PLANT AGEING ADJUSTMENTS TO MAINTAIN REACTOR POWER AT THE POINT LEPREAU GENERATING STATION

W.J. Hartmann, M. Cormier

Atomic Energy of Canada Limited
2251 Speakman Drive
Mississauga , Ontario, L5K 1B2
Phone: (905) 823-9040

B. Willemsen, T. Whynot, P.D. Thompson

New Brunswick Power, PLGS

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ABSTRACT

This paper deals with a portion of the Point Lepreau ageing program that investigates the ageing behavior of the primary heat transport system (HTS) and its potential impact on the Critical Channel Power (CCP). It summarizes the current understanding and the actions taken to date to monitor, understand, correct and combat this aspect of plant ageing. It also shows that the program has led to improved analysis tools, enabling plant optimization as well as prediction of future requirements. The focus is on accurate tracking of ageing and periodic ageing adjustments such that safe reactor operation is ensured without unnecessary economic penalties.

1. INTRODUCTION

All industrial plants undergo changes with time and nuclear plants are no exception. The CANDU 6 reactor follows earlier CANDU designs such as NPD, Douglas Point, Pickering A, and Bruce A. As such, certain aspects of plant ageing were addressed directly in the design process. This was reflected in the selection of materials, design allowances and provisions, and operating margins. In addition, certain maintenance practices have been developed over time at the stations to address ageing issues. In terms of Special Safety System availability, Reliability Studies were performed to determine the required tests and frequency of testing to ensure that necessary targets were achieved. Nonetheless, it was not possible to precisely predict how the plants would change with time at the design stage. In order to ensure that such changes would not compromise public safety, a program was developed at Point Lepreau to provide a continued assurance of Nuclear Station Safety.

The program is based on the principles of monitoring/detecting, anticipating, understanding and then correcting or compensating. It has components involving R&D, analysis and assessment, maintenance, operational changes and design modifications. The program is risk-based, and grew from an initial ad-hoc series of activities into an integrated plan (Thompson, 1998). Much of the work on developing the understanding and strategies to combat the adverse effects of ageing has been a joint effort with Hydro Quebec, which operates a sister CANDU 6 plant (G-2), and with Atomic Energy Of Canada Ltd. (AECL), the designer.

This paper deals with the portion of the program that investigates the ageing behavior of the heat transport system (HTS) and its potential impact on channel flows and Critical Channel Power (CCP), one of the major components in the Regional Overpower Protection (ROP) system. Critical channel power is defined here as the channel power at which the fuel elements experience dryout conditions. In this manner, the margin to CCP can be tracked with time, and appropriate corrective actions can be determined to ensure that adequate margins continue to exist throughout the life of the station.

2. CANDU-6 REACTOR SYSTEM DESCRIPTION

Point Lepreau is a CANDU 6 PHWR. It has 380 horizontal fuel channels surrounded by a cool low pressure heavy water moderator. Each fuel channel is six meters long and contains twelve fuel bundles within a pressure tube. A bundle is made up of 37 elements which contain natural uranium in the form of compacted sintered cylindrical pellets of uranium dioxide (UO_2). Each channel has an End Fitting at each end, which allows for the Fueling Machines to attach to facilitate on power refueling. Coolant enters the channel from an inlet feeder pipe which is connected to the inlet end fitting. The coolant then enters the fuel string, flowing within the subchannels between the fuel elements inside the pressure tube. The coolant leaves the channel via the outlet feeder pipe which is attached to the outlet end fitting. The coolant enters the channel at about 11 MPa and 265 °C. It leaves slightly above 10 MPa and 310 °C.

When a channel is fueled, its power is increased and the coolant at the end of the channel contains D_2O vapor. As the fuel in the channel burns up, the channel power decreases, and the length of the boiling region decreases. When the power has reduced sufficiently, the channel drops out of boiling until it is refueled again.

Figure 1 shows a simplified overview of the HTS. The core is subdivided into two symmetrically located figure of eight loops. Each loop consists of two core passes of 95 channels each. Each pass contains a pump which feeds an inlet header which is connected to 95 inlet feeders which connect to the channels as described above. The 95 outlet feeders connect to an outlet header. Two riser pipes connect the outlet header to the hot leg side of a steam generator. Coolant flows through the vertical steam generator u-tubes to the steam generator outlet where it flows to the pump suction line of the pump in the other core pass in the loop. Because it is a figure of eight loop, flow in channels in one core pass is in the opposite direction to the other. As the channels in each core pass are laid out adjacently, it follows that the flow in adjacent channels are in opposite directions, forming a checker-board pattern. The two HTS loops are connected at each end of the reactor through the pressurizer interconnect line and the purification and feed interconnect lines. The pressurizer is connected to the discharge pipes of outlet headers 3 and 7. The purification feed flow is associated with inlet headers 2 and 6, while the purification return flow enters the HTS at the suction of pumps 1 and 3. The HTS also contains stability pipes which connect outlet headers 1 and 3 in loop 1, and outlet headers 5 and 7 in loop 2.

The intent of the reactor design was to have the flow rates in each channel proportional to power. Channels in the central region of the core have a higher time average power than those near the periphery, and hence require a higher flow. Required flows are achieved by use of different diameter feeder pipes, and through the use of flow restriction orifices in the inlet feeders of channels in the outer core region.

The design utilized the NUCIRC computer program together with an appropriate CCP model. NUCIRC models the various steady state homogeneous thermalhydraulic processes that take place in the HTS. Through the use of CHF correlations based on out-reactor testing, it also predicts the channel power at which fuel goes into dryout. The modeling associated with NUCIRC (Hartmann, 1987, Soulard, Hau, 1991, Soulard, Dam, et al, 1995, and Harvel, Soulard, 1996) has been improved over the years based on a cooperative program between NBP, Hydro Quebec and AECL. A further improvement was the generation of a CCP model based on 1995 cold and hot ultrasonic flow measurements and the latest Point Lepreau pressure tube creep measurements during 1998. This model was implemented at Point Lepreau in January 1999 (5000 EFPD). An update of this model including the extension of the CHF correlation to include 5.1% pressure tube diametral creep in 1999 is used in this analysis. This improved modeling incorporates the up-to-date important aspects of ageing for Point Lepreau.

