Fuel String Thermal Expansion at Refurbished Point Lepreau Generating Station

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Abstract

AECL has undertaken a multi-disciplinary analysis and design task that examines the axial expansion of the fuel string during postulated accidents for the refurbished Point Lepreau Generating Station. During refurbishment at the Point Lepreau Generating Station the fuel channels will be replaced. The replacement fuel channel includes a change to the axial location of the shield plugs that will increase the initial axial gap between the fuel and shield plugs compared to the existing design. The analysis has demonstrated that the replacement fuel channel axial gap between the fuel string and the shield plugs is sufficient to preclude any constrained expansion during postulated critical accidents. Fuel channel integrity with respect to fuel string axial expansion is therefore assured for safety defence-in-depth with design and analysis support.

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1. Introduction

In a CANDU® fuel channel, fuel bundles along the channel form a fuel string located between the shield plugs. A certain clearance (gap) between the fuel string and shield plugs is designed, which is at a minimum with new (uncrept) pressure tubes. Fuel heat-up occurs during a postulated large-break loss-of-coolant accident (LOCA) as a result of the over-power transient during the early phase of the accident, and degradation in heat removal during the later phases. Following the postulated accident, the fuel string would thermally expand in the axial direction. As the fuel string expands, the available gap between the fuel string and the shield plugs decreases. If that available gap is too short, fuel string thermal expansion may be constrained. Constrained expansion results in a combination of compressive strain on fuel elements and an outward load on the channel end components, which would potentially lead to fuel element deformation to contact with the pressure tube and then challenge fuel channel integrity, typically when the channel pressure is still high. It is therefore important to preclude constrained fuel string expansion. In order to demonstrate fuel channel integrity, the net expansion of the fuel string in the channel should not exceed the length of the available gap between the fuel string and the shield plugs in the fuel channels.

The available axial gap between the fuel string and the shield plugs in the fuel channels increases with time as the pressure tubes creep axially during normal operation. An analysis effort at the

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Point Lepreau Generating Station (PLGS) was made to study the likelihood and consequences of fuel string compression [1]. The study shows that to preclude constrained expansion is sufficient but not necessary. For the ‘as-built’ (new) core at the station, with some postulated large break LOCA conditions, the fuel string in some channels would be in axial compression but the axial load is not large enough to challenge the fuel channel integrity [1]. The study also indicates that constrained expansion of the fuel string, as a result of the increase in fuel temperature caused during a postulated large break LOCA, does not take place due to the fuel channel elongation that had taken place since the station went into operation [1].

For safety defence-in-depth for the refurbished PLGS, AECL has undertaken a multi-disciplinary analysis and design task that examines the axial expansion of the fuel string during postulated critical accidents. During refurbishment, the fuel channels, including the crept pressure tubes, will be replaced at the station. The proposed replacement fuel channels include a change to the axial location of the shield plugs that will increase the initial axial gap between the fuel and shield plugs compared to the existing design. The analysis of fuel string expansion due to the accident, which is presented here, was performed to support the proposed design to preclude constrained fuel string expansion for the refurbished PLGS core.

2. Critical Event Determination

2.1. Critical event

The intent of the fuel string expansion analysis is to examine the most limiting design basis accident. This provides reasonable assurance that analysis and design work capture a near-worst-case scenario for the refurbished PLGS. The axial fuel string expansion due to an accident depends mainly on the fuel temperature excursion (the temperature increase from the normal operating conditions). The magnitude of the fuel temperature excursion depends on the net energy deposited during the transient. Among the design basis accidents, a large break LOCA tends to result in the largest and most rapid over-power transients. Also, a large break LOCA tends to result in the most degraded cooling conditions among the design basis accidents except for the postulated flow blockage or feeder stagnation break events. However, such single channel events are terminated by pressure tube failure, so it is not necessary to assess whether fuel string expansion would result in an earlier channel failure. Therefore, a limiting large break LOCA will bound all other design basis accident events with respect to fuel string expansion.

The fuel string expansion following a LOCA coincident with total loss of Emergency Core Cooling (LOECC) injection is not a safety concern. First, such a LOCA/LOECC event is due to a consequence of the postulated multi-failures, and the initial stage of LOCA/LOECC is covered by the LOCA blowdown phase. Also, with LOECC scenarios, fuel reheats up slowly at decay power with a very flat temperature profile. With that temperature characteristic, it is unlikely that the bundles will retain their original geometry before the fuel temperature would heat up again to reach the level during the LOCA blowdown phase. As the bundles slump with hot junctions, their structure strength is negligible to pass a significant axial load along the channel. Furthermore, the primary heat transport system pressure will be very low near atmospheric with LOECC scenarios. Hence, there will be little or no load upon the pressure tube to challenge fuel channel integrity due to fuel thermal expansion with a postulated LOCA/LOECC.

2.2. Event sequence and effect on fuel string expansion

A general description of the event sequence during a large break LOCA and its consequential effects as applicable to fuel string expansion are given here focusing on the initial over-power phase, and subsequent degraded fuel cooling phase of a large break LOCA event.

Following a postulated large break in a primary circuit pipe of the heat transport system, coolant in the circuit rapidly discharges into the reactor vault or boiler room. The void in the circuit begins to increase as the primary circuit depressurizes. The decreasing coolant density in some fuel channels, mainly downstream of the break, introduces positive reactivity at a rate and depth for which the reactor regulating system cannot compensate. This leads to a rise in reactor power in all fuel channels, with the most rapid increase in the high-power region of the broken loop, where coolant voiding is most rapid.

Degraded heat transfer to the coolant in some channels accompanies increased heat generation in the fuel; hence, fuel temperatures begin to rise. The rate at which the fuel temperature increases depends mainly on the over-power generation in the fuel elements, and on the effectiveness of cooling of the fuel sheath. The latter depends on coolant void and flow, which in turn depend on the particular break size and location. The highest fuel and pressure tube temperatures occur for breaks where the flow is stagnant or reduced to near zero while the power generation in the fuel is still large. As the fuel and sheath experience a temperature rise, they start to increase in length due to thermal expansion at a faster and greater extent than pressure tube thermal expansion. As a consequence, the gap between the fuel string and shield plugs is reduced. Meanwhile, fuel in the intact loop remains well cooled with little or no expansion since the heat removal remains effective there.

With two independent shutdown systems, the reactor will trip within one or two seconds following the accident depending on break size and initial power level. Following the reactor trip, fission power drops quickly, and the average fuel temperature decreases as the heat generation rate decays. Also, isolation of the two heat transport system loops, operation of the Emergency Core-Cooling System and other safety related systems will occur and function as designed subject to a certain time delay. The isolation of the two loops prevents further inventory loss from the intact loop. Fuel cooling in this loop is provided by forced circulation if the heat transport pumps are still operating, or by thermosyphoning if the pumps are tripped. Soon after loop isolation and steam generator crash cooldown, emergency coolant begins to refill the entire heat transport system. With the introduction of cool injection water into the feeders and channels in the broken loop, fuel and sheath temperatures start to decrease.

