MANAGING THE SAFETY ASPECTS OF PLANT AGEING AT THE POINT LEPREAU GENERATING STATION

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ABSTRACT

This paper deals with a portion of the Point Lepreau ageing program that is investigating 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 lead to improved analysis tools, enabling plant optimization as well as prediction of future requirements.

1. INTRODUCTION

All industrial plants undergo changes with time and nuclear plants are no exception. The CANDU 6 reactor follows on from 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 & 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 the necessary targets were achieved. Nonetheless, it was not possible to precisely predict how the plants would change with time at the design stage. 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 principals of monitoring/detecting, anticipating, understanding and then correcting or compensating. It has components involving R&D, analysis & 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. 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, who operates a sister CANDU 6 plant (G-2), and with Atomic Energy Of Canada LTD (AECL), the designer.

It should be realized that a good ageing management program should be more than a plan. It is an attitude in which the institution recognizes the importance of:

- staying within specifications and proven operating envelopes
- not cutting back on preventive maintenance due to short term budgetary pressures
- following up on unusual failures, observations, or findings
- the trade off between proactive action versus equipment problems and unit down time

If unchecked, ageing has the potential to adversely effect safety margins due to changes in conditions and equipment, and due to “institutional degradation”. Changes in plant conditions and equipment can potentially lead to either an increased probability of event initiation due to component failures, or an increased severity of consequences given an unrelated process failure. The latter may result from either reduced availability or effectiveness of the special safety systems. Institutional ageing relates to plant staff turn over affecting knowledge, skills, and safety culture, as well as configuration management. Unless there is a well managed on-going safety assessment program (or periodic safety reviews), confidence in the safety basis of the station will erode with time.

The paper deals with the portion of the program that is investigating the ageing behavior of the heat transport system (HTS) and its potential impact on channel flows and Critical Channel Power (CCP). Critical channel power being defined 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 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 ^{o}C. It leaves slightly above 10 MPa and 310 ^{o}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. 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 in NUCIRC (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. This improved modeling incorporates the important aspects of ageing.

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.

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 and pressure profiles in addition to fuel pressure tube diameter measurements, and cold and hot ultrasonic flow measurements for individual channels as well as individual passes. Inconsistencies in Equation (1) have to be resolved by addressing ageing in all three main parameters. 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 15 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 is 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.

- increase in hydraulic resistance due to redistribution of iron in the HTS. Dissolution of iron and flow accelerated corrosion 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 and 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 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 associated 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 indicator of flow degradation has been measured coolant flow trends as summarized in Figure 3, and HTS pressure profile trends such as pump differential pressure trends and header to header differential pressure trends, the latter being shown in Figure 4. One of the main observations in Figure 4 is that each pass is ageing at a different rate. Pass specific ageing is also supported by coolant flow measurement comparisons as summarized in Figure 5.

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.

Figure 3 shows the single phase inverse heat balance flow prediction 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 consideration (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. The change in trend from 2000 EFPD to 4200 EFPD is mainly due to a combination of pressure tube diametral creep, steam generator divider-plate leakage, steam generator pressure reduction, steam generator cleaning, and thermal power calibration.

Figure 4 shows the core pass specific growth in operating asymmetries. The differential pressure of each pass is represented relative to the average of all passes, showing substantial increases in flow resistance for the pass defined by headers 2 and 3 (HD 2-3) after about 2000 EFPD. Figure 5 shows that the conclusions derived from pressure profile considerations are well supported by inverse heat balance as well as ultrasonic flow measurement trends. Recognition of this plant specific ageing characteristic has lead to pass specific operation optimization. It is, however, regarded to be beneficial to reduce pass specific asymmetries by appropriate HTS modifications.