3. INITIAL REACTOR CONDITIONS

For new reactors, component geometry and corresponding thermalhydraulic characteristics are well known from controlled laboratory tests and analysis producing accurate flow prediction capabilities. These flow predictions are then used in plant commissioning to demonstrate that design flow rates are consistent with measured plant coolant flow rates. Cold ultrasonic channel flow measurements are central to this demonstration. The accuracy of core channel power predictions can be validated using already validated flow rate predictions and rise in temperature measurements according to the following relationship:

$$P_i = \Delta H_i * W_i \qquad (1)$$

where P_i is the component power to or from the coolant for the HTS component "i", while ΔH_i is the corresponding change in enthalpy. For single phase coolant conditions, typically below 80% full power (FP), ΔH_i can be obtained directly from the measured temperature rise across the core or temperature drop across the steam generator. W_i is the component specific coolant flow rate. Equation (1) is generally used for inverse heat balance flow measurements when the reactor is operated at power levels in which the coolant fluid in channels and in outlet feeders are in single phase. In addition to inverse heat balance flow measurements, pass flow can be confirmed from differential pressure measured from the HTS pump suction to pump discharge by using an appropriate differential-pressure to flow correlation. The pump derived flow then has to be adjusted by minor peripheral flow to yield pass flow.

As the reactor ages, the component properties change (e.g. pipe roughness, orifice characteristics, fouling layers, pressure tube diameter, etc.). Furthermore, it is difficult and costly to obtain this information directly for all components. The characteristics of most components therefore have to be derived from the analysis of representative components as well as site measurements. Once components are characterized, generic models can be generated from laboratory tests. Of main importance here are nominal and high power coolant pressure drop correlations and critical heat flux (CHF) correlations for pressure tubes exhibiting typical diametral pressure tube creep and coolant flow conditions. The main measurement emphasis is on HTS circuit temperature, pressure and flow profiles in addition to fuel pressure tube diameter measurements. Inconsistencies in the three independent flow measurements have to be resolved to yield a best estimate flow. Once ageing trends are tracked with an appropriate neutronic and thermalhydraulic code set, future reactor performance can be estimated. Remedial actions can then be taken to ensure continued safe operation at full power.

4. POINT LEPREAU REACTOR AGEING DESCRIPTION

Subtle changes have occurred within the HTS at the Point Lepreau Generating Station (PLGS) over the 17 years the station has been in operation. Some of these effects are counter balancing. This makes the individual contributing mechanisms and effects more difficult to understand. The most visible change in the HTS is the increase in inlet header temperature that has taken place over time as shown in Figure 2. The PLGS program is aimed at understanding the mechanisms which lead to changes in channel flow and in critical channel power, and then taking action to ensure adequate safety margins are maintained. Close tracking of the principal hydraulic parameters in the HTS and appropriate CCP model adjustments are a critical part of the program.

The following is a list of the currently known ageing processes that are occurring within the HTS that can affect CCP:

- increase in pressure tube diameter due to irradiation creep. This reduces the hydraulic resistance in the channel, hence increases flows, but causes the coolant to preferentially bypass the interior subchannels of the bundle, reducing CCP. 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 inner channels. This mitigates the effect of pressure tube diametral creep on CCP for the central, ROP most limiting channels.

- increase in hydraulic resistance due to redistribution of iron in the HTS. Dissolution of iron and flow accelerated corrosion (FAC) is occurring in the outlet feeders. Iron is being removed from the outlet feeders and being redeposited in the cold part of the circuit, including the cold leg of the steam generators, the inlet feeders, and possibly the first section of the channel. 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 has a negative effect on core flow, on inlet header temperature and, consequently, on CCP.

- leakage across the divider plate in the steam generators. This was a more significant concern before the plates were replaced in 1995 with an all welded design. This leakage allows a portion of the flow to bypass the steam generators, resulting in reduced heat transfer but decreased pressure drop and increased flow for single phase (80%FP) conditions.

- erosion of the edges of flow reducing orifices. This can lead to a relative flow redistribution from inner to outer core.

Other potential ageing mechanisms, such as fouling on the external surface of steam generator tubes, and HTS pump impeller wear, have been shown to have negligible effects up to this point in time.

5. OVERVIEW OF HTS DATA AND ASSESSMENT

The main safety related HTS ageing indicators are associated with heat transfer and flow degradation. The most prominent indicator of heat transfer degradation has been inlet header temperature trends as shown in Figure 2. The most prominent indicators of flow degradation have been: measured coolant flow trends, as summarized in Figures 3 and 4, and HTS pressure profile trends, such as pump differential pressure trends and header to header differential pressure trends, the latter being shown in Figure 6.

Figure 2 shows a characteristic increase in inlet header temperature associated mainly with steam generator fouling and divider-plate leakage. At about 3400 Effective Full Power Days (EFPD) steam generator pressure was reduced, thus lowering the steam saturation temperature, which resulted in lowering the inlet header temperature. This was a temporary measure to reduce HTS temperature until work could be performed to replace the steam generator divider-plates, and to clean both the primary and secondary side of the steam generators which were undertaken during the 1995 outage (approximately 4200 EFPD). Following this corrective action, steam generator pressure was restored to its original setpoint, and HTS inlet header temperature returned to levels seen earlier at 2000 EFPD. At about 4970 EFPD steam generator pressure was reduced again to reduce inlet header temperature, resulting in increased operating margin allowing reactor power to be returned to full power as shown in Figure 12.

