MONITORING AND MANAGING COMPONENT FATIGUE IN CANDU STATIONS

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Abstract

Many CANDU® plants are now approaching their end of their design lives and are being considered for extended operation beyond their design life. The Canadian Nuclear Safety Commission (CNSC), has asked utilities to consider component fatigue issues in plant life extension (PLEX) applications. In particular, environmental effects on fatigue was identified as an issue that needs to be addressed, similar to being addressed for license renewal for U.S. nuclear power plants. To address CNSC concerns, the CANDU Owners Group (COG) has initiated a program to help utilities develop component fatigue management programs for PLEX operation. A pilot study conducted at the Point Lepreau Generating Station (PLGS) showed that

- Only 10 to 15% of the design transients have been used after 25 effective full power years (EFPY) of operation. Hence, a significant amount of original design margin for fatigue usage margin remains available for PLEX operation.
- Environmental fatigue considerations in heavy water (D_2O) were included in the assessment. Only warm-up transients were assessed to have dissolved oxygen concentrations that can result in a significant environmental effect for the ferritic steels used in the CANDU primary systems.
- Due to the low accumulation of transients and the absence of known thermal stratification mechanisms, thermal fatigue is not as significant an issue in CANDU plants as in pressurized water reactor (PWR) and boiling water reactor (BWR) plants.

This paper summarizes the results of the pilot study conducted for the CANDU plants.

1. Introduction

In recent years, significant industry attention has been devoted to metal fatigue and its impact on the design qualification and serviceability of operating nuclear power plant components. It has been estimated that anywhere from 50% to 90% of all nuclear component failures are caused by fatigue [1]. The interest in fatigue has also increased as the majority of operating reactors are in the last half of their original design lives, and many are approaching the end of design life with the expectation of life extension.

Some fatigue failures in safety-related systems and components have occurred throughout the industry. Fatigue damage in pressure-retaining equipment has typically manifested itself as small cracks or leaks. If not detected early, a small crack or leak can grow to a critical size where a major pressure boundary rupture becomes possible. Thus, fatigue can eventually become a safety issue if

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periodic inspection and leak detection are not adequate. However, fatigue has been identified as primarily a shorter-term economic issue because pressure boundary leaks lead to forced plant shutdowns long before rupture or failure is predicted to occur. Confirmation of the adequacy of the fatigue life of metal components is therefore a pressing economic and a safety issue, especially for operation beyond the originally planned operating life (i.e., license renewal or PLEX period), where the risk of fatigue damage increases.

CANDU nuclear generating stations and their components were originally designed for 30 years of operation at 80% power (equivalent to 24 EFPY of operation) including adequate margins on fatigue life. Many CANDU plants are now approaching their design end-of-lives and are being considered for extended operation beyond their design lives. The CNSC has expressed concern on component fatigue for extended plant life. The United States Nuclear Regulatory Commission (U.S. NRC) requires U.S. nuclear generating stations to commit to implementing a comprehensive component fatigue management program as part of their license renewal applications for extended operation.

To help CANDU utilities develop comprehensive fatigue management programs for PLEX, the CANDU Owners Group, Inc. (COG) initiated a work package in 2005. Under this work package, a strategy was developed and the elements of an effective fatigue management program for PLEX were identified. A pilot study was then conducted at a CANDU station (PLGS) to identify the specific steps necessary to implement a comprehensive fatigue management program.

2. Scope of the Fatigue Management Program

A phased approach is proposed. The initial scope of the program is limited to Class 1 piping at the PLGS. Other Class 1 components (fuel channel components, vessels, etc.) will be later included in the program. Additionally, some known fatigue locations beyond this scope are included if they are judged to be a safety or an economic issue.

Following the Canadian standard CSA-N285.0-95 [2] for CANDU systems and components, the following systems are classified as Class 1 systems and are considered for the fatigue management program: Heat Transport System (HTS), Pressure and Inventory Control System (P&ICS), Fuel Channel Assemblies, Feed and Bleed System, Gland Seal System, Main Purification System, Shutdown Cooling System, Emergency Core Cooling System and Recovery Circuit, Liquid Injection Shutdown System, Fuelling Machine Head, Fuelling Machine D_2O, Main Steam System, Gross Activity Monitoring, and Failed Fuel Detection System. Fatigue evaluation is required for Class 1 components, and is generally performed in accordance with Section III of the American Society of Mechanical Engineers (ASME) Boiler & Pressure Vessel (B&PV) Code.

3. Regulatory Requirements

In Canada, the CNSC issues regulations concerned with the safety of Canadian nuclear plants, and licenses nuclear plants to be built and operated. The CNSC has no special requirements for fatigue other than what is specified in the CSA Standards, which often refer to the ASME Code and/or ANSI Standards. According to CSA Standard N285.0, fatigue evaluation is required for all Class 1 components.