3. Analysis Approach and Analysis Cases

The fuel string expansion analysis was performed in a sequence of analysis steps corresponding to a number of contributing disciplines: reactor physics, thermalhydraulics (circuit), fuel and fuel channel. The interactions between the different disciplines and computer codes are illustrated in the flow diagrams given in Figure 1 to Figure 3.

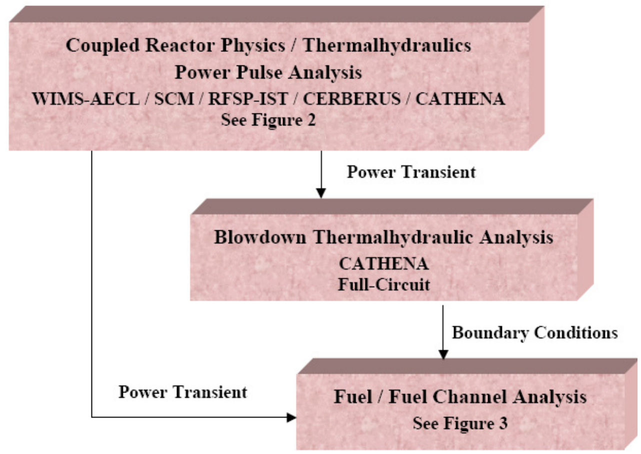

Figure 1 Flow Diagram Illustrating Interactions between Different Disciplines and Codes

Conditions analysed were to cover the limiting over-power range and the limiting degraded coolant condition range, the most influential parameters for the fuel string expansion based on understanding of the critical event discussed previously. From recently assessed LOCA scenarios, a postulated 100% pump suction header (PSH) break results in the most critical conditions in terms of fuel energy deposition, meanwhile postulated 30% reactor inlet header (RIH) and 60% PSH breaks tend to have the worst flow stagnation in terms of the maximum fuel sheath temperature. Therefore, the 100% PSH break is considered for the over-power transient analysis. A break survey was conducted around 30% RIH and 60% PSH breaks applying the same over-power transient to identify the worst break for limiting degraded cooling conditons in terms of the worst (largest) fuel string expansion.

The break survey is summarized in the following steps.

- Over-power Base Case: Five-second coupled reactor physics/thermalhydraulics simulations (first block in Figure 1 or more in Figure 2) that calculated the over-power transient of the equilibrium core for the expected limiting break size and location (i.e., 100% PSH break) in terms of fuel energy deposition calculated at best-estimate modelling parameters and limit of the operating envelope operational parameters.

- The Base Case over-power transient was then used to perform a full break survey to determine the critical break size and location in terms of expected axial expansion. The number of simulation sets with different break sizes and locations were determined during the break survey depending on the results. Each set of simulations in the break survey included the following steps:
    - Full-circuit thermalhydraulic calculations (second block in Figure 1) for each break to provide transient header boundary conditions for single fuel channel analysis.
    - Coupled fuel and single channel thermalhydraulic calculations (third block in Figure 1 or more in Figure 3) for selected breaks. These simulations provided fuel element temperature transients for subsequent fuel elongation calculations.

- Fuel elongation calculations (Figure 3) to identify the critical break in terms of fuel string elongation (expansion).

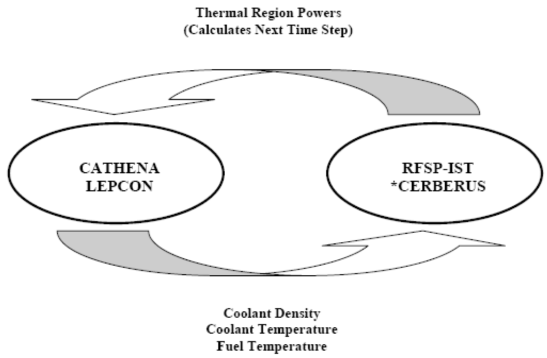

Figure 2 Coupling Physics and Thermal-hydraulic Simulation for FSE Analysis

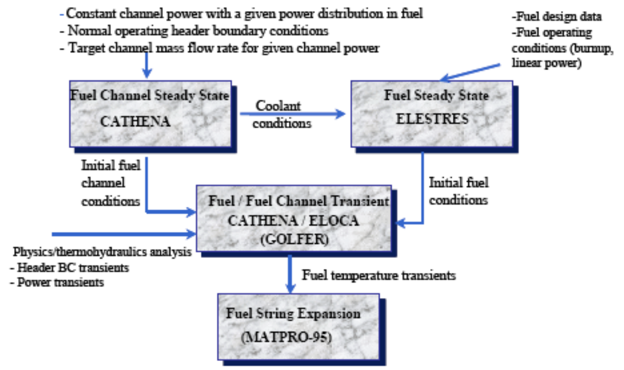

Figure 3 A Flow Diagram Illustrating Fuel and Fuel Channel Analysis for FSE Analysis

With conditions of the over-power Base Case, 8 sets of the circuit and channel simulations were performed for

- 25%, 30% and 35% of RIH breaks, and
- 55%, 60%, 65%, 70% and 75% of PSH breaks.

Two types of additional over-power cases for modelling sensitivity analyses were also planned and performed with similar steps used in the results of the break survey mentioned above:

1. An equilibrium core case with limit of the operating envelope operational parameters and accounting for uncertainties in modelling parameters identified as the over-power modelling sensitivity (over-power MS Case); and
2. A fresh core case at plutonium peak calculated at best-estimate modelling parameters and limit of the operating envelope operational parameters (over-power Fresh Core Case).

4. Computer Codes

Physics computer codes, RFSP-IST and WIMS-IST ([2] and [3]), including WIMS utilities and libraries, were used for the over-power calculation. Thermalhydraulic computer code, CATHENA [4], was used for the coupled physics calculation, circuit blowdown analysis and single channel analysis. The fuel performance codes, ELESTRES-IST and ELOCA-IST ([5] and [6]), were used for fuel behaviour calculations. All codes used in the analysis are of industry standard toolsets (IST) and/or CANDU Owner Group (COG) supported, developed, verified and validated following the CSA N286.7 standards [7].

