NUCIRC Thermal-Hydraulic Applications in Support of CANDU© Plant Design and Operation

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Abstract

The NUCIRC thermal-hydraulic code is used during the design stage, the commissioning stage, the operational stage, and the refurbishment stage of CANDU© nuclear plants. It provides the basis for many subsequent applications ranging from general safety analysis to main Heat Transport System (HTS) component integrity analyses. Most often the code is used for Critical Channel Power (CCP) analysis in support of Regional Overpower Protection (ROP) trip setpoint analysis. Thermal-hydraulic models representative of the HTS can be updated using site measured data, by employing the many options of this code. Different code options allow the determination of aging for each HTS component and, once combined, all components result in the thermal-hydraulic model of the entire HTS. This paper will discuss each NUCIRC option and how each option is used to define a thermal-hydraulic model representative of site data and how observed HTS component aging trends can be obtained and used for accurate extrapolation to future operation.

1. Introduction

The NUCIRC (A **NU**clear Heat Transport System Code for **CIRC**uit analysis) thermal-hydraulic code has reached a high level of maturity since its first development in the 1970s. With its highly developed modelling detail, it has many applications in reactor design support as well as reactor operational support. NUCIRC is the only steady-state thermal-hydraulic code that is being used throughout the life time of a nuclear plant from the initial design stage, to the licensing stage and the commissioning stage in addition to the continuous operational support it offers. It is also the only thermal-hydraulic code which models the plant's main HTS system in detail for single and two-phase operating conditions as compared to any other existing code. It has a wide range of applicability which varies from operational applications to design and experimental applications. The fundamental structures of these applications will be discussed in this paper. The NUCIRC code is used to perform CANDU1 design, and/or maintenance analysis calculations in Canada, Korea, Argentina, China and Romania.

2. NUCIRC code development

The NUCIRC code has been under development, verification, and validation activities since its conception in the 1970s. These code development activities have conformed to AECL scientific code quality assurance procedures. These procedures have recently been

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revised in accordance with the new N286.7-99 standard and AECL's Quality Assurance Manual (QAM) requirements for analytical, scientific, and design computer programs.

Generic NUCIRC code developments have included:

- Updating form losses associated with bends, elbows, diffusers, etc. in feeders,
- Updating the pressure drop and heat transfer models for the Steam-Generator (SG),
- Including a new manifold model to account for pressure gradients in the inlet and outlet headers.

The key developments in the code in terms of the application of NUICIRC are as follows:

- Update fuel element pressure drop and Critical Heat Flux (CHF) models for non-crept and crept pressure-tubes (PT),
- Modelling of pressure-tube diametral-creep,
- Modelling of component roughnesses.
- Modelling of the feeder-orifice performance, and
- Performing inverse heat balance calculation to obtain measured HTS flow

These modelling capabilities allow the code to capture the key known aging phenomena of pressure-tube diametral-creep, magnetite transport, and orifice hydraulic geometric degradation. As such, this code is capable of performing different design and analysis calculations, which are discussed in detail in Reference [1].

To enable the user to solve specific physical problems, nine "ITYPE" options have been set up in NUCIRC. The geometrical boundaries of the main options are shown in Figure 1. The nine options are summarised as follows:

ITYPE 1 Determines the header to header pressure drop for a given channel flow and feeder geometry.

ITYPE 2 Determines the channel flow for a given header-to-header differential pressure drop and feeder geometry.

ITYPE 3 Selects the size of the inlet and outlet feeders in accordance with established feeder sizing criteria.

ITYPE 4 Predicts the temperature/pressure/flow distribution in the HTS: In this option, the HTS model consists of a single core-pass with one HTS pump and one SG bounded by one reactor inlet header and one reactor outlet header. This option is generally used in design for sizing the HTS pumps and SG of a given HTS design. This ITYPE option is further evolved to ITYPE 6 which includes all the capabilities that ITYPE4 offers and more.

ITYPE 5 Determines the channel flow for a given header-to-header pressure drop, feeder geometry and fuelling machine boundary conditions. It is a fuelling extension of ITYPE 2.

ITYPE 6 Predicts the temperature/pressure/flow distribution in the complete figure-of-eight HTS loop: This ITYPE option is widely used since it includes all the capabilities of ITYPE 4. Beside the two-quadrant HTS model included in this option, an inter-connect balance line between the outlet headers of a common loop is also modelled. Loop interconnect flows and interactions with the purification, pressure and feed/bleed systems can be modelled via user-defined boundary conditions (i.e., coolant flow, pressure, and temperature) at specified nodes. This option is generally used for the detailed performance evaluation of HTS operating conditions.

ITYPE 7 Focuses on the SG/Pump models. This option is used to predict the performance of the SG for given geometrical and thermal-hydraulic characteristics.

ITYPE 8 Predicts the magnitude and direction of loop interconnect flows between HTS loops via the common purification system piping between the associated HTS pumps.

ITYPE 9 Predicts the magnitude and direction of loop interconnect flows between HTS loops via the common pressurizer piping between the associated reactor outlet headers.