Currently, there is no formal document that defines the CNSC requirements for component fatigue for PLEX operation. As the lead CANDU 6 plant, PLGS is preparing to apply for permission to extend operating life beyond 30 years. As a result, the CNSC has requested that PLGS identify critical components with high fatigue usage factors, address the effect of the water environment on component fatigue life, and provide an acceptance criterion for PLEX operation. CNSC referred to Section 4.3 of the US NRC document NUREG-1800, "Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants" [3], for guidance and details.

In the absence of any CNSC regulations and formal guidance, the proposed approach for Canadian nuclear generating stations closely follows that of the existing U.S. approach.

Section 4.3 of NUREG-1800 includes the Standard Review Plan (SRP) for metal fatigue analysis of nuclear components. According to this document, license renewal applicants are required to demonstrate that the existing fatigue calculations remain valid either by: (1) demonstrating that the number of assumed transients will not be exceeded during the extended period of operation, (2) re-evaluating the fatigue calculations based on an increased number of assumed transients that bound the period of extended operation, or (3) manage the effects of fatigue using an aging management program. The SRP also refers to the Generic Aging Lessons Learned (GALL) Report [4] to ensure that applicants have developed a program to monitor and track the number of critical thermal and pressure transients.

The GALL report contains the U.S. NRC's generic evaluation of existing plant programs and documents the technical basis for determining where existing programs are adequate without modification and where existing programs should be augmented for the extended period of operation for PWR and BWR plants in the U.S. The relevant section of the GALL Report for fatigue is Chapter X.M1 – Metal Fatigue of Reactor Coolant Pressure Boundary.

4. CANDU Fatigue Failure Operating Experience

Information on fatigue failures of CANDU piping is available in various sources and databases. However, there is no convenient way to compile, query and report this data. This makes it difficult to analyze the data and to establish patterns and trends. Using various sources, a review of all available data was performed. The review shows that only a very small number of thermal fatigue failures have occurred in Class 1 piping in CANDU generating stations. Literature review yielded the following Class 1 piping failures in CANDU plants:

- 1993 Bruce 1 – Thermal mixing tee connecting the feed line to the HTS.
- 1991 Embalse – Thermal mixing tee connecting the purification return line to the HTS.
- 1999 Bruce 8 – Thermal mixing tee connecting the feed line to the HTS.
- 2001 A CANDU 6 – Mixing tee weld failure at the tee connecting the purification line to the HTS.
- 2003 A CANDU 6 – The weldolet previously discovered in 2001 was found cracked again.

All of these fatigue failures were either at or downstream of the thermal mixing tees connecting the D_2O feed pump outlet lines to the Main HTS. Hence, the likely cause for these failures was cyclic thermal loads not originally considered in the plant design. In the CANDU 6 design, the abovementioned thermal tee connects the feed line to the purification return line en route to the HTS.

In addition to the above locations, a number of large-diameter (>2”) non-Class 1 piping fatigue failures occurred in the primary-side of various CANDU units. Almost all of these failures occurred in the Feed and Bleed system along the reflux lines. In older CANDU designs, reflux lines route part of the feed flow through the bleed condenser to reheat D_2O prior to feeding it into the HTS. The cause for fatigue in these lines is postulated to be flow-induced vibration caused by large pressure losses across valves and orifices.

5. Environmental Fatigue

Environmental fatigue multiplication factors, F_{en}, are used to assess the effects of the reactor water environment on fatigue. The impact of the reactor water environment on the fatigue life of components in CANDUs has been considered within the Canadian industry in the context of insights gained from the investigation of the PLGS feeder failures. Since the discovery of the first cracked feeder at PLGS in 1997, the primary HTS water chemistry, specifically the presence of oxidizing species, and their effect upon environmentally assisted fatigue cracking in the HTS have been extensively investigated [5]. In particular, warm-up transients have been identified as periods where elevated levels of oxidizing species may be present. This situation occurs primarily in the earlier stages of warm-up transients before water chemistry is controlled to a level where environmental effects are minimized.

The behaviour of carbon steels subject to environmentally assisted cracking in oxygenated heavy water is expected to be the same in CANDUs as it is for light water reactor systems (BWR/PWR). The only major difference in water chemistry, in the context of corrosion, between CANDU primary coolant and BWR/PWR systems is the relatively high pH maintained with LiOH addition (specified pH[1] range is 10.2 to 10.4 at 25°C). Experimental work at Chalk River Laboratories (CRL) has shown that there is no difference in cracking susceptibility and cracking behaviour of carbon steel in water up to pH 11 and pH 7 water for strain-induced corrosion cracking. The mechanism of strain-induced corrosion cracking can be viewed as a very-low cycle (1 to 100 cycles) form of corrosion fatigue. As a result, it is reasonable to apply the environmental fatigue multiplication factor directly to the low-cycle fatigue assessments in CANDU HTS.

