Transactions of the Korean Nuclear Society Autumn Meeting
PyeongChang, Korea, October 25-26, 2007

Development of Safety Margin Analysis Methodology on Aging Effect for CANDU Reactors (II)

Manwoong Kim, Sang Kyu Lee, Hyun Koon Kim, Kun Joong Yoo, Yong-Ho Ryu
Korea Institute of Nuclear Safety, Goosong-dong, Yusong-gu, Daejeon, 19, mwkim@kins.re..kr

Yong Won Choi,a Un Chul Lee
Dept. Nuclear Engr. Seoul Nat'l Univ., Shillim-dong, Kwanak-gu, Seoul, 151-744, kaks2000@snu.ac.kr

1. Introduction

Considering that operating year of Wolsong Unit 1 gets close to the design life, 30 years, the aging effect due to the component degradation takes into consideration as an important safety issue. However, since the thermal-hydraulic effect due to the aging did not identify clearly, the safety analysis methodology is not be well established so far. Therefore, in this study, the aging effect affected by thermal-hydraulic characteristics was investigated and a safety margin analysis methodology considering aging effect was proposed

2. Evaluation Model Method for Aging Effect

In the best estimate bounding approach, the best estimate computer code is used while the uncertain input parameter values are selected conservatively to bound the parameter of interest. This approach represents the uncertainties by taking upper bounds for the ranges of uncertain parameter values. The approach has many similarities with best estimate plus uncertainties. However, the major difference is that instead of quantifying the impact of input uncertainties the result is expected to be bounded. One of the major limitations of such methods is that they may involve unquantifiable over conservatism due to the linear combination or bounding of all conservative assumptions.

In this study, after identifying aging phenomena by component degradation in CANDU reactors, various elements are determined to analysis the thermal-hydraulic phenomena using RELAP-CANDU code. In addition, an uncertainty analysis also conducted using the statistical method like a random sampling analysis. Thereafter the major influence aging components on the safety is identified for CANDU reactors.

2.1 Ageing Effect Components and Elements

As for CANDU reactors, the following ageing components and elements are considered in Reference 1.Based on Table 1, the ageing functions for ageing elements are assumed as a Weibull statistical distribution type shown in Table 2.

Table 1: Ageing effects and applicable code input

Aging Component	**Code Input**	
Neutron irradiation embitterment	Conductivity change	Not considered
SCC	Small leak junction	Not considered
Corrosion	Roughness Junction loss coefficient Hydraulic diameter	Applied to sub-channel
Fatigue	-	Not considered
Stress relaxation	Hydraulic diameter	Not considered
Creep, Growth and Sag	Junction loss coefficient Hydraulic diameter Junction model option	Applied to sub-channel
Wear	Roughness	Applied to average channel
Pump degradation	Pump head Rated flow	Not considered
etc	etc	-

Table 2: Ageing components and assumed ageing function

Aging Component	**Agine Element**	**Assumed degree of ageing**	**Assumed ageing function**
Corrosion	roughness	1000.0% for 60year	$f = e^{0.039965x}$
	Junction loss coefficient	200.0% for 60year	$f = e^{0.0183102x}$
	Hydraulic diameter	5.0% for 60year	$f = e^{-0.000855x}$
Creep, Growth, and Sag	Volume area	5.0% for 60year	$f = e^{-0.00085}$
	Junction area	5.0% for 60year	$f = e^{-0.00085}$
	Junction loss coefficient	200.0% for 60year	$f = e^{0.0183102x}$
	Hydraulic diameter	2.5% for 60year	$f = e^{-0.000422x}$
Wear	roughness	50.0% for 60year	$f = e^{-0.01155x}$
	Junction area	2.0% for 60year	$f = e^{-0.000336}$
	Volume area	2.0% for 60year	$f = e^{-0.000336}$

2.2 Evaluation Model Analysis

The evaluation model methods may be used to provide more realistic estimates of plant safety margins, provided the licensee quantifies the uncertainty of the estimates and includes the uncertainty when comparing the calculated results with prescribed acceptance limits.

