Structural Integrity Assessment of Pressure Tubes for Wolsung Unit 1 Based on Operational Experiences

Youn Won Park\textsuperscript{1)}, Sung Sik Kang\textsuperscript{1)} and BongSup Han\textsuperscript{2)}

\textsuperscript{1)} Korea Institute of Nuclear Safety, Korea
\textsuperscript{2)} Korea Electric Power Company, Korea

ABSTRACT

Pressure tube integrity has been considered as a key issue since the first operation of the CANDU reactor. Wolsung unit 1 has been in service since 1983 and subjected to inspections three times covering 44 tubes. The in-service inspections revealed that a major portion of inspected tubes was in contact with Calandria tubes. This is likely to increase the probability of blister formation which is a potential threat to pressure tube integrity. Fortunately, the inspection results indicated that no tube has been affected by blister formation so far. Nevertheless, to reduce the undue risk of blister formation the utility has decided to conduct spacer relocation every year until the entire core is covered. On the other hand, LBB analysis of pressure tubes using AGS performance measured at Wolsung unit 2 indicated that the operational safety margin was marginal when using 15-year operational data. This raises the concern of pressure tube integrity at Wolsung unit 1 which has had a more than 15-year operation. This paper presents the overall integrity evaluation of Wolsung unit 1 pressure tubes considering AGS performance test results and operating experience data.

INTRODUCTION

Since several leak incidents in the pressure tubes of CANDU reactors have been experienced, significant efforts have been made to improve their integrity through design, material, and fabrication upgrades in the last decade.

However, delayed hydride cracking (DHC) is still believed to be the major potential threat to pressure tube integrity. To cope with the likelihood of loss of pressure tube integrity due to DHC, Leak-Before-Break (LBB) is used as a defense-in-depth. Even though this is a very critical issue for old CANDU reactors, the operating procedure of Wolsong unit 1 has not been prepared to take LBB into account until 1998 when a set of complete LBB assessments have been carried out - due mainly to the lack of appropriate information.

The LBB concept requires timely detection, confirmation of leak, and appropriate operator action to shut the reactor down to a cold depressurized state before the growing crack exceeds the critical crack length (CCL). Since the Wolsong reactor consists of 380 pressure tubes, it is not easy to locate the troubled tube even though a leak is detected in
time and the reactor is cooled down. This is because once the reactor is in a
depressurized condition, the crack closes and the leak is almost impossible to find.
Unless the leaking tube is identified, the entire core inspection is necessary at a cold
state. This is very difficult, time consuming, and expensive. Therefore, leak detection
and leak location are important not only from an integrity viewpoint but also from an
economical standpoint. Normally, only leak detection is considered in the LBB analysis
because there is not enough data for leak location tests. During the pre-operational test
of Wolsong unit 2, a complete set of tests was conducted for leak detection and location
using simulated moisture injection. Since all of the functional capabilities of the leak
detection system of Wolsong unit 1 are the same as those of Wolsong units 2, 3, and 4,
the LBB assessment results were incorporated into the operating procedure of Wolsong
unit 1 in 1998.

CONFIGURATION OF FUEL CHANNEL IN THE CANDU REACTOR

A CANDU reactor consists of a large tank, called calandria, containing D₂O moderator
at 70°C, and is penetrated by 380 horizontal fuel channels each 6 m long. Each channel
consists of a pressure tube containing fuel and coolant D₂O at a pressure of 10Mpa and
at a temperature ranging from 260°C at inlet to 310°C at outlet. Fig. 1 shows a typical
fuel channel of CANDU reactor. The pressure tubes are surrounded and insulated from
the cold moderator by a calandria tube. The space between the pressure tube and the
calandria tube is filled with recirculating CO₂ gas, which is called the Annulus Gas
System (AGS). The separation of the pressure tube from the Calandria tube is
maintained by four evenly located spacers, called garter springs. The pressure tubes are
made from cold-worked Zr-2.5Nb with a wall thickness of 4 mm and an inside diameter
of 103 mm. The calandria tubes are made from annealed Zircaloy-2 with a wall
thickness of 1.4 mm and an inside diameter of 129 mm. The pressure tubes are rolled
into the end fittings at each end of the fuel channel. The residual stress produced by the
rolled joint fabrication process is still considered as a potential cause of developing
DHC, even though it has been remarkably reduced by using a zero clearance rolled joint
[1].