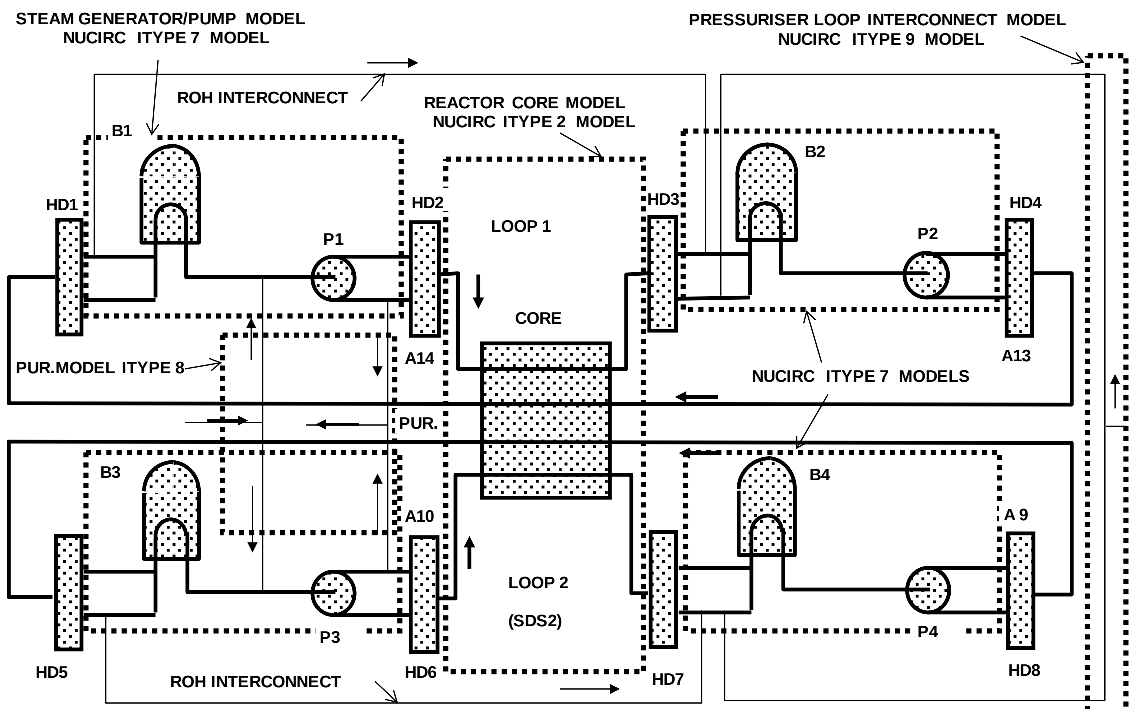

Figure 1 CANDU simplified circuit diagram (NUCIRC ITYPE 6 Model)

For ITYPE 1, 2, 3 or 5 options, either critical channel power for dryout and fuel centre-line melting (CCP) or critical channel flow for dryout (CCF) can be included if required.

The NUCIRC code is maintained and developed at Sheridan Park under the administrative supervision of the Fuel Design Branch, CANDU Core Design, Engineering Technology Delivery, and under the technical supervision of the Thermal-hydraulics Branch, Chalk River, Reactor Safety Division.

3. NUCIRC code interface

NUCIRC is used through the life time of a nuclear plant at the design stage, the commissioning stage, the operational stage where constant monitoring of the HTS is performed, as well as the refurbishment stage. NUCIRC is also used extensively for safety analysis. Figure 2 shows how this code relates to the different fields of expertise required for the design and safe operation of CANDU reactors.