In carbon-steel CANDU components, high environmentally-assisted fatigue usage factors may exist for warm-up transients based on high concentrations of oxygen that might exist in the HTS. Other transients are expected to contribute very little (i.e., a factor of two or less) to environmental fatigue of carbon steel because the oxygen levels are maintained at very low levels (<0.01 ppm) under normal operating conditions. In recent years, PLGS has implemented new operating procedures that minimize adverse environmental conditions during warm-up transients [6].

6. Fatigue Critical Components

For the PLGS pilot study, the components with possible high fatigue usage factors are identified using the following process:

[1] This is the apparent pH of heavy water, which is referred as pHa in CANDU stations.

- A review of all relevant design basis stress reports was performed for PLGS plant components that possess a Class 1 fatigue analysis. Fatigue-susceptible locations from design stress analysis reports were identified. Class 1 piping locations with cumulative fatigue usage factors (CUFs) greater than 0.5 were identified as possible locations. Where there were multiple locations on a line that satisfied these requirements, the highest CUF location was selected. High CUF locations from stress analysis reports are rarely seen to fail, as they are typically calculated to satisfy the code compliance (i.e. CUF smaller than 1) rather than estimating the CUF as close to reality as possible, resulting in very conservative estimates.

- An additional review was also performed to evaluate thermal fatigue effects not included in the original plant design. Non-design thermal loads have received considerable attention after some Class 1 fatigue failures in the LWR community. A review of the CANDU 6 design using the EPRI/MRP-146 thermal cycling screening methodology indicated that the PLGS design has a relatively low propensity for thermal-fatigue of normally-stagnant branch lines. Two possible locations of mixing tees along the return D_2O line to the HTS were identified for potential thermal fatigue mixing effects.

- CANDU industry experience with fatigue failures was also reviewed. The review shows that only a very small number of thermal fatigue failures have occurred in the Class 1 piping of CANDU generating stations. All of these fatigue failures were either at or downstream of the thermal mixing tee connecting the D_2O feed pump outlet lines to the HTS. The cause of these failures was mostly due to vibration and thermal mixing. The thermal mixing tee experience is consistent with the findings of the thermal screening discussed in the previous bullet.

- The past PLGS experience was reviewed. No fatigue failures have occurred in the HTS at PLGS. Some vibration problems have been encountered in some Pressure and Inventory Control System (P&IC) lines, which is identical to the location of some failures that have occurred in other CANDU plants. PLGS has taken measures to reduce vibration levels in those lines, but these lines are nevertheless identified for fatigue monitoring.

- The adverse effect of reactor water environment on fatigue was evaluated. The worst environmental location was identified to be hot-leg of the HTS. Although PLGS chemistry specifications minimize the environmental effect under normal operating conditions, it was assessed that oxygen levels can be very high during the initial part of the warm-up transient, potentially resulting in a high environmental fatigue multiplication factor. PLGS has modified its warm-up procedures to minimize dissolved oxygen and, hence, the detrimental environmental effect.

7. Fatigue Monitoring Options

For components identified with a high CUF potential, it was recommended to manage fatigue through a combination of monitoring, inspection and analysis. Depending on the location and the mechanism causing the fatigue usage, the following fatigue monitoring options were recommended for consideration:

- *Cycle-Based Fatigue Monitoring*: Cycle-based fatigue monitoring consists of computing fatigue directly from counted transients and parameters using a pre-defined fatigue algorithm (i.e., the fatigue table from the governing stress report). Components in Class 1 pressure boundary are selected for monitoring that bound or represent all other components. A variation of this approach is to simply count actual transient cycles and to ensure that the severity and the numbers of transient cycles do not exceed those specified in the design reports.

- *Stress-Based Fatigue Monitoring*: Stress-based fatigue monitoring consists of computing a "real time" stress history for a given component from actual temperature, pressure, and flow histories that are available from existing plant instruments. Fatigue usage is then computed from the computed stress history using appropriate techniques. Stress-based fatigue methodology requires more effort, but provides the most extensive, refined fatigue analysis for a component without the need for categorizing and counting plant transients. Stress-based fatigue monitoring is intended for those high-fatigue components where a more refined approach is necessary to show long-term structural acceptability.