4.1. Physics computer codes

RFSP-IST [2] is a computer program that performs fuel management and other full-core neutronics calculations for CANDU reactors. Its main function is to calculate neutron flux and power distributions using two-group, three-dimensional neutron diffusion theory. RFSP-IST can calculate both instantaneous and time-average flux distributions and simulate reactor operations, including refuelling and burnup steps. The RFSP-IST code is also capable of performing kinetics calculations (power transients) by solving the neutron kinetics diffusion equations, with delayed-neutron terms, using the Improved Quasi-static method. This latter approach is used by the *CERBERUS module that was used to simulate the transient in the present analysis.

RFSP-IST calculates flux and power distributions in three dimensions using two-group finite difference diffusion theory. It obtains cross sections as functions of fuel irradiation from a table calculated by the neutron-transport code WIMS-IST [3]. It simulates reactor operation by taking time steps with the irradiation distribution updated from the previous step.

4.2. Thermalhydraulic computer codes

CATHENA is a two-fluid thermalhydraulic computer code, developed by AECL for analysis of flow transients in reactors and piping networks. CATHENA was developed primarily for thermalhydraulic analysis of postulated accident scenarios in CANDU reactors.

CATHENA uses a one-dimensional, non-equilibrium, two-fluid thermalhydraulic representation of two-phase flow. Conservation equations for mass, momentum and energy, together with flow-regime dependent constitutive relationships describing the interfacial transfer of mass, momentum and energy are solved for each phase (liquid and vapour). The code uses a staggered-mesh, semi-implicit, and finite-difference numerical solution technique to solve the conservation equations.

The CATHENA code includes component models such as pumps, valves, pipes, generalized tank models and point-kinetics models, and has extensive control modelling capability. It also contains a GENeralized Heat Transfer Package (GENHTP) which allows the modelling of solid components such as pipes and fuel elements, heat generation within these components and heat transfer between these components and the surrounding fluid.

4.3. Fuel performance codes

ELESTRES-IST (ELEment Simulation and sTRESses) [5] is a fuel performance code that models the axisymmetric thermal, micro-structural and mechanical behaviour of a CANDU fuel element during the element's irradiation life (such as power/burnup histories) under normal operating conditions. The code contains one-dimensional models of heat generation, temperature distribution, fission-gas release, and pellet-to-sheath heat transfer. A two-dimensional axisymmetric deformation model is used to calculate the stresses and strain in the pellet and in the sheath. The radioactive isotopes to the fuel-element voidage are also estimated.

ELESTRES-IST calculations provide fuel conditions (such as fuel pellet temperature distribution, heat generation, sheath temperature, sheath strain, internal gas pressure, and fission gas volume) under normal operating conditions at the time just prior to a postulated accident. This information is then used as initial conditions for estimating the transient fuel behaviour.

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ELOCA-IST [6] is a FORTRAN-90 code that calculates the thermo-mechanical response of

CANDU fuel element under transient conditions. Given the power history of the reactor transient, and the boundary condition transients (i.e., coolant temperature and pressure, sheath-to-coolant heat transfer coefficient), the ELOCA-IST code calculates the thermo-mechanical response of the fuel element during the transient.

The ELOCA code models a single cylindrical fuel element of a specified length and containing a specified number of fuel pellets. The axial cross-section of a typical CANDU fuel pellet is shown in Figure 4. For the purposes of the thermal calculation, the fuel pellets are divided into up to 100 radial annuli, covering fuel dish, land and chamfer regions Figure 4. The fuel-to-sheath gap is modelled with a single annulus with a node at the fuel surface and a node at the inner surface of the sheath. The fuel sheath is also modelled as a single annulus with a node at the inside and outside surfaces of the sheath.

5. System Modelling and Analysis Assumptions

5.1. Over-power and blowdown calculations

To achieve the highest over-power transient and energy deposition into the fuel for fuel string expansion analysis for the selected break, the over-power calculations were performed at full power from a core state with a pre-existing side-to-side flux tilt, where the initial flux is higher on the affected loop side. Both simulations of the steady state (reflecting the normal operating conditions prior to postulated accident) and transient were performed for three over-power cases (Base Case, MS Case and Fresh Core Case), respectively. The response of the PLGS plant controllers is accounted for by including important functions such as Boiler Level Control, Boiler Pressure Control, Pressure and Inventory Control and reactor shutdown action of both shutdown systems through the coupling of CATHENA (with the PLGS controller code LEPCON) with RFSP-IST.

For the Base Case, the over-power calculation was performed with limit of the operating envelope operational parameters and best-estimate modelling parameters.

For the MS Case, the modelling parameters were adjusted to accommodate a more conservative uncertainty bias from the base case best-estimate values. As such, coolant void reactivity (CVR) bias was set to the 95^{th} percentile value of + 0.21 mk, based on a 1-sigma uncertainty of 1.1 mk. Other significant modelling parameters (sheath-to-coolant critical heat flux, fuel-to-sheath heat transfer coefficient (HTC), sheath-to-coolant HTC, and delayed neutron fraction) were simultaneously analysed at the 2-sigma conservative value.

It is expected that an equilibrium core with crept PTs is a more limiting core condition for large break LOCA (i.e., that producing the largest over-power transient) than a fresh core, at the same assumed modelling and operational parameter conditions. This core condition could not be reached without some fuel-channel elongation (creep), which would increase the available gap for fuel string expansion due to the accident. However, in order to cover all possible core conditions from fresh to equilibrium core, the assumption on radial creep without any channel axial elongation (creep) is made as the analyzed equilibrium core condition for the over-power Base Case as well as the over-power MS case. Furthermore, both the Base Case and the MS case were assumed to restart following a long shutdown with a pre-existing side-to-side flux tilt, as well as with limit of the operating envelope operational parameters.

For the Fresh Core Case (plutonium-peak core), another potential limiting core condition is assessed. This is because it is the core condition with maximum poison in the moderator, which enhances the void reactivity. The fresh-fuel void reactivity is also greater than the equilibrium value. However, the delayed-neutron fraction is significantly greater than at equilibrium (by about 20 percent) and this is expected to reduce the over-power transient. This case is assumed to restart following a long shutdown with a pre-existing side-to-side flux tilt, as well as with limit of the operating envelope operational parameters, and best-estimate modelling parameters. The difference from the equilibrium Base Case is that the core was modelled at the plutonium peak, at about 45 Equivalent Full Power Days, with the incumbent extra poison load and no pressure tube diametral creep.

The core coolant density, coolant temperature, and fuel temperature data (input to RFSP, see Figure 2) were calculated using CATHENA thermalhydraulic plant circuit models derived from the CATHENA plant model of the PLGS. In both the aged (for Base and MS Cases) and fresh core (for Fresh Core Case) CATHENA circuit models, the reactor core representation contains a seven-average channel representation of each core pass, for a total of 28 channel groups in the core. The CATHENA integrated plant models are also used to generate the header boundary conditions for the blowdown phases applying the over-power transient and decay power transient for the single channel analysis and fuel analysis (Figure 1 and Figure 3).