6. CHANNEL FLOW AND CCP CHANGES

Figure 6 summarizes measured and NUCIRC predicted HTS flow rates associated with the margin to dryout of the fuel bundles. For comparison the inverse heat balance measurement trend and the estimated flow trend assuming only pressure tube creep as an ageing parameter are reproduced from Figure 3. It should be noted that 75% FP to 77% FP NUCIRC flow predictions are based on site model development using cold and hot ultrasonic flow measurements. The NUCIRC predictions are in close agreement with inverse heat balance measurements. NUCIRC 100% FP predictions show a decreased flow with respect to the inverse heat balance measurements, typical of two phase versus single phase flow conditions. With respect to inaugural predictions, the 100% flow predictions show a maximum flow decrease of about 7% at around 3400 EFPD, just before steam generator pressure reduction. The steam generator pressure reduction increased the flow to about -5.5%, but inlet header temperature increases reduced the flow to about -6.5% by 4200 EFPD. The steam generator divider plate replacement and steam generator cleaning have been shown to have compensating effects on flow (Hartmann, Thompson, et al, 1996) at 77% FP (single phase HTS conditions). At 100% FP (two phase HTS conditions) a substantial increase in flow from -6.5% to -3% is observed. The subsequent steam generator pressure readjustment reduces the flow to about -4%. Safety operating margin to fuel sheath dryout is generally expressed as the critical power ratio, the ratio of dryout channel power to nominal 100% channel power. As power increases to dryout conditions the associated channel flow also decreases steadily as a function of channel power or the amount of quality in the HTS. For convenience the margin to fuel dryout can therefore be expressed in relative terms as the difference between the NUCIRC predicted 100% FP flow and the NUCIRC predicted channel flow at dryout conditions. This channel flow trend at fuel dryout is shown in Figure 6 to show the trend in safety operating margin. The trend in dryout channel flow is generally constant for the first 2000 EFPD and then increases steadily as more and more flow bypasses the fuel due to pressure tube diametral creep. It can be seen that there was a steady decrease in operating margin from 29% at 0 EFPD to 21% at 3000 EFPD. Subsequent steam generator maintenance and reduction of HTS parameter measurement uncertainties yielded a constant operating margin of about 21% to 4500 EFPD. This margin is expected to be reduced further to about 20% by 5000 EFPD. Further loss of margin beyond 5000 EFPD is expected if no additional remedial actions are taken.

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 15 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 on line 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:

- temporary 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 (RTDs) and other activities to better characterize instrument loop uncertainties, and
- improvements to the purification system to improve purification flow rates.

It should be noted that although a number of these activities were principally performed for other reasons, their benefit on 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, 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:

- reducing steam generator secondary side pressure (permanently),
- 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) is scheduled for 1998),
- refining the fuel dryout criterion
- further development and application of on-line 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 economic operation.

9. ACKNOWLEDGMENTS

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D. Edgar of SYSTEC

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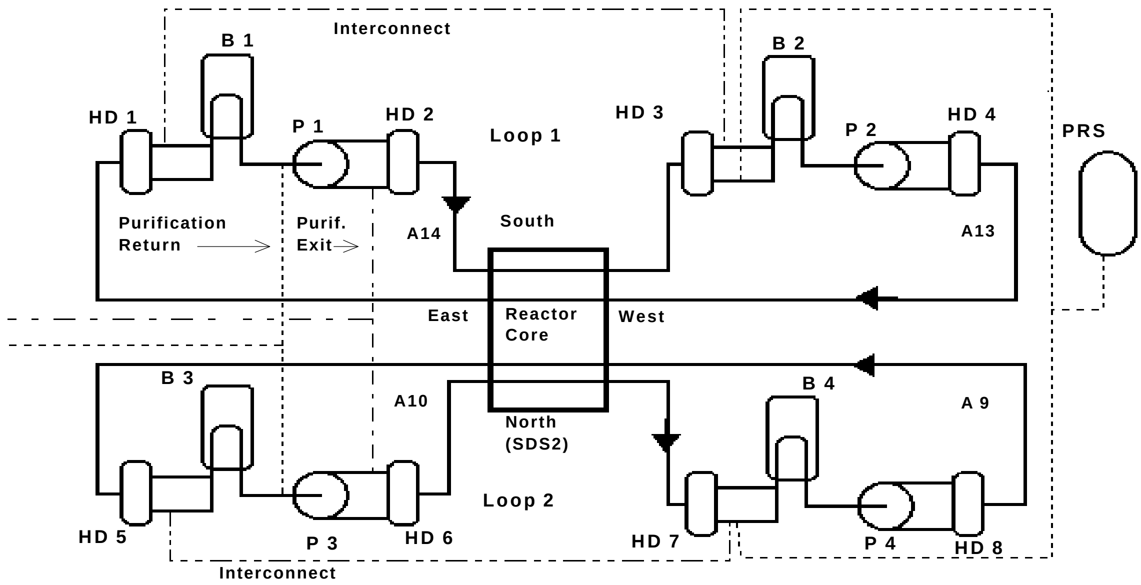