It has been recognized that feeder surface degradation due to iron transport will affect core flow distribution. Between 1982 and 1995 this effect resulted in an about 5% relative bulk core flow decrease (Thompson, 1998) as summarized in Figure 3. Figure 3 shows the single phase inverse heat balance flow calculation trend mostly at about 75% full power (FP) to 80%FP. It is the sum of all 380 individual channel flow predictions, typically done about 4 times each year, mainly for flow verification purposes. It includes flow uncertainties associated with thermal power calibration uncertainties. A trend assuming pressure tube diametral creep as the only ageing phenomenon, obtained from theoretical considerations (NUCIRC code simulations), is also shown for reference. At 4200 EFPD a 5% decrease in core bulk flow is therefore associated with general HTS degradation. The inverse heat balance measurement trends are well supported by cold and hot ultrasonic flow measurements at 0 EFPD and at 4200 EFPD. Figure 3 shows flow degradation predominantly between 0 EFPD and 2000 EFPD and again between 4200 EFPD and 4700 EFPD. It should be noted that high HTS-pH_a (about 10.8) approximately coincides with both periods of relative flow degradation. High pH is associated with high magnetite solubility. The change in trend from 2500 EFPD (about May 1990) to 4200 EFPD (December 1995) is mainly due to a combination of pressure tube diametral creep, steam generator divider-plate leakage, steam generator pressure reduction, steam generator cleaning, changes in HTS chemistry, and thermal power calibration. Similar arguments can be made for the period between 5000 EFPD (about January 1999) and 5500 EFPD (about August 2000).

Figure 3 also summarize the cold (40°C) ultrasonic flow measurements (RPC) taken in 1982 and 1995 for all channels and the cold and hot ultrasonic flow measurements (AMAG) taken in 1995 for selected feeder and pump suction flow. The ultrasonic flow measurements are used to formulate CCP models consistent with the aged plant, generally referred to as the reference CCP model for this analysis. The ultrasonic based CCP model produces core-pass, radial and top to bottom flow distributions independent of uncertainties and biases associated with inverse heat balance flow measurements.

Figure 4 shows the average core pass specific measured flow between 1995 and 2000 for pump based flow and inverse heat balance based flow. The 1995 ultrasonic based flow-derivation for hot conditions, the CCP reference, is also shown. It can be concluded that inverse heat balance and pump based flow derivations are in basic agreement and are well represented by the 1995 ultrasonic based flow derivation. Flow differences up to 2%, with respect to the ultrasonic reference, are observed. The differences may be due to actual flow trends or measurement uncertainties. Uncertainties in pump based flow and heat balance based flow range between ±2.0% and ±2.5% (1 sigma).

It has been recognized that pressure tube radial creep and feeder surface degradation will affect the radial and top to bottom flow distribution. Central core channels have higher creep and therefore have relatively less flow resistance. Similarly longer feeders, servicing lower core channels, may show relatively higher flow resistance increases due to feeder surface degradation. Figures 5a and 5b summarize the radial and top-bottom core flow redistribution trends. The scatter in data is reduced by presenting data points representing an average over an interval. The 250 Effective Full Power Day (EFPD) interval is chosen as reference. This interval corresponds to about one year of data with, on average, about 4 data points. Predictions based on measured pressure tube diametral creep rates show a gradual relative increase in inner core flow as shown in Figure 5a. Both ultrasonic measurements (1982 and 1995) and recent (1995 to 2000) inverse heat balance predictions are in agreement with analysis based predictions, taking associated measurement uncertainties into account. The quality of a best fit linear trend representing data can be expressed by the coefficient of determination (R^2). Here, generally, a coefficient (R^2) above 0.90 is preferred. The observed coefficient of 0.81 is marginal. Flow differences up to 0.1%, with respect to the ultrasonic reference, are observed. Figure 5b shows trends for top to bottom core flow tilts. Flow tilts are observed for inverse heat balance calculations, but the coefficient of determination is too small to show a defensible characteristic trend (R^2 = 0.30). The ultrasonic flow based measurements show a top to bottom core tilt change of about 0.1% over 13 years of operation (1982 to 1995). The average flow of top core channels remains basically equal to the average flow of bottom core channels. Inverse heat balance based flow differences up to 0.1%, with respect to the constant ultrasonic reference, are observed. The reference CCP model therefore keeps top to bottom flow tilts constant with respect to time.

From the discussion of flow trends it is evident that bulk flow changes and associated bulk flow resistance changes may be substantial and warrant tracking at about three months intervals. Radial and top to bottom flow tilt trends are well represented by the reference models. Model adjustments are not expected to be required for several years. The tracking and associated adjustment of the reference CCP model with respect to bulk pass flow, therefore, is described in greater detail in the following sections.

The measured header to header differential pressure of each pass is represented in Figure 6. It can be seen that both flow and header to header drop are about constant, yielding an about constant header to header flow resistance "k" defined by:

$$k_j = dP_{hhj} / (W_j^2) \qquad (2)$$

According to Equation 2, flow-resistance trends can be obtained for each pass (j) and are summarized by Figures 7 to 10 for the passes defined by headers 2-3, 4-1, 6-7, and 8-5 respectively. The figures show that the flow resistance below the headers is about constant, justifying a HTS model of constant pass resistance . The pass between headers 2 and 3 (Loop 1) has the highest flow resistance, consistent with onset of boiling at relatively low channel powers during reactor startup (here different outlet header pressure profiles are taken into account) and relatively low ultrasonic flow measurements in 1995.
The effect of high pH and high temperature on increased iron (magnetite) transport has been recognized. To reduce FAC, the primary coolant pH was reduced from 10.8 pHa to about 10.4 pHa in 1996. Further, starting in 1997, reactor power has decreased steadily, decreasing HTS temperature. A direct consequence of reduced operating margin due to plant ageing. In 1999 steam generator pressure was reduced, further lowering HTS temperature. As can be seen in Figures 7 to 10 this may have resulted in a slight change in below header flow resistance trend from slightly increasing or being constant to about 1997 to slightly decreasing between 1997 to 2000. The below header flow resistance trend between 1997 and 2000 is consistent with the decreasing flow resistance trend associated with measured pressure tube diametral creep, also shown for comparison. Comparing the end points (1995 and 2000) in Figures 7 to 10 it is observed that average below header flow resistance is about the same.