5.2. Single channel and fuel analyses

5.2.1. Channel Model

Figure 3 illustrates the interactions between the different disciplines and computer codes for the fuel and fuel channel analysis. The fuel string expansion due to the accident depends mainly on the fuel temperature excursion (the temperature increase from the normal operating conditions) along the channel. The magnitude of the fuel temperature excursion depends on the net energy deposited during the transient.

The CATHENA single channel model represents an individual fuel channel (or a channel group) that defines a thermalhydraulic path from the reactor inlet header (RIH) through the fuel channel to the reactor outlet header (ROH). The components of a single-channel model include RIH and ROH boundary conditions, feeders, end fittings, pressure tube, and the fuel bundles (37-element design). The limiting high channel power of 7.3 MW was used for all single channel simulations. The limiting high channel power is used to maximize the heat generation that results in a bounding fuel string temperature increase, hence, a bounding fuel string expansion during transient.

The single channel models contained detailed fuel models. A typical 37-element fuel bundle cross-section inside pressure tube is modelled as shown in Figure 5. Based on the element radial power factors (RPF) and radial burnup factors (RBF), the 37 elements were grouped in four rings with one centre element, 6 inner elements, 12 intermediate elements and 18 outer elements, respectively. The RPF and RBF are element power and burnup normalized to the bundle average, respectively. The RPF of 0.7585, 0.7968, 0.9105 and 1.1408, and the RBF of 0.7746, 0.8161, 0.9239, and 1.124, from the centre to outer elements, respectively, were used. They are for a typical CANDU 37-element fuel bundle at the Plutonium peak giving a limiting element power and burnup for outer elements.

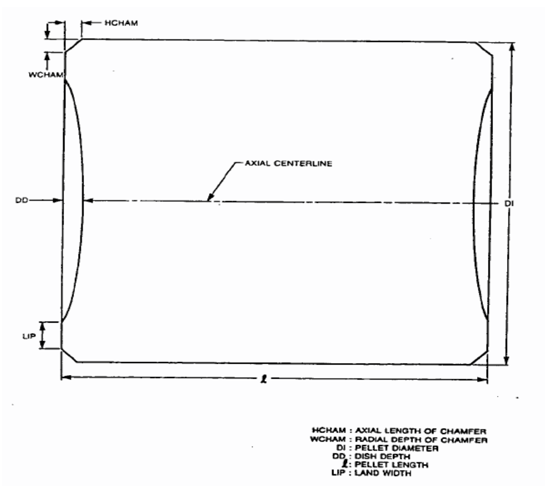

Figure 4 Cross-section of a Typical CANDU Fuel Pellet

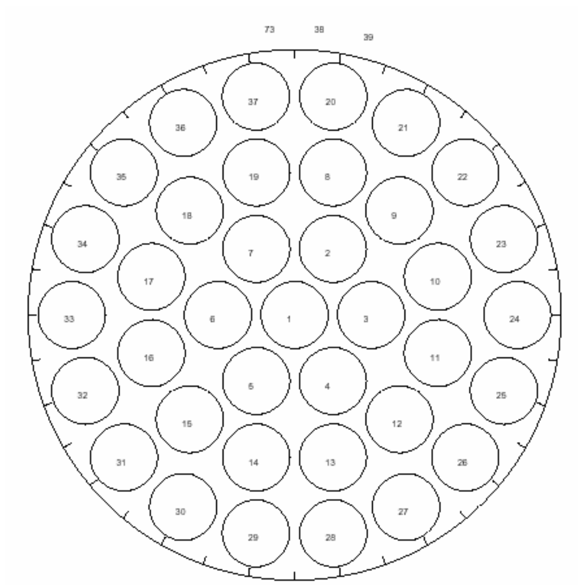

Figure 5 Modelling of Pressure Tube Containing 37 fuel elements in four rings

5.2.2. Axial power and burnup distribution

A typical axial burnup (TB) distribution for a high power channel at PLGS was considered as reference for the limiting channel power of 7.3 MW in this fuel string expansion calculation. The TB distribution, based on equilibrium bundle burnup at a given bundle location of a typical high power channel, ranged from 19.6 to 180.8 MW·h/(kgU) varying with axial bundle location.

In this fuel string expansion calculation, another two axial limiting burnup (LB) distributions were also used, which were determined by searching the maximum bundle-average burnup for each bundle location in the entire core. One LB distribution was for the end-of-cycle burnup just before refuelling of a channel from the PLGS Time Average Model (for Base and MS Cases), ranged from 42.1 to 240.6 MW·h/(kgU) varying with axial bundle location. The other LB distribution was for a channel from the PLGS Plutonium peak model (for Fresh Core Case), ranged from 12.26 to 40.37 MW·h/(kgU) varying with axial bundle location.

The limiting burnup distribution used for the fuel string elongation calculation is the distribution of the maximum bundle average burnup for each bundle location in the entire core. However, at higher bundle burnup, the bundle power decreases, and the high burnup case results in a lower power channel. Although the use of maximum bundle average burnup for each bundle location for the limiting high power channel in the analysis is not realistic, the limiting bundle burnup distribution was introduced to assess the safety margin of the typical bundle burnup distribution.

During the transient, for the same power, elements with higher fuel burnup tend to have higher initial fuel temperature. This also causes early dryout as well as higher fission-gas pressure that result in earlier sheath lift-off and lower fuel-to-sheath gap conductance, leading to higher fuel temperature increase hence larger fuel elongation.

In this fuel string expansion task, the element power/burnup histories are assumed to follow the limiting over-power envelope shown in Figure 6. This limiting history is raised from the reference envelope (based on fuel management simulation of reactor operation from startup until the time that the last remaining bundle from the initial core is discharged), such that the maximum power (900 kW) goes up to the limiting bundle power (935 kW). The power/burnup histories of all fuel elements are assumed to be proportional to, but limited up to the limiting bundle power envelope subject to the relative RPF and RBF mentioned previously. Applying the limiting power history, for the same burnup used, tends to result in worse degraded fuel conditions, similar to applying higher burnup for the same power.

