Gentilly-2 Full Power Operation History and Future Challenges

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Abstract

The Gentilly-2 nuclear generating station performance has been affected, in the recent past, by the reduced safety margins resulting from heat transport aging mechanisms. Margins provided at the design of the CANDU-6 stations, to cater for the in-service aging degradation that was expected to occur, have eroded to a point where remedial actions needed to be taken in order to continue operation at full power. Steam generator fouling was originally considered by plant designers as the single most important aging mechanism affecting safety margins in CANDU PHW reactors. Changes were expected for other heat transport components such as that resulting from pipe roughness degradation and pressure tube diametral creep; however these were not deemed to be unduly detrimental to safety margins and were not explicitly considered in the safety margins determined at the design of the stations. Monitoring of Gentilly-2 plant operating data has indicated that these aging mechanisms significantly affect the heat transport system hydraulic characteristic. As a consequence, effective Regional OverPower (ROP) setpoints have been reduced in order to mitigate the impact of these aging mechanisms on safety margins. Also a number of actions have been taken to restore safety margins and maximize operating reactor power; these actions include replacement of the steam generator bolted divider plates by the welded design, reduction of secondary heat transport steam pressure, refinement in channel selection rules for refueling and steam generator primary side cleaning using Siemens-SIVABLAST. In spite of the success of these actions, safety margins are expected to be further eroded by the recently identified aging mechanism such that operation at full power may no longer be possible in the near future. This paper summarizes the evolution of Gentilly-2 data since first commissioning of Gentilly-2 and presents the relative contribution of the various aging mechanisms which impact on the safe operating envelope. Challenges to the future full power operation of the Gentilly-2 nuclear generating station are identified and avenues are identified for possible corrective actions which could be taken to restore full power operation or to minimize production losses.

Introduction

Steam generator fouling was originally considered by plant designers as the single most important aging mechanism affecting safety margins in CANDU PHW reactors. Accordingly, reactor shutdown system trip setpoints were determined with sufficient margins to ensure trip effectiveness to end-of-life conditions, defined in terms of a limiting reactor inlet header temperature or reactor outlet header quality which could be expected for plants with fully fouled steam generators. Changes were expected for other heat transport components such as pipe roughness degradation, pressure breakdown orifice erosion and pressure tube diametral creep; however these were not deemed to be unduly detrimental to safety margins and were not explicitly considered in the design safety analyses. As a consequence, allowances provided at the design of the CANDU-6 stations were deemed to be sufficient so that the shutdown systems could prevent onset of intermittent dryout (OID), in slow loss of regulation events, therefore meeting the design requirements for these systems [1] with a significant margin during the entire operating life of the stations.

Monitoring of Gentilly-2 plant operating data and of heat transport flows inferred from heat balances, pump head-flow characteristics and from theoretical calculations show that other aging mechanisms significantly affect the heat transport system (HTS) hydraulic characteristic [2]. Also recent experimental measurements [3] have indicated that, for given inlet coolant conditions and channel flow, the channel power at OID critical heat flux (CHF) was considerably lower for the crept pressure tube geometry than for the uncrept geometry, with power decreasing further as the amount of creep increased. As a result, HTS aging mechanisms combined with pressure tube diametral creep contributed to erode allowances provided at the design to a point where remedial actions needed to be taken in order to ensure that operation at full power remains within the safe operating envelope originally defined by the designer.

The first part of this paper summarizes the evolution of the HTS parameters since first commissioning of Gentilly-2 and discusses the relative contribution of the various aging mechanisms on margin reductions, that can be inferred from this data. A second part summarizes the impact of the aging mechanisms on safety margins and presents the result of actions taken at Gentilly-2 to ensure that operation remains within the safe operating envelope originally defined by the designer and to restore margins for full power operation. Current estimates of the possible impact of the combined effect of the identified HTS aging mechanism on maximum achievable operating power are given. Finally, possible actions that are being considered to maintain full power operation or to minimize production losses are summarized.

Evolution of Gentilly-2 Heat Transport System Hydraulic Characteristic

In CANDU reactors, the primary heat transport system is controlled at constant reactor outlet header pressure and the secondary heat transport is controlled at constant steam pressure. Evolution of the primary heat transport system characteristics is therefore indicated by variations in the reactor inlet coolant subcooling and the reactor core pass and channel flows. Primary and secondary heat transport systems are instrumented with a number temperature, pressure, differential pressure and flow measurements used by the Digital Computer Control and by the Reactor Shutdown systems computers. Core pass flows are not directly measured in CANDU-6 reactors, but flow variations can be inferred from reactor inlet and channel outlet temperature measurements, from reactor inlet to outlet differential pressure measurements and from pump head measurements, although these inferences add some uncertainty to the value. Instrumentation used to monitor this data is illustrated on the heat transport schematic diagram shown in Figure 1.

The station computers were not originally designed to collect and log all of the available instrument signals. In 1994, Gentilly-2 installed a system which continuously collects and logs all the process and shutdown system instrument signals. Since then, complete sets of data are available to identify trends in the different HTS parameters; in particular, we have compiled comprehensive sets of single phase HTS operating data taken during deratings (to single phase flow conditions - at around 80% of full power) for flow verification, made to confirm no channel blockage. These are done on a monthly basis since initial commissioning.

Prior to 1994, a limited number of instrument measurements could be retrieved from the plant control computers on demand for various specific needs. Among these, a large number of data were collected and logged during the initial commissioning tests; all of these measurements are for single phase coolant conditions. Since first commissioning in 1983, reactor inlet header temperatures, reactor outlet header pressures and reactor header to header differential pressure drops have been recorded a few times a week for the shutdown systems Regional OverPower Trip (ROPT) calibration; most of these data can not be easily used to track the evolution of the heat transport system characteristic, because, after 1988, the coolant boiling conditions were different. However, all of the 380 channel outlet temperatures and reactor inlet header temperatures have been logged on a monthly basis, along with neutron detector measurements, at reduced reactor power for channel single phase monthly flow verifications. This constitutes a comprehensive set of data since the station initial commissioning, allowing to trend the heat transport system flow evolution, as determined from heat balances computed for each of the primary heat transport loop (each reactor core and steam generator passes). Pump suction to reactor inlet header differential pressure measurements are also available for some of the monthly power reductions prior to 1994 and for all monthly power reductions after that date. This data constitutes a second consistent set of data since the station first commissioning, permitting to trend the heat transport system flow evolution, as determined from the pump head-flow characteristics. This data set provides an independent verification of the flows inferred from heat balance calculations. The methodology and accuracy of the heat balance and pump head-flow characteristics HTS flow calculations are documented in [2]. The accuracy is inside 2,6% .

Reactor Inlet Header Temperature

Figure 2 illustrates the evolution of reactor inlet header temperature taken at full power, except for some of more recent measurements where power had to be reduced to between 95 % and 98 % full power due to reduced safety margins. This figure shows that steam generator performance degradation is indeed an important aging mechanism in CANDU reactors.