An about 2% drop in flow resistance, equivalent to the pressure tube creep component of below header ageing is not observed. Some ageing effects associated with iron transport, therefore, cannot be ruled out. This is independently supported by measured steadily increasing steam generator flow resistance ($k_{sg} = dP_{sg} / (W_{sg}^2)$, see Figure 11) from 1995 to 2000, which is explained by continued magnetite deposition.

The steam generator divider plate replacement (increasing SG flow resistance by eliminating a flow by-pass) and steam generator cleaning have been shown to have offsetting effects on flow (Hartmann, Thompson, et al, 1996) at 77% FP (single phase HTS conditions), as shown in Figure 3. At 100% FP a substantial increase in flow by about 3.5% is observed (Thompson, 1998) due to reduced inlet header temperature and corresponding reduction in two-phase flow.

6. CCP AGEING ADJUSTMENTS

The CCP model is based on 1995 cold and hot ultrasonic flow measurements and includes the latest Point Lepreau pressure tube creep measurements during 1998. This model was first implemented at Point Lepreau in January 1999 (5000 EFPD). An update of this model, including the extension of the CHF correlation to include 5.1% pressure tube diametral creep in 1999, is used in this analysis. This improved modeling incorporates the up-to-date important aspects of ageing for Point Lepreau. The Regional Overpower Protection (ROP) system ensures that an appropriate safety margin exists between operating power and channel dryout power (CCP). It takes into account CCP distribution for different reactor core configurations as well as system uncertainties. The CCP model is a below header model (inlet to outlet header slave channel model) with reference header conditions as boundary conditions, specifically:

- inlet header temperature (Trih),
- header to header differential pressure (dPhh),
- outlet header pressure (Proh).

Further the CCP model consists of a thermalhydraulic model including a single-phase flow resistance model for:

- inlet feeder,
- inlet feeder orifice
- inlet end fitting,
- fuel channel
- outlet end fitting, and
- outlet feeder.

The CCP model also includes:

- appropriate two-phase flow pressure drop and
- critical heat flux (CHF) correlations.

In a fast neutron flux and high temperature environment the pressure tube diameter increases and coolant flow bypassing the fuel bundle consequently increases as well. Iron transport, FAC and magnetite deposition increases flow resistance. Both pressure tube creep and flow resistance increases lower the flow through the fuel bundle. Without excess operating margin this results in core power derating as was the case for PLGS since about 1997, as shown in Figure 12.

Historically the CCP model is defined by constant header conditions and constant thermalhydraulic model. The current (1999/2000) analysis done here makes the same assumptions, but requires only the flow resistance below the header to be constant. It models the effect of pressure tube diametral creep decreasing flow resistance and therefore increases feeder roughness to keep overall flow resistance constant. This model improvement gives radial flow distribution changes consistent with measurements as shown in Figure 5a as well as keeping flow resistance constant, consistent with about constant observed flow and header differential pressure drop as shown in Figures 4 and 6 respectively.

The reference CCP model is associated with reference header conditions. During reactor operation these header conditions change. Changes in header condition affect CCP and therefore also affect the margin between reactor power and CCP or correspondingly ROP detector reading and ROP trip setpoint. To maintain correct margin to trip, detectors are appropriately calibrated. Corrections obtained for each parameter are combined for each pass. The most restrictive pass correction is then applied over the entire core. This is therefore a conservative methodology. Although the reference CCP model is representative for long periods of time, frequent adjustments (once every two to three days) have to be made to keep the CCP header conditions in line with site measurements. For each pass the difference between reference conditions and actual measured conditions is obtained and, using appropriate CCP sensitivity multipliers, an equivalent CCP adjustment is calculated. Similarly it is proposed that adjustments are necessary for below header flow resistance changes due to changes in HTS chemistry as an example. The difference between reference CCP below header flow resistance and measured flow resistance is obtained. An appropriate CCP sensitivity factor is then used to obtain the CCP equivalent adjustment. These adjustments are then combined with the header condition adjustments for each pass. The integrated adjustments are then applied as part of the already existing ROP-detector calibration methodology to ensure adequate operating margin to fuel dryout. Specifically, the most restrictive pass correction, including header condition changes and below header flow resistance changes, is applied over the entire core. This approach integrates ageing flow resistance changes into the existing header condition adjustment methodology. Flow resistance correction is expected to be small. Changes less than 0.1% for a three month period are expected. Corrections for below header flow resistance changes therefore do not have to be made more frequent than about once every three months.

The main difficulty in CCP-model flow resistance adjustments is the high uncertainty in flow measurements associated with Equation 2. Several steps are therefore taken to consider only long term small flow resistance changes established by a long term trend. Flow trends are collected with at least 2 independent methods. Out of trend data points not supported by the second independent method are not included in the trend or are adjusted. An average of the two or more independent flow measurement trends is then used together with measured header differential pressures to obtain derived average flow resistances. The difference between reference-CCP resistance and derived flow resistance is then taken and the CCP sensitivity applied to obtain the CCP equivalent flow model resistance difference. This resistance difference (hbal-pump average) is summarized in Figures 13 to 16 for each core pass. It should be noted that the data trends are normalized to yield the average of inverse heat balance flow, pump flow, and ultrasonic flow based resistances at 4218 EFPD (December 1995). These data points are then smoothed using a 5 point running average. With this method two data points at each end of the trend are lost . The current data point is then regenerated by extrapolation, as shown in Figures 13 to 16. For June 2000 the adjustment for the ROP most limiting pass (Figure 13, headers 2-3) is small , about +0.1%. For the other passes the adjustments are 0.0%, +0.7%, and +0.1% (Figures 14, 15 and 16) relative to the CCP reference model. It should be noted that core flow resistance increases are associated with high pHa, high HTS temperature and high power. It is also noted that recent flow resistance reduction trends (1998 to 2000) are consistent with flow resistance reduction trends predicted by assuming only pressure tube diametral creep ageing (Figures 13 to 16, hd x-y creep only). This suggests that pressure tube diametral creep may be the dominant ageing mechanism for this period.

