Probabilistic Evaluation in Support of Risk-Informed Pressure Tube Maintenance

Eric Nadeau¹, Tom Byrne², Siavash Khajehpour¹, Eduardo Lupia¹, Eric Araujo¹

¹) Atomic Energy of Canada (AECL), Reactor Engineering, Mississauga, Ontario, Canada
²) DRT Technical Services Ltd., Mississauga, Ontario, Canada

ABSTRACT

AECL has recently developed probabilistic tools to support nuclear plant operators with a risk-based fuel channel management strategy. One such tool is used to evaluate the probability of pressure tube (PT) rupture resulting from pressure tube to calandria tube (CT) contact and hydride blisters. This tool assumes that PT rupture occurs when delayed hydride cracking (DHC) initiates in a blister. The objectives of the probabilistic assessments are to:

1. Determine the overall risk of PT rupture in the reactor core for comparison with the acceptance criteria.
2. Determine the risk of PT rupture for specific fuel channels to assist in the development of an inspection/maintenance strategy.
3. Evaluate the risk reduction that would result from different fuel channels inspection/maintenance scenarios.
4. Optimize inspection/maintenance programs.

BACKGROUND

CANDU® 6 nuclear reactors are comprised of 380 fuel channels (FC), each having a pressure tube installed concentrically inside a calandria tube. Figure 1 shows that helicoidal annular springs, also known as spacers, provide support to the PT with the purpose of preventing contact between the two tubes. The reactor core length is approximately 6000 mm, and the PT is approximately 6150 mm long.

Figure 1 – Schematic of CANDU® Fuel Channel

In the older CANDU® design, it has been observed that lightly loaded loose-fitting spacers, illustrated in Figure 2, could move large distances from their design locations resulting in long unsupported PT spans. This eventually leads to the PT contacting the CT, with the contact area spreading axially over time. Because the moderator on the outside of the CT is much cooler (approximately 70°C) than the PT (>260°C), PT to CT contact results in a cold region at the location of contact on the PT outer surface.

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The combination of the initial hydrogen (from the manufacturing process) and the deuterium accumulated during operation (through a corrosion process) is called the equivalent hydrogen (H_{eq}) concentration. These species tend to migrate from the warmer regions of the PT to the cooler region at the contact area. The resulting combination of higher local concentrations and reduced solubility limits in the contact region may result in zirconium hydrides being formed on the outside surface of the PT. This will occur if the bulk H_{eq} in the PT (at the axial location of contact) exceeds a limit called the blister formation threshold (BFT).

As more hydrides precipitate at the contact location, blister growth begins to occur. An example of a blister in Zr 2.5% Nb PT material is shown in Figure 3. More rapid blister growth will occur with higher PT temperatures, lower PT contact temperatures and the bulk H_{eq} exceeding BFT by a large amount. Once the blister depth reaches a critical value, the blister may crack through to the parent material resulting in an initiation site for delayed hydride cracking. The resulting crack growth in the radial and axial directions can lead to the rupture of the PT. In August 1983, a Zircalloy-2 (Zr-2) pressure tube ruptured during reactor operation according this degradation mechanism at the Pickering Nuclear Generating Station (NGS) “A”, Unit 2 [1][2].

BLISTER PREVENTION

Since the Pickering 2 incident, AECL strategies to prevent blister formation have included both contact avoidance and a number of changes to reduce the amount of hydrogen dissolved in the pressure tube material. These strategies are summarized as follows:

- To avoid contact, AECL designed tight fitting spacers to prevent spacer movement.
- To reduce the hydrogen content in the pressure tube, AECL relied on using Zr 2.5% Nb PT material, which has a deuterium uptake rate significantly lower than the Zr-2 PT material originally used in Pickering Unit 2 [2]. Also, the manufacturing process has continually been improved over the years to reduce the initial hydrogen concentration in pressure tubes. The level of initial hydrogen concentration in the latest generation of PT is close to zero.
For existing reactors with loose-fitting spacers, the industry has introduced maintenance strategies to ensure that, at all times, at least one of the blister formation conditions is not met. Chiefly, the industry has utilized the Spacer Location and Repositioning (SLAR) program to prevent PT to CT contact and, in certain cases where contact cannot be prevented, ensuring through measurements that the bulk $H_{eq}$ in the contact region remains below BFT during operation.

DETERMINISTIC ASSESSMENTS VERSUS PROBABILISTIC ASSESSMENTS

Reactor inspection capabilities are not available to detect blister initiation in pressure tubes. Therefore, assessment methods rely on modelling predictions [3] once contact occurs. The selection of channels for SLAR maintenance has historically been made using a deterministic methodology for blister susceptibility assessments based on a set of inputs and acceptance criteria designed to provide a conservative evaluation of the reactor core. While this methodology has been very successful in preventing contact and blisters, it suffers from a number of limitations, the main ones being:

1. Neither the relative risk of PT failure associated with blister formation in individual channels nor the overall risk to the reactor core can be calculated.
2. There is no single “worst-case” analysis because of the multi-parametric nature of the problem.
3. It is very difficult to select and prioritize channels for inspection, because the relative importance of violating different acceptance criteria is subjective.