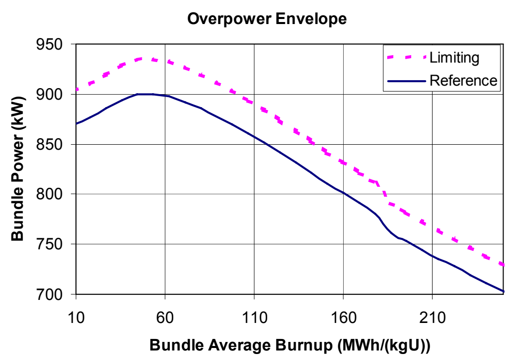

Figure 6 Bundle Power Envelopes / History

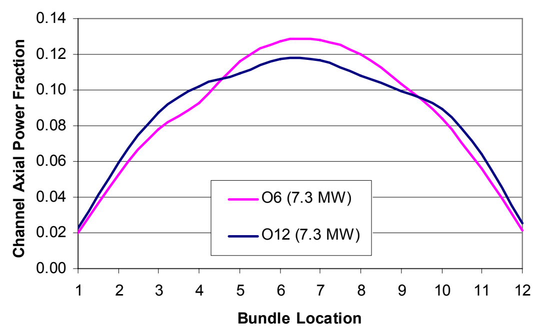

Figure 7 Channel Axial Peaked and Flat Power Distributions

In the single-channel analysis, several CATHENA single-channel models were used. These models have the same hydraulic components, and modelling details, except for the use of different combinations of the axial power distribution (either Peaked or Flat, Figure 7) and the axial burnup distribution (either TB or LB), to cover the possible variation of power and burnup distributions in the core. The Peaked distribution tends to give a limiting local fuel temperature, whereas the Flat distribution tends to give a hot region with a longer axial length after the large break, which may result in a larger fuel string expansion overall.

5.2.3. Fuel grouping and coupled CATHENA-ELOCA simulation

Following a large LOCA, the top elements of a fuel bundle ring tend to be hotter than the other elements in the same ring due to stratification effect, which is accounted for in the CATHENA single channel model. Therefore, one of the top elements of each ring was selected to represent the other fuel elements in the same ring. These top elements (such as elements 1, 2, 8 and 20 in Figure 5) were examined in detail through fuel simulations using the fuel performance codes, ELESTRES-IST and ELOCA-IST. ELOCA-IST is coupled with CATHENA through GOLFER (Gain Open Link to Fuel Element Response), a communication interface for ELOCA. The

CATHENA fuel element models (except the sheath surface heat transfer) for the top elements were automatically replaced by the ELESTRES/ELOCA fuel element models during the coupled fuel and fuel channel simulation. However, the power generation in each fuel element was ensured to be consistent between the CATHENA, ELESTRES and ELOCA fuel models. By using coupled CATHENA-ELOCA, the data of coolant conditions and fuel response are effectively transferred between two computer codes at every time step during channel simulation with 48 ELOCA fuel models.

5.3. Fuel thermal expansion calculation

The material properties (thermal expansion equations) for the fuel are based on the correlation provided in MATPRO [8]. The MATPRO correlation is used for fuel temperatures below the melting point, which is the case in this fuel string expansion task. The comparison of the correlation with its database [8] is shown in Figure 8**.** This correlation is used in the fuel performance codes (ELESTRES and ELOCA ([5] and [6]). It is also used in US nuclear safety and licensing [8]. In this correlation, material properties used correspond to those for un-irradiated material. The effects of fuel burnup and oxygen to metal ratio are neglected, as are the effects of irradiation swelling prior and during the short duration of the accidents. Error bands [8], calculated from the sum of the squared residuals, are shown in Figure 8 as dotted lines. They reflect a standard error of ± 10% of the calculated value found from the UO_2 dataset.

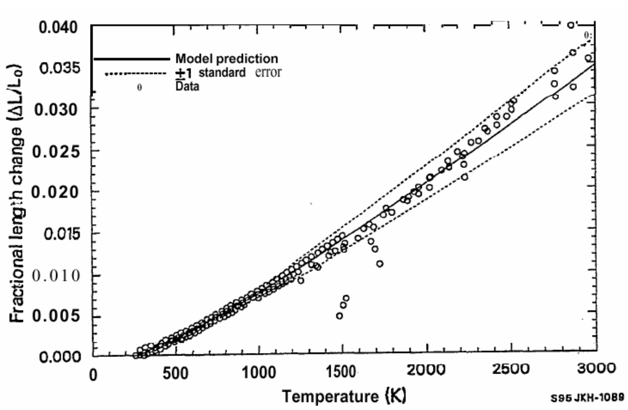

Figure 8 Correlations for Thermal Expansion Strain of UO_2 Compared with Its Data [8]

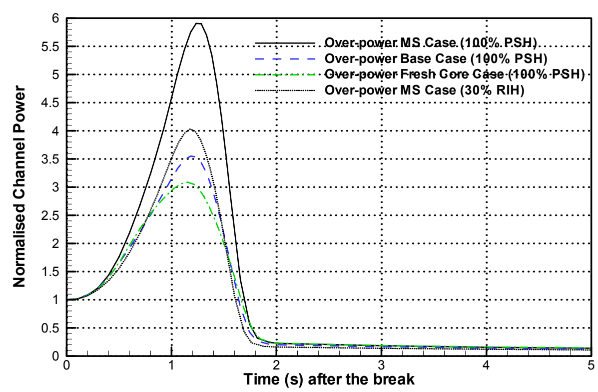

Figure 9 Comparisons of Channel Power Transients of Three Over-Power Cases

Since the objective of the present work is to preclude any constrained fuel string expansion during postulated critical accidents, free thermal expansion is assumed for each fuel annulus modelled as well as for each fuel element.

In each channel simulation, the element axial expansion (in strain due to the break) was first calculated by applying the MATPRO correlation with the fuel temperature transient and the initial temperature for each of 48 selected elements. Those 48 element thermal strain transients formed four sets of fuel-element string expansion data, each for elements on one of four fuel rings along the channel. The peak fuel string expansion strain was then obtained for the channel. The limiting fuel string expansion strain was determined from the peak strains from all channel simulations, which would translate to the maximum elongation length due to the postulated accidents. The reason calculating the elongation strain rather than the actual elongation length is to allow for the application of the actual initial fuel string length only once.

The initial fuel string length is assumed the same as the maximum fuel bundle string length in the design. The maximum axial elongation of the longest fuel pellet annulus is taken as the axial elongation of the fuel element. It is expected that the shoulder annulus (at the outer edge of fuel land, Figure 4) have the largest elongation. Applying the initial fuel string length assumption for the fuel string expansion due to the accident treats all of the fuel string materials (the fuel stacks, about 97% of bundle length in cold, plus the bundle junctions) as fuel (UO_2). The bundle junctions (end plates, end caps and welds) are not expected to have as high thermal expansion coefficients and experience as high temperature increases as the fuel stack. Therefore, the above assumption leads to a slight overestimation of the overall fuel string expansion due to the break.

The methodology and approach used in the fuel string expansion calculations for the refurbished PLGS is applicable for other CANDU reactors, subject to the station and fuel data adjustment.