As already pointed out, adjustments in the pressure tube creep rate or the orifice degradation rate need not be made at this time and are not expected to be necessary in the future since agreement between reference and measured radial flow redistribution is maintained (see Figure 5a).

The key measurable parameters affecting the CCP model and needing ageing tracking with corresponding adjustments when required, therefore, are:

- inlet header temperature,
- header to header differential pressure,
- outlet header pressure,
- channel and pass coolant flow resistance (k), and
- pressure tube diameter (pressure tube diametral creep) .

The last two items adjust for ageing changes below the headers while the first three parameters adjust for ageing and other changes above the headers, such as steam generator ageing and pump characteristic changes. The adjustment frequencies range from days for HTS header temperature and pressure adjustments, to months for pass bulk flow related adjustments and, finally, to years for adjustments related to radial and top to bottom flow tilt trends.

7. PAST FIELD RELATED ACTIVITIES

PLGS has had a detailed monitoring program of HTS system hydraulic parameters in place since the station went into service 17 years ago. Understanding of the ageing mechanisms is based on information inferred from detailed simulations of the hydraulic conditions, in conjunction with an overall understanding of potential degradation mechanisms. A number of the key mechanisms have been supported by component characterization, such as:

- pressure tube diameter measurements,
- removal and study of sections of steam generator tubes,
- outlet feeder pipe thickness measurements,
- cold ultrasonic flow measurements on all channels at about 4200 EFPD,
- installation of online ultrasonic flow measurement devices on a number of feeders and pump suction lines,
- pump impeller inspection,
- removal of one section of one outlet feeder, and
- water CHF and pressure drop experiments in crept pressure tubes (in cooperation with our Canadian industry partners).

In addition to these activities, a number of corrective actions were also undertaken. These include:

- reduction in secondary side steam generator pressure,
- replacement of steam generator divider plates with a more leak tight design,
- cleaning of about 60% of the inside of the steam generator tubes,
- cleaning and lancing of the external surface of steam generator tubes,
- calibration of inlet header temperature instrumentation (RTD's) and other activities to better characterize instrument loop uncertainties,
- improvements to the purification system to improve purification flow rates,
- collapsing ROP trip setpoint plateaus (here uniform ROP detector setpoints improve the ROP trip probability for all flux shapes), and
- re-evaluating cases covered under HSP-1.

Implementing the last two improvements would generate a gain in ROP margin (4% to 5%) allowing reactor power operation to return to close to 100% full power operation during 2000.

It should be noted that although a number of these activities were principally performed for other reasons, their benefit to the hydraulic performance of the HTS has been significant, and hence were included in the list. The combination of the analytical and field activities has allowed for continued full power operation up to 1997, with the exception of minor power reductions to ensure adequate operating margin during fueling.

8. FUTURE ACTIVITIES AND POSSIBLE BENEFITS TO FUTURE DESIGNS

Depending on the developing requirements arising from the ageing program, future activities may include:

- bundle power redistribution (4 bundle refueling shifts, channel power optimization (reform) etc.),
- additional pressure tube diametral creep measurements (based on a yet to be developed quick measurement scheme),
- introduction of a CHF enhanced fuel bundle design (a demonstration irradiation of the new CANFLEX bundle (Lane, Dimmick, et al, 1996) has recently been completed at Point Lepreau),
- refining the fuel dryout criterion
- further development and application of online ultrasonic flow measuring devices,
- removal of an inlet feeder and flow reduction orifice for inspection and testing,
- further primary side steam generator cleaning,
- HTS cleaning as well as pressure tube and feeder replacement during the refurbishment outage.

In addition to ensuring continued safe and economic operation of PLGS, information gained from the program will be of benefit to plant designers as well. By better understanding the ageing mechanisms, improved material selection and component design can be performed, thus continuing the evolution of the design. Refinements to the operating envelope (chemistry of the HTS and secondary side for example) can be made. Practices for monitoring, maintenance, and analysis can also be recommended as a guide to ensure continued safe and economic operation.

9. SUMMARY

The proposed ageing tracking and periodic ageing adjustment methodology expands the current methodology, which adjusts for HTS header condition changes, to include changes in below header flow resistance. These below header ageing changes are due to pressure tube diametral creep and iron transport related HTS geometry changes. The improved 1998 CCP reference model is based on ultrasonic flow measurements and the assumption of constant below header flow resistance ageing, taking the main ageing components into account. The tracking and adjustment methodology has been applied to PLGS data for the period between 1995 and 2000. Between 1995 and 2000 only minor adjustments are necessary to keep the 1998 CCP reference ageing model representative of plant operating conditions. Once CCP reference model and tracking methodology are implemented it ensures continued future safe reactor operation without unnecessary economic penalties. By showing that the CCP reference ageing model describes PLGS ageing well, credible predictions can be made for future reactor requirements to mitigate ageing. Options to gain operating margin are summarized as well.