- *Fatigue Crack Growth Evaluation (Flaw-Tolerance Approach)*: Fatigue crack growth may also be monitored in relevant components. Fatigue crack growth monitoring is part of a damage tolerance-based examination strategy designed to assure that the component will operate reliably between subsequent inspections even though a design fatigue limit has been exceeded or a relevant fatigue analysis is not available. These "flaw tolerance" evaluations are used to demonstrate that actual or postulated flaws remain within allowable limits. Components in the plant that are expected to accumulate abnormally high CUFs, or that lack a fatigue design basis calculation, may be monitored using a fatigue crack growth fracture mechanics methodology. Fatigue crack growth monitoring functions similarly to stress-based fatigue monitoring, except that fatigue crack growth is computed from a real-time stress intensity history rather than computing CUF from a real-time stress history.

8. Fatigue Monitoring at the PLGS

The thermal duty experienced by the Class 1 pressure boundary components is specified by thermal transient definitions. These definitions specify the temperature, pressure, and flow rate time history experienced by the component for a series of events postulated by the designer. One easy way of "measuring" a plant's actual thermal duty is by counting thermal transients as they occur and categorizing them as one of the thermal transients assumed by the designer. The transients specified in the plant design basis are therefore a key element in any fatigue management program. In addition, which transients to monitor are important, as counting benign transients that are very numerous may make the program overly complicated, while adding little value. It is important to understand which transients are most important to thermal duty (fatigue) accumulation and must be counted. Finally, design basis transient severity versus actual transient severity should be confirmed.

8.1 Plant Transients

An essential component of a fatigue management program is to classify major transients and to keep track of the number of each of these defined transients that occur during the lifetime of the plant. Typical CANDU transients for the HTS are listed in Table 1. In this table, the transients are classified according to various service conditions, consistent with ASME Code Section III definitions. Level A loads are experienced during normal operation, such as start-up, shutdown, and operation at power. Level B loads occur during upset conditions that are caused by loss of power or the activation of safety systems. Level C loads are defined as those that deviate from normal conditions and that occur infrequently during the life of the system and that might necessitate a shutdown for correction of the condition or repair of damage. Level D loads are those that significantly affect the integrity and the operation of the system. Level C and D transients need not be monitored, because they rarely occur and do not contribute to fatigue usage. The significant transients that require monitoring are identified in Table 1.

Table 1
Typical CANDU Transients of the HTS System

Load Type	# of Design Transients	To be Monitored
Test Loads		
Hydrostatic Test	10	Yes
Leak Test	10	Yes
Commissioning		
Hot Conditioning	1	No
Normal Operating Condition (Level A)		
Warm-up and Cooldown	500	Yes
Start-up and Shutdown	1000	Yes
Power Manoeuvring	10000	Yes
Upset Conditions (Level B)		
Turbine Trip / Loss of Offsite Load	100	Yes
Reactor Stepback	500	Yes
Reactor Trip from 100% Full Power	500	Yes
Loss of Class IV Power	50	No
Reactor Overpower / Loss of Regulation	200	No
Loss of Feedwater Supply from 100%	100	No
Emergency Conditions (Level C)		
Rapid Cooldown	15	No
Rapid Pressurization followed by a trip	1	No
Faulted Conditions (Level D)		
Steam Line Break	1	No
Loss of Coolant (Large break)	1	No

8.2 Automatic Monitoring of Plant Transients – ThermAND Software

The recommended approach for counting the cycles identified in Table 1 is with the use of an automated cycle counting (ACC) program. In addition to detecting events, the ACC can also collect descriptive statistics and profiles for each event (e.g., duration, maximum and/or minimum temperature, rates of change, etc.), thereby providing a means to compare design basis transient severity versus actual transient severity. Several ACC techniques have been developed over the course of time for the nuclear plant industry. Two of these techniques are developed and refined to the point that they are readily available for implementation in CANDU reactors: (1) ThermAND™, and (2) FatiguePro. FatiguePro was developed by EPRI and is widely used in over 50 nuclear units worldwide for comprehensive cycle counting and fatigue monitoring, and has become the standard for addressing license renewal Fatigue Management Program needs in the U.S. ThermAND was developed by AECL for system health monitoring and automatic cycle counting, and has been selected for automated counting of transients at the PLGS. ThermAND is suitable for cycle-based monitoring, whereas FatiguePro is a more appropriate tool for stress-based fatigue monitoring.

ThermAND™, AECL's system and component health monitor for heat transport systems, has been developed to monitor key operating parameters within these systems, and to integrate measured station data with models to provide predictable performance. ThermAND was developed by AECL using the same design as ChemAND®, the system health monitor for chemistry control, installed at Gentilly-2, PLGS, and Wolsong Unit 3. ThermAND has been on field trial at PLGS during which time it has been refined and improved. ThermAND permits users to be proactive by providing alarming on out-of-specification conditions and warnings when parameters are trending towards being out of specification. ThermAND permits the users to trend the time evolution of parameters calculated based on the values stored in the station data historian. Examples of these include calculating motor power from the measured pump current and bus voltage, and measuring the sensor drift of triplicated channels within the CANDU control logic. The model for reactor inlet header temperature (RIHT) increase has also been implemented to model this station performance parameter and to account for variations caused by changes in process parameters, e.g., secondary side pressure, reactor power.