ANALYSES OF OPERATING EXPERIENCE DATA

Inspection Summary
The first CANDU reactor, Wolsong unit 1, has been operating since 1983. It was
required that all of the components covered by CSA CAN3-N285.4-M83 be subjected
to In-Service Inspection (ISI) within 5 years from the time of commercial operation.
The pressure tubes were supposed to be inspected before March of 1988. However,
since the CIGAR-special inspection tool for pressure tubes was not available at that
time, the continued operation was allowed by the authority until April of 1990 when the
first overall inspection of pressure tubes was carried out at Wolsong unit 1. The initial
two channel inspection (four high power and one low power) indicated an ultrasonic
response greater than the 0.15 mm deep reference notch in channel M-11. The scope of
the inspection was increased by an additional six channels. A total of 26 reportable
indications were found over 11 fuel channel inspections in 1990.
During the overhaul of 1992, nineteen pressure tubes were inspected with CIGAR. This consisted of full length volumetric inspections, inspections for sag and garter spring location, and gauging inside diameter and wall thickness. Four of the channels (M-11, O-14, Q-06 and S-15) were the repeats taken out of the 1990 CIGAR inspection and the others were selected for various reasons discussed between the regulatory body and the utility. The 1992 inspection revealed that there were 14 reportable indications at 8 channels and two flaws greater than code allowable at M-11 and O-08, and contact between pressure tube and Calandria tube took place at seven channels.

Two conclusions have been made: firstly, from an operation standpoint, the reactor could be put into service with the condition that total duration of operation could not exceed 10,500 EFPH or the number of cooldowns were limited to four; secondly, from safety concerns’ view point, two pressure tubes with flaws exceeding code allowances would be replaced and another 14 pressure tubes would be inspected in the 1994 overhaul.

The inspection conducted in February 1994 revealed that one pressure tube had many reportable indications including ones exceeding code limit. Consequently, this tube was replaced so that a total of three tubes were replaced. They were pulled out in March 1994. Table 1 shows the summary of pressure tube inspections and table 2 the analysis of the inspection results.

As shown in table 2, 95% of the inspected tubes were found to have mislocated garter springs and 33% of pressure tubes were in contact with Calandria tubes. 59% of them were affected by inner surface debris damage, however, most of the debris damage turned out to be minor defects except the flaws exceeding code allowances in size.

KEPCO decided that it was necessary to reposition the garter springs by performing Space Location And Reposition (SLAR) on all the fuel channels to minimize the likelihood of blister formation on pressure tubes contacting with Calandria tubes. Between 70 and 80 pressure tubes have been subjected to SLAR every year since 1995 and full coverage will be done by 2000.

As determined from the inspection results, three pressure tubes were replaced during the scheduled outage in 1994 and the removed pressure tubes were subject to metallurgical and mechanical investigation in Canada.
Table 1 Summary of pressure tube inspection

<table>
<thead>
<tr>
<th></th>
<th></th>
<th></th>
<th></th>
<th></th>
</tr>
</thead>
<tbody>
<tr>
<td>High Hydrogen</td>
<td></td>
<td>1</td>
<td>4</td>
<td>5</td>
</tr>
<tr>
<td>Debris Defect</td>
<td></td>
<td>1</td>
<td>7</td>
<td>8</td>
</tr>
<tr>
<td>PT/CT contact</td>
<td></td>
<td>2</td>
<td></td>
<td>2</td>
</tr>
<tr>
<td>Abnormal Elongation</td>
<td></td>
<td>3</td>
<td></td>
<td>3</td>
</tr>
<tr>
<td>Offcut Analysis or Ingot</td>
<td></td>
<td>1</td>
<td>2</td>
<td>3</td>
</tr>
<tr>
<td>Repeat</td>
<td></td>
<td>1</td>
<td>1</td>
<td>2</td>
</tr>
<tr>
<td>Extension</td>
<td>5:H, 1:L</td>
<td>6</td>
<td></td>
<td>12</td>
</tr>
<tr>
<td>Total</td>
<td>11</td>
<td>19</td>
<td>14</td>
<td>44(39)*</td>
</tr>
</tbody>
</table>

(39)* indicates the number of inspected tubes excluding 5 repeats from 44.