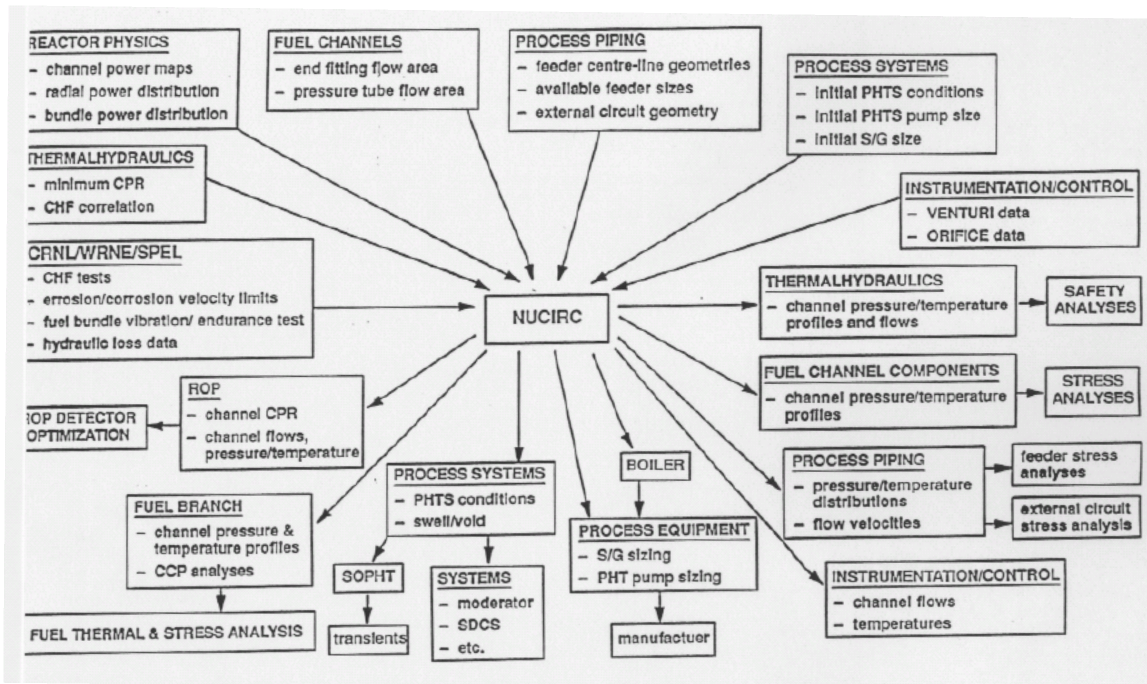

Figure 2 NUCIRC code interfaces

The inputs required by NUCIRC to create the NUCIRC model consist of the design specific geometries and properties from the different disciplines of reactor design. Site specific thermal-hydraulic properties are also used as an input to create the thermal-hydraulic models for the different HTS components. For a new reactor, the site specific thermal-hydraulic properties are based on the HTS component manufacturing specifications. For an operational plant data is collected and further processed through NUCIRC to capture the actual performance of the plant. Consequently, the thermal-hydraulic characteristics of an aged plant, obtained through NUCIRC-assisted data analysis, are further used as an input for other analysis that ensure the safe operation of the plant while meeting the regulatory safety standards.

4. NUCIRC code applications

To be able to understand the different applicability of this code, the fundamental structure of the model development has to be defined. Detailed discussion of model development is included in Reference [1]. Therefore, this paper focuses on how the NUCIRC code uses site data to develop the thermal-hydraulic models of the main HTS.

Site data collection is performed to give an insight into the performance of the plant in terms of thermal-hydraulic properties of the HTS components and to ensure compliance with licensing analysis. These data can be used by the NUCIRC CCP/ROP analysis in different ways ranging from off-line trend analysis to on-line immediate consideration as discussed in References [2] and [3]. The same data can also be used differently by the NUCIRC code to give an insight into how the reactor core geometry, the SG/pumps, the purification systems, and the inter-connect pressurized piping are behaving as the plant ages, allowing trend extrapolation to future operation. The associated HTS aging models are then used as a basis for other analysis such as the analysis of accident scenarios analysed using the CATHENA thermal-hydraulic transient code.

It is very crucial to confirm that the actual thermal-hydraulic site behaviour is consistent with safety analysis through frequent comparison with site data, as it becomes available. The aging trend of every plant is unique and defined by its measurable parameters [2]. It is possible to achieve consistency between site-specific data and NUCIRC thermal-hydraulic representative models once data is available. Furthermore, it is possible to extrapolate these trends to estimate future aging characteristics for these parameters. However, analysing a newly designed plant or a refurbished plant, thermal-hydraulic aging trends, such as defined by header boundary conditions and the HTS reactor core geometry, have to be available in order to demonstrate compliance with regulatory requirements. Operational and aging related changes of the HTS throughout its lifetime may lead to restrictions in certain safety system settings. The NUCRIC code applications allow for an assessment of the HTS at key points of interest throughout the plant life cycle.

4.1. Use of site measurements for NUCIRC HTS model development

CANDU 6 plants keep track of key measurable parameters [2] as a mean to track the aging of the plant and make adjustments as appropriate. Typical site data measurements include inlet header temperature, header-to-header differential pressure, outlet header pressure, pump suction pressure, coolant flow rate, and PT diametral creep. Some of the aging trends associated with these parameters are captured in Figure 3, 4, 5 and 8. Similar site specific data are included in References [4] and [5]. The careful selection of the variables used to define the HTS component aging, allows NUCIRC HTS component aging models to be generated in terms of HTS component geometries, the SG and HTS pump performance, and purification flow.