A module of ThermAND is developed to permit cycle-counting based CUF monitoring within CANDU systems. A typical ThermAND screen showing refuelling transients is shown in Figure 1. By combining the results in ThermAND with the data in ChemAND, the environmental factor (see Section 5) can be assessed and used to modify the fatigue usage factor, where such chemistry (e.g., dissolved oxygen) and stress data exist.

8.3 Actual Number of Transient Cycles

The numbers of design cycles given in Table 1 are chosen very conservatively at the design stage. They were originally established for the intended plant life of 30 years. Although they were selected with conservative assumptions, the adequacy of the numbers of transient cycles needs to be demonstrated for life extension of the plant for 30 more years of operation. Hence, establishing the past number of transient cycles at the time of life extension, and estimating the numbers of transient cycles at the end of the extended operating period are key elements of a Fatigue Management Program for PLEX.

The numbers of normal operating transients that have occurred at PLGS as of 2006 June is graphically presented in Figure 2. A count of upset transients (i.e., trips or stepbacks) is plotted in Figure 3. Upset transients are harder to classify, because the events causing set-back, step-back, reactor trip or turbine trip are not determined ahead of time. As a result, in many upset transients more than one of these events occurs. Also, the design-upset transients are defined for shutdowns from 100% power. In Figure 3, any upset transient from 40% power level and higher are included, and all transients from less than 40% power level are ignored. The ignored transients are assessed to be insignificant from a fatigue point-of-view.

Figures 2 and 3 reflect a thorough attempt to summarize all of the plant duty experienced by PLGS since original plant start-up in 1982. The information used for transient counts came from normal shutdown records maintained for the Steam Generator (SG) System, from upset events recorded in the PLGS Reliability Database for Upset Events and from Event Reports.