Figure 1 PLGS Simplified Circuit Diagram

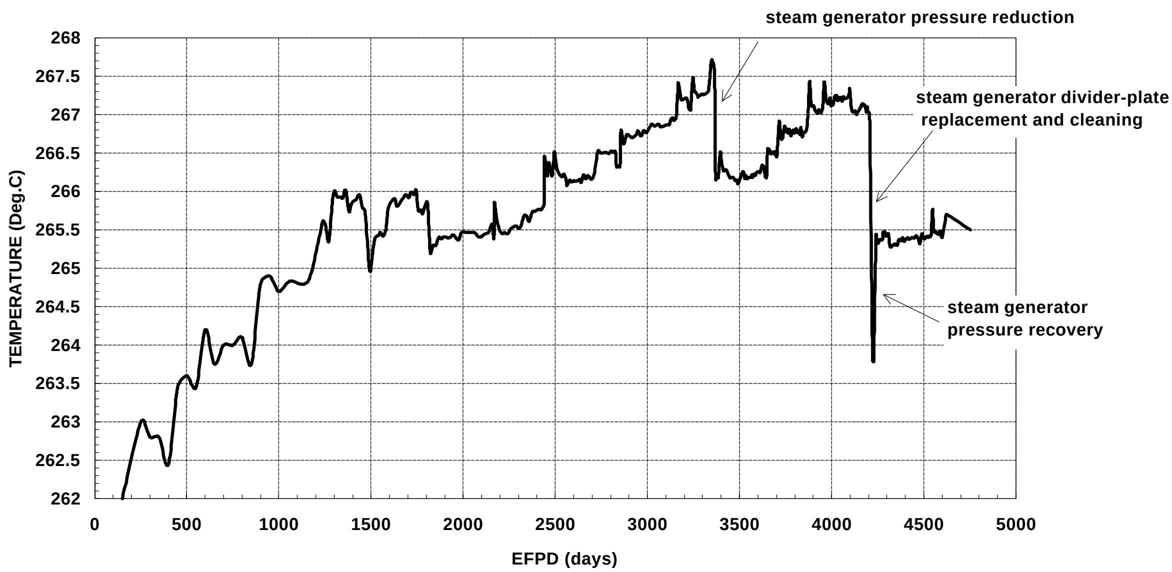

Figure 2 PLGS Average Inlet Header Temperature Trend (up to 97-11-17)