These limitations may lead to unnecessarily large inspection and maintenance programs. Therefore, to assist the CANDU® operators with a risk-informed fuel channel management program, AECL developed a probabilistic methodology to assess the probability of pressure tube rupture resulting from contact and blisters. This new tool provides the operators with the information needed for ranking individual fuel channels and making risk-informed comparisons with other plant components. It can be used in combination with other available information for the development of an optimized inspection strategy.

EVOLUTION OF PROBABILISTIC METHODOLOGY FOR CONTACT AND BLISTERS

Probabilistic assessments for contact and blisters were first used in the late 80’s and early 90’s at the Ontario Hydro Research Division [4]. A computer program called RABIT (Reactor Assessment for Blisters in Tubes) was developed to perform the assessments. In those days, there were few in-reactor measurements available. Therefore, RABIT was designed to explore the effect of the different parameters affecting contact and blister growth. It was found that the controlling parameters were the deuterium uptake model, the annular spacer distribution (representing spacer movement), and to a lesser degree, the initial hydrogen concentration. RABIT allowed for the evaluation of different scenarios, and it was thus instrumental in the development and prioritization of inspection programs.

In the following years, a large amount of data was obtained through inspections. In many units, almost all the fuel channels were inspected at least once. Therefore, the spacer locations became known inputs. At the same time, further development on the finite element code CDEPTH (Creep Deformation Evaluation Program of Tubes – Hydro) proceeded rapidly and helped increase the understanding of creep and its effect of contact. A pre and post-processor code called SLARON (SLAR On-Line) was developed to automatically generate CDEPTH input files and report the results in a user-friendly format. All these advances led to the development from 1995 to 2001 of a new probabilistic program called UPOCA (Uncertainty Prediction of Contact Analysis). It was designed to perform probabilistic assessments of contact with the intention of adding a blister module later on. UPOCA supported a large number of distributed input parameters and a variety of statistical distributions. Detailed sensitivity studies found that PT creep and end-slopes have the largest impact on the time of first contact. Another sensitivity study provided the justification for the 35% margin on contact time currently used in deterministic assessments. UPOCA was used in support of reactor operation in a number of instances.

The same development team moved to AECL and developed the next generation probabilistic code called PROBE (PRObabilistic Blister Evaluation). PROBE is based on CDEPTH and SLARADE™ (Spacer Location And Repositioning Advanced Deformation Evaluation), the industry replacement for SLARON. PROBE currently supports probabilistic assessment of contact; the development of the blister module is underway and is close to completion. Like UPOCA, PROBE supports a large number of distributed input parameters and a variety of statistical distributions. In addition, it can model spacer movement in simulations where spacers become unloaded.

ACCEPTANCE CRITERION

The Informative Annex C of CSA Standard N285.8-05 [5] defines an acceptance criterion based on the allowable pressure tube failure frequency per known in-service pressure tube degradation mechanism per reactor year. The acceptance criterion for probabilistic assessments applies to the entire core rather than to individual pressure tubes. It is therefore necessary to perform a core assessment for each particular degradation mechanism. Table C.1 of the CSA Standard provides
acceptable failure frequencies in terms of the number of known degradation mechanisms and core types. Section C.3.3.4 stipulates that: *it is acceptable to combine the allowable frequency for multiple degradation mechanisms provided that the contribution has been quantified for each known mechanism.*

Although generally not accepted in Canada, the USNRC Regulatory Guide 1.174 [6] also provides acceptance guidelines for risk-based decision-making. This guidance includes the general recommendation that a change to the plant may be considered acceptable without performing an overall probabilistic analysis of the plant if the increase in Severe Core Damage Frequency (SCDF) associated with the change is below $10^{-6}$ per reactor year, and provided that there is no indication that the overall SCDF is above $10^{-4}$ per reactor year. For CANDU® reactors, the combined effect of increases in PT deuterium levels, increases in creep induced PT sag and the SLAR maintenance can be considered as such a change to the plant. The concept of SCDF is useful in comparing the relative risk associated with different reactor components, such as fuel channel and feeders, as part of a risk-informed management program.

**PROBABILISTIC METHODOLOGY**

The probabilistic methodology is based on the Monte-Carlo technique with a large number of realizations (or trials) per channel. Random sampling is performed on input parameters affecting the FC bending deformation and thus the PT to CT contact. The input parameters affecting blister formation and growth as well as the failure criterion also have distributed values. The selection of fixed and distributed parameters was based on previous probabilistic experience in the field and on the availability of data. The following paragraphs provide a brief description of the parameters.