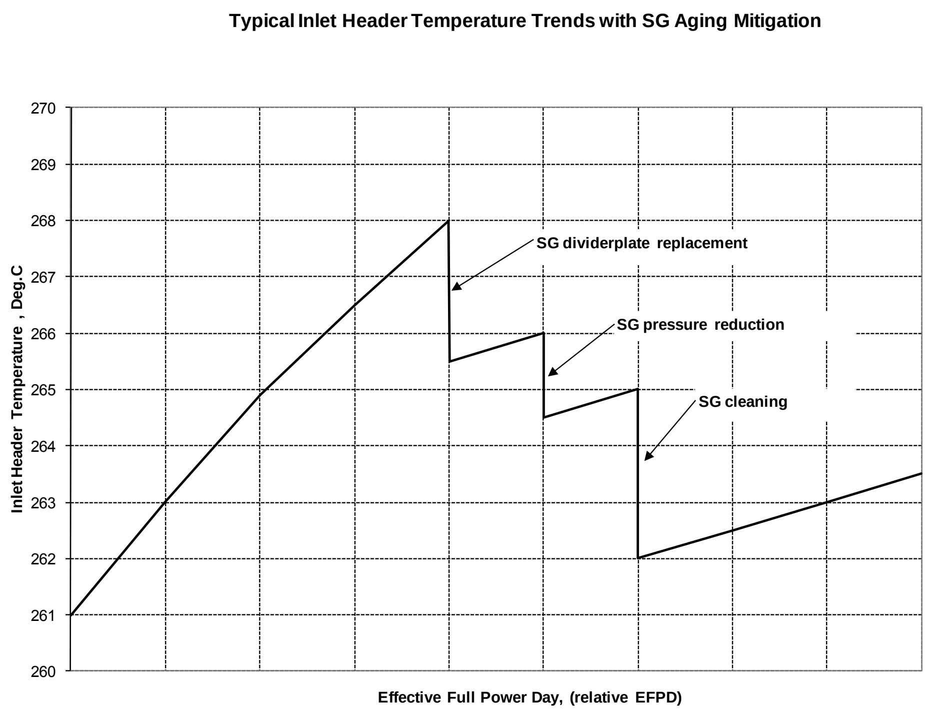

Figure 3 A typical inlet header temperature history of a CANDU plant at full power

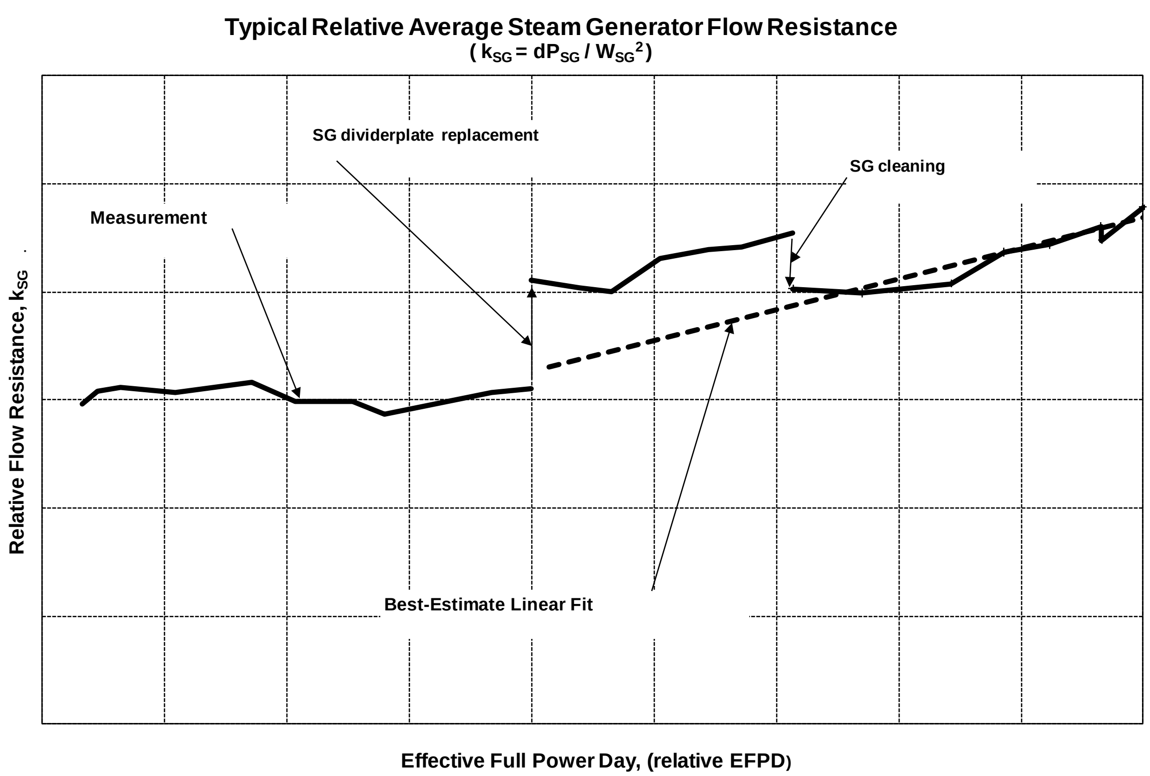

Figure 4 A typical relative average steam generator flow resistance of a CANDU plant