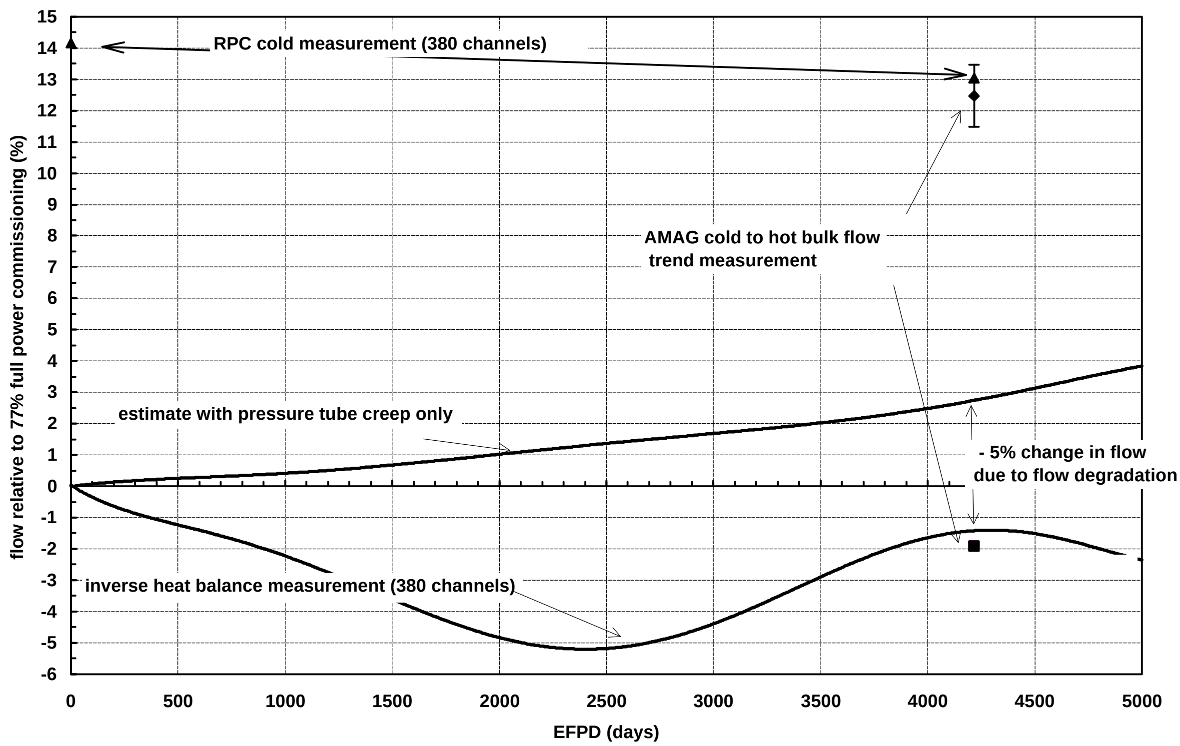

Figure 3 PLGS Site Bulk Flow Rates

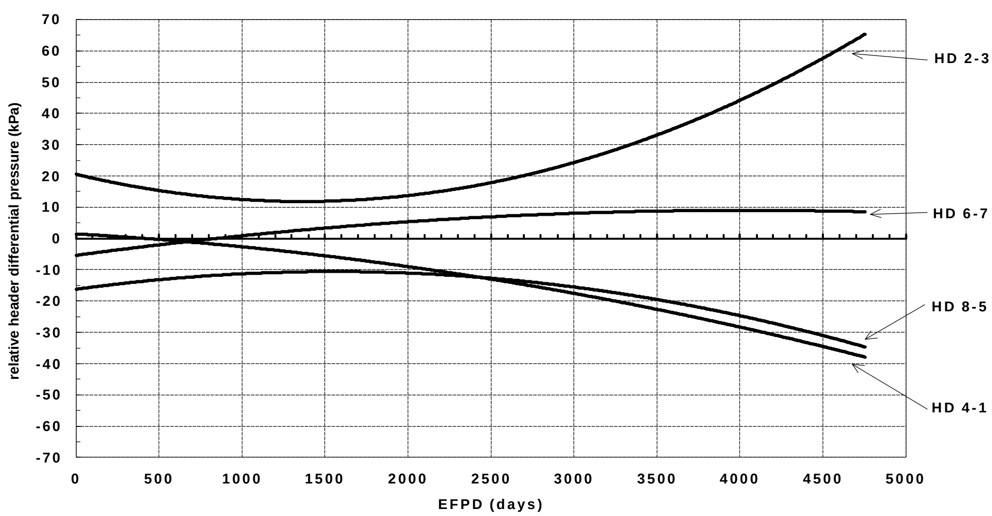

Figure 4 PLGS Header Differential Pressure Asymmetry