6. Analysis Results

6.1. Over-power transient results

The resulting highest relative channel power transients are shown in Figure 9 for all three of the over-power cases analysed, over the first 5 seconds following the break (i.e., 100% PSH break). The MS Case results in the highest full power seconds (FPS) for the first 5 s after the break. The FPS is the integrated relative power over a given period (e.g. over the first 5 s), which indicates the over-power level. The high power channel also contains a 'hot bundle', the highest FPS in the core. The bundle power transient data of the high power channel were extracted from results of coupling physics-thermalhydraulic simulations. The integrated relative power of each bundle along the high power channel over the first 5 seconds is between 4.19 to 4.46 FPS for the Base Case; between 5.65 to 6.33 FPS for MS Case; and between 4.16 to 4.29 FPS for Fresh Core Case. As comparison, with the same over-power modelling sensitivity applied for 100% PSH break, 30% RIH break gives a low over-power level (between 4.10 to 4.50 FPS), which is much lower than the over-power level of MS Case following the 100% PSH break.

Blowdown analyses were then carried out for each case using the over-power transient calculated from the coupled analysis with a typical power rundown curve after trip for the next 100 seconds.

6.2. Typical results of fuel annuli expansion

Applying the power transients and reactor header conditions, the fuel and fuel channel simulations were performed for the fuel string expansion calculation. Results in Figure 10 for the 70% PSH break with the Base Case show that the peak temperatures at the fuel centre line, fuel land edge and on the fuel surface appear within the first 15 s after the break. All the fuel temperatures reported are calculated by ELOCA. Figure 11 shows the typical fuel temperature radial profiles at 0, 10, and 50 s after the break. With the degradation of the coolant heat removal capacity and the power decrease after the peak of the over-power transient, the fuel temperature profile flattens (Figure 11). As expected, the outer edge of the fuel land (Figure 4) has the maximum temperature increment (Figure 11) and experiences the largest expansion (Figure 12) with the longest axial length after the break. With fuel dish and chamfer designed (Figure 4), the axial lengths of fuel centreline and fuel surface remain shorter than the fuel land axial length (Figure 12). Figure 13 shows that among elements of four rings, the outer element has the largest thermal expansion. Therefore, the thermal strain of the fuel land edge of the outer element is to be used for the fuel string expansion calculation.

These results show that the peaked axial power distribution together with the limiting axial burnup distribution (Peaked_LB) gives the highest fuel string expansion strain, among the different combination of axial power and burnup distributions. For 70% PSH break with Base Case, Peaked_LB results in the peak strain about 1.5% more than the same peaked axial power distribution but with the typical axial burnup distribution. Therefore, the channel model for Peaked_LB is used for the break survey and further calculation of the fuel string expansion.

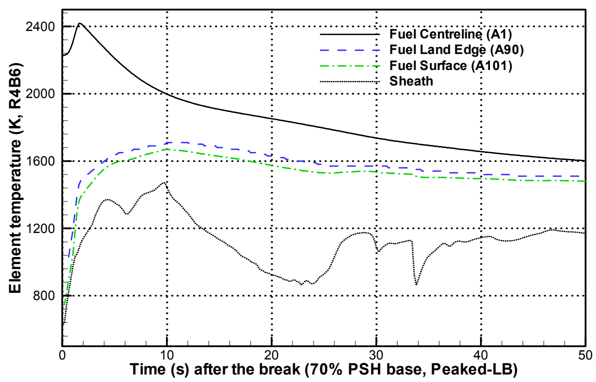

Figure 10 Temperature Transient of An Outer Element of Bundle 6 after A 70% PSH Break (Base Case)

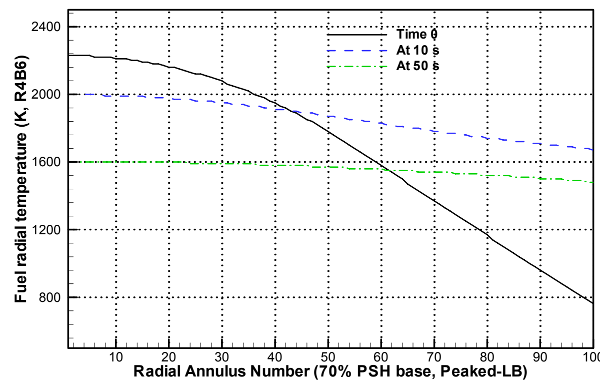

Figure 11 Typical Temperature Radial Profiles of An Outer Element of Bundle 6 after A 70% PSH Break (Base Case)

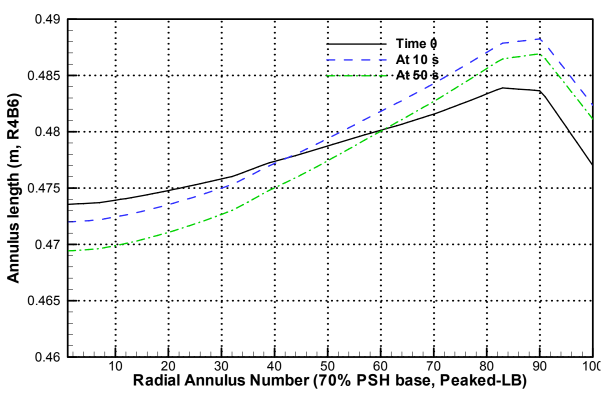

Figure 12 Fuel Annuli Length Radial Profiles in An Outer Element of Bundle 6 after A 70% PSH Break (Base Case)

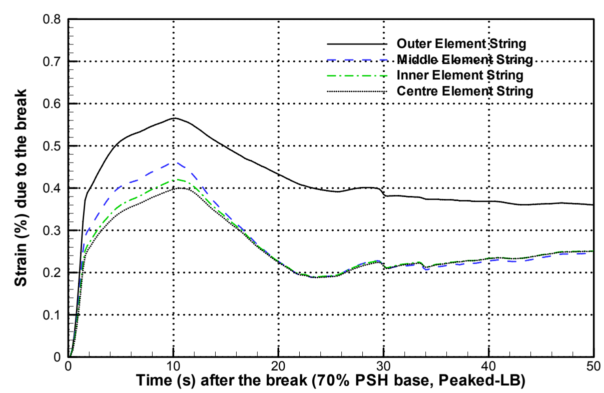

Figure 13 Fuel Land Thermal Expansion Strain due to the 70% PSH Break for Four Element Strings (Base Case)

6.3. Break survey results

In the break survey of RIH break events (Base Case), with the combined effects of the break discharge and the main pump forced circulation, the 30% RIH break resulted in a longer period of near–zero flow (flow stagnation) among the examined 25%, 30% and 35% RIH break sizes (Figure 14). Figure 15 compares the calculated transient fuel thermal expansion strain with the three RIH break sizes in the break survey. The results show that the 30% break is the limiting RIH break since it is associated with the highest peak of fuel string expansion strain (0.574% at 25.4 s) due to the break (Figure 15). The 25% RIH break gives the earliest strain peak (0.544% at 5.8 s), which corresponds to the early but shorter period of channel flow stagnation.