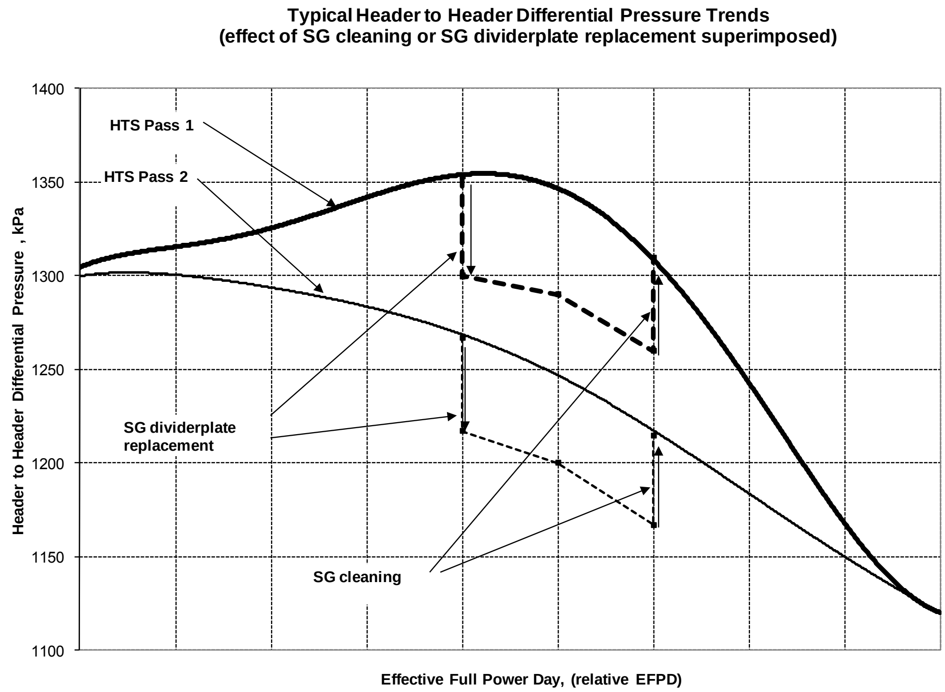

Figure 5 A typical header-to-header differential pressure trends of a CANDU plant

During the design stages NUCIRC HTS models are generated from known HTS component geometry and performance as supplied by the component manufacturers. The analysis of aging trends is generally limited to best-estimate PT diametral creep predictions [6], although conservative estimates may be used for SG, reactor core feeder roughness, and inlet feeder flow-reducing pressure break-down orifice aging. For these conditions the NUCIRC ITYPE 6 option is used to produce reference header boundary conditions used for NUCIRC ITYPE 2 (CCP) analysis.

As shown in Figure 1, the NUCIRC ITYPE 6 option is used to predict the detailed temperature, pressure, and flow distribution in the complete figure-of-eight HTS loop. This model includes the two-quadrant HTS model which consists of both core passes of a HTS loop, each core pass consisting of one PHT pump and one steam generator bounded by the associated reactor inlet and outlet headers and all the channels. An inter-connect balance line between the outlet headers of a common loop is also modelled. Loop interconnect flows and interactions with the purification and feed/bleed systems can be modelled using user-defined boundary conditions (i.e., coolant flow, pressure, and temperature) at specified HTS model nodes.

As the HTS ages, the main tracked parameters of temperature, pressure and coolant flow rate may change considerably over time. These three main tracked parameters are being affected by the aging of all HTS components. SG aging not only reflects in the inlet header temperature trend(Figure 3) and the SG flow resistance trend (Figure 4) but also in the header to header differential pressure trend (Figure 5) which, together with core flow rate, defines reactor core hydraulic characteristics. Therefore in order to develop the most accurate aged HTS circuit model (ITYPE 6 model) one has to analyse each HTS component, produce a best estimate thermal-hydraulic component model and then combine all HTS component models to obtain a full HTS circuit model. The NUCIRC ITYPE options allow this systematic approach. The entire HTS model is split up into the core model and the external circuit model. The core model is split into its 4 passes each analysed with an appropriate ITYPE 2 model. The external circuit is split into the four SG-pump combinations, each analysed with an appropriate ITYPE 7 model. The boundaries of the ITYPE 2 and ITYPE 7 models are shown in Figure 1. Each of the SG – pump combinations (ITYPE 7 models) can be split further into a SG component and a HTS pump component per pass. This yields 12 HTS components with associated measurement-based temperature and pressure model boundary conditions and measured coolant flow rate. The model defining aging parameters of the 12 components can now be adjusted until the measured temperatures, pressures and coolant flow rate are achieved for each component. The 12 adjusted HTS model components are then combined to yield a best estimate full HTS circuit model (ITYPE 6 model). Peripheral, generally second order flow rates associated with outlet header to outlet header HTS Pass interconnect flow (ROH interconnect as shown in Figure 1), purification HTS Loop interconnect flow rates and the pressurizer HTS Loop interconnect flow rate (see Figure 1) can now be obtained by ITYPE 6, ITYPE 8, and ITYPE 9 analysis respectively. These peripheral coolant flow rates may change the temperature, pressure, and coolant flow rate associated with each of the 12 HTS components, resulting in the need for an iterative process until consistency is achieved between ITYPE 6, ITYPE 7, ITYPE 2, ITYPE 8, and ITYPE 9 analyses. This analysis procedure will result in best-estimate aging parameter evaluations for all HTS components. Analysing at different historical times will allow the determination of trends associated with component aging. These trends can then in turn be used to predict best estimate HTS operating conditions at future operating points.