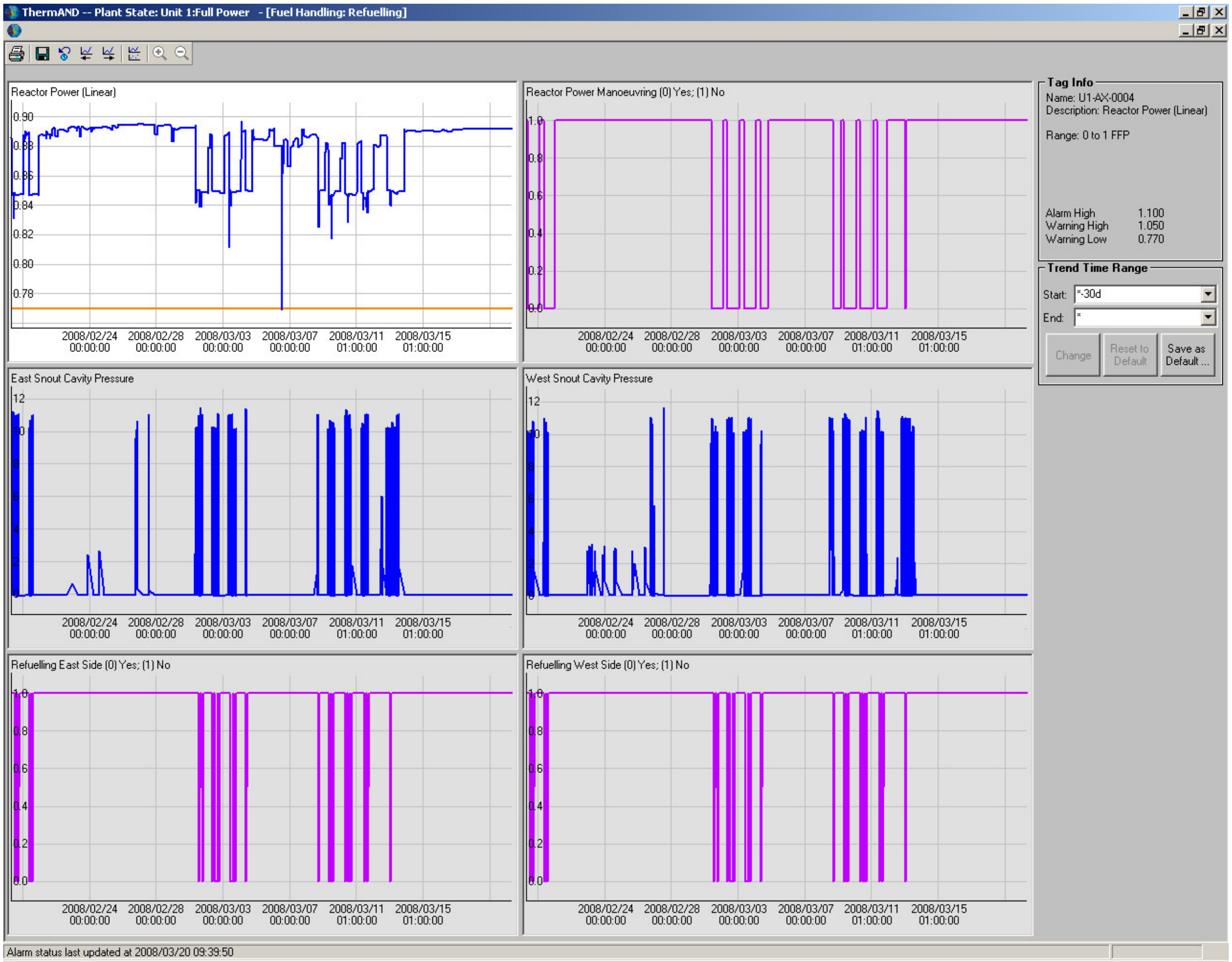


Figure 1 Example of a ThermAND screen showing refuelling transients.

8.4 Actual Severity of Transients

In addition to monitoring the actual numbers of transients, the fatigue monitoring program should demonstrate that the transient severity is the same or less benign than the assumed design transients. The transient severity is defined by the magnitude of the temperature change, the rate of temperature change, and the magnitude of the pressure change. Industry experience has demonstrated that actual transients are generally less numerous than assumed in the design basis, and are almost always significantly less severe.

Using ThermAND, the severity of actual transients was investigated for PLGS for the major transients listed in Table 1. As an example, the design warm-up transient and a an actual warm-up transient are plotted in Figure 4. In assumed design transients, temperature and pressure changes are typically linear and, hence, the definition of "severity" is well defined. In actual transients, the rate of change of temperature and pressure varies with time. In those cases, the highest rate of change (typically at the beginning of the transient) is conservatively used as the value to quantify transient severity. This is shown in Figure 4(b). As shown in this figure, the rate of change of actual warm-up transients is significantly smaller than the assumed design warm-up transient.

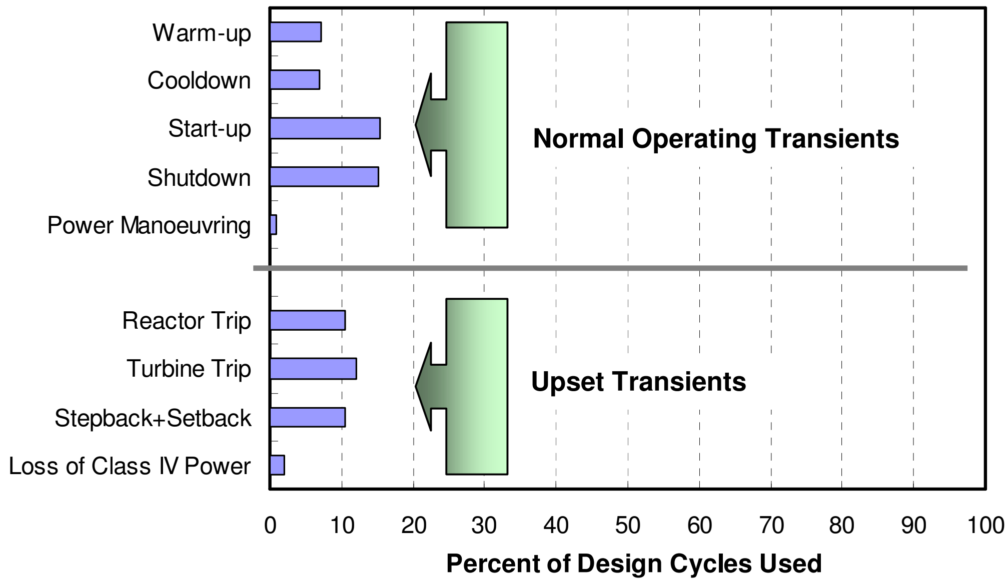


Figure 2 PLGS Transients and Usage as of 2006 June (24 Calendar Years of Operation)