Table 2 Analysis of Inspection Results

<table>
<thead>
<tr>
<th>Defect type</th>
<th>Number of tubes</th>
<th>Rate (%)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Inner surface debris damage</td>
<td>23</td>
<td>59</td>
</tr>
<tr>
<td>PT/CT contact</td>
<td>13</td>
<td>33</td>
</tr>
<tr>
<td>Garter spring mislocation</td>
<td>37</td>
<td>95</td>
</tr>
<tr>
<td>Flawed tubes (&gt; code allowables)</td>
<td>3</td>
<td>8</td>
</tr>
</tbody>
</table>

**Inspection Data**

The foregoing inspection data provide two very important information in view of integrity assessment of pressure tubes: firstly, dimensional changes in diameter and in thickness and, secondly, deuterium ingress rate along the tube length from inlet to outlet. The base line data of the above parameters were taken from tube installation history dockets. Measurements of inner diameter were performed on 12 tubes in 1990, 18 in 1992, and 14 in 1994. Of the 44 inspected tubes, three were measured, repeatedly, in 1990, 1992, and 1994. The comparison between the measured data showed that the diameter values of the tubes inspected over three consecutive inspections increased at a higher rate than those of the others, which was, therefore, conservative from an integrity evaluation standpoint. Figure 2 shows the dimensional change in inner diameter versus operating years and we can see that the diametral increase is linearly proportional to operating years.

Thickness measurements were included during the 1994 inspection. The decrease rate in thickness can be drawn out from the comparison between the installed values and the inspection results over 14 tubes in 1994. The average and minimum thickness values are shown in Fig. 3 that indicates a linear decrease of thickness as the operating time increases.
Fig. 2 Inner diameter of pressure tube versus operating years

Fig. 3 Thickness reduction versus operating years

**Surveillance Examination Data**

The removal of three pressure tubes and the investigation of their material and mechanical properties provided important information in terms of integrity assessment of the Wolsong unit 1 reactor core. At the time when the replacement work was performed, Wolsong unit 1 reactor had operated for approximately 81,160 EFPH or 9.3 EFPY (8,760 EFPH = 1 EFPY). The scope of the work done for tube removal was similar to the standard tasks for the surveillance examination for pressure tubes previously removed from other CANDU reactors.

Of the examination results, fracture toughness test, tensile test, and measurement of delayed hydrided crack (DHC) velocity were used in the LBB assessment. A typical fracture toughness measured at 250°C on irradiated material would be a dI/da of less than 50 MPa. The measured fracture toughness of the removed tube is consistent with previous test results [2, 3]. The tensile strength increases whereas the fracture toughness drops during the first $2 \times 10^{25} \text{ n/m}^2$ of fast neutron fluence [2]. Both the yield strength and ultimate tensile strength reach a plateau value of about 820 MPa at 250°C. These values are consistent with the range of values measured in other irradiated pressure tubes removed from CANDU reactors [2, 3].
In addition to the above work, the evaluation of DHC properties was investigated since pressure tube integrity is dependent on the DHC properties. The DHC velocities in axial and radial direction were measured because these data are essential for LBB calculation. The axial and radial DHC velocities measured in the M-11 pressure tube at 130°C were within the 95% confidence limits of the previous data [4].

As a result of the surveillance examination, it was concluded that the fracture toughness and tensile strength values for Wolsong unit 1 reactor were consistent with other pressure tube data and the same values could be used for LBB assessment as those of reference [5].

LBB ASSESSMENT

Parameters and Assumptions for Leak-Before-Break Analysis

The LBB analysis means that when a leak occurs at a through-wall DHC crack, throughout the period of time needed for leak detection, confirmation, location and finally placing the reactor in a cold depressurized state, the growing crack length is always less than the critical crack length in order to assure that the probability of an unstable pressure tube rupture is extremely low.

