Value Addition Initiatives for CANDU® Reactor Operation Performance

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ABSTRACT

Recently, AMEC NSS initiated projects for CANDU® station performance engineering with potentially high returns for the utilities. This paper discusses three initiatives. Firstly, optimization of instrument calibration interval from 1 to 3 years will reduce time commitments on the maintenance resources on top of financial savings ~$3,500 per instrument. Secondly, Reactor thermal power uncertainty assessment shows the level of operation which is believed to have an over-conservative margin that can be used to increase power by up to 0.75%. Finally, as an alternative means for controlling Reactor Inlet Header Temperature (RIHT), physical modifications to the High Pressure (HP) feedwater heaters can be useful for partially recovering RIHT resulting in increased production by 10-12 MWe.

Key Words: Calibration, Feedwater, Heat Balance, Instrument, IUC, Maintenance, Operation, Optimization, Performance, PFU, SOE, Uncertainty.

1. Instrument Calibration Frequency Optimization

Most instruments in a nuclear power plant are calibrated at regular intervals to ensure that the assumptions in the plant Technical Specifications and/or Safe Operating Envelope (SOE) compliance limits (e.g., As-Found Tolerance) are satisfied. In the Instrument Uncertainty Calculations (IUC), As-Found Tolerance for instrument drift is estimated based on statistical analysis of As-Found (AF) and As-Left (AL) calibration data such as that carried out for Bruce NGS by the Electric Power Research Institute (EPRI) in 1998.

The use of statistical analysis techniques to evaluate instrument drift based on calibration history and implementation of on-line instrument monitoring has enabled some Pressurized Water Reactor (PWR) nuclear plants to extend calibration intervals and, thus, move from an 18-month fuel cycle to a 24-month fuel cycle, since calibration of most instruments can be done only when the reactor is shut down.

Instrument calibration at CANDU® Nuclear Generating Station (NGS) is not tied to unit outages, and, therefore, annual savings are likely to be significantly lower. However, in addition to the actual savings, reduction in calibration frequency will reduce time commitments on the part of authorized nuclear operators (ANOs) and safety system qualified control maintenance (CM) staff, and will allow more schedule flexibility.

1.1 SOE Assessment for Emergency Coolant Injection (ECI) Instrument Loops

As mentioned before EPRI conducted a study in 1998 to quantify instrument drift behaviour [1, 2]. The purpose of the EPRI study was to evaluate the expected performance of the instruments as installed and maintained in the station. More specifically, the study focused on determining a statistically derived tolerance interval for instrumentation drift with a 95% probability and 95% confidence by using historical instrument AF and AL calibration data collected from Bruce B.

The EPRI study included a large amount of AF/AL calibration data i.e. 7636 calibrations for 1924 instruments. The extensive data quantity and coverage ensured that the quantified drift behaviour had a sound statistical basis.

In the EPRI study, instrument drift is quantified as the difference between the AF output from the current calibration and the AL output from the last calibration, i.e.:

Drift = AF Output current calibration – AL Output last calibration

The statistics for the drift data (mean and 95/95 tolerance interval) were calculated and tabulated for different instrument type, application, make, model, and calibration points. The EPRI drift study is currently used as the basis for specifying drift uncertainties in the SOE IUC.

Instrument Uncertainty Calculations

IUCs are performed on two types of SOE parameters: actuation and indication parameters. Actuation parameters are those that will trigger an automatic actuation when monitored signal (e.g., reactor power) exceeds the set point of Special Safety Systems following a design basis accident. Indication parameters are those that require routine monitoring (e.g., reactor inlet header temperature) during normal operation.

In IUC, Total Uncertainty (TU) is calculated for both actuation and indication parameters. The TU includes random and bias uncertainty components from various sources, including instrument drift. Consistent with the industry practice, the TU is evaluated at 95% probability with 95% confidence (95/95). An important assumption for the 95/95 uncertainty evaluation is that random instrument uncertainty, including drift, is normally distributed. In the EPRI instrument drift study, the normality assumption is checked for the drift data. It is concluded that drift data distribution is more peaked than an ideal normal distribution, and is therefore bounded by the normality assumption. Random instrument drift beyond 3σ is highly unlikely to occur. If it does occur, a non-random failure mechanism may be the cause for the drift and a repair or replacement would be recommended.

Actuation Loops

In order to ensure compliance with SOE, the setpoint design criterion for an actuation loop is that the Total Allowance (TA) must be greater than or equal to the TU, where TA is the difference between the Safety Limit (SL) and the nominal setpoint, and TU is the combination of all loop uncertainties (random and bias):

Margin = TA - TU $\geq$ 0

A zero margin is the minimum that the loop is still in compliance with SOE (See Figure 1). Therefore, a sufficient margin (to accommodate for potential instrument drift due to extended calibration period) for any given loop, is used as a selection criterion to identify loops that are qualified for calibration interval extension.

Indication Loops

For an indication parameter, the SOE compliance is defined by Surveillance Limit which equals to the Safety Limit minus (or plus) the Total Uncertainty, depending on how the Safety Limit is defined (high or low) relative to the normal operating point range, i.e.

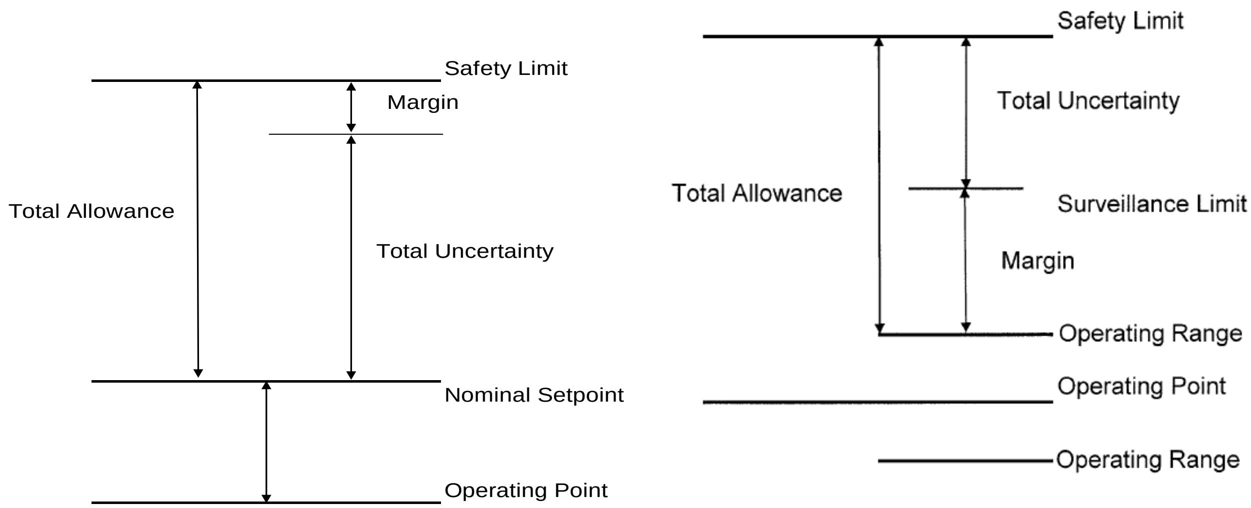

Figure 1&2: SOE Compliance for an Actuation (L) & Indication (R) Parameter

Surveillance Limit = Safety Limit ± Total Uncertainty

To ensure compliance within SOE, an indication loop should have sufficient allowance to accommodate instrument uncertainty, such that at a minimum the surveillance limit does not intercept with normal operating range (See Figure 2). Similar to the criterion established for actuation loops, if the margin for an indication loop is significant to accommodate potentially increased instrument drift (3σ) due to extended calibration interval, then the instrument would be qualified for a calibration interval extension of up to 3 years.

1.2 Impact on ECI Unavailability from Reduced Instrumentation Calibration Frequency

The methodology employed for the sensitivity analysis of the effect of the reduced calibration intervals on ECI system unavailability consists of the following steps:

(a) Solving the latest ECI unavailability model (UM) using baseline failure rates and a 3-year test interval (TI)

(b) Identification of prior drift failures and additional instrumentation failure events, and associated pre-defined maintenance (PMIDs) in the ECI UM

(c) Drift failure probability re-estimation for the two sensitivity cases

(d) Simple human interaction event probability re-calculations

(e) Solving the ECI UM

(f) Impact on severe core damage frequency

Only the loops deemed eligible for the calibration interval extension by the SOE assessment were considered for the sensitivity analysis.

The impact of reducing the calibration frequency on ECI system unavailability for SOE related instruments was assessed by revising the calibration interval to three years and re-calculating ECI system Predicted Future Unavailability (PFU).

2. Reactor Thermal Power Optimization

At Bruce Power, and for reactors operated by other utilities in Canada, the licensed reactor power limit is specified by the Canadian Nuclear Safety Commission (CNSC). The licence limit is established based on a design value and the safety limit is a value used in the safety analysis. The difference between the two is based on our understanding of the uncertainties associated with any selected level of reactor power. The licensed limit for each reactor is about 3%FP lower than the safety limit established for that reactor based on safety analysis work. It has resulted in a level of operation which is believed to be overly conservative in that it forces the reactors to operate at power levels that are lower than they could be, while still ensuring safe operation. The target "Limit" desired by Bruce Power is the safety limit. .

This requires establishing the resolution of two issues:

- Deriving mathematical models to fully describe the behavior of the true reactor power. The models need to be consistent with operating data and assessments of reactor thermal power uncertainties. Included in this work would be a determination on whether the current value of the safety limit is adequately supported, since a firm value for this limit will be needed for any compliance work.

- Using the developed models to support justification of any proposed change in reactor power level. Justification will need to demonstrate meeting all the requirements associated with the proposed change, including any changes to existing methods and procedures, such as changes to the way reactor thermal power is calibrated and measured.

This initiative considers the first issue and presents the development of a mathematical model for reactor power and compliance. This is based on the recent modeling experience in channel power compliance and Neutron Overpower Protection (NOP) work. The model is then used to analyze historical operating data for reactor thermal power for Bruce Power units 5 and 8 reactors, over a five year period from 2006 to 2010**.** Model equations have been solved using maximum likelihood estimation, and preliminary results indicate that for the reactor power set point at 93%FP, the true reactor power is below 94.5%, with 95% confidence, or 1.5% less than the safety limit over the entire period of time. The model is currently being applied to analyze the behavior of the reactor thermal power in Bruce Power, units 6 and 7, and can be used for other CANDU® plants.

2.1 Reactor Thermal Power Measurements and Compliance

2.1.1 Methods of Reactor Power Calculation

It is well known that fissioning of uranium and plutonium nuclides within the core releases quantities of energy which have been well characterized over the past 60 years. The fission energy appears in various forms: kinetic energy of fission fragments, kinetic energy of neutrons, energy in the form of gammas and other radiation in the electromagnetic spectrum, energy carried by neutrinos, and energy carried by beta and other particles.

It is also well understood that not all these energy carriers result in enthalpy increases in the primary heat transport system coolant. Some are lost completely (e.g. the neutrinos). Some are deposited in the moderator (e.g. gammas and neutrons), in structural materials, in end shields, in biological shield (gammas and neutrons), and in structures outside the core altogether, such as containment equipment and containment walls (primarily gammas). Any decision on which of these processes to include in and to exclude from the definition of reactor power will affect how we interpret a model statement of "reactor power" and how well we can reproduce, through calculations, a physical measure of reactor power.

There are only two practical means of measuring reactor power that are used in the industry. They are measurement of primary side reactor power based on the Reactor Regulating System (RRS) algorithm, also referred to as reactor "indicated thermal" power, and measurement of reactor power as obtained by the secondary side heat balance. These two measurements reflect the third indicator, which is the postulated "true" reactor power, and it is an established practice in most nuclear plants around the world to calibrate the indicated reactor power to the value obtained from the secondary side heat balance. Although the secondary side heat balance is considered the best and most accurate measurement of the reactor thermal power, it excludes some energy loss from the core that does not pass to the secondary side. Other reactor side measurements must therefore be used to take into account all significant energy leaving the core, and added to the secondary side heat balance.

2.1.2 Reactor Power Compliance

The word "compliance" normally implies comparison of a measured or calculated value to a standard to determine whether or not a specific criterion has been met.

The Power Reactor Operating Licence (PROL) for a CANDU reactor states that

- the total power generated in any fuel bundle shall not exceed P_b (max bundle power)
- the total power generated in any fuel channel shall not exceed P_{ch} (max channel power)
- the total thermal power from reactor fuel shall not exceed P_{th} (max thermal power)

However, there is an inconsistency between the current compliance approach for P_{th} compared to P_b and P_{ch}. The values for P_b and P_{ch} specified in the PROL are the same as in the Safety Analysis. But the value for P_{th} is equal to design reactor thermal power, which is about 3% lower. Implementation of a modified compliance approach will not only result in more consistent compliance practices but will also allow increase in reactor power operating levels.

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2.1.4 Equations for Estimating True Reactor Power

Definitions and Measurement Errors

For the work being discussed here, the entity of interest (reactor power) has to do with processes occurring within the core. As discussed in the previous section, there are various measures for quantifying this entity, but no two of them look at exactly the same set of processes, and all of them have associated errors. A good model can, in principle, encompass all these processes and measurements and can be used to make estimates of the associated errors. Let us introduce the following definitions:

True Reactor Power (‘un-observable’): RP

Indicated Reactor Power (based on Primary Heat Transport System):

$RP^{ind} = RP\,(1 + e^{rp})$ where, e = estimated error (1)

Heat Balance Reactor Power (based on Secondary Heat Transport System):

$RP^{hb} = RP\,(1 + e^{hb})$ (2)

RP^{ind} is controlled to set point (SP):

$RP^{ind} = RP\,(1 + e^{rp}) \approx SP$ or $RP = SP\,(1 - e^{rp})$ (3)

and is based on Fully INstrumented CHannel (FINCH) measurements.

The “≈” sign reflects the fact that RP^{ind} fluctuates around the SP; however, these fluctuations are small and can be neglected in the derivation of the error equations.

$RP^{ind} = (1/f) \sum CP_i^{m} / N_i$ (4)

where CP_i^{m} are the measured FINCH powers, N_i are the called FINCH nominals and f is the number of FINCH’s.

FINCH measurement error is then given by

$CP_i^{m} = CP_i\,(1 + e^{m})$ (5)

Here we introduce “d”, which is referred to as the so-called Channel Power Drift and can be also expressed as

$ave(CP) / ave(CP_0) = (RP / RP_0)\,(1 + d)$ (6)

CP_0 and RP_0 are taken at time zero, when changes in nominals take place and therefore the drift is measured, and CP and RP are taken at a later time. If all reactor channels were FINCH’s, then d would be equal to zero.

In this case the expression for the true reactor power can be written as

$RP = SP\,(1 - e^{rp}) = SP\,(1 + b_0 - d - e^{m})$ (7)

b_0 is the average error in FINCH nominals. This expression is valid only on a time interval with constant FINCH nominals. A description of RP across the time of nominal change is obtained by modeling station procedures when changing nominals, which results in a constraint on b_0.

Regression Model

Heat Balance data:

$RP^{hb} = RP(1 + e^{hb}) = SP(1 + b_0 - d - e^m)(1 + e^{hb})$ Or (8)

$(RP^{hb} / SP - 1) = (b_0 - d) + (e^{hb} - e^m)$ (9)

SORO (Simulation Of Reactor Operation) data:

$S = CP(1 + e^{rp} + e^{rfs})$, $RP^{ind} = RP(1 + e^{rp})$ (10)

Additional equations can be obtained for b_0 and d by replacing CP with S and RP with $RP^{ind.}$

Solution of Regression Model

Solve Equations (8) – (10) using Maximum Likelihood Estimation [3]:

b^{est}, d^{est}, $\sigma^2 = (\sigma_m)^2 + (\sigma_{hb})^2$, where, σ = standard deviation

$RP^{est} = SP(1 + b^{est} - d^{est} - e^m)$ (11)

Confidence Limit (RP) $= SP(1 + b^{est} - d^{est} \pm t_{n,a}[(\sigma_{est})^2 + (\sigma_m)^2]^{1/2})$ (12)

RP^{ind} and HB (Heat Balance) data for Units 5 and 8 over a 5 year period (2006-2010) were used

Unit 5: SP = 90% (2006 – 2007), SP = 93% (2008 – 2010)

Unit 8: SP = 90% (2006 – 2009), SP = 93% (2010 – 2010)

The estimate of the mean true RP and 95% bounds on the true RP were performed for this period.

2.2 Reactor Power Uncertainty Analysis Results

Reactor power obtained from Heat Balance and the estimated true reactor power are shown in Figures 3&4. Comparison between the sets of results confirms that the true reactor power is not sensitive to the random error in the value of the Reactor Power obtained from Heat Balance. A possible systematic error is normally addressed by regular verification and calibration of station instruments such as feedwater flow and temperature against more accurate methods of measuring the same parameters. Finally, it is important to emphasize that, at 95% confidence level, the true reactor was within about 1% of the license limit and never exceeded the safety limit.

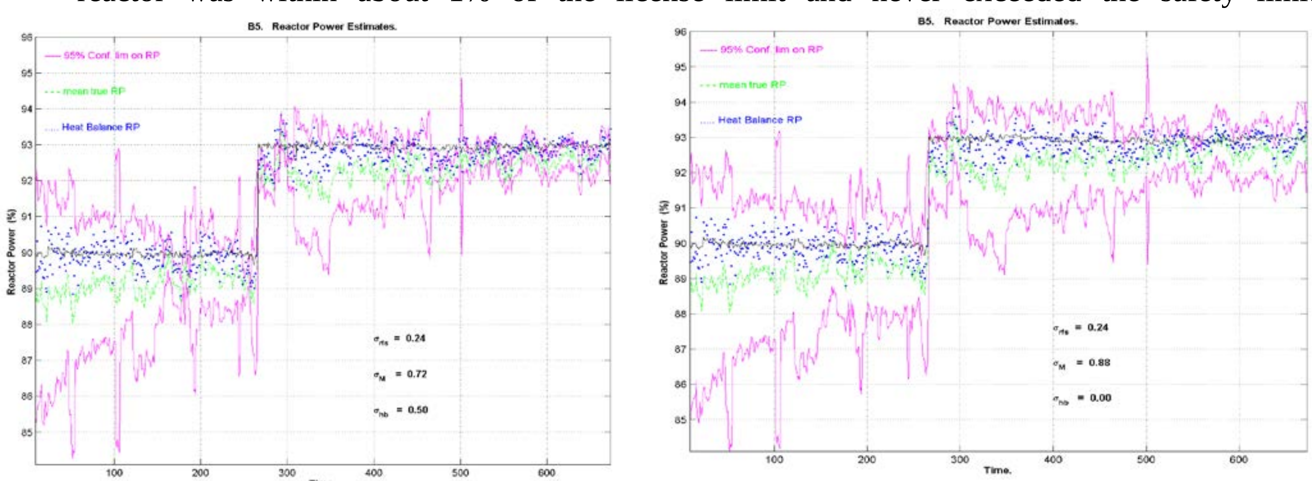

Figure 3&4: Reactor Power for 2006-2010 for σ_{HB}=0 & σ_{HB}=0.5

3. Minor Design Changes for Controlling Reactor Inlet Header Temperatures in CANDU® Reactors

In general, the Reactor Inlet Header Temperature (RIHT) has risen more rapidly in CANDU units, compared to the original aging predictions. A typical example of the RIHT behaviour in recent years is shown in Figure 5. In this example, the RIHT increases by approximately 0.3°C/year. Thus, RIHT are monitored and high temperature alarms are required to prevent operation outside the safe operating envelope as supported by the safety analysis.

To prevent RIHT alarms, increasing RIHT are being mitigated through changes in unit operating conditions. Boiler secondary side pressures have been lowered and high pressure (HP) feedwater (FW) heater steam supply has been isolated, often leading to electrical output loss.

The main contributor to the RIHT increase is the ongoing accumulation of magnetite deposits on the Inner Diameter (ID) surfaces of both the boiler and preheater tubes.

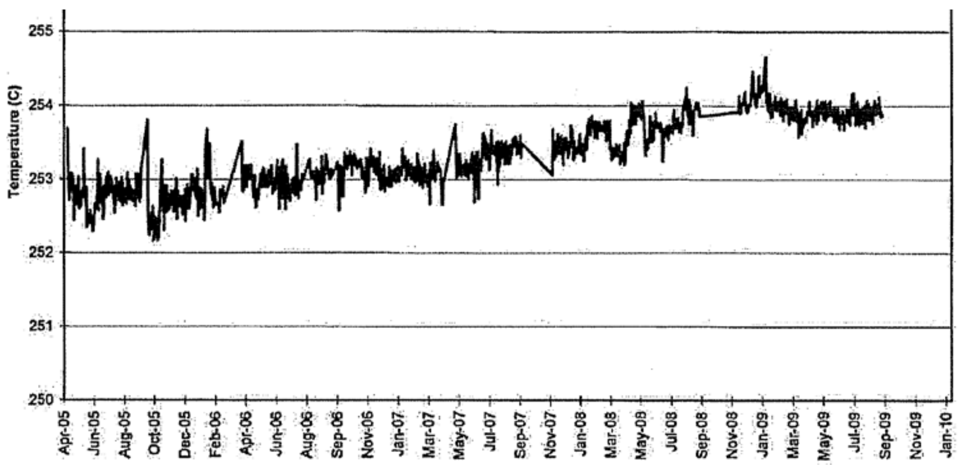

Figure 5: Typical RIHT Trend with Time

The increasing RIHT trends are reviewed annually, and corrective actions such as ID cleaning of the boiler and preheater tubes can be performed to recover lost heat transfer efficiency. However, the execution of ID boiler tube cleaning has not always been successful in providing long-term relief from high RIH temperature alarms. In some cases RIHT have returned to pre-cleaning levels within as little as three years of operation. Repeat ID boiler tube cleaning is a very expensive means for controlling RIHT. Minor design changes may provide a cost effective means of controlling RIHT as an alternative to ID boiler tube cleaning, and it may also be a cost effective approach to maximize production prior to refurbishment outages. The objective of the work presented in this paper was to evaluate two potential minor design changes in the high pressure feedwater (HP FW) system for the purpose of achieving lower RIHT by reducing preheater / steam generator feedwater inlet temperatures.

3.1 Feed Water (FW) Flow Bypass of High Pressure (HP) FW Heater Tube Bundle

In a typical CANDU turbine cycle, the HP FW heaters increase the FW temperature from around 150°C after leaving the boiler feed pumps to about 175°C before entering the shell side of the preheaters / steam generators. A 10 °C drop in FW temperature will result in a RIHT drop ranging from 0.5 to 1 °C depending on the preheater/steam generator design. The heating supply to the HP FW heaters is a normally a combination of extraction steam from the HP turbine outlet and drain flow from the moisture separator/reheaters (MSR). A path for the MSR drains must be maintained at high power operation and therefore if the HP heaters are not available to accept the drains, the flow must be diverted to the condenser resulting in high thermal losses. The extraction steam flow to the HP heater however can be reduced or isolated in such a way that the steam remains in the steam path minimizing thermal losses. Therefore, in order to reduce HP FW heating with the least impact on overall cycle efficiency, normal MSR drain paths must be maintained while reducing the extraction flow to the heaters. In some plants isolating extraction steam to the HP heaters while maintaining normal drain paths is possible. In the latter case, minor design changes could be a better approach for controlling RIHT.

The first minor design change considered was to provide a means of bypassing a portion of the FW flow around the tube side of the HP FW heater, as shown in Figure 6. The objective is to manage high RIH temperatures by reducing FW temperatures while minimizing trade-off in lower cycle efficiency.

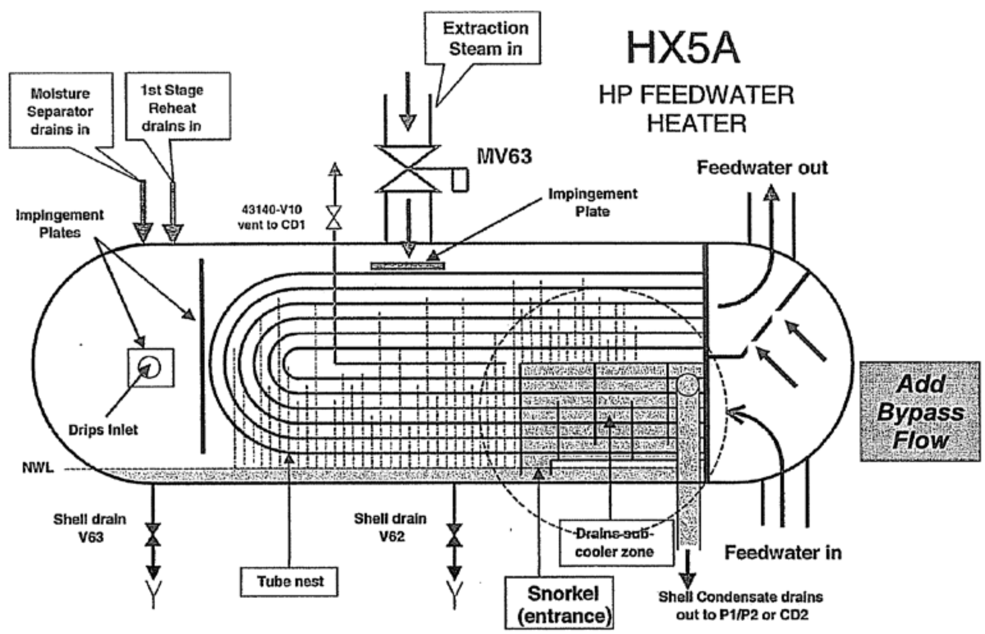

Figure 6: FW bypass flow of HP FW heater tube bundle

The feasibility of the design change was studied using the Performance Evaluation of Power System Efficiencies (PEPSE) turbine cycle modelling software [4] to simulate this minor design change. Due to the complexity of modeling moisture separator drain piping, two bounding cases were simulated with respect to drain flow response to changing conditions in the HP FW heater and impact on cycle efficiency as Best Case and Worst Case. The Best

Case assumes that drain flows remain unchanged either by control action or compensating changes in the drain line water column. The Worst Case assumes no control action or change in the drain line water column. In this case, reheating steam load is expected to increase and super heat to the LP turbines is expected to drop resulting in increased cycle efficiency losses. Two sensitivity studies were therefore carried out with varying degrees of FW bypass flow from 0% to 50% of the total FW flow under Worst Case and Best Case conditions.

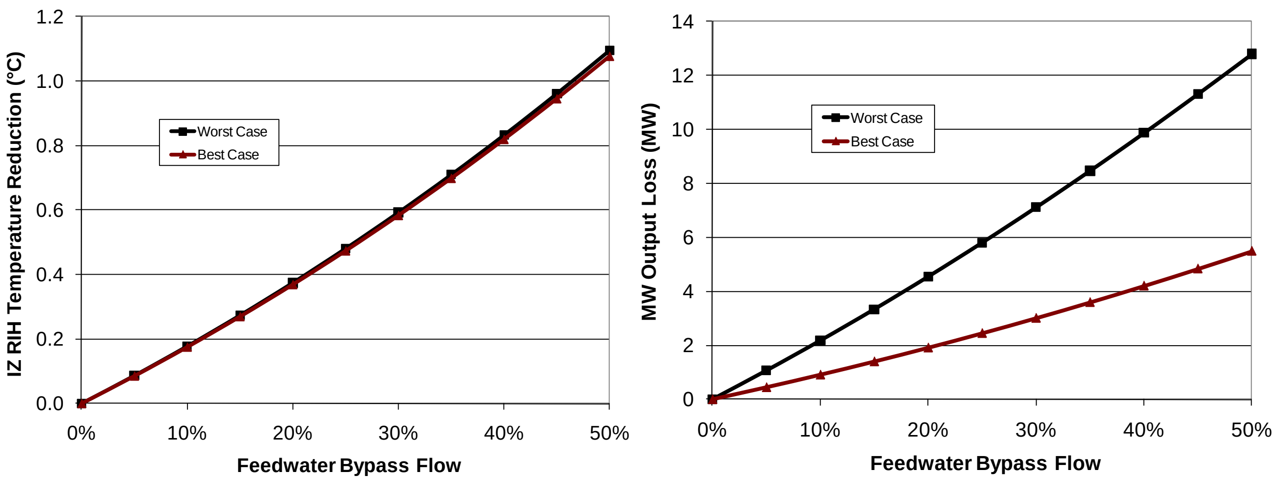

Figure 7&8 RIHT Drop & Gross Electrical Output Loss due to FW Bypass Flow

For one case studied, the trends of the RIH temperature drop and MW output loss due to increased FW bypass flow are shown in Figures 7&8. If there is 50% FW flow bypassing the HP heater tube bundle, the RIH temperature will drop by about 1.1°C. However, the gross electrical output will also be reduced by between 5.5 MW and 12.8 MW depending on the overall impact on cycle efficiency.

It was noted that the worst case production losses are similar in magnitude to the production losses incurred using boiler pressure reduction when turbine governor valves are at maximum opening which is typical of units combating high RIH temperatures. However it was also observed that by reducing feedwater temperatures, less steam is generated in the boilers allowing additional boiler pressure reduction without additional production losses and essentially doubling the achievable RIH temperature reduction. In the one example studied, by combining a 25% flow bypass with additional boiler pressure reduction, a 1.0°C reduction was achieved with a 4 to 6 MW output loss compared to a 16 MW output loss using boiler pressure reduction alone.

The design envelope for the HP heaters must be considered as part of any design change in the plant. The HP heater shell pressure will be increased due to FW bypass flow, but is expected to be well within the shell design pressure. The FW temperature at the exit of the HP heater tube bundle will also increase and in one case studied the tube design temperature was exceeded if there is more than 30% bypass flow. In one worst case study, with more than 40% bypass flow, the reheater drains flow increased beyond the reheater drains pump performance capacity.

3.2 Partial flooding of HP FW Heater Shell Side

Shell side flooding of the HP FW heaters is an undesirable condition from the thermal performance point of view, but it can provide another means of reducing final FW temperatures for controlling RIH temperatures. As more and more tubes are covered in the horizontal heaters, as shown in Figure 9, the condensing area of the tube bundle is reduced, resulting in less heat transfer and increasing shell side pressure. This option may be effective alone or in combination with a fixed FW bypass flow providing a means of fine tuning final FW temperatures through changes in the heater operating levels.

The PEPSE turbine cycle model was used again to simulate this design changes and the aforementioned Worst Case and Best Case conditions were also applicable. Two sensitivity studies were carried out with varying amounts of tubing length assigned to the condensing section from 90% (normal operating state) to 70% of the total tubing length of the HP heaters under Worst Case and Best Case conditions, respectively. Structural considerations within the heater shell limited the reduction in condensing tube area to 70%.

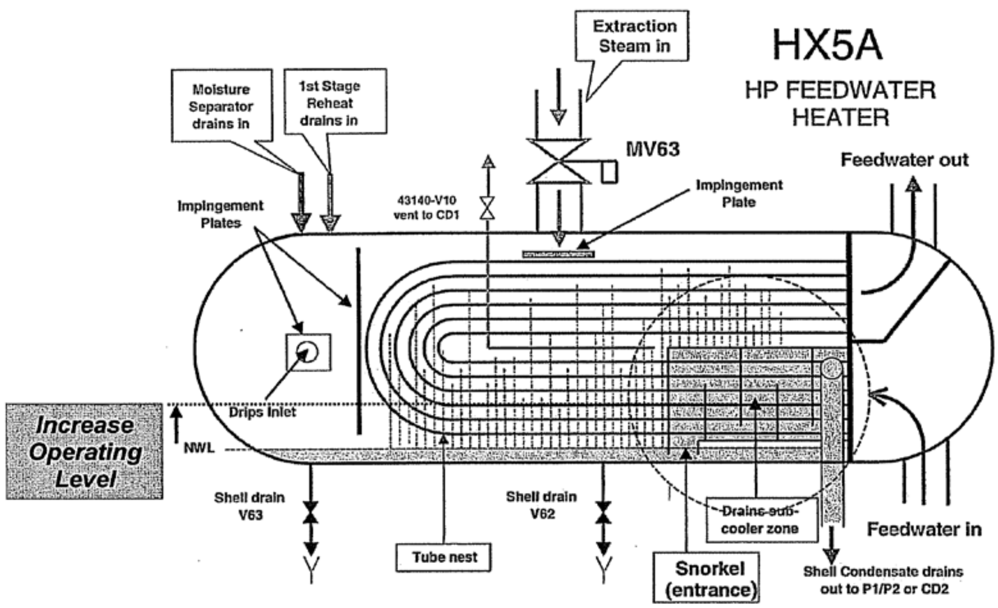

Figure 9: Partial flooding of HP FW heater shell side

The trends of the RIH temperature drop and MW output loss due to increased flooding are shown in Figures 10&11. If the tubing length assigned to the condensing section is reduced from 90% (normal operating state) to 70%, i.e., the flooded tube length is increased from 10% to 30%, the RIH temperature will drop by about 0.17°C. However, the gross electrical output will also be reduced by between 0.8 MW and 2.3 MW. Since the turbine inlet steam flow will also drop as the condensing tube length is reduced, as discussed previously, additional RIH temperature drop can be achieved for the same amount of flooding and MW loss. The HP heater shell pressure and reheater drains flow will be increased due to flooding, but are still expected to be below the shell design pressure and the reheater drains pump performance capacity.

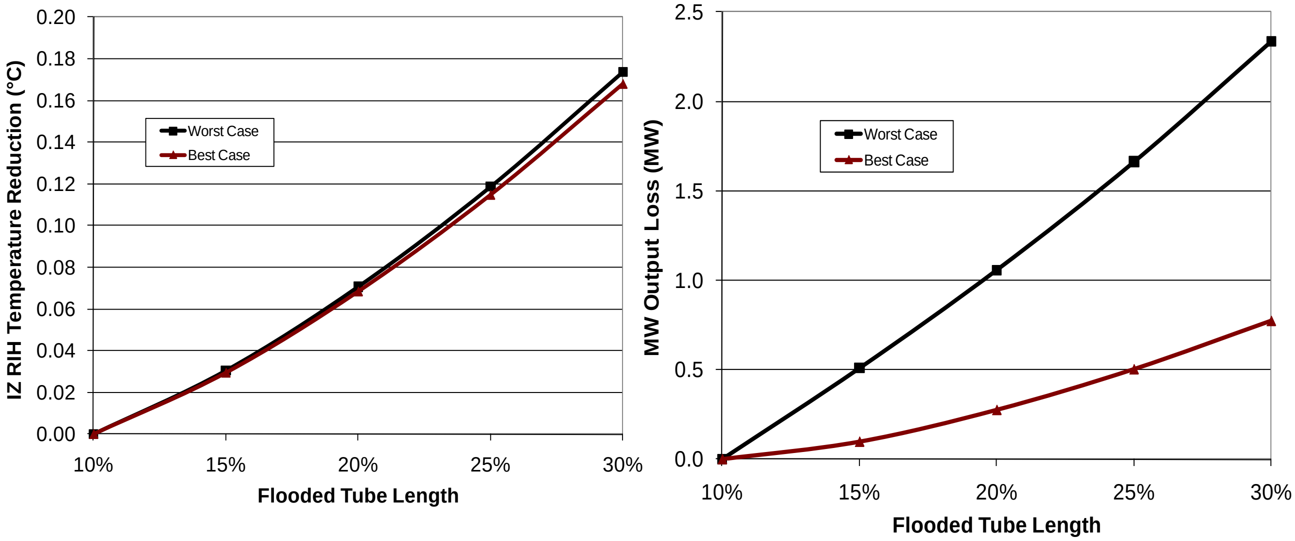

Figure 10&11: RIHT Drop & Electrical Output Loss due to Partial Flooding

4. Conclusions

4.1 Instrument Calibration Interval Optimization

To establish the proof of concept, As-Left/As-Found tolerances and available margins have been evaluated for the Bruce B Emergency Coolant Injection system instrument loops. This determines whether an extension of the current calibration period from one or two years to be extended to three years is justifiable. The incremental risk of non-compliance with SOE is small and is acceptable. Out of a total of 277 instruments that were analyzed, 166 instruments (60%) are qualified for calibration interval extension up to three years. Sensitivity assessment of the effect of proposed changes in calibration intervals for 60% of the instruments on the ECI system unavailability has also been performed using the current Bruce Power ECI unavailability model. The results are shown in Table 1 below. BB ECI alone can result in ~$150k/year savings and maintenance scheduling flexibility at a fraction of cost.

Table 1: Bruce B ECI PFUs

Baseline ECI PFU and Sensitivity Cases (Limit 1.00E-03) (For Instruments Eligible for Calibration Interval Extension by SOE Assessment)			
Accident Scenario	PFU at TI=1 Year (2010 Failure Rates)	PFU at TI=3 Years	
		2010 Failure Rates	2011 Failure Rates
Large LOCA	7.8E-04	8.8E-04	7.71E-04
Very Small LOCA	4.2E-04	5.3E-04	4.55E-04
In-core LOCA	7.9E-04	9.2E-04	7.51E-04

4.2 Reactor Thermal Power Optimization

A statistical model for reactor power compliance has been developed and applied to Bruce Power Units 5 and 8. The results are promising showing feasibility of the model since it fits data well

- Preliminary results indicate that, with 95% confidence, we can state that the true reactor power is below 94.5% over the entire period of time when Set Power = 93%.
- The robustness of the methodology is demonstrated by the fact that the average "sigma" estimates for Unit 5 and 8 are the same.
- 0.5-0.75% of additional power production is predicted to be available once this study is completed.

4.3 Minor Design Changes for Controlling Reactor Inlet Header Temperatures in CANDU® Reactors

Based on the study findings, the design change feasibility is ranked high for HP FW heater bypass flow modification but marginal for HP FW heater level increase due to the significantly lower impact on RIH temperatures. The physical modification to the HP heater can be as simple as drilling holes in the channel partition plate to provide a bypass path for the flow or new bypass piping and valving could be installed around the HP heater to provide a means of adjusting the bypass flow for optimum RIH temperature control. Implementation of the minor design modification can provide significant increases in production over using boiler pressure reduction alone and can also be used to extend operating periods between steam generator cleanings for optimizing outage planning. In the one example studied, by combining a 25% flow bypass with additional boiler pressure reduction, a 1.0°C reduction in RIHT was achieved with a 4 to 6 MWe output loss compared to a 16 MWe output loss using boiler pressure reduction alone.

5 References

[1]. EPRI Report TR-111348, "Instrument Drift Study: Bruce Nuclear Generating Station", November 1998.

[2]. EPRI Report 103335-R1, "Guidelines for Instrument Calibration Extension/Reduction -- Revision 1: Statistical Analysis of Instrument Calibration Data", October 1998.

[3]. Stuart , A. and Ord, J.K., "Kendall's Advanced Theory of Statistics", Vol. 2, Oxford University Press, 1991.

[4]. G.L. Minner, et al, "PEPSE Use Input Description, representing version 73 of PEPSE – Volume 1", SCIENTECH a Curtiss Wright Flow Control Company, August , 2008.

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