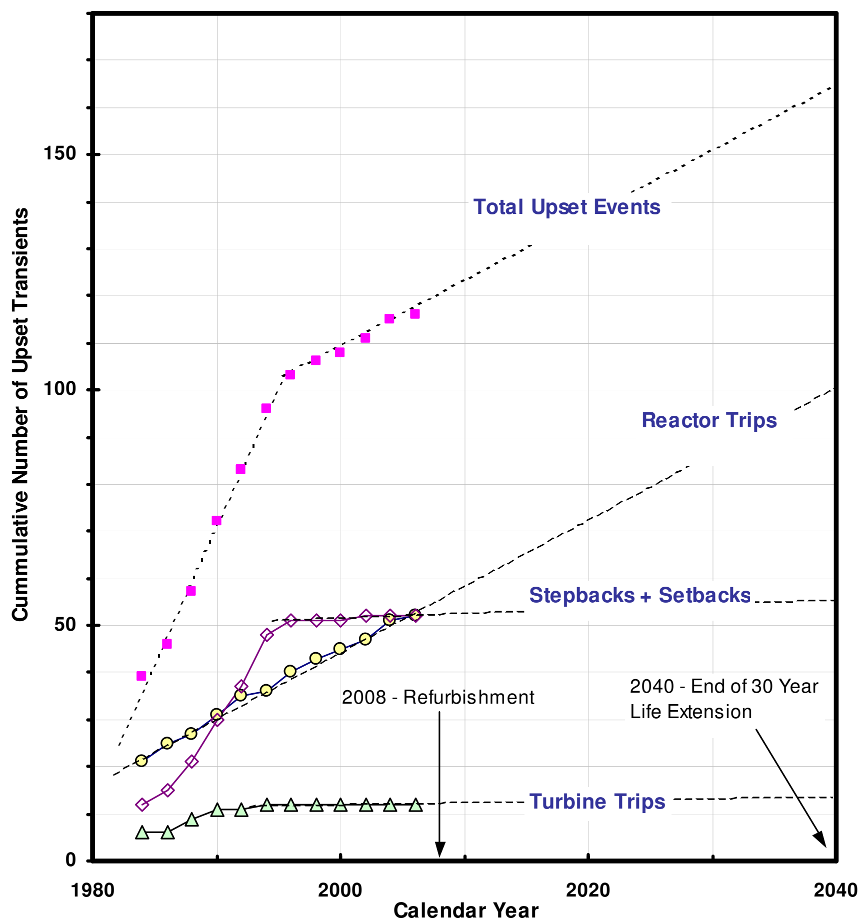
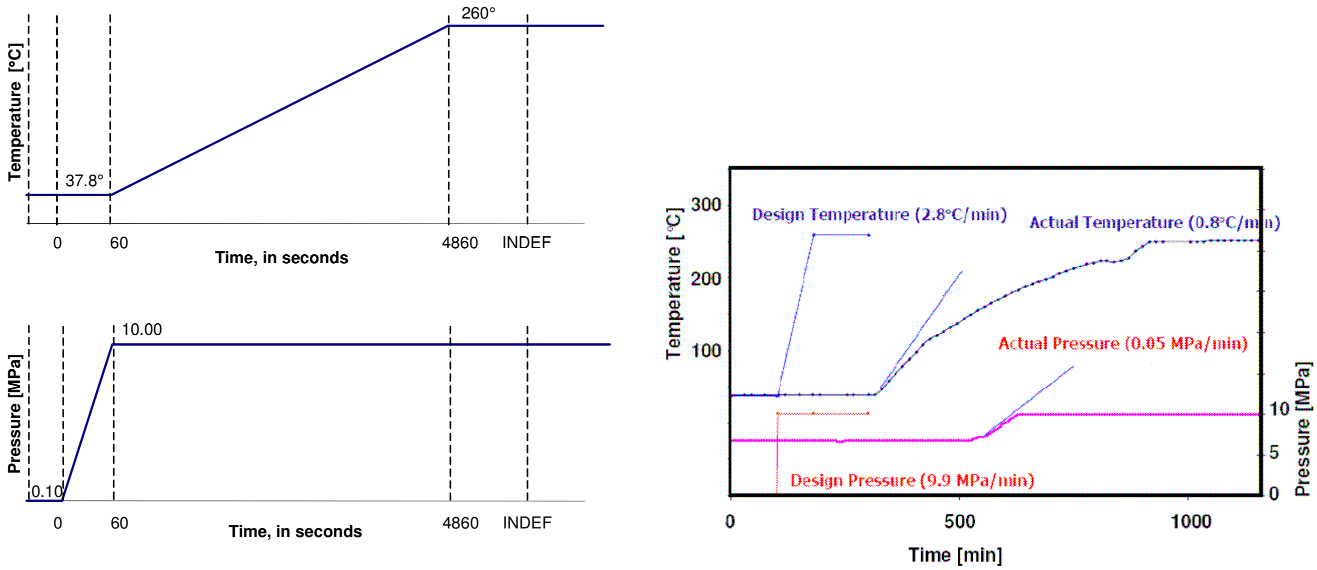


Figure 3 Trend of the Cumulative Number of Upset Events. *Other Upset Events* in the plot includes turbine trips, setbacks and stepbacks. *Total Upset Events* is the summation of *Turbine Trips* and *Other Upset Events.*

It was observed that normal operation transients, i.e., warm-up and cooldown, are easy to identify and follow consistent patterns because they are generally well-controlled (by procedure) planned events. The severities of upset transients are more variable as they are generally unplanned.