The uncertainty in PT creep is the most critical parameter with respect to predicted contact time. Although the CT creep properties predominantly govern the overall fuel channel sag, these have a relatively small effect on PT to CT contact. Therefore, the creep uncertainty of the CT is currently accounted for through the creep uncertainties of the PT. The PT creep uncertainty is modelled as a single factor uniformly affecting the deformation equation along the entire length of the PT. The distribution was derived using PT to CT gap measurements from CANDU® 6 plants.

The end-slope uncertainty also has a dominant effect on the predicted contact time, although this effect is observed mostly in the first and last spans. End-slopes, as illustrated in Figure 4, are the angles between the axis of the PT at End Fitting (EF) locations and the line that connects the two ends of PT (i.e. line AB). The end slopes at the two ends of the PT are defined by Equation 1 and Equation 2. Where $\alpha$ is the angle between the axis of the end fitting and the horizontal plane, $\beta$ is the PT angle with the end fitting axis, and $\theta$ is the PT centre line angle with the horizontal plane. The end slopes as defined by Equation 1 and Equation 2 are very difficult to measure in-situ. In the absence of measured end slopes, the corrected PT sag measurements as shown Figure 4 and the corrected PT slope along the length of the channel are used to derive the PT ends slopes. This process is repeated for numerous sag measurements to define the PT end slope distribution used in the Monte Carlo simulations.

![Figure 4 – Schematic of End-Slopes](image-url)

**Equation 1**

\[
\text{PT Slope at A} = \alpha_1 + \beta_1 + \theta_{cl}
\]

**Equation 2**

\[
\text{PT Slope at B} = \alpha_2 + \beta_2 - \theta_{cl}
\]

In the probabilistic assessment, some combinations of the distributed parameters may result in a spacer being unloaded (a spacer is considered unloaded when the predicted load is less than 5 lbf). In these cases, PROBE determines the maximum distance that the spacer can move in either direction based on the applicability of the following geometric constraints:

1. Location of the next loaded spacer.
2. Locations on each side of the spacer where the gap is less than the spacer coil diameter.
3. Distance to the end bell.
The amplitude of the spacer displacements is considered through the ratio of observed displacement to the maximum possible displacement (based on the application of the above constraints). The distribution of the displacement ratio is based on a relationship between observed spacer movement and predicted gap profiles in CANDU® 6 reactors.

The uncertainty in the spacer locations is due to three effects. Each time that a SLAR is performed in a pressure tube, there can be an error in the location of the zero reference position, which affects all spacer locations measured in that pressure tube with a systematic bias (datum error). The accuracy of the axial position of the SLAR tool also affects the spacer locations (measurement error). Finally, since the spacer can tilt either towards the inlet or the outlet and the tool measures the centre of gravity of the spacer, there is uncertainty in the spacer contact location at the bottom of the pressure tube (tilt error). The amount of error is based on the tilt direction and the clearance between the coil inner diameter and the pressure tube outer diameter. All three sources of spacer location uncertainty are modelled as distributed inputs.

The PT and CT outside radius and wall thickness are treated as distributed parameters based on the design drawing dimensions. The sampled PT and CT dimensions are applied in each Monte-Carlo realisation to the calculation of the initial gap clearance between the two tubes. It is currently assumed that there is no correlation between the outside radius and the wall thickness of both the PT and CT.

The deuterium uptake rate is the most dominant parameter affecting the formation and growth of blisters (assuming contact between the PT and CT), and it is treated as a distributed parameter. The lifetime-average (LTA) temperature profile is treated as a deterministic input to the deuterium uptake rate model when calculating the deuterium concentration at axial locations along the PT. The variability in LTA temperature has a neutral effect on the probabilistic assessment since, if the LTA temperature were to change, the deuterium uptake rate model would have to be updated based on the new LTA temperature distribution. The initial hydrogen concentration varies little along the length and around the circumference of the PT and was treated deterministically.

The BFT values are treated deterministically and are obtained from the extrapolation of tabulated values [5], which are based on a set of complex calculations and benchmarked with experiments. The current PHTS operating temperature profile and the moderator saturation temperature are used as deterministic inputs in the calculation of BFT. In future assessments, more formal sensitivity studies may be performed.

Pressure tube failure is considered to occur when DHC initiates at a blister. One instance of DHC initiation at a blister is assumed sufficient to cause PT failure. This definition is somewhat conservative as a crack initiated by DHC would still have to propagate through the parent PT material and reach sufficient dimensions to rupture the PT. Blister cracking experiments have shown that, for a particular applied stress, the probability of DHC initiation is a function of the equivalent blister depth as illustrated in Figure 5. Therefore, a distributed failure model was used for DHC initiation.

For each PT in the core, the probability of blister failure is calculated at each of the potential sites of contact along the PT axis (240 sites with a 25 mm spacing; the lowest observed distance between blisters) for each Monte Carlo simulation. The probability of PT failure for each trial is calculated using the probability of blister failure at each axial location and the fact that PT failure is deemed to occur with one blister failure. The effect of tube dependency for the deuterium ingress is included in the calculations (i.e. it is unlikely that PTs with high ingress rates at one location would have low ingress rates at a neighbouring location). The total probability of failure for each channel is calculated as the average of the probabilities from each simulation.

For axial locations with contact, the probability of blister failure will depend on many factors including the initial hydrogen concentration, the lifetime average and current operating PHTS temperatures, the moderator saturation temperature and the assessment time. Figure 6 shows that, for all other conditions being equal, the relative probability of blister failure increases with temperature (i.e. axial location from inlet to outlet) and initial hydrogen concentration.

PROBABILISTIC EVALUATION

In a typical CANDU® 6 assessment, the probability of pressure tube rupture is estimated for each of the 380 fuel channels at one-year intervals, normally 7,000 Effective Full Power Hours (EFPH) apart. Fuel channels are sorted in order of decreasing probability of rupture. If the probability of rupture for the ith fuel channel at the end of the evaluation period is denoted \( p_i \), the aggregate probability of rupture for the \((i + 1)\)th, \((i + 2)\)th, … 380th fuel channels is denoted by \( p_{aggreg} \):

\[
p_{aggreg} = 1 - \prod_{i=1}^{380} (1 - p_i)
\]
Figure 5 - Results of Experimental Blister Fracture

Figure 6 – Example of Variation of Blister Failure Probability as a Function of Temperature and Initial Hydrogen Assuming PT to CT Contact

Figure 7 shows the normalized aggregate probability of pressure tube rupture for a typical CANDU® 6 reactor core assessment as a function of the number of inspected or maintained fuel channel assemblies (in order of risk). In this figure, it is assumed that the inspection or maintenance of each fuel channel reduced the probability of pressure tube rupture to zero. Therefore, Figure 7 provides an indication of the maximum risk reduction that can be achieved by inspection or maintenance of a number of the top channels.
The estimated frequency of pressure tube rupture per reactor year in the reactor core is calculated as the difference of the probabilities for each of the one-year intervals considered. Both the frequency and the probability of pressure tube rupture increases with time as shown in Figure 7 and Figure 8. In CANDU® 6 reactors with loose-fitting spacers, the frequency of PT rupture is typically dominated by a small number of channels. It is not uncommon to see 7 to 10 channels representing about 75% to 90% of the probability of PT rupture. Figure 7 and Figure 8 allow operators to plan outages to achieve their safety goals.

CONSERVATISMS

The general procedure during any probabilistic assessment is to use best estimates for the various inputs with realistic uncertainty distributions whenever possible. However, it is not always possible to develop best estimates either due to the data structure or the lack of data. In these cases, it is appropriate to use assumptions that tend to shift the calculated probabilities in a conservative direction. The conservatisms in this assessment are summarized as follows:

1. Pressure tube rupture (or failure) is assumed to occur when a blister grows to a critical size causing DHC initiation. It is assumed that once DHC is initiated, failure will occur instantaneously. This failure model takes no credit for the possibility that DHC propagation may become arrested, due to the steep temperature gradient and lack of Hg needed for DHC, as it progresses through the thickness of the PT.
2. Pressure tube rupture is deemed to occur with blister failure at one location.
3. Blisters are assumed to grow instantaneously to their equilibrium depth.
4. The possibility of leak before break (LBB) was not considered.
5. The distribution of PT creep rate includes the uncertainties due to measurement errors, which tends to widen the distribution.

CONCLUSIONS

Risk-informed tools provide the necessary information to manage nuclear plants in an optimized manner. Using these tools, operators can focus their interventions (inspection and maintenance) on the greatest sources of risk and obtain the best return on investment in term of additional safety achieved. The expectation is that the use of probabilistic tools in the nuclear industry will increase substantially in the upcoming years.
The probabilistic tool for contact and blister presented in this paper has been successfully applied to two different CANDU® 6 reactor core assessments. A number of activities are underway to gather additional inspection data and/or update inspection tools to improve the quality of the databases that are used to derive the distributions for the various input parameters. AECL is also developing new methodologies for benchmarking against inspection data, and thus yield input distributions that better represent the state of the reactor under study. These improvements will increase the confidence in the results obtained using the probabilistic methodology for contact and blisters.

**Figure 8 – Normalized Probability and Frequency of Pressure Tube Rupture as a Function of Operating Time for a Typical CANDU® 6 Reactor Core**

**REFERENCES**