Depending on the application requirements, the HTS component model evaluations may have considerable detail. For SG modelling the ITYPE 7 option may be used to predict the performance of the SG for given geometrical and thermal-hydraulic characteristics or determine the parametric curves which define the relationships between tube inner diameter, tube roughness, tube fouling , tube plugging, divider plate leakage, thermal plate leakage, recirculation ratio and primary separator pressure drop. As a plant ages, the SGs exhibit aging signs in terms of these parameters. Consequently, the inlet header temperature (Figure 3) increases due mainly to two effects:

- Lower heat transfer coefficients in the SG tubes as residual builds up; and
- Divider plate leakage which leads the coolant flow to by-pass the SG tubes.

As these two effects are overcome by SG divider plates replacement or SG cleaning, respectively, the inlet header temperature drops as noted in Figure 3.

Hydraulic resistance or flow resistance is a property representative of the physical geometry of the system at hand. Figure 4 demonstrate a typical SG flow resistance. In this particular situation, even though the residual build up is expected to increase as the plant ages, the SG flow resistance is noted to be relatively constant. This can be an indication there is no steady residual build-up in the SG or that the flow resistance increase due to increased roughness in the SG tubes is compensated by a flow resistance reduction due to SG divider plate leakage. Once the divider plates are replaced with a non-leaking type, the flow resistance increases as shown in Figure 4, since flow by-passing the SG tubes is eliminated or significantly reduced. Similarly, once the SG tubes are cleaned, i.e. less residual build up, the flow passing through the SG tubes increases due to lower flow resistance in the SG. In both cases, where SG divider plate is replaced or the SG is cleaned, the flow resistance expresses a physical change in the SG.

Similar to the SG flow resistance defined in Figure 4 a reactor core pass flow resistance can be defined by

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

Where ΔP_{hh} is the header to header differential pressure across the reactor core and Q_{hh} is the pass flow rate. Considering Equation (1), for single phase flow conditions, the flow resistance of the reactor core is practically not affected by a SG hydraulic changes. Therefore, a decrease in flow due to divider plate replacement will necessarily also decrease the header to header differential pressure across the core and an increase in flow due to SG cleaning will necessarily increase the header to header differential pressure as observed in Figure 5. SG aging, therefore, directly affects the header boundary conditions of header to header differential pressure as well as inlet header temperature (Figure 3) of the reactor core NUCIRC hydraulic model (NUCIRC ITYPE 2 model).

The aging behaviour of the below header geometry can, in part, be expressed in terms of the header to header differential pressure, as shown in Figure 5. As the plant ages, residual build up along the feeders is expected to increase. Consequently, the header to header differential pressure is also expected to increase as seen in Figure 5. However, it is noted that PT diametral creep reduces the reactor core flow resistance and consequently, as per Equation (1), also tends to reduce the header to header differential pressure as seen in Figure 5 as well, late in plant life.

4.2. Header manifold model in NUCIRC code

The header manifold model included in the circuit simulation option of NUCIRC (ITYPE 6) is a very unique capability as compared to any other code. This model is not only used to simulate the axial pressure gradients in the inlet and outlet headers but it is also used to calculate the recoverable and non-recoverable loss coefficients of the entrance and exit nozzles corresponding to the axial and lateral flow distributions simulated in the inlet and outlet headers. Site data offers pressure measurements at some locations along the inlet and outlet headers as seen in Figure 6. Such data is not sufficient to provide detailed description of the axial pressure gradient along the headers, which in turn affects the channel differential pressure and consequently affects the channel flow distribution. The use of the header manifold model results in a more accurate axial pressure gradient, as seen in Figure 7, which in return leads to accurate prediction of the radial flow distribution in the core. The result of this model is further confirmed by comparison to pressure and flow distribution measurements. The code therefore, makes it possible to correlate measurements with channel specific performance. This is a necessary requirement for improved analysis agreement with actual site performance.

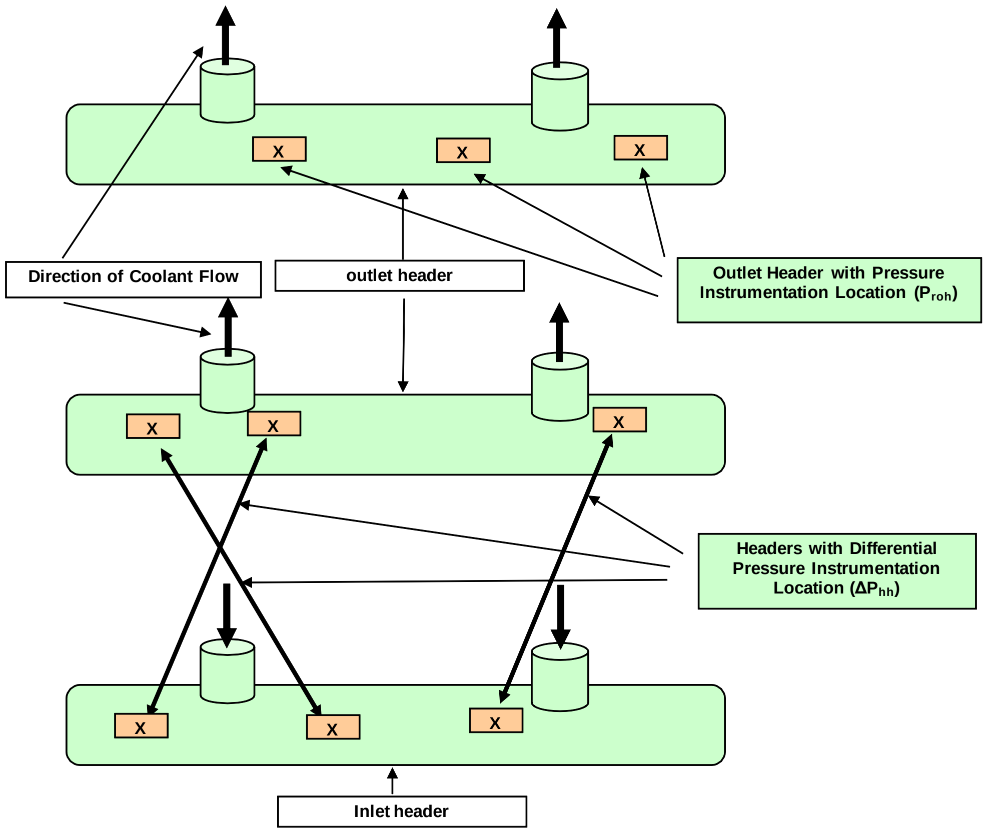

Figure 6 A typical header pressure measurement methodology of a CANDU plant

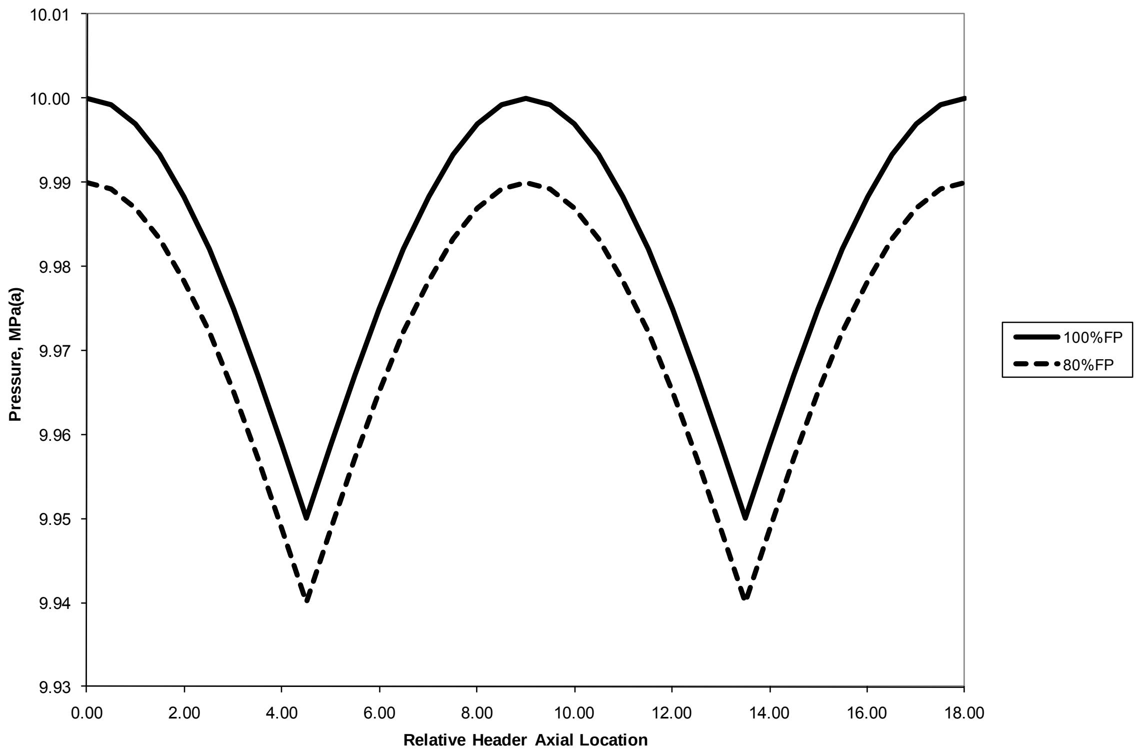

Figure 7 A typical outlet header pressure distribution of a CANDU plant

4.3. Pressure tube diametral creep model

In an iterative process between the RC-1980 based code [**6**] and the ITYPE 2 option of the NUCIRC code, the PT diametral creep profile of the pressure tube can be derived to reflect the aging of the pressure tubes. Using the output of ITYPE 2 simulations in terms of fuel bundle specific conditions and flow distribution profile, the RC-1980 based code calculates the creep profile for each point in time at each fuel bundle location in the reactor core, where hydraulic or CCP analyses are required. Further the code generated results are scaled to the measured site data as seen in Figure 8. The obtained PT creep profile is then applied as an input to ITYPE 2 and ITYPE 6 options of the NUCIRC code.

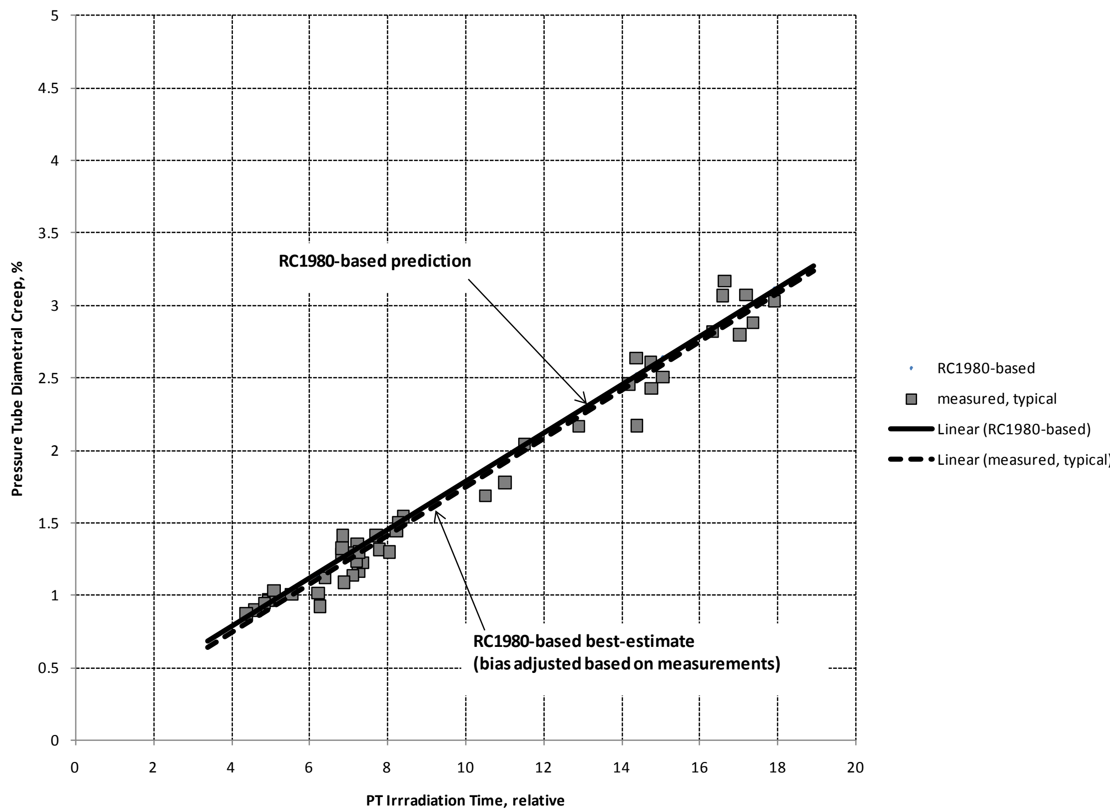

Figure 8 A typical Pressure Tube Diametral Creep measurements VS. Prediction in a CANDU Plant

5. Conclusion

NUCIRC code is mainly used for the CCP calculation required for the ROP analysis during the operational stage as a means to monitor and adjust the Trip Set Point (TSP) required for compliance with prevention of intermittent fuel dryout during normal and non-normal operating scenarios. However, the fundamental ability of the NUCIRC code is to predict the coolant conditions in terms of pressure gradients, flow rate, temperature and quality at any location of the primary HTS. This capability does not only contribute to the CCP calculations, but also can be used to build other thermal-hydraulic models required for the different disciplines in design and operational analyses.

Because of the different options which NUCIRC offers, detailed aging trends for all the HTS circuit components can be tracked and modeled accurately to match the collected site data. The header manifold model, the fuel bundle pressure drop model and the pressure tube diametral creep model are among the unique capabilities of NUCIRC compared to any other existing thermal-hydraulic codes. Through these models, detailed pressure and temperature distributions along the axial direction of the headers and along the PT, per bundle location, are achieved. These properties are further used to yield the PT diametral creep models through RC-1980 based codes and to analyze the integrity of fuel bundles, feeders, and fuel channel pressure tubes among other qualities.

6. 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 development of the NUCIRC code, understanding of the operational behaviour of reactors and/or to implementations of various aspects of the CANDU program, specifically D. J. Wallace the most recent NUCIRC code developer deserves special mentioning. Special appreciation goes to Peter White of AECL for his contributions to this paper.

7. References

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