Severities of assumed design basis transients and actual transients are compared in Table 2. The selected transients were obtained during the time period of 1998 through 2006. The data in this table demonstrates that actual transients are almost always less severe than the corresponding design basis transients.



(a) Assumed Transient by Design

(b) Actual Transient

Figure 4 Assumed and Actual Warm-up Transient at the PLGS

Table 2 Actual Severity of CANDU Transients[2]

Transient Type	Number of Occurrences (1998 to 2006)	RATE OF TEMPERTURE TRANSIENT (°C /min)		
		DESIGN	ACTUAL	
			Average	Max.
Warmup	13	2.8	0.85	0.98
Startup	20	37.5	0.85	1.6
Shutdown	20	-37.5	-8.7	-26.8
Cooldown	13	-2.8	-2.6	-2.8
Turbine Trip	1	-55.9 and 69.9	-19.7 and 2.3	-19.7 and 2.3
Reactor Trip	6	-114.3 and 37.5	-14.3 and 0.52	-35.1 and 0.83
Setback	1	-333.3	-0.98	-1.8

2 Based on selected number of transients

9. Conclusions

This paper summarizes the ongoing efforts in CANDU industry to address the management of component aging by fatigue in CANDU generating stations. The recommendations summarized in this paper provide a framework and methodology for the development of comprehensive Fatigue Management Programs in CANDU generating stations that are intended to assess the adequacy of fatigue life margins for PLEX operation. In addition, these programs are intended to identify any fatigue issues not covered by design analyses.

As a specific application, plant-specific study for PLGS was presented. The conclusions of this study are summarized below. These conclusions are expected to be applicable for all CANDUs.

- Only a small percentage of design transients have been experienced in CANDUs to-date. Numbers of actual transients at the end of design life are expected to only be about 10 to 15% of the numbers of design transients. Hence, a significant amount of original design fatigue usage margin remains available for PLEX operation.

- Environmental fatigue considerations in heavy water (D_2O) are included in the assessment. Existing environmental fatigue data obtained for PWR coolant (H_2O) is assessed to be valid for heavy water applications. Only warm-up transients are assessed to have sufficiently high dissolved oxygen concentrations that can result in a significant environmental effect (for ferritic steels).

- Worldwide nuclear industry experience (PWR, BWR and CANDU) indicates that most fatigue failures have occurred due to the presence of non-design basis loads, such as those caused by thermal cycling, stratification, or vibrational loads. In CANDU designs, almost all Class 1 piping fatigue failures have occurred along D_2O feed pump outlet lines leading to the HTS have been caused by non-design basis loads as a result of either vibration or thermal cycles at mixing tees.

- Thermal fatigue of normally stagnant and non-isolable branch lines has received considerable attention in the U.S. PWR community because a number of failures have occurred in these branch lines connected to the reactor coolant system. A review of the CANDU 6 design using the EPRI/MRP-146 and MRP-132 thermal cycling screening methodology indicated that CANDU designs have a relatively low propensity for thermal-fatigue of normally-stagnant branch lines.

A review of PLGS transients was performed for all relevant design basis transients, as well as any additional transients experienced during past plant operation. Transients that are important contributors to fatigue were screened and selected for monitoring. All major plant transients were counted as of 2006 June. Extrapolating these data to the future, about 30% of the numbers of design transients are estimated to be used at the end of a 30-year life extension (60 years of total plant operation).

10. References

[1] EPRI Fatigue Management Handbook, TR-104534-Volumes 1 to 4, 2005 November.

[2] CSA-N285.0-95, "General Requirements for Pressure-Retaining Systems and Components in CANDU Nuclear Power Plants".

[3] NUREG-1800, Rev.1, "Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants", U.S. Nuclear Regulatory Commission, 2005 September.

[4] NUREG-1801, Rev.1, "Generic Aging Lessons Learned (GALL) Report", U.S. Nuclear Regulatory Commission, 2005 September.

[5] Slade, J. P. and Gendron, T. S., "Flow Accelerated Corrosion and Cracking of Carbon Steel Piping in Primary Water – Operating Experience at the Point Lepreau Generating Station", Proceedings of the 12th International Conference on Environmental Degradation of Materials in Nuclear Power Systems – Water Reactors, TMS, 2005, pp. 773-784.

[6] Stuart, C.R., Reid, V., and Turner, C.W., "Optimization of Chemistry Control in the Primary Heat Transport System of CANDU Reactors" International Conference on Water Chemistry of Nuclear Reactor Systems 2006, 2006 October 23-26, Jeju Island, Korea.