In this study, LBLOCA is considered as an accident scenario because it enables to identify thermal-hydraulic effects most obviously among most accidents. To analysis

LBLOCA, an average channel, composed of 95 assemblies, is divided into 1 average channel and 7 sub channels to observe the local ageing effects with assumptions estimated aging model as shown in Table 3.

Table 3: Identified ageing components and elements

Ageing Component	Ageing element	Ageing Mechanism
Fuel Channel	roughness	Corrosion
	loss coefficient	Pressure Tube (PT) Creep and Sagging
	hydraulic diameter	PT Creep and Sagging Corrosion
	flow area	PT Creep and Sagging
Pump	pump head	Degradation
	pump flow	Degradation
Steam Generator	roughness	Corrosion
	hydraulic diameter	Corrosion
Feeder Inlet+ End Fitting	roughness	Corrosion

2.3 Statistical Sampling Method

To develop a major ageing components and elements mapping, a probabilistically based sampling is used This mapping then provides a basis for both the evaluation of the probability (i.e. uncertainty analysis) and the evaluation of the effects of individual input parameters on output parameters (sensitivity analysis). In this study, A Latin Hypercube Sampling (LHS) method was used to identify mapping for ageing components and their related ageing elements. This mapping then provides a basis for both the evaluation of the probability (i.e. uncertainty analysis) and the evaluation of the ageing effects of individual ageing components on output parameters (sensitivity analysis).

3. Sensitivity Evaluation Results

To investigate the sensitivity of ageing components and elements, the analysis is conducted in terms of cladding temperatures with ageing period.. In this study, RELAP-CANDU model was developed that incorporated all the mitigation systems and evaluate the cladding temperature for cross section and roughness in the fuel channel, pump head, and roughness of steam generator(SG) as shown in Figs 1-4. It is shown that the ageing element and components is sensible in terms of pump head and roughness of SG, while ageing elements of fuel channels are not.

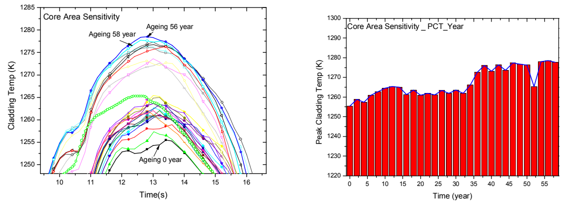

Figure 1 Sensitivity of cross section in fuel channel

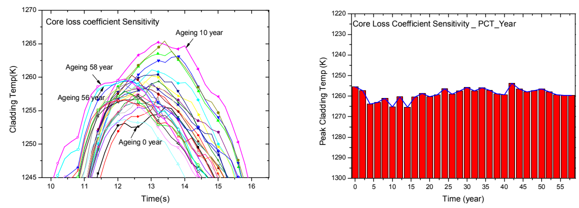

Figure 2 Sensitivity of roughness in fuel channel

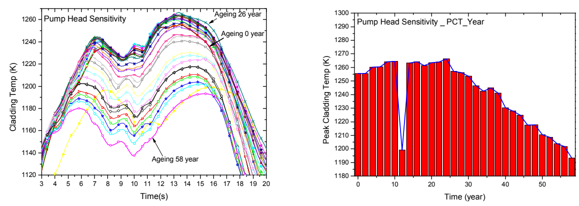

Figure 3 Sensitivity of pump head

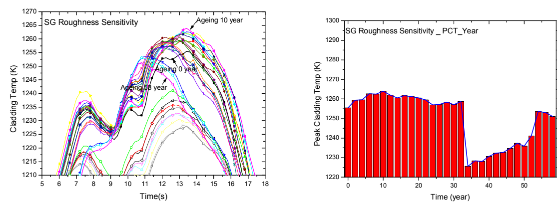

Figure 4 Sensitivity of roughness in SG

4. Conