

Figure 3 Steam generator hydraulic resistance

5.2 HTS reactor core model application

The first step in a ROP-CCP analysis is the gathering of data associated with

1. inlet header temperature (at 100% FP),
2. outlet header pressure (at 100% FP),
3. header to header differential pressure (at 100% FP), and
4. reactor core hydraulic resistance (at about 89% FP, single phase, the surrogate for feeder-pipe inner surface roughness)

This is followed by a best estimate reactor-core hydraulic model development based on this site data. The best estimate model can be used for extrapolation to future operating conditions. A ROP-CCP trip setpoint analysis may then be performed. Future differences between best estimate model extrapolation and actual future site measurements may be adjusted by the use of a pre-analyzed CCP analysis Response Surface [4], ΔRS, defined by:

$$\Delta RS = (P - P_{ref}) * k_P + (T - T_{ref}) * k_T + (\Delta P_k - \Delta P_{ref}) * k_{\Delta P} + (K - K_{ref}) / K_{ref} * k_K \quad (3)$$

where P, T, ΔP, and K refer to the parameters of outlet header pressure, inlet header temperature, header to header differential pressure and reactor-core hydraulic resistance and the subscript "ref" refers to the associated ROP analysis reference (it is noted, that for ease of writing, the subscripts "roh" (reactor outlet header),"rih"(reactor inlet header), "hh"(header-to-header) have been omitted, as presented in Figure 2, but are implied). The sensitivity factors k_P, k_T, $k_{\Delta P}$, and k_K are associated with the parameter sensitivity generally with respect to the average central channel CCPs. The subscript "k" in "ΔP_k" emphasizes that a reactor-core hydraulic-resistance measurement, K, is associated with this parameter. This methodology can be expanded to evaluate performance between flow verifications as outlined in Reference [4]. The sensitivity factor k_K is appropriately obtained from perturbations in feeder roughness.

5.3 Qinshan CANDU 6 HTS reactor-core data and thermalhydraulic model development

As outlined in Section 5.2 the following data is of primary importance in reactor-core hydraulic model development and plant aging tracking and adjustment:

1. The inlet header temperature, (obtained from measurements, T_{rih}),
2. The outlet header pressure, (obtained from measurements, P_{roh}),
3. The header to header differential pressure, (obtained from measurements, ΔP_{hh}), and
4. The HTS geometry of the reactor core.
   a) PT diameter (obtained from prediction, see Reference [5])

b) Feeder orifice adjustment factors (obtained from radial flow tilt considerations)
c) Feeder surface roughness (obtained from measurement based surrogate: K_{hh})

The HTS data for these parameters as well as their model application are presented in the following Sections.

5.3.1 HTS inlet header temperature and hydraulic model development

The measured HTS pass-averaged inlet header temperature is presented in Figure 4. The inlet header temperature trend has remained constant during the time interval of interest. This suggests negligible degradation of heat transfer in the SG. In contrast, from [1] and [2], other CANDU6 reactors showed increasing inlet header temperatures of about˚C/kEFPD due to SG fouling and divider plate leakage. This finding is consistent with little SG hydraulic resistance changes (Figure 3). A new analysis reference, used in CCP analysis, (T_{ref}) at 261.5 ˚C is established.

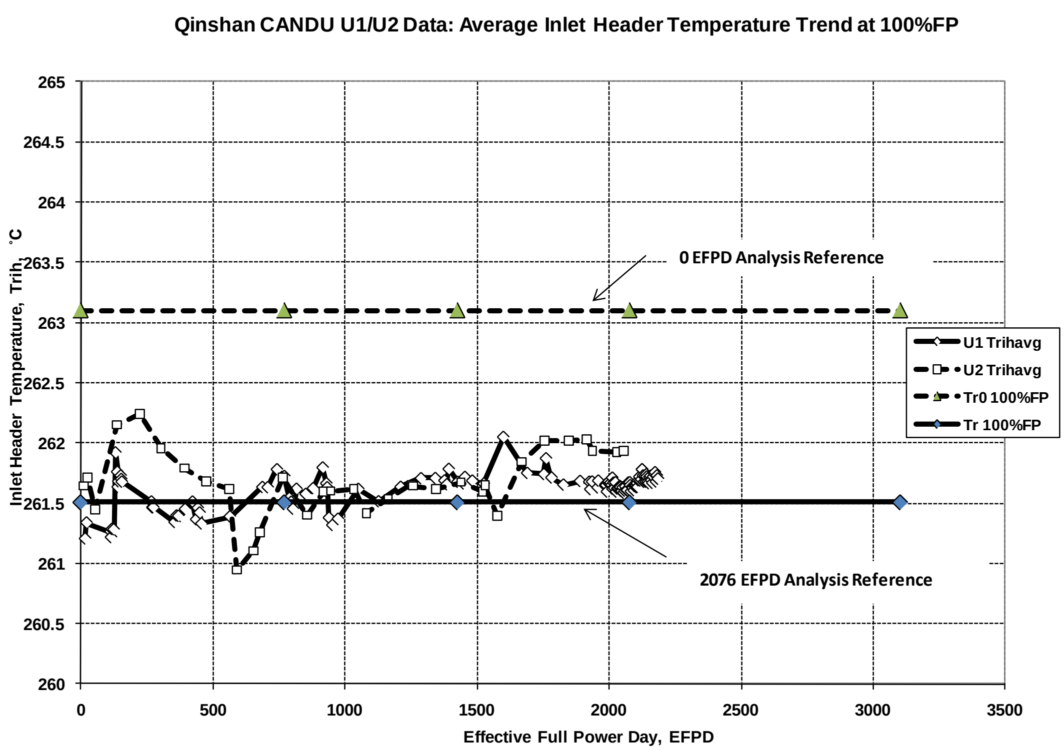

Figure 4 Inlet header temperature and model references

5.3.2 HTS outlet header pressure data and hydraulic model development

The measured HTS pass-averaged outlet header pressure is presented in Figure 5. Outlet header pressure remained constant during the time interval of interest leading to the conclusion that significant aging related hydraulic pass asymmetric magnetite transfer deposits were not observed. This supports that typical pass-asymmetric magnetite deposition characteristics specifically decreasing pass-averaged outlet header pressures as observed for other CANDUs [1] and [2] are not evident here. The reference outlet header pressure (P_{ref}) remains practically unchanged.

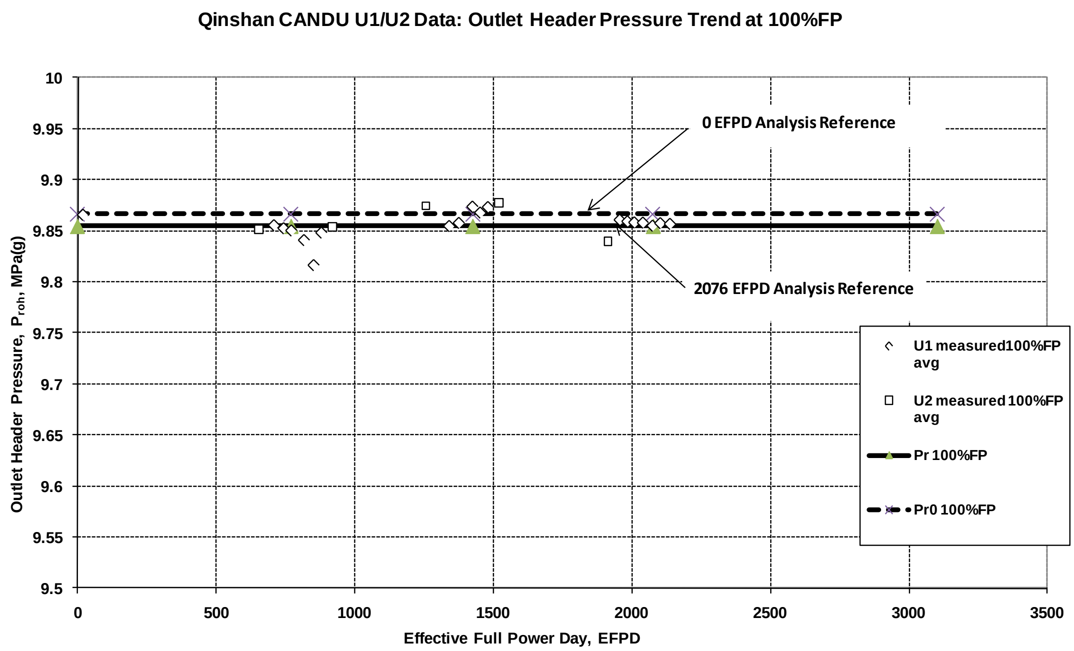

Figure 5 Outlet header pressure and model references

5.3.3 HTS header to header differential pressure data and hydraulic model development

The measured HTS pass-averaged header to header differential pressure is presented in Figure 6. Header to header differential pressure significantly decreased with an average rate of about –46kPa/kEFPD for the first about 6.5 years of operation, with HTS pass to pass differences less than the corresponding measurement uncertainty. This can be compared to older CANDU 6 reactors, specifically an about constant differential pressure trend for some HTS passes, while the corresponding pass in the same HTS loop shows an increasing differential pressure trend of about +24kPa/kEFPD for the first 6.5 years of operation as reported in Reference [2]. Similar characteristics are observed for other older CANDU 6 reactors (see Reference [1]). The large pass-asymmetry within a HTS loop for older reactors is attributed to pass asymmetric magnetite deposition in the inlet feeders leading to increased feeder roughness and increased reactor core hydraulic resistance. The pass-symmetric reduction in differential pressure drop observed for Qinshan is typical for PT diametral creep characteristics. As shown in Figure 6 a new analysis reference, ΔP_{ref}, has been established at about 2076 EFPD.

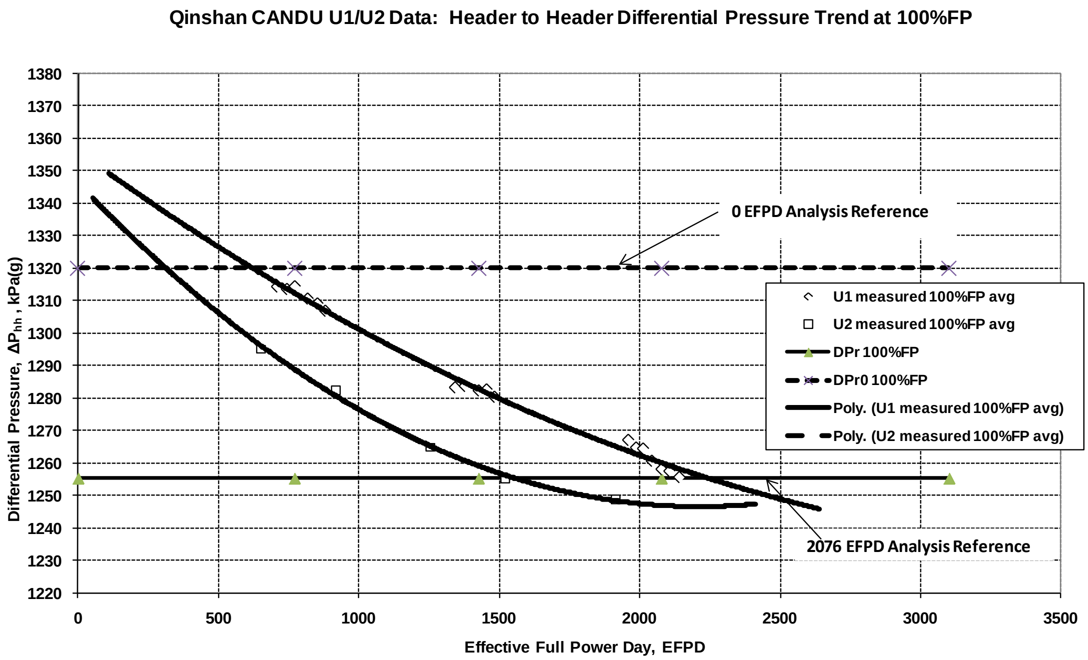

Figure 6 Header to header differential pressure and model references

5.3.4 HTS reactor-core data and hydraulic model development

The measured-based HTS pass-averaged hydraulic resistance is presented in Figure 7. It is obtained from Equation (1). The single-phase (89% Full Power (FP)) header to header differential pressure drop is generally less than 10kPa lower than the data reported for 100%FP in Figure 6. This indicates that there is little two-phase flow at 100%FP. The single-phase flow has been obtained from inverse heat balance flow which corresponds to the sum of channel specific inverse heat balance flow measurements calculated using the formula:

$$Q_{ch} = Power_{channel} \,/\, \Delta Enthalpy_{channel}(T_{rih} \text{ to } T_{chroh}) \quad (4)$$

Figure 7 also shows the best-estimate analysis hydraulic resistance K_{ref} (see Equation (3)). In the absence of significant feeder roughness increases and outer core feeder orifice degradation this decreasing hydraulic resistance trend is consistent with PT diametral increases. In Figure 7 it is noted that a decrease in reactor-core hydraulic resistance of about -3% is observed. This can be compared to older CANDU 6 reactors [1],[2] where an increase in hydraulic resistance of about +10% is estimated during the same reactor operational time interval. It can be concluded that hydraulic resistance increases due to feeder roughness increases have remained relatively insignificant during the first 6 years of operation for the Qinshan CANDU reactors as compared to older CANDU reactors.

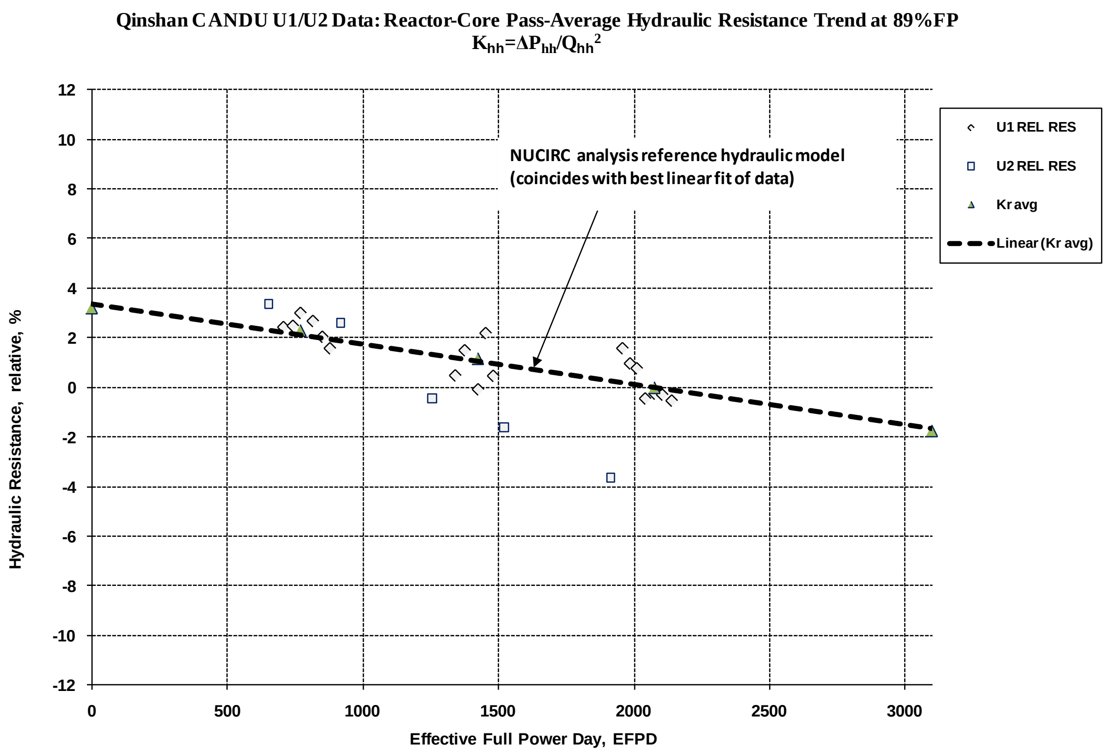

Figure 7 Reactor core hydraulic resistance at 89% FP and model reference

The best estimate PT diametral creep has been obtained by the methodology described in Reference [5]. PT diametral creep predictions have to be used at this time since accurate measurements to establish creep rates cannot be made until about 7 years of reactor operation. This methodology has been verified and validated for CANDU 6 specific application. The PT diametral creep used in generating the reactor core hydraulic model is summarized in Figure 8. It is noted that PT diameter increases faster in the central core than in the outer core due to higher fast neutron flux in the central core. This results in a progressively lower hydraulic resistance in the central core channels resulting in coolant flow redistribution from outer to inner core channels. A radial flow tilt can be defined to quantify this effect. Figure 9 summarizes the measured radial flow tilt trend. It is consistent with the NUCIRC thermalhydraulic model with PT being the only core geometry aging parameter. Specifically negligible outer core feeder orifice adjustments were necessary for the analytical model to reproduce the best estimate, measurement-based, linear trend. This measured and modeled radial flow tilt trend is consistent with the trend observed at other CANDU 6 reactors (see Reference [1]).

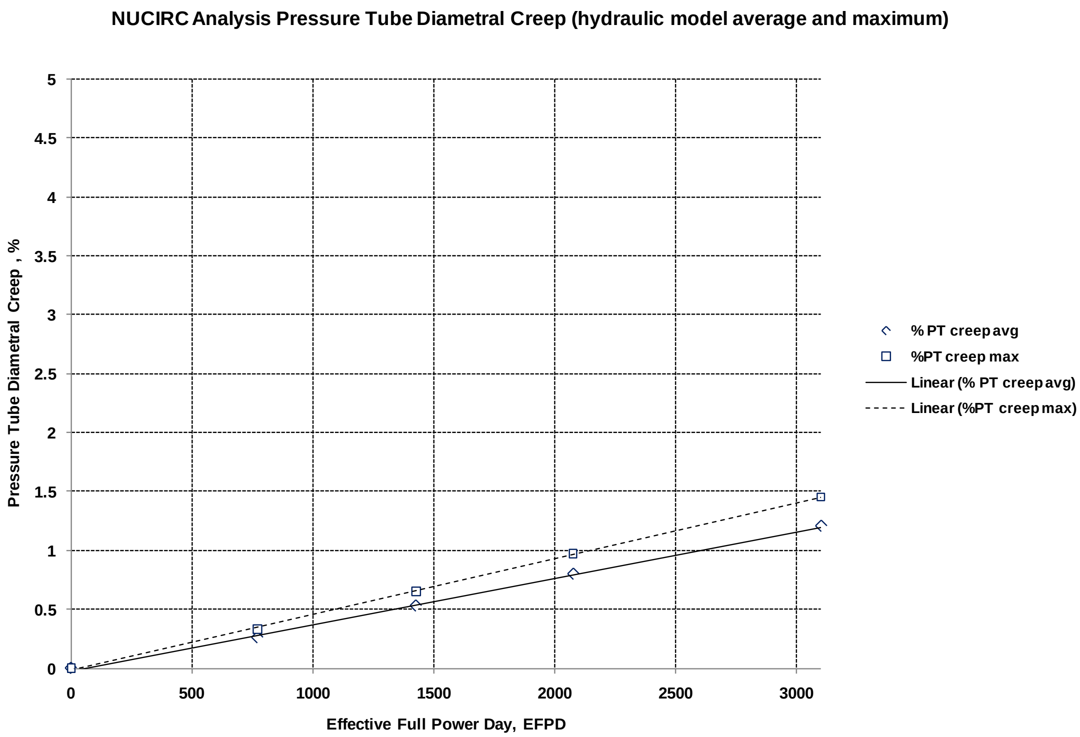

Figure 8 Pressure tube diametral creep best estimate model references

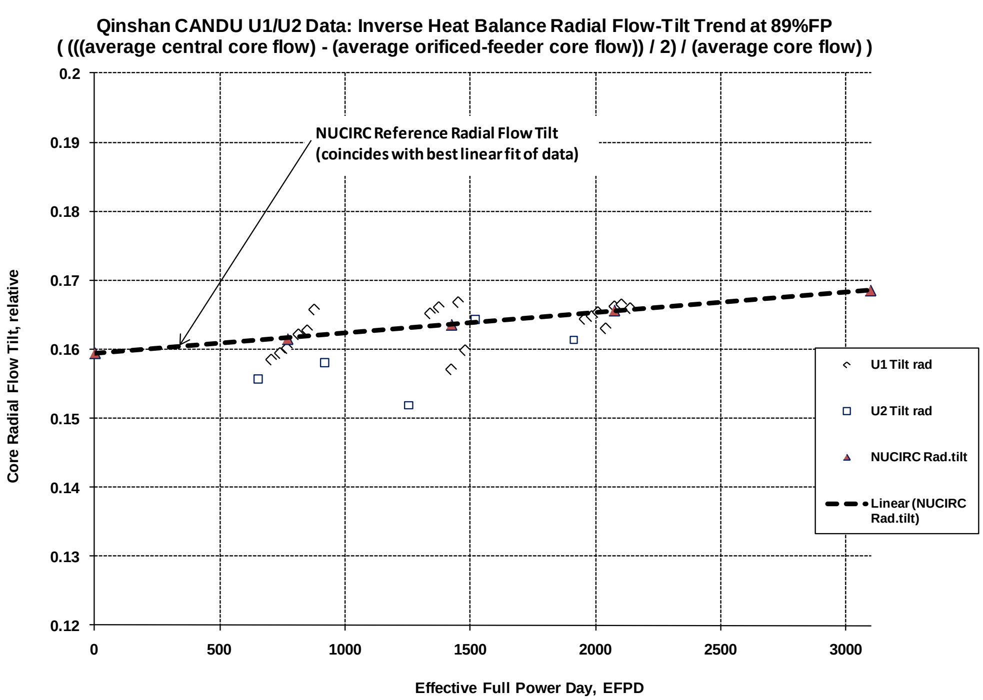

Figure 9 Core radial flow tilt at 89% FP and model reference

5.3.5 HTS reactor-core hydraulic model and critical channel power analysis results

Figure 10 summarizes the NUCIRC [3] generated CCP trend of a representative core channel. It is an indicator of required ROP trip setpoint changes due to PT diametral creep between 0 EFPD and 3100 EFPD for constant header to header differential pressure. In the absence of significant indicated feeder roughness increases as well as orifice degradation, it is in part an excellent indicator of the CCP effect due to reactor core geometry changes. It is noted that a decreasing reactor core hydraulic resistance will result in decreasing header to header differential pressure and the associated ΔP_{hh} parameter adjustment as well.

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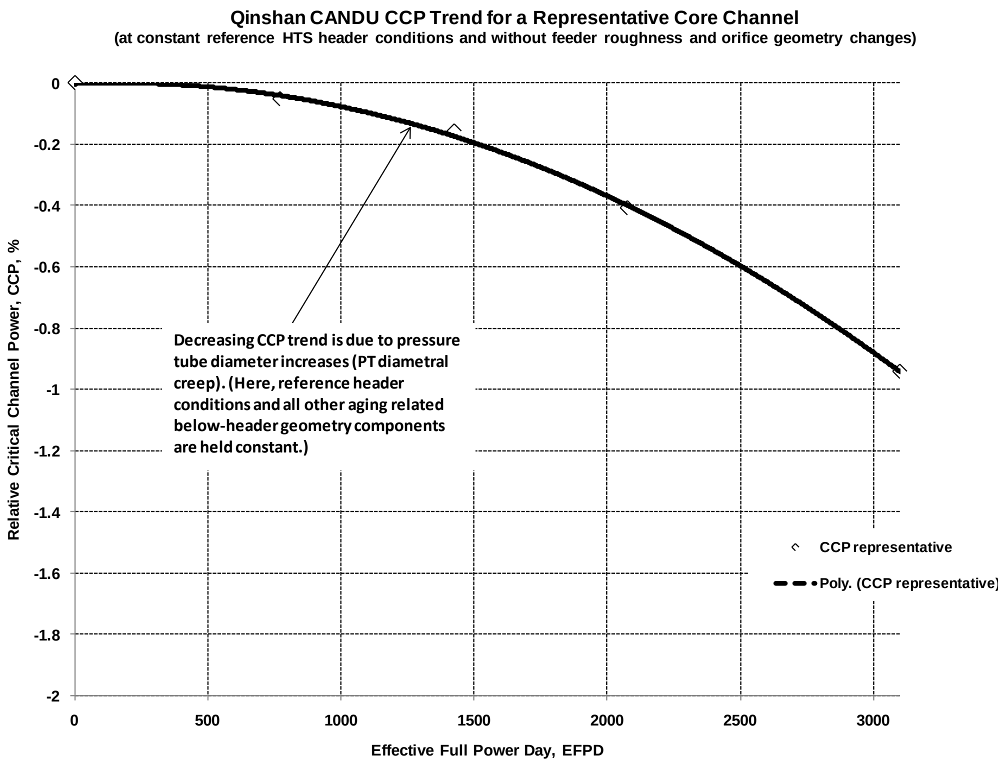

Figure 10 CCP trend of a representative core channel

6. Summary and conclusion

This paper deals with the Qinshan CANDU® 6 Critical Channel Power (CCP) associated performance up to about 6 years of operation. Operational and aging related changes of the primary Heat Transport System (HTS) throughout its lifetime may lead to restrictions in certain safety system settings and hence some restriction in performance under certain conditions. A step in confirming safe reactor operation is the tracking of relevant data and their corresponding interpretation by the use of appropriate thermalhydraulic analytic models. Based on these analytic models up to 10 years of reactor operation are predicted and presented. These predictions, in association with an optimized parameter tracking and adjustment methodology, confirm continued safe reactor operation. This paper demonstrates that Qinshan CANDU Units 1 and 2, as compared to other CANDU 6 nuclear reactors of earlier design, exhibit significantly improved performance with respect to flow assisted corrosion and magnetite transport. This resulted in relatively constant inlet header temperatures and significantly reduced magnetite transport related hydraulic resistance increases in the steam generators and in the reactor core (the HTS feeders). This high performance may in part be attributed to design improvements as well as improved operating practices. These performance improvements can also be expected for both new and refurbished CANDU 6 type nuclear reactors.

7. Acknowledgements

The authors would like to acknowledge the contributions of many experts associated with CANDU utilities and CANDU designers who have made significant contributions to the understanding of the operational behavior of reactors and/or to implementations of various aspects of the CANDU program. Specifically, the authors would like to acknowledge the significant contributions of X. Mou, X. Peng, and C. Su of TQNPC and of D.J. Wallace, N. Lee, A. Elalami, and N. Barkman of AECL.

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