Therefore, the LBB analysis integrates the Annulus Gas System (AGS) response to leakage from an assumed through-wall DHC crack with the change in crack velocity and critical crack length as the reactor goes from full power to a cold depressurized condition. The actual station operating procedures should be used to analytically simulate the sequence of events (SOE) following the crack penetration of pressure tube. The parameters needed for the LBB analysis are taken as follows [6]:

- Maximum crack length at penetration (Lp) of 20 mm is used in this analysis based on the recommendation of FFSG.
- The upper bound values of DHCV in axial direction is, conservatively, used.
- The half-critical crack length (c), as a function of temperature, stress, fluence, and hydrogen concentration, can be determined from the following equation:

\[
\sigma_f = \frac{\sigma_y + \sigma_w}{2}, \quad \sigma_h = P\left(\frac{r}{w} + 1\right)
\]

\[
m = [1 + 1.255\phi^2/(\phi_w^2)] - 0.0135\phi^2/(\phi_w^2) - 1.5^\circ
\]

- Station Response to Moisture in AGS; The time needed to detect, confirm, and locate the leak should be provided. Following the first leak at penetration, the time necessary for initiating the rate of rise alarm is taken from the AGS performance test carried out at Wolsung unit 2 in May 1997 [5].
- Pressure tube thickness and inner diameter equivalent to 15-year operation and 30-year operation are determined from Fig. 2 and Fig. 1, respectively.
- DHC crack is to occur in the most likely area which is the inboard of the rolled joint because of relatively high residual stress.
- DHC crack growth is assumed in both directions throughout the event.

V-316
- A lower bound estimate of critical crack length and fracture toughness is used.
- The pressure tube leak occurs in the longest AGS channel and the AGS is operated in very low recirculation flow. This leads to a longer moisture detection time.
- There are sufficient hydrides at power for crack growth to occur.

Based on the current operating procedures, a complete sequence of events are established from leak detection to the time of the reactor being placed in a cold depressurized state.

**Sequence of Events based on Current Operating Procedure**

The Sequence of Event (SOE) including the leak detection, confirmation, leak location and reactor shutdown should be developed based on station operating procedure. Based on the AGS test results [6], a rate of rise alarm is to be initiated less than 2 hours after the start of a leak, when the AGS is under very low flow operation.

Operators should reduce power from Full Power Hot (FPH) to Zero Power Hot (ZPH : 261°C and 7MPa). Reactor temperature drop with a maximum allowable cooldown rate, 2.8°C/min, is followed by a pressure drop with a maximum rate of 1bar/min. After reaching ZPH, the AGS should be put into the stagnant mode and should be followed by the collection of leakage and the leak location process, according to the AGS abnormal operation procedure. As soon as the leaking source is located or total allowable duration time elapses, the reactor should be cooled down to a cold depressurized condition.

**LBB Analysis**

The crack length at each time step (every 0.1 hour), when time=t, has been calculated using the following expression [6]:

\[ L_t = L_{t-1} + (V_t + V_{t-1})(t_t - t_{t-1}) \]  \hspace{1cm} (2)

Where, \( L_t \) = length of crack at time \( t \), \( V_t \) = DHC velocity at time \( t \).

The critical crack length at each time step was also generated from equation (1) and the growing crack was calculated using equation (2). The LBB assessment of Wolsong unit 1 reactor is shown in Fig. 4. As shown in this figure, the LBB can be satisfied with relatively ample margin when 15-year operation data were used, however, it is barely marginal for a 30-year operation case. It is, therefore, necessary to take appropriate action to assure the LBB of Wolsong unit 1 reactor to the end of design life even though the CCL is not exceeded by the growing crack throughout the SOE because of the too small safety margin.
CONCLUSIONS

The inspection data of Wolsong unit 1 pressure tubes were reviewed and the surveillance examination results were compared with those of previous investigations. As a result, it is concluded that Wolsong unit 1 pressure tubes show a behavior consistent with the other CANDU pressure tubes and exhibit more safety to a certain degree. The LBB assessment was performed using inspection and surveillance data. The conclusion is that the LBB is satisfied when 15-year operation data are used, however, it is marginal when the operating year approaches the end of life.

REFERENCES