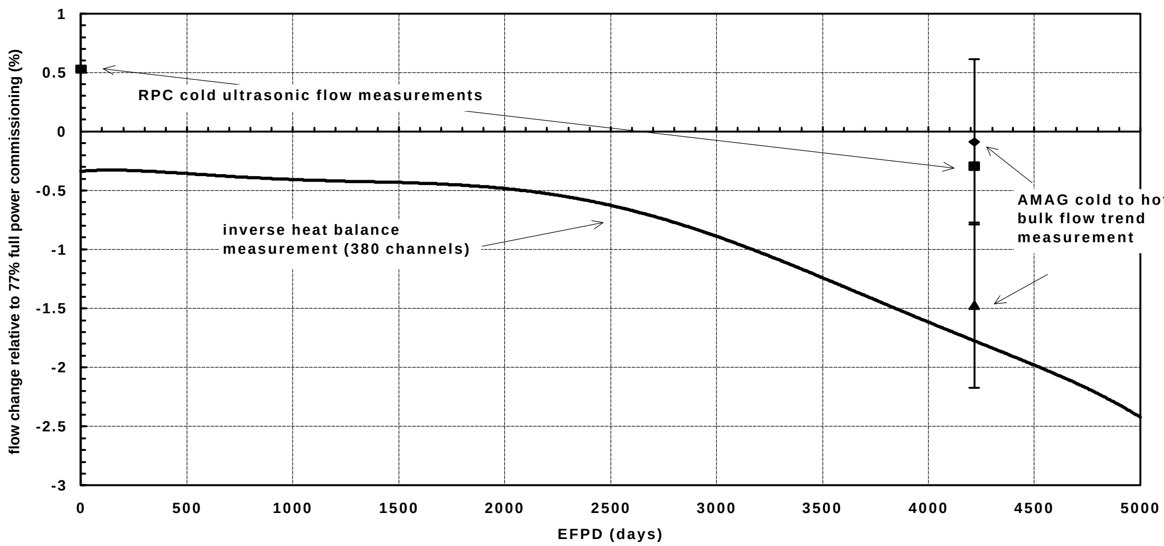

Figure 5 PLGS Loop2-Loop1 Bulk Flow Rate Differences

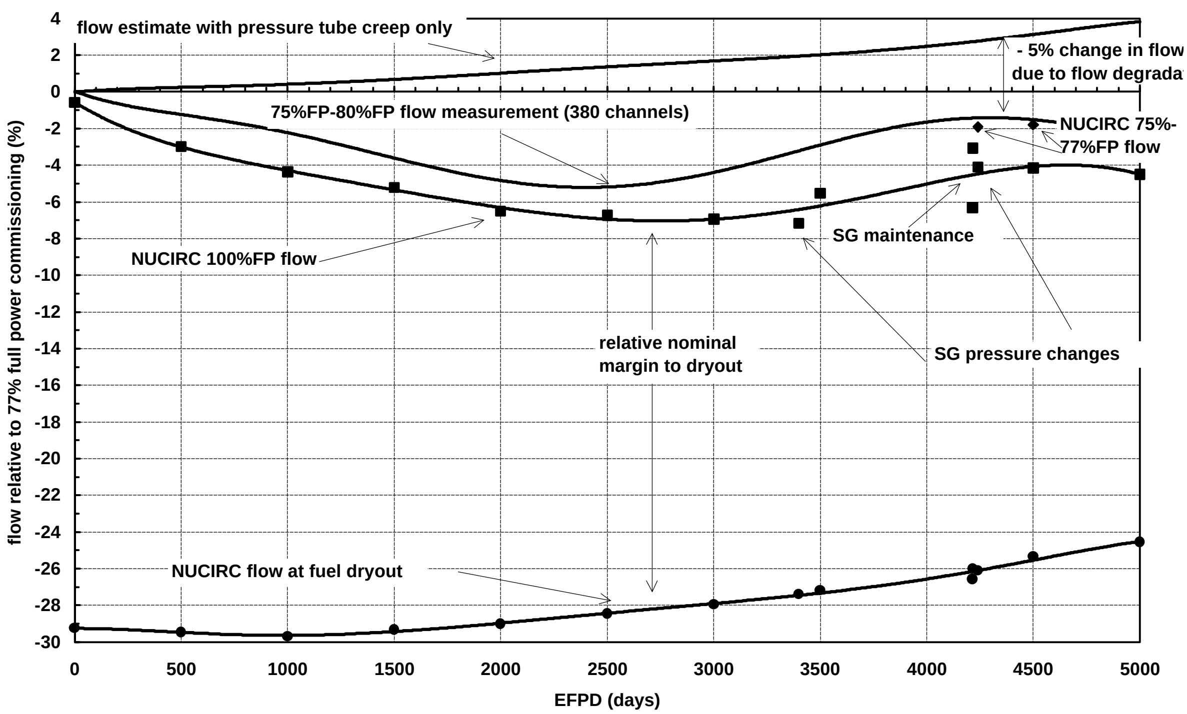

Figure 6 PLGS Site Bulk Flow Rates, Margin to Fuel Dryout