Gentilly-2 has experienced a continuous increase, from 262 °C to 268 °C, of the reactor inlet header temperature at full power. This is a direct result of steam generator performance degradation. In 1993, after 2750 EFPD, the maximum inlet header temperature had reached a level where action had to be taken to ensure that continued increases would not result in reactor operation outside the safe operating envelope analyzed limit of 270°C. At that time, action was taken to reduce the secondary side heat transport system steam pressure in order to obtain reduced primary side inlet header temperatures.

Inspection of the steam generator during annual outages indicated that the divider plates ensuring the separation between the steam generator primary side inlet and outlet plenums were leaking. This leaking, caused by erosion, was slowly increasing with time and steam generator pressure had to be reduced further in 1995, before the divider plates could be changed from a bolted to a welded design. This design change was very successful and the reactor inlet temperature dropped to around 263 °C for the reduced steam pressure and to around 265 °C when the steam pressure was restored to it's nominal design value. This replacement indicated that divider plate leakage was an important factor contributing to steam generator performance degradation; this flow by-pass was not considered in the original design [4].

Since the divider plate replacement, reactor inlet header temperatures have remained nearly constant, with a slow increase of around 0.5°C/1000 EFPD[1] attributed to steam generator fouling. Steam pressure has recently been decreased again when it was realized that inlet header temperatures had to be limited to approximately 265 °C at or near full power to ensure SDS2 high pressure trip effectiveness for loss of single even numbered pump events. SDS-2 original design did not consider single PHT trip as a limiting design condition.

Primary Heat Transport System Flow

Figure 3 illustrates the evolution of the single phase primary heat transport system coolant flow. This figure shows that there is more to heat transport system aging than steam generator performance degradation

Figure 3 shows the evolution of the total primary coolant flow obtained from pump head characteristic and from reactor and steam generator heat balances; the ±2.6% uncertainty in steam generator heat balance flow [2] and the ±3.1% uncertainty in reactor heat balance flow [2] are shown by error bars. Error bars are not included for the pump head flows as their uncertainty has not yet been formally established. Also shown in figure 3 is the sum of the ultrasonic measurement of channel flows taken during the 1982 commissioning at 40 °C, 0% FP and extrapolated to 262 °C; the uncertainty of the ultrasonic measurements, estimated at ±2.6% at 40 °C, is also illustrated for that point in figure 3. As shown, all measurements agree within the measurements uncertainties, with the sum of ultrasonic measurements underestimating by around 3% the flows established at first commissioning based on reactor heat balance and pump head characteristic.

Figure 3 shows that Gentilly-2 has experienced a significant decrease in primary heat transport flows, since first commissioning. Although the data is sparse prior to 1987-88, the flow decrease seems to have started from the 1987-88 period. The data shows that the flow reduction has subsided since 1995. Although there may be no correlation, the period of flow reduction coincides with a period of operation with average reactor outlet qualities greater than around 2%.

The reactor designer, AECL, has linked the flow reduction to magnetite transport in heat transport system which increases the heat transport system piping roughness [5]. Replacement of the steam generator bolted divider plates by the welded design in 1995 resulted in an additional 2% to 3% decrease in flow due to increase resistance in the Steam Generator tubes (no more flow by-pass), after which the core pass flows remained approximately constant indicating that the effect of magnetite transport may have subsided. Steam generator primary side cleaning using Siemens-SIVABLAST restored around 5% of the core pass flow.

The increase in pipe roughness associated to magnetite transport is in fact much greater than that suggested by the flow reductions. Indeed, pressure tube diametral creep increases the flow area in the channel as the heat transport system ages. Figure 4 illustrates the range of maximum pressure tube diametral deformation obtained from CIGAR measurements taken at G2 and compares them to those predicted by the RC-1980 prediction model, which is based on all available CANDU 6 CIGAR measurements [6]. Figure 5 show the diametral strains obtained for the maximum creep rate predicted by the RC-1980 model. Diametral strains of Gentilly-2 pressure tubes were in the range of

[1] EFPD stands for Effective Full Power Days

around 2% to 3%, in 1998, for high power channels. Experimental measurements taken at Stern Laboratories [3], on a full scale simulated bundle string for creep profiles typical of that seen in Gentilly-2, show that channel flows should be 10% to 15% higher, at constant fuel string pressure drop, in a channel with 2% to 3% maximum diametral creep than that in a channel with no creep. Although, the reactor is not operated at constant fuel string pressure drop, such a large decrease in channel hydraulic resistance should result in higher core pass flows with aging, assuming creep is the only mechanism affecting the heat transport hydraulic characteristic.

Impact and Mitigation of Heat Transport Aging Mechanisms Gentilly-2

Channel flow reduction and temperature increases reduce the minimum margin to OID CHF. Full scale experiments [3] show that, at constant fuel string pressure drop, diametral expansion of the pressure tube due to creep does not affect the margin to dryout, because the flow increases as a result of the pressure tube creep. However if we consider more realistic reactor condition, for given coolant inlet conditions and channel flow, full scale experiments show that the OID CHF is significantly lower for the crept geometry than for the uncrept geometry.

Based on recent AECL analyses and recommendations [5,7], Gentilly-2 has seen a reduction in effective ROP setpoints from 121% in early 1985 to 112% at the end of 1998, as a consequence of heat transport system aging. The replacement of the steam generator bolted divider plates by the welded design in 1995 and the reduction of secondary heat transport steam pressure in 1998 have restored the reactor inlet header temperatures to the level seen in 1985 and therefore the 8% setpoint reduction (9%/121%) is entirely due to the reduction in margin to dryout resulting from the combined effects of flow reductions associated with pipe roughness degradation due to magnetite transport and of pressure tube diametral creep. With the minimum operating margin to ROP trip setpoints at around 6% for an average CPPF of 1,07, even with careful channel selection and improved operating procedures, the setpoint reduction had reached a level where operation at full power was compromised and frequent reactor deratings were required.

In 1999, cleaning of the primary side of the steam generator was attempted using Siemens-SIVABLAST. The cleaning was quite successful as this decreased reactor inlet temperatures by around 3 °C and increased reactor flows by around 5%. As a result, 3% operating margin was immediately recovered from the ROPT calibration because of the decrease of the RIH temperatures. However, increase in margin to dryout associated with flow increases is not yet included in the ROPT calibration procedure. This gain in operating margin has to be approved by AECB

Future Challenges to Full Power Operation

The increased flow obtained from the cleaning of the primary side of the steam generators provides around 5% increase in margin to OID CHF. However, in order to determine the net impact on station performance, this margin needs to be discounted for the margin erosion due to the combined effects of continuing pressure tube diametral creep and pipe roughness degradation and due to the increased uncertainties associated with prediction of pressure tube diametral creep rates.

Pressure Tube Deformation by Diametral Creep

Figure 6[2] illustrates our estimate of the impact of pressure tube diametral creep on the ROP trip setpoint, based on the most recent NUCIRC calculations, performed by AECL for

[2] The dates shown in Figure 6 are based on an 82% availability factor at expected reduced operating power limits

37-element fuel strings. Specifically, Figure 6 shows the decrease in ROPT margins determined for the current ROP setpoint limiting case set, for the maximum pressure tube deformation and channel axial deformation profile, as predicted based on the average of all available CIGAR diameter measurements taken in CANDU-6 reactors. Setpoint corrections are given for the effect of pressure tube creep alone and for the combined effects of pressure tube creep and HTS aging. As shown by the 95% confidence limits, illustrated in Figures 4 and 5, established on the basis of the CIGAR data through 1997 from all the CANDU-6 reactors, there is a significant variation in the channel creep rates. The ROPT margin reductions include the effect of a ±16% uncertainty in channel creep rate to account for this variance in creep rates. The decreases are given relative to the 1997 ROP setpoint update for G2. The 1997 ROP setpoints included the effect of pressure tube diametral deformation, based on CIGAR measurements taken at G2, as determined by an interim model developed at AECL taking into consideration the limited number of experimental measurements available at that time [3,5]. The decrease from the 1997 ROP setpoint includes an initial correction, which result form the new models predicting a slightly greater impact of diametral creep than the interim model, followed by a gradual decrease in setpoint. The curve showing the effect of pressure tube diametral creep alone is determined assuming no further fouling of the steam generator and core flow increases as a result of diametral creep, such that reactor inlet header temperatures and reactor header-to-header pressure drop remain constant. As shown, in those ideal conditions, the 5% margin for flow increases obtained from the steam generator cleaning would be entirely eroded by pressure tube diametral creep alone around 2004 and this assuming acceptance by the regulator of our submission for increased setpoints for steam generator cleaning.

Fouling and Roughness Degradation

Fouling of the steam generators and pipe roughness degradation is expected to continue from magnetite formation and deposition. The combined effect of steam generator fouling and roughness degradation will be to increase inlet header temperature and to limit flow increases expected from pressure tube diametral creep, resulting in a decrease in reactor header-to-header differential pressure. Both of these effects have indeed been observed in recent G2 operating data, as shown in Figures 7 and 8. The combined effects expected from fouling and roughness degradation is illustrated in the second curve given in Figure 6. This additional setpoint reduction corresponds to expected increases in reactor inlet header temperature at a rate of 0.5°C/1000 EFPD and decreases in header-to-header pressure drop of 30 kPa/1000 EFPD (based on past experience). A shown, the 5% margin for flow increase obtained from the steam generator cleaning, if accepted by the regulator, would be entirely eroded by the combined effects of fouling, roughness degradation and pressure tube diametral creep no later than 2002.

Further Mitigation of HTS Aging Mechanism

The setpoint reductions shown in Figure 6 illustrate the future challenges to full power operation of the Gentilly-2 nuclear generating station. Meeting these challenges is important both for operation until current end-of-life of the station and for possible plant life extension; indeed, after a plant refurbishment, HTS aging effects will eventually result in power limitations which will affect the return on investments. Solutions will need to provide significant increases in margin to OID critical heat fluxes to ensure full power operation in the current regulatory context, as shown in Figure 6. Remaining available corrective actions, even when combined, may not provide enough additional margin to permit full power operation until the expected end-of-life of Gentilly-2. In addition, most remaining options are complex and expensive to implement and the actual gains expected from their implementation are, in some cases, uncertain. Possible corrective actions include the following in order of increasing level of complexity:

- optimization of the channel power distribution to maximize ROP setpoints,
- comprehensive CIGAR measurements to reduce the uncertainty in diametral creep rates variations,
- four-bundle shift refueling,
- use of CANFLEX bundles expected to provide higher CHF,
- alignment of the fuel bundle string using bundles with modified endplates,
- chemical cleaning of the primary heat transport system.

ROP Trip Confidence and Effectiveness

Current power limitations due to aging mechanisms are the result of reduced margin to ROP trip setpoints. ROP setpoints are determined in order to prevent OID CHF in the event of a postulated slow loss of regulation accident. Both in-reactor and out-of-reactor tests [3,8] have shown that there is a significant margin between OID CHF and the actual regulatory design requirement for CANDU reactor shutdown systems, which is to prevent "failure of the primary heat transport system due to over pressure, excessive fuel temperatures or fuel break-up" [1]. Accordingly shutdown system trips are shown to be effective after limited periods of post-dryout operation (PDO), as long as sheath temperatures remain under temperature limits at which fuel sheath integrity has been demonstrated in the experiments. This has been recognized by the AECB in the C-144 regulatory guide [9]. ROPT setpoints are particular since they are designed to ensure shutdown system trip for slow loss of regulation accident scenarios, in which time duration in PDO can be long and for which a large number of neutron flux shapes need to be covered. ROPT setpoints are therefore determined in a best-estimate probabilistic assessment ensuring that reactor trip will prevent OID dryout with a 98% confidence level for slow loss of reactor regulation events. The ROP system was designed to be highly reliable, with a significant safety margin at reactor trip.

With the current levels of operating margins to ROP trip setpoints, any reduction in margins to OID CHF will result in a corresponding limitation in reactor power, even though this power limit may not be justified on the basis of the shutdown system requirements set by the regulator. Indeed, there would still be a significant margin to OID CHF, at the original design ROP trip setpoints, for a significant number of the possible flux shapes that could result of a loss of regulation. In most postulated LOR scenarios, dryout would be of limited duration, for the most limiting flux shapes, with a significant margin to fuel overheating that could result in failure of the primary heat transport system. Experiments on full scale bundle test assemblies, such as the tests recently completed at Stern Laboratories [3], show that, for LOR flow conditions, the channel power has to be increased by more than 15% above that which results in OID dryout to get the sheath temperatures to around 600°C. The tests further show that, at these sheath temperatures and fuel element powers, the heat transfer from fuel to coolant is high enough to ensure a significant margin to fuel center line melting. Time in dryout could only be unacceptable for very improbable slow loss of regulation accidents; in these events there would be a number of indications of the LOR occurrence in the control room so that operator action could be credited to limit the period in post dryout to an acceptable duration.

It is not unreasonable to suggest that changes to the ROPT design could permit operation of the reactors at full power, even when the aging mechanisms not included in the original design are considered. This will surely be complicated and controversial, as this may require hardware changes for new instrumentation, such as an increased number of neutron flux detectors, and trip control logic, such as automatic setpoint reductions. Although, these changes may not be easily introduced in operating reactors, the challenge of producing an ROPT system, which could provide operating margin such that full power operation would be achieved during the entire expected life of the stations, should not be disregarded, if we intend to operate and extend the life of CANDU-6 reactors.