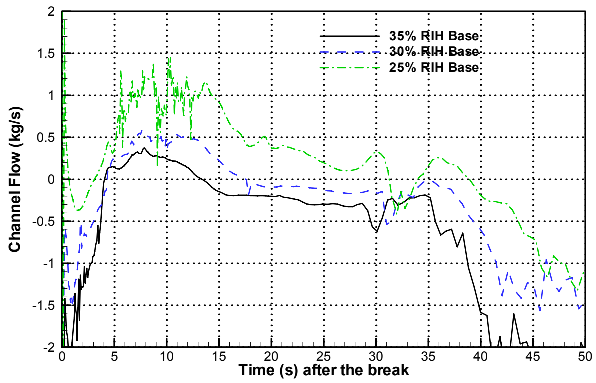

Figure 14 Comparison of Channel Flows with Different RIH Break Sizes (Base Case)

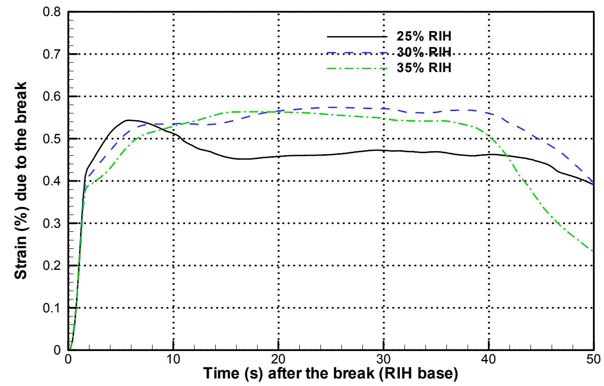

Figure 15 Comparison of Fuel Land Thermal Expansion Strain due to the Break with Different RIH Break Sizes (Base Case)

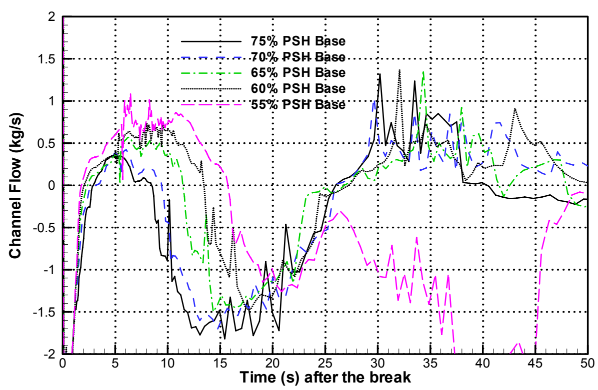

Figure 16 Comparison of Channel Flows with Different PSH Break Sizes (Base Case)

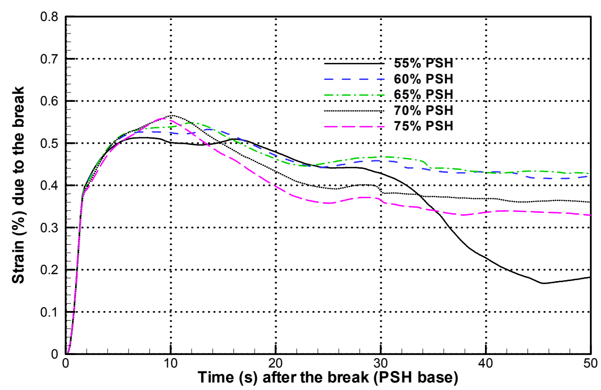

Figure 17 Comparison of Fuel Land Thermal Expansion Strain due to the Break with Different PSH Break Sizes (Base Case)

In the break survey of PSH break events (Base Case), with the combined effects of the break discharge and the main pump forced circulation, the 70% PSH break results in the longest period of near–zero flow (flow stagnation) among the examined 55%, 60%, 65%, 70% and 75% of PSH break sizes (Figure 16). The second flow reversal period appears earlier than for the critical RIH break (Figure 14 and Figure 16) due to faster pump head degradation with the PSH break as the break is closer to the pumps. Figure 17 compares the calculated fuel thermal expansion strain due to the five PSH break sizes in the break survey. All the peaks of thermal expansion appear before the end of the second flow reversal period (around 10 s). After the peak, the thermal expansion transient flattens (similar to the decay power) with values significant lower than the peak for the different break sizes (Figure 17). The results show that the 70% PSH break is the limiting PSH break since it is associated with the highest peak of fuel string expansion strain (0.565% at 10.2 s) due to the break (Figure 17).

Together with break survey results and consideration with the same 100% PSH over-power power transient used, the 70% PSH break is selected as a representative limiting break in terms of fuel string expansion. The 30% RIH break is also considered, since it tends to give a long period of high fuel string expansion strain due to degraded channel cooling conditions.

6.4. Limiting fuel string expansion

In addition to the fuel string expansion strain of the Base Case for a 70% PSH break, the fuel string expansion strains of the MS Case and Fresh Core Case were obtained (Figure 18) by applying the relative over-power transients for 100% PSH break and header boundary conditions of a 70% PSH break. The results show:

- MS Case resulted in a fuel land expansion peak strain of 0.718% at 10.0 s, and.
- Fresh Core Case resulted in a fuel land expansion peak strain of 0.412% at 4.4 s.

These peak strains translate into fuel string axial elongation due to the accident of 33.6 mm with the Base Case, 42.8 mm with the MS Case, and 24.5 mm with the Fresh Core Case, respectively.