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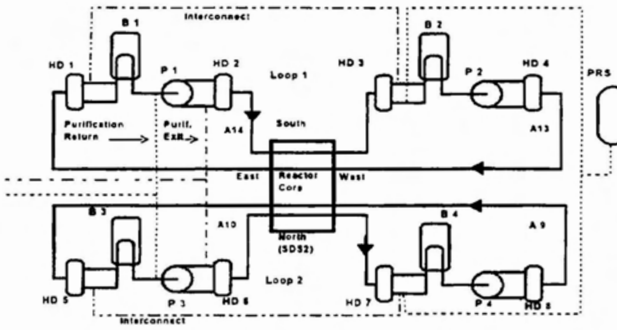

Figure 1:
PLGS Simplified Circuit Diagram

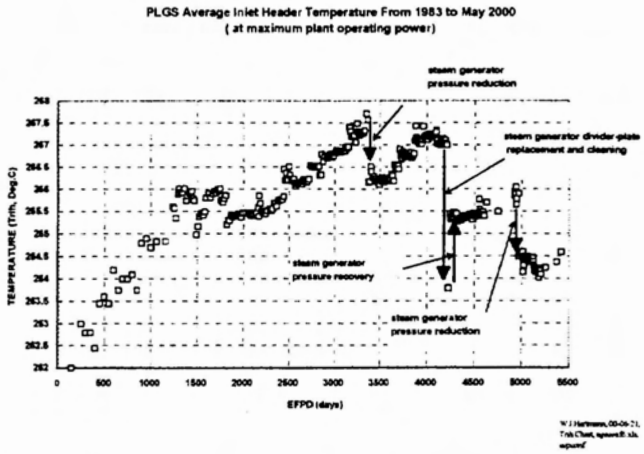

Figure 2:
PLGS Average Inlet Header Temperature Trend (up to May 2000)

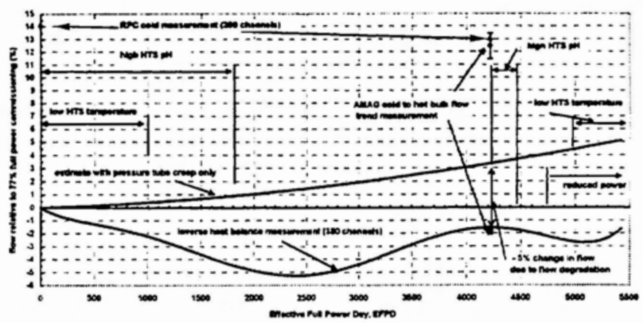

Figure 3:
PLGS Site Bulk Flow Rates

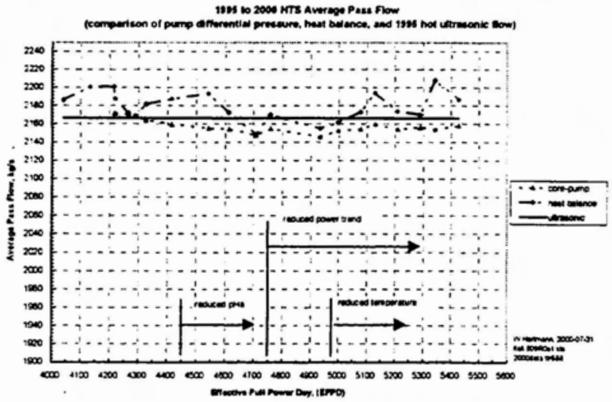

Figure 4:
PLGS 1995 to 2000 Detailed Bulk Flow Rates

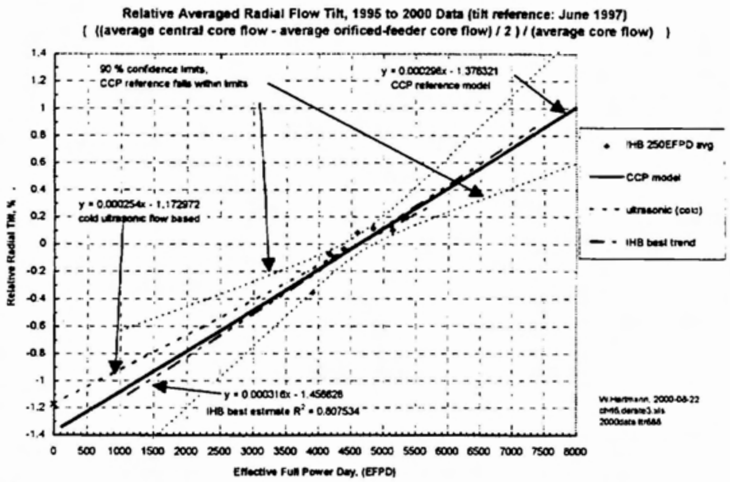

Figure 5a:
PLGS Core Radial Flow Distribution Trends

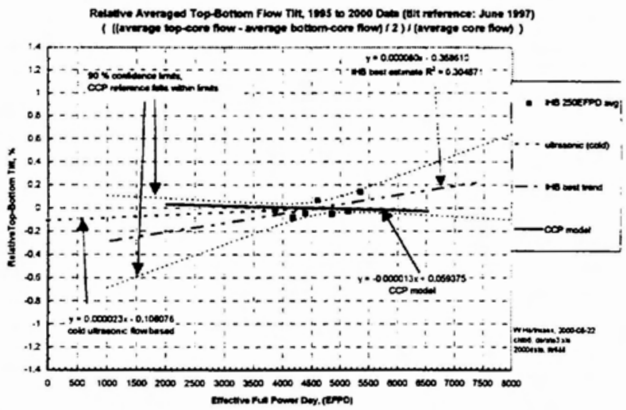

Figure 5b:
PLGS Core Top-Bottom Flow Distribution Trends

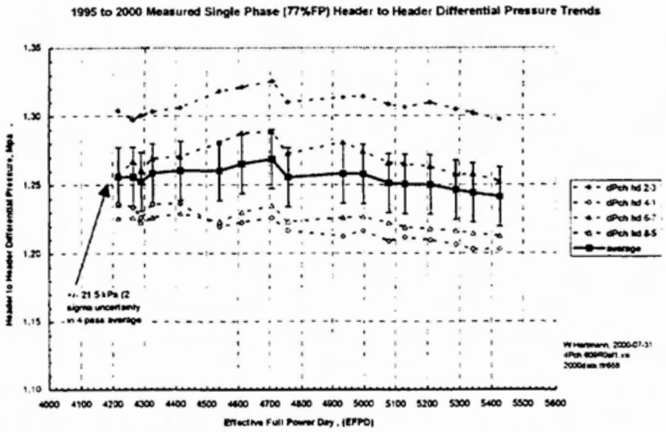

Figure 6:
PLGS 1995 to 2000 Detailed Header to Header Differential Pressure

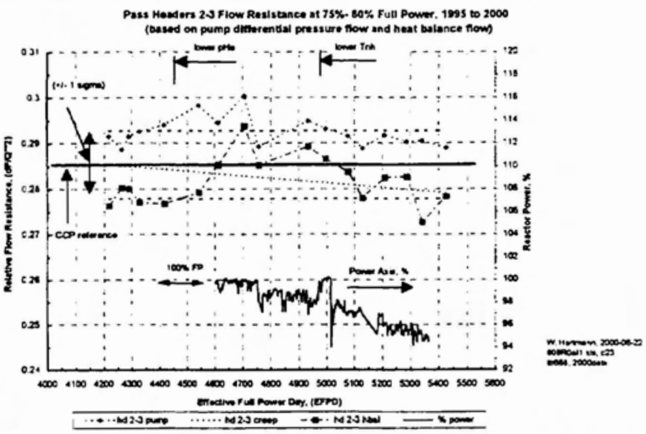

Figure 7 :
Single Phase Flow Resistance for Pass Defined by Headers 2-3

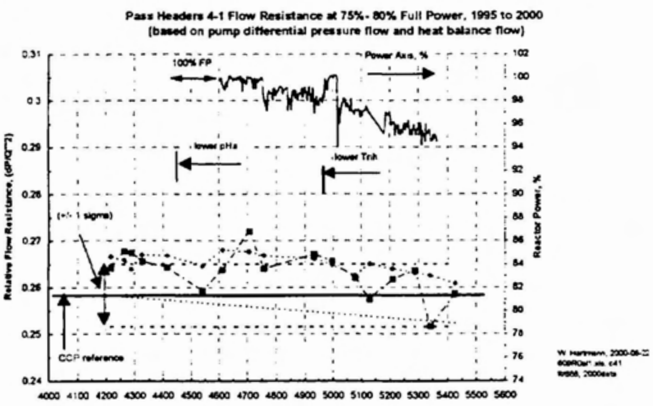

Figure 8:
Single Phase Flow Resistance for Pass Defined by Headers 4-1

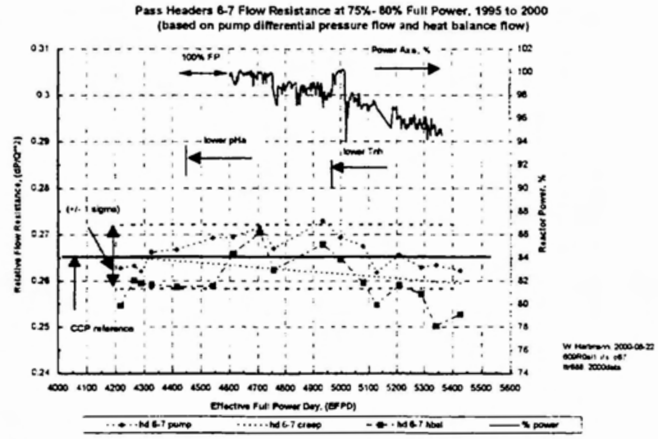

Figure 9:
Single Phase Flow Resistance for Pass Defined by Headers 6-7

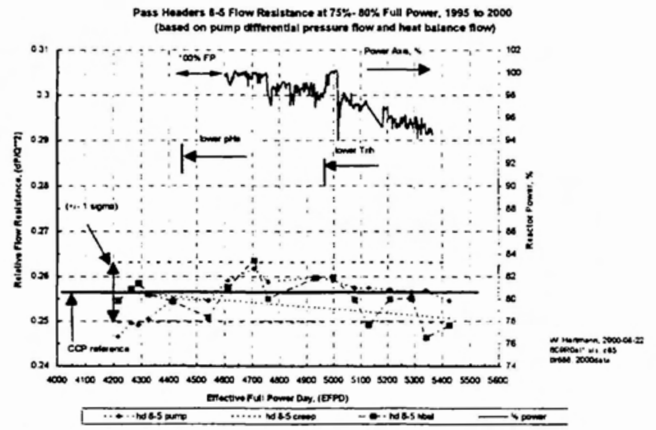

Figure 10:
Single Phase Flow Resistance for Pass Defined by Headers 8-5

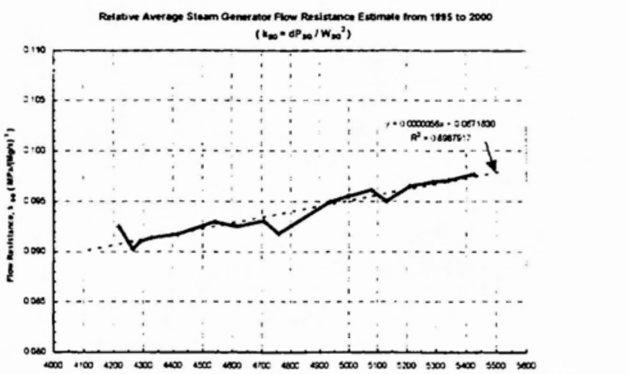

Figure 11:
Estimated Steam Generator Flow Resistance Trend

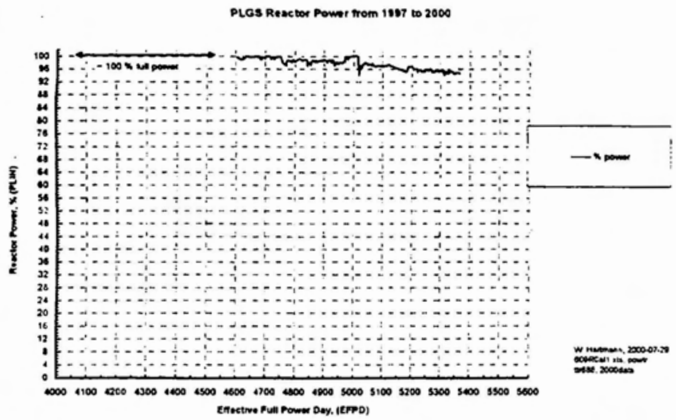

Figure 12:
Reactor Power Trend from 1997 to 2000

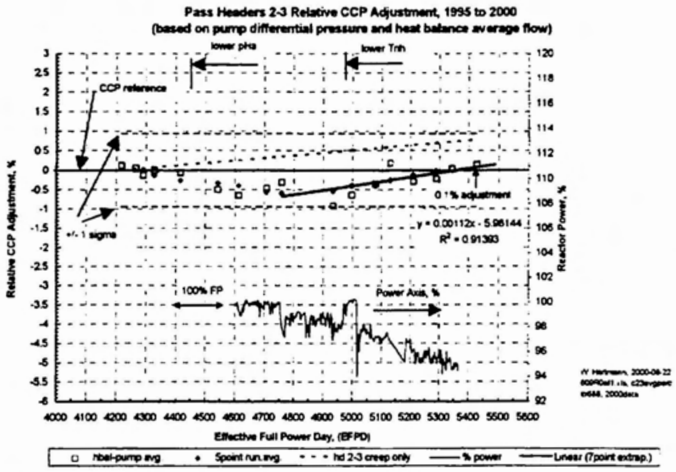

Figure 13:
CCP Reference Model Adjustment for Pass Headers 2-3

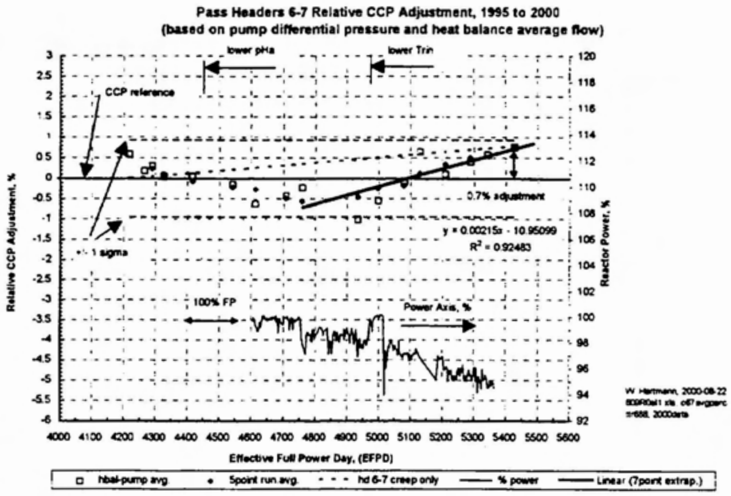

Figure 15:
CCP Reference Model Adjustment for Pass Headers 6-7

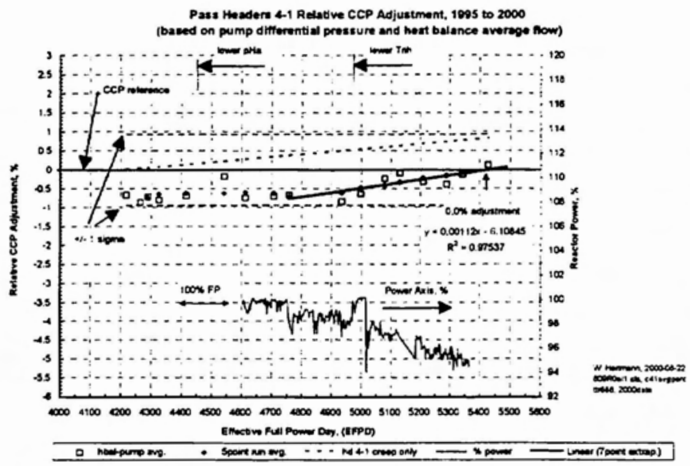

Figure 14:
CCP Reference Model Adjustment for Pass Headers 4-1

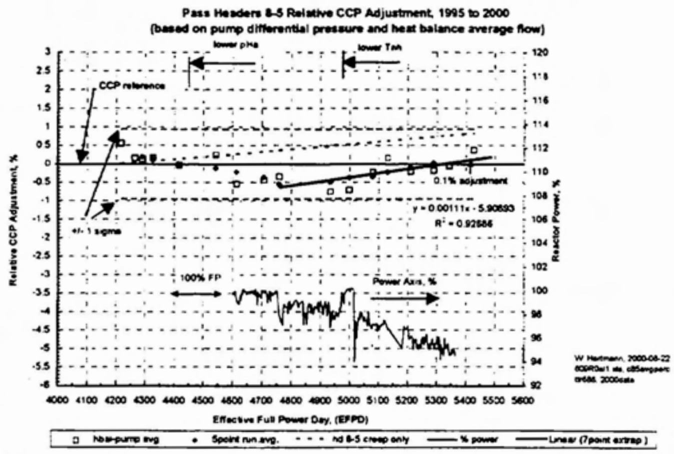

Figure 16:
CCP Reference Model Adjustment for Pass Headers 8-5