Conclusions

Monitoring of Gentilly-2 heat transport system data has indicated that the following aging mechanism have occurred in the station since first commissioning:

- reactor inlet header temperatures increased due to steam generator performance degradation caused by fouling in the steam generator and steam generator divider plate bypass flow;
- diametral expansion of pressure tubes resulted in an increase in maximum pressure tube diameter; in 1998, the range of this expansion was around 2% to 3% for high power channels;
- magnetite deposition has led to an increase in the hydraulic resistance of the heat transport piping, which has led to a decrease in core pass flows.

A number of actions have been taken to mitigate the impact of these aging mechanisms on safety margins and to maximize allowable operating reactor power. These include:

- reduction of effective ROP setpoints from 121% to 112%,
- replacement of the four steam generator bolted divider plate by the welded design,
- reduction of the secondary heat transport steam pressure,
- careful channel selection in order to minimize channel power peaking factor,
- cleaning of the primary side of the steam generators using SIVABLAST.

The observed evolution of the Gentilly-2 heat transport system hydraulic characteristic has indicated the importance of systematic monitoring of heat transport system data in order to ensure that safety margins remain acceptable. To this end, we are integrating a number of activities at G2 to monitor the heat transport aging and to take appropriate actions to ensure that the plant is maintained in the safe operating envelope.

In spite of the success of these actions, ROPT margins are expected to be further eroded by aging mechanism such that, in the near future, operation at full power may no longer be possible. Corrective actions need to provide significant increases in margin to OID critical heat fluxes to ensure full power operation up to the end of station life, if there is no change to the current regulatory context. Most remaining options are complex to implement and the actual gains expected from their implementation are still to be confirmed. These include the following :

- optimization of the channel power distribution to maximize ROP setpoints,
- CIGAR measurement to reduce the impact of variations in diametral creep rates,
- four-bundle shift refueling
- use of CANFLEX bundles expected to provide higher CHF,
- alignment of the fuel bundle string using bundles with modified endplates,
- chemical cleaning of the primary heat transport system.

Hydro-Québec will implement some of those measures in the near future, more specifically the optimization of the channel power distribution and CIGAR measurements. The other alternatives are still under evaluation. Of course, change in the Regulatory requirements are not expected unless there is a joint effort to resolve this issue by all the concerned parties.

ROPT setpoints are designed to ensure shutdown system trip for slow loss of regulation accident scenarios in which time duration in PDO may be long and for which a large number of neutron flux shapes need to be covered. ROPT setpoints were therefore determined in a probabilistic assessment ensuring that reactor trip will prevent OID CHF with a 98% confidence for slow loss of reactor regulation events. It has been recognized that shutdown system trips remain effective after limited periods of post-dryout operation (PDO) as long as sheath temperatures remain under temperature limits at which fuel sheath integrity has been demonstrated in the experiments [9]. It is important to note that prevention of OID dryout for all shutdown system process trip setpoints would require limiting channel and bundle powers to a level where full power operation would also not be possible, even with the additional margins that could result from a change to CANFLEX fuel. Possible changes to the ROPT system design, which could provide greater operating margin, should not be overlooked as a possible solution to ensure full power operation during the entire life of the stations.

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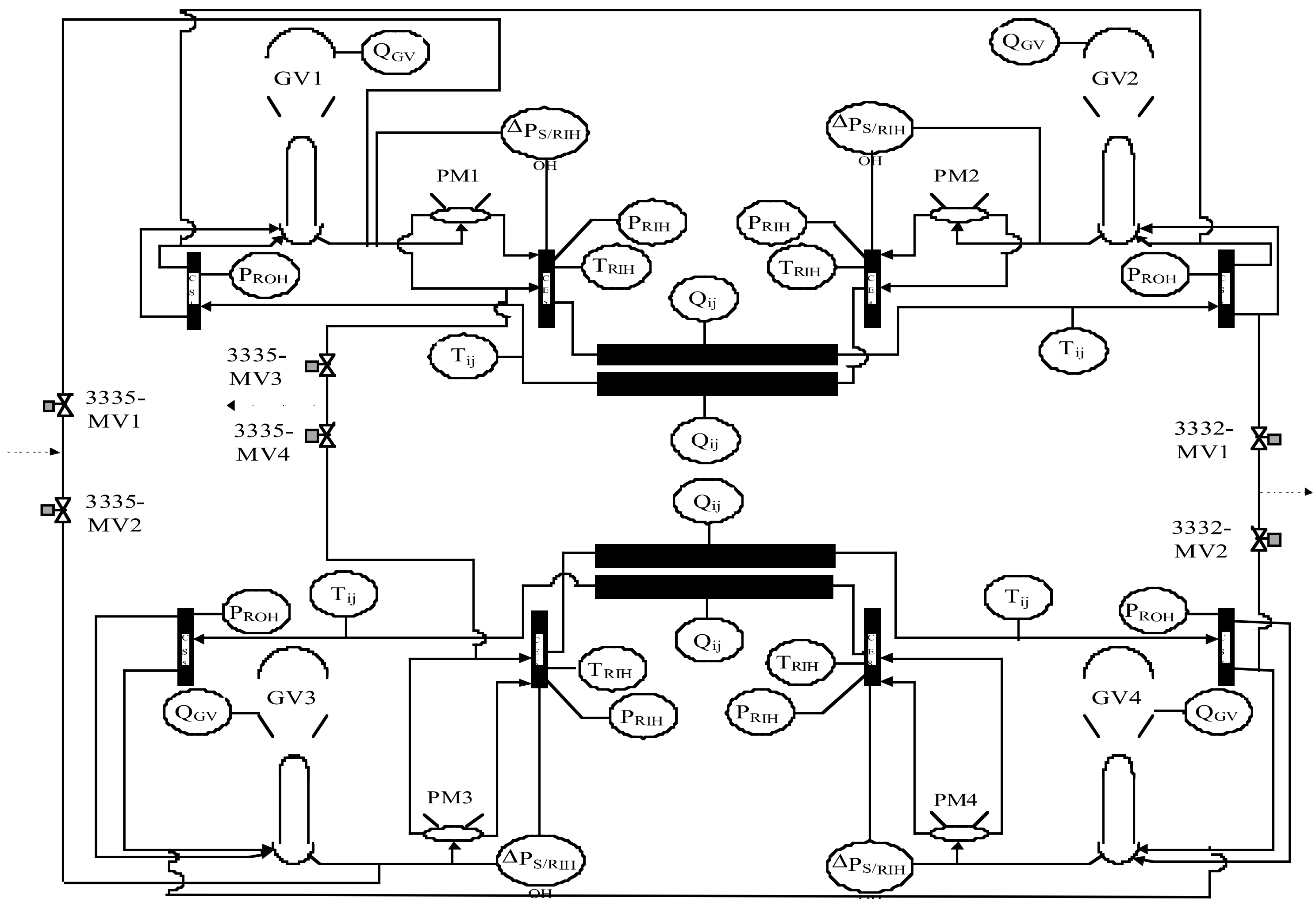

Figure 1
Heat Transport Instrumentation used to Monitor Coolant Flow and Subcooling

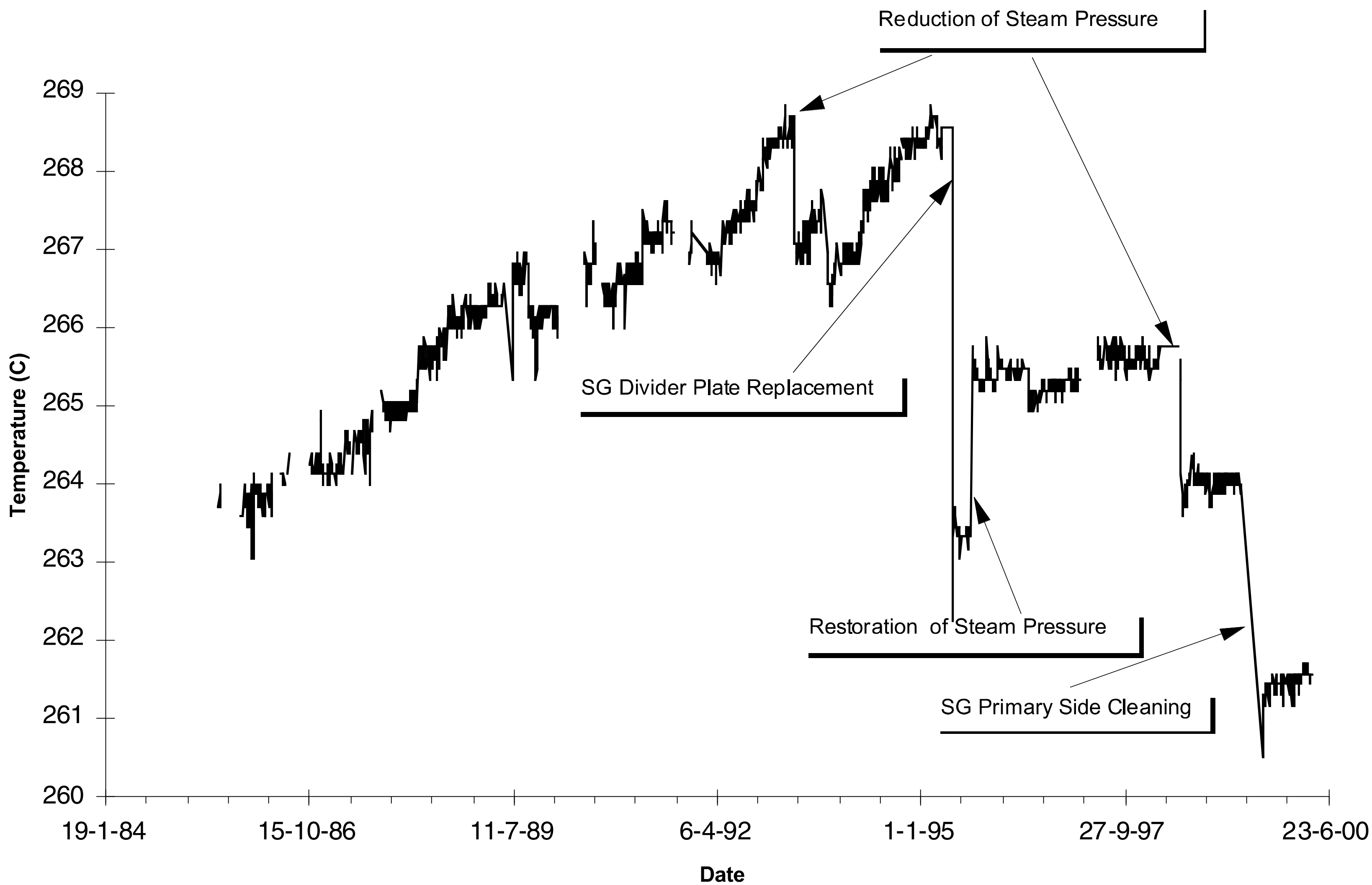

Figure 2
Maximum Reactor Inlet Header Temperature

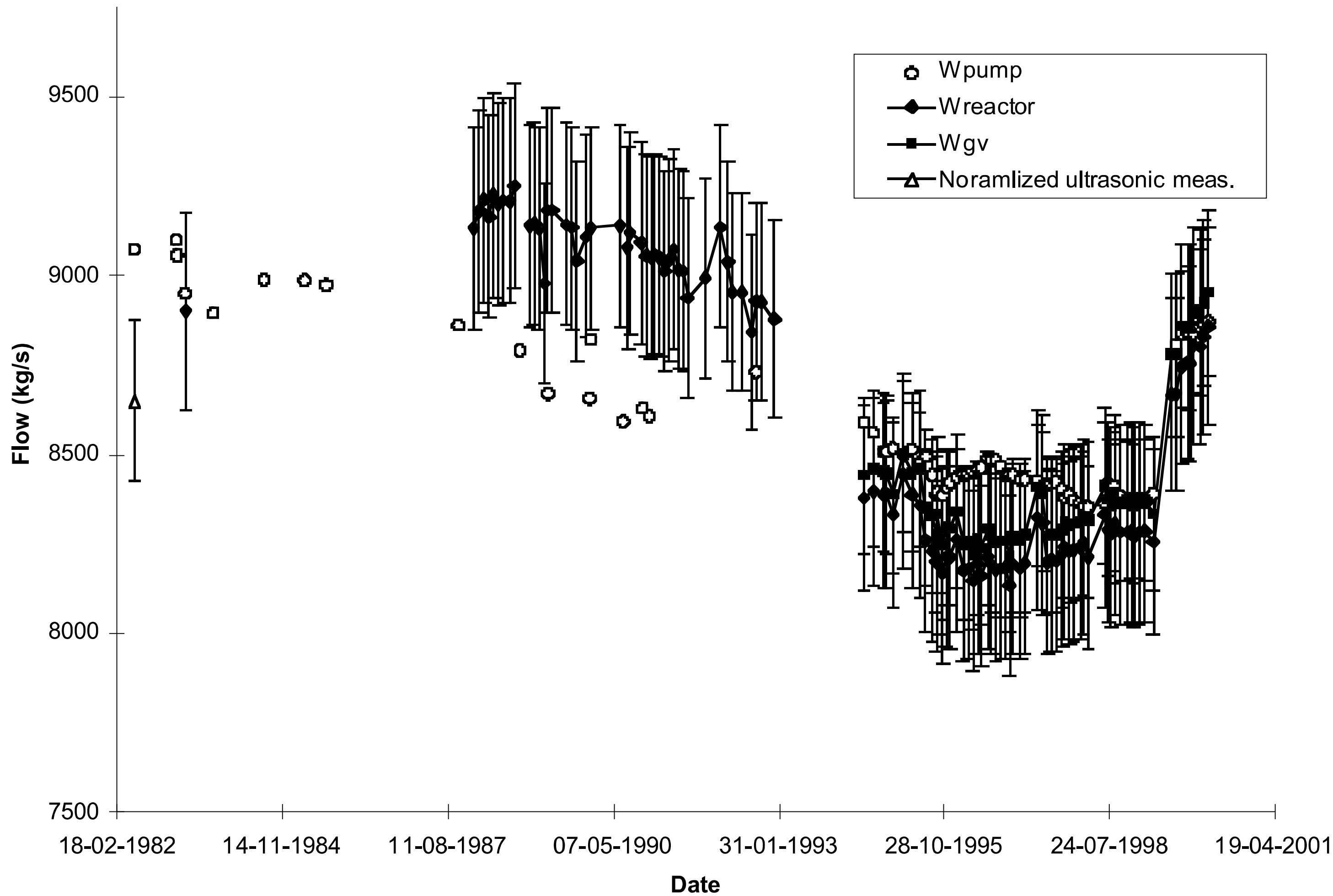

Figure 3
Evolution of Total Single Phase Core Flow Based on Pump Head and Heat Balance Calculations

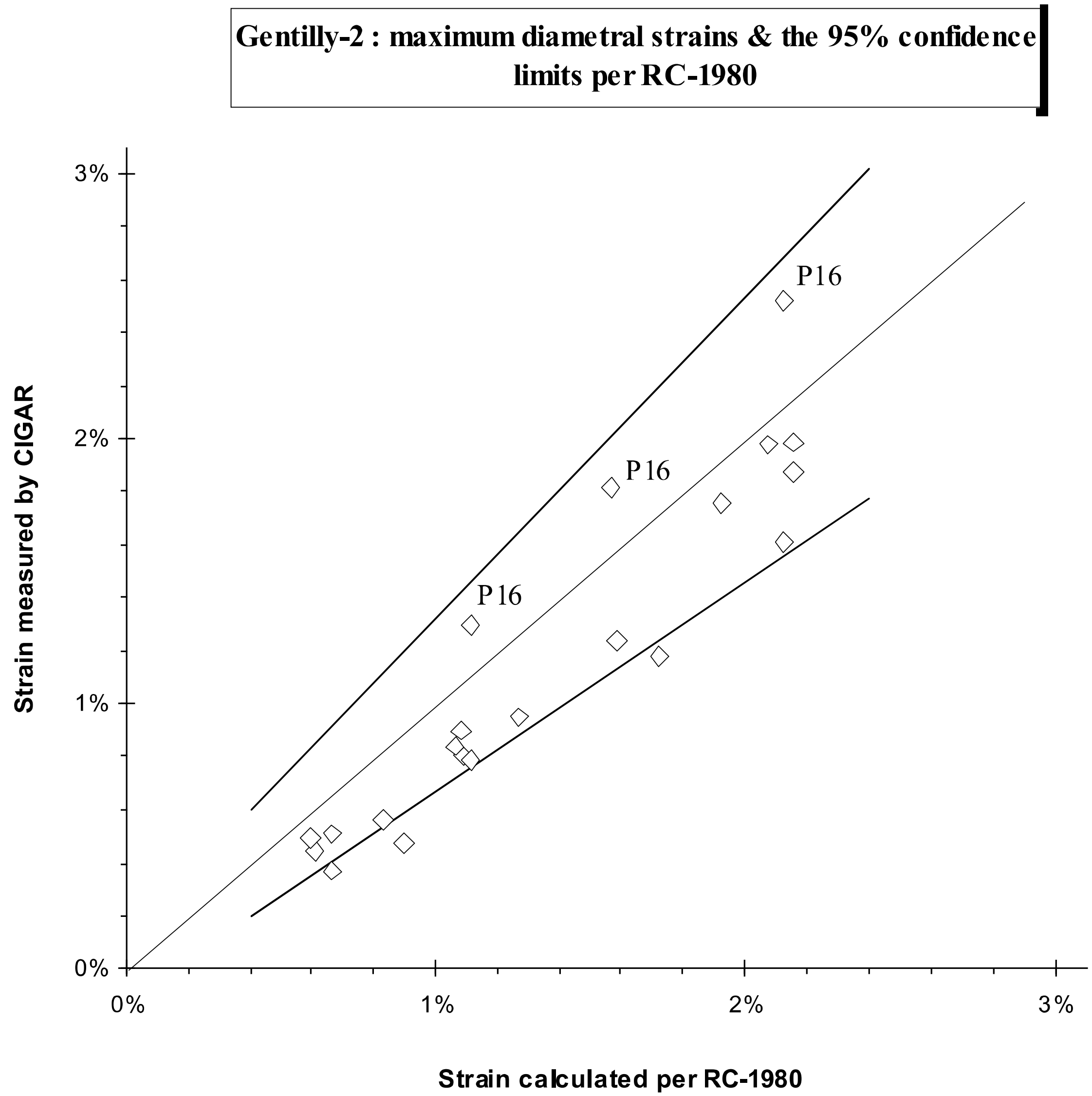

Figure 4

Maximum Measured Diametral Strains for the Gentilly-2 Pressure Tubes as a Function of the Maximum Diametral Strains Calculated per Equation RC-1980
(Compared to the 95% confidence limits based on all the data through 1997 from CANDU 6 reactors)

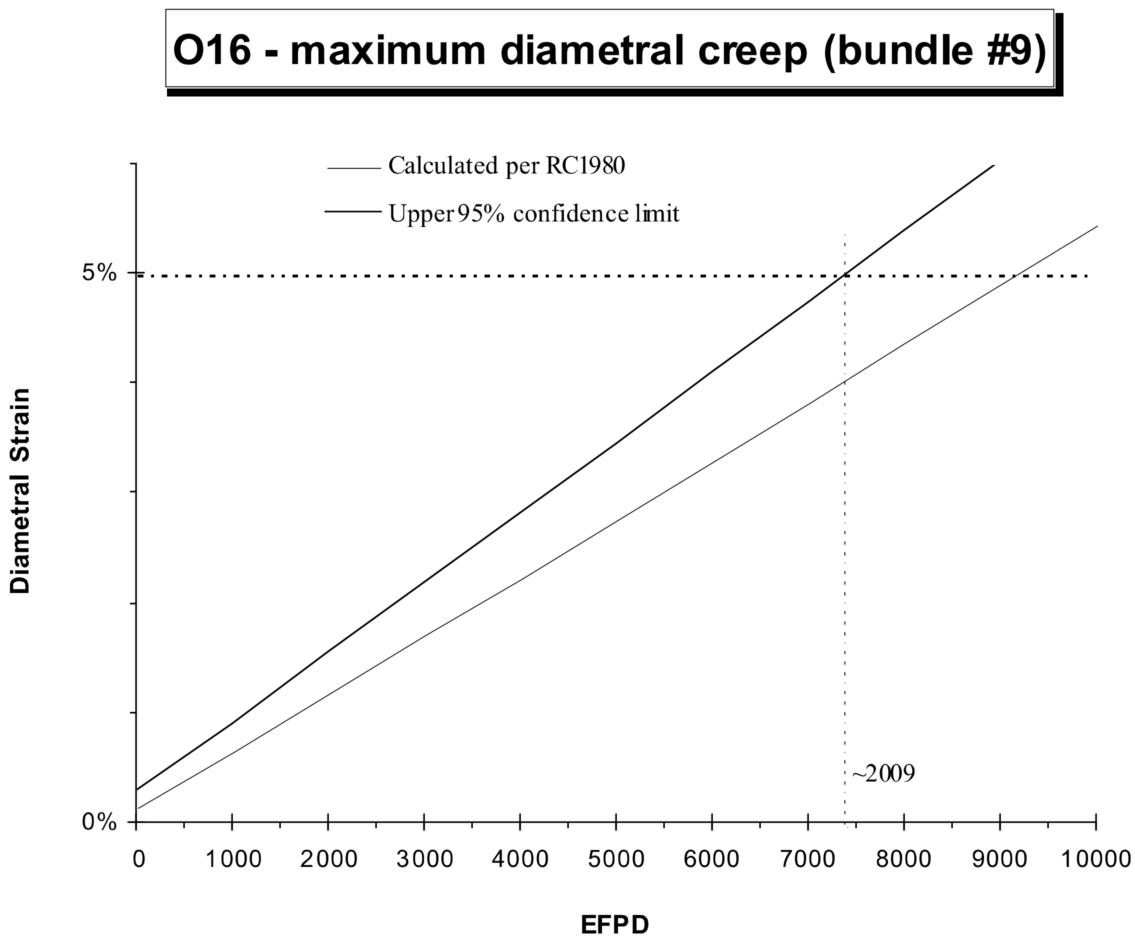

Figure 5

Maximum Strain (i.e. the channel with the highest predicted strain rate - Channel O16) Calculated per Equation RC-1980
(The upper 95% confidence limit is established on the basis of the data through 1997 from all the CANDU 6 reactors)

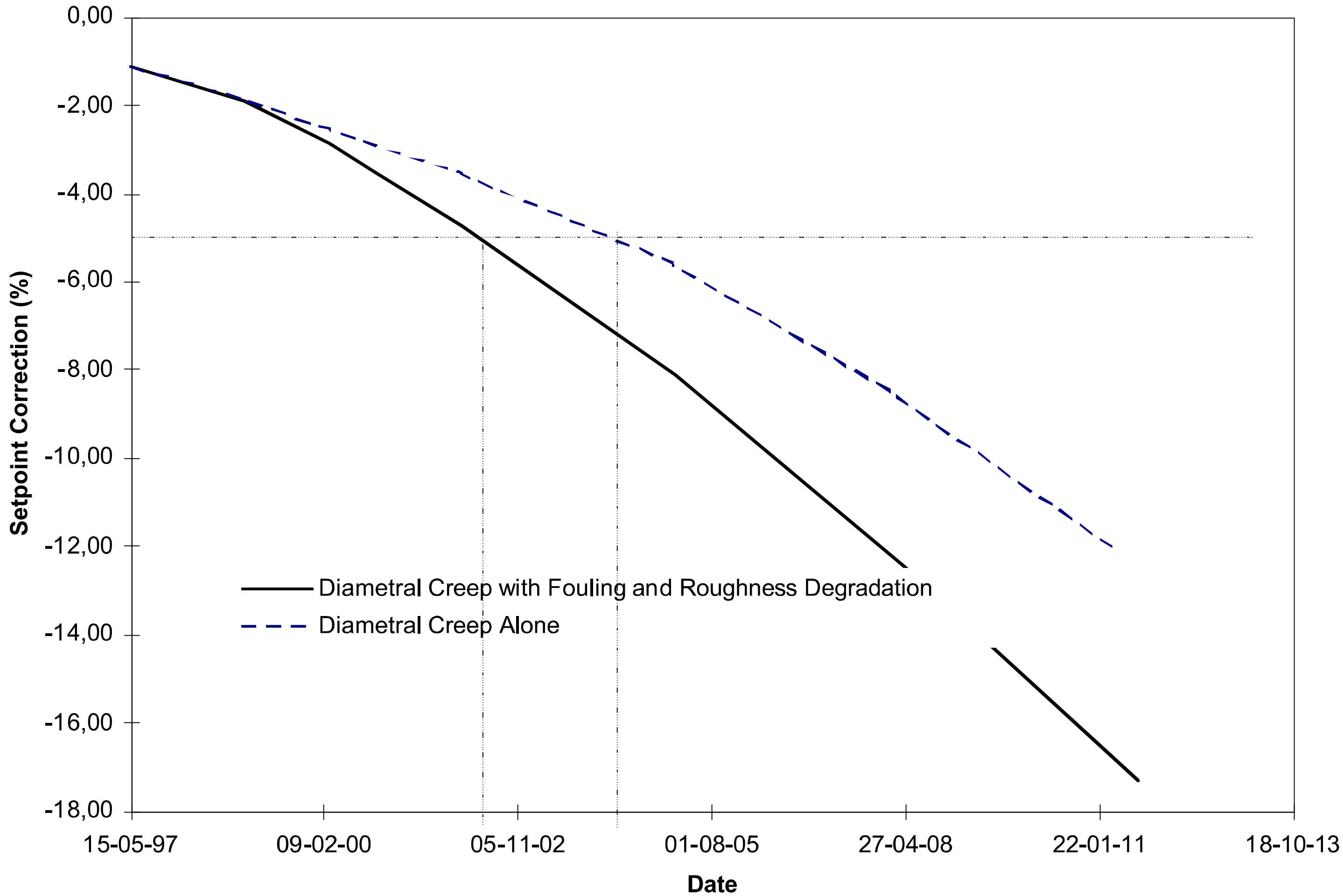

Figure 6
ROPT Setpoint Correction for PT Diametral Creep and HTS Aging

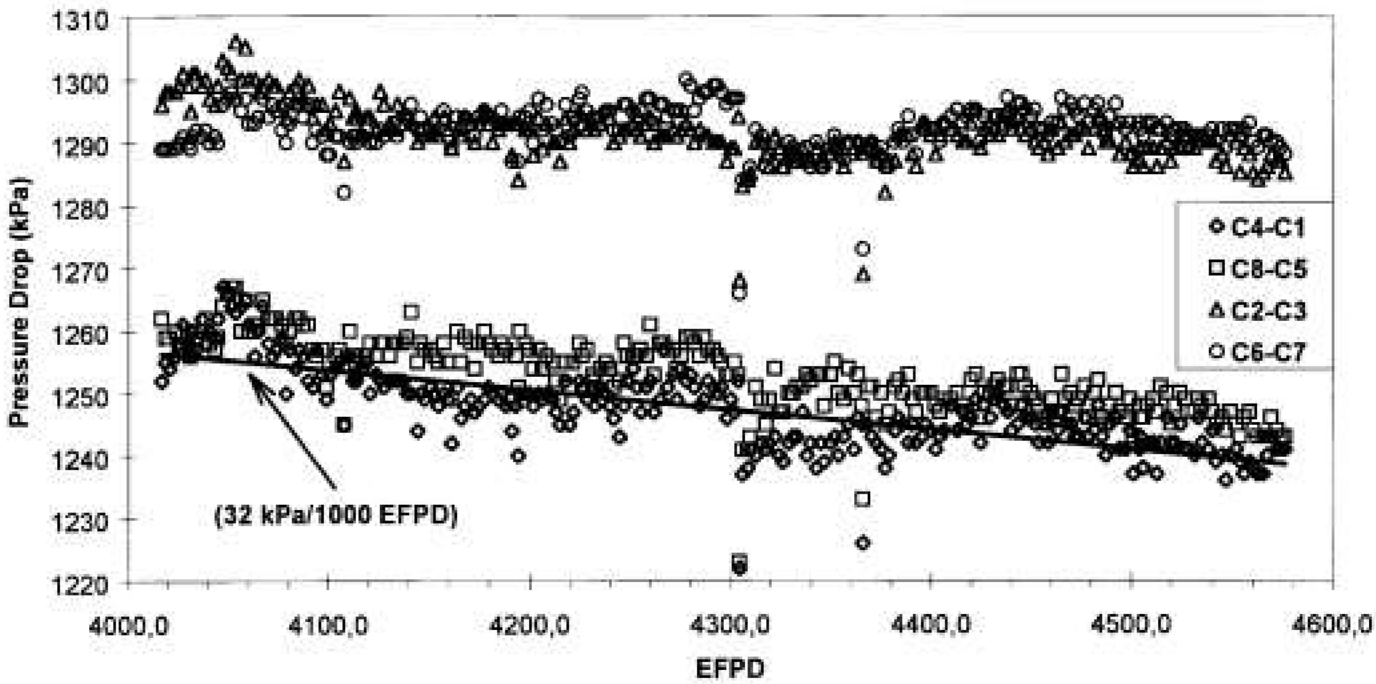

Figure 7
Typical Recent Evolution of Header-to-Header Pressure Drop

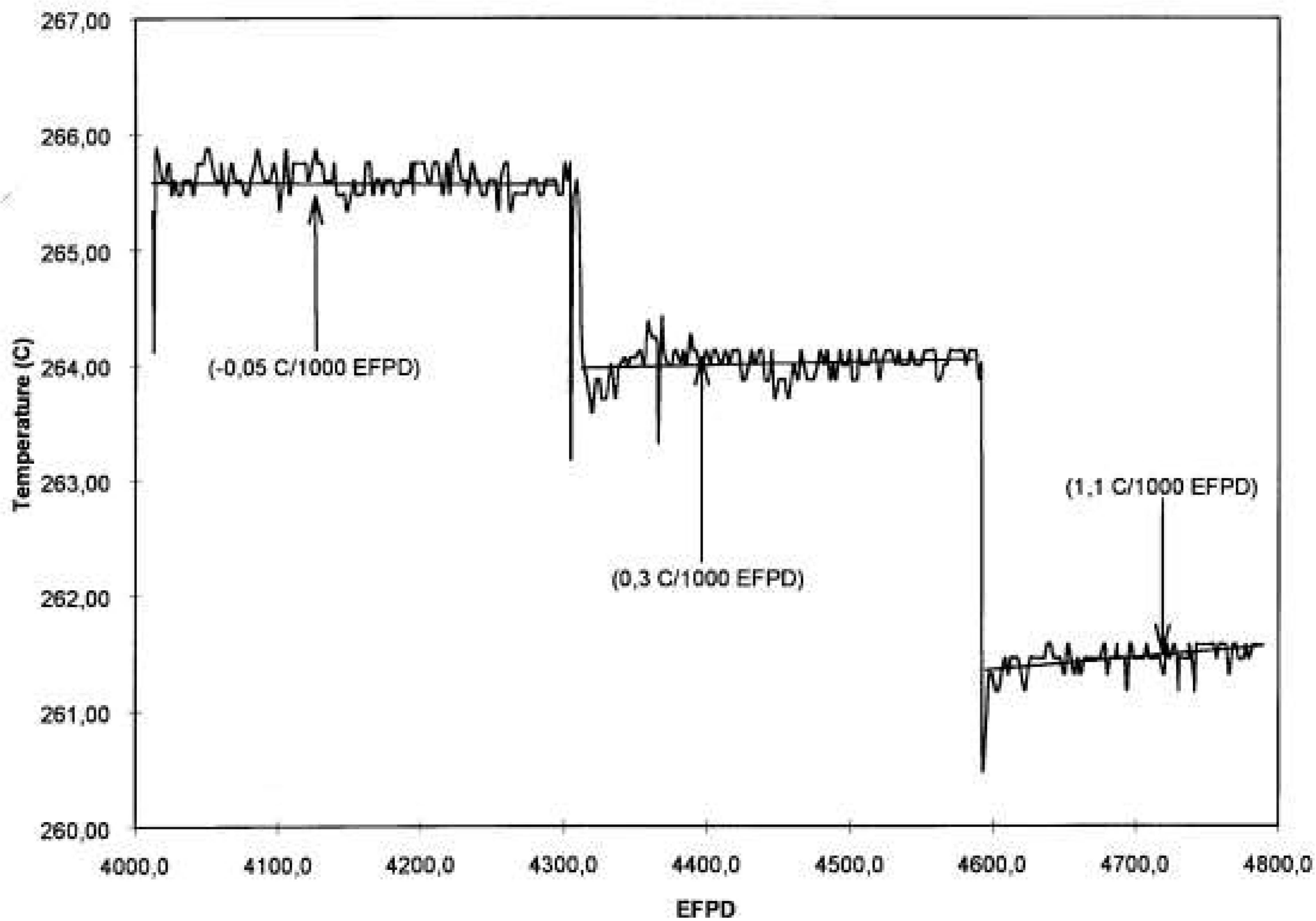

Figure 8
Typical Recent Evolution of Reactor Inlet Header Temperature