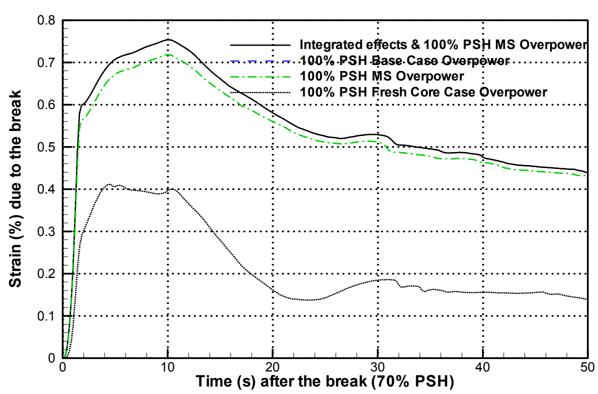

Figure 18 Comparison of Fuel String Expansion Strain (Fuel Land Edge) due to the 70% PSH Break for Three Over-power Cases

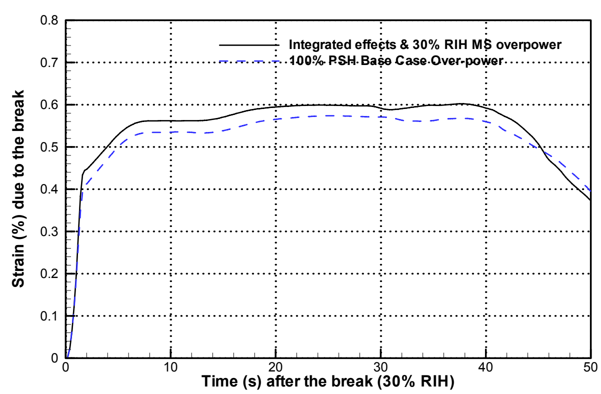

Figure 19 Comparison of Fuel String Expansion Strain (Fuel Land Edge) due to the 30% RIH break

To assess the impact of additional conservatisms, the integrated effects of crept channels (with a maximum creep of 3.3%), fuel density (with a minimum of 10.4 g/cm^3) and sheath wall thickness (with a minimum of 0.38 mm) were further considered in fuel and fuel channel analysis with the over-power MS Case, which resulted in a peak strain of 0.754% at 10.0 s or a fuel string elongation of 44.9 mm due to the 70% PSH break. Furthermore, considering the MATPRO correlation uncertainty of ± 10% or ± 4.5 mm, a limiting fuel string elongation of 49.4 mm is calculated due to the 70% PSH break

With the same over-power modelling sensitivity and integrated effects considered for the 70% PSH break, the 30% RIH break results in a peak strain of 0.602% at 37.6 s (Figure 19) or a fuel string elongation of 36.0 mm due to the break. This tends be the worst one due to the RIH break (Figure 15 and Figure 19); however, it is well below the above limiting results due to the PSH break (Figure 17 and Figure 18).

With the many conservative assumptions and calculation uncertainties used in this task, the minimum required gap between the fuel string and shield plugs is determined as 49.4 mm during full power normal operation. This requirement has conservatively not credited the pressure tube elongation due to the accidents.

6.5. Fuel string cavity length and minimum gap between the fuel string and shield plugs

The length of the fuel cavity provided in the fuel channel was carefully designed to be greater than the length of the fuel string during all service conditions and design basis accident conditions over the channel design life. The fuel cavity length is the distance between the inboard faces of the shield plugs inside the fuel channel. The minimum required fuel cavity length is established based on the maximum fuel string length (including manufacturing tolerance, thermal expansion, and irradiation growth of fuel) plus a suitable clearance. The proposed replacement fuel channel in the Refurbished PLGS includes a change to the axial location of the shield plugs that increase the initial axial gap between the fuel and shield plugs compared to the existing design.

The calculation of the minimum fuel cavity length for the proposed PLGS refurbished condition during normal operation is based on the worst fuel channel component tolerances and the shortest fuel channel to be installed at the Refurbished PLGS. The dimension for the replacement fuel channel is derived based on the measurement of the existing calandria vessel, which will not be replaced. Hence, the minimum fuel channel cavity length at the refurbished PLGS for full power operation is obtained.

The initial (pre-accident) fuel string length is calculated with the maximum initial bundle length in design plus a calculated length to account for thermal expansion including permanent growth at full power hot conditions. Comparing this initial fuel string length with the minimum fuel channel cavity length at the refurbished PLGS for full power operation, this leaves a minimum gap between the shield plugs and the fuel string under the normal operating conditions of 59.2 mm. The maximum axial expansion of the fuel string due to a postulated worst-case accident scenario is estimated to 49.4 mm. Thus, the 59.2 mm gap will be large enough to cover this axial expansion with a margin of about 9.8 mm. There is sufficient margin to preclude constrained expansion during a critical large break LOCA; hence, the results are applicable to all design basis accidents, particularly with the many conservative assumptions and calculation uncertainties applied in these calculations.

7. Conclusions

A multi-disciplinary safety analysis and engineering design task was carried out to examine the axial expansion of the fuel string of the refurbished PLGS to determine the extent of fuel string expansion under accident conditions. The analysis was conducted with generally sound methodology and incorporates conservative assumptions. This analysis has demonstrated the existence of margins at the refurbished PLGS to prevent the constrained fuel string expansion for a critical large break LOCA, and hence for all design basis accidents. Fuel channel integrity with respect to fuel sting axial expansion is therefore assured for safety defence-in-depth with design and analysis support.

The methodology and approach used in the fuel string expansion calculations for the refurbished PLGS is applicable for other CANDU reactors, subject to the station and fuel data adjustment.

8. References

[1] Reid, P.J., and Gibb. R.A., "Likelihood and Consequences of Fuel String Compression at Point Lepreau Generating Station", Proceedings of the 17th Annual Conference of the Canadian Nuclear Society , Fredericton, New Brunswick, Canada, June 9-12, 1996.

[2] Rouben, B., "RFSP-IST, the Industry Standard Tool Computer Program for CANDU Reactor Core Design and Analysis", Proceedings of the 13th Pacific Basin Nuclear Conference, Shenzhen, China, October 21-25, 2002.

[3] Irish, J.D., and Douglas, S.R., "Validation of WIMS-IST", Proceedings of the 23rd Annual Conference of the Canadian Nuclear Society, Toronto, Ontario, Canada, June 2-5, 2002.

[4] Hanna, B.N., "CATHENA: a Thermalhydraulic Code for CANDU Analysis", Nuclear Engineering and Design, 180(2), pp. 113-131, 1998.

[5] Sim, K-S., G.G. Chassie, G.G., Xu, Z., Tayal, M., Westbye, C., "Progress in Qualifying ELESTRES-IST 1.0 Code: Verification and Interim Results of Validation", Proceedings of the Canadian Nuclear Society 7th International Conference on CANDU Fuel, Kingston, Ontario, Canada, September 23-27, 2001.Canada, September 2001.

[6] Williams, A.F., "The ELOCA Fuel Modelling Code: Past, Present and Future", Proceedings of the 9th International Canadian Nuclear Society Conference on CANDU Fuel, Belleville, Ontario, Canada, September 2005.

[7] Canadian Standards Association, "Quality Assurance of Analytical, Scientific and Design Computer Programs for Nuclear Power Plants", CSA N286.7-99, March 1999.

[8] Hagrman, D.T., "SCDAP/RELAP5/MOD 3.1 Code Manual: MATPRO – A Library of Materials Properties for Light-Water-Reactor Accident Analysis", NUREG/CR–6150, EGG-2720, Volume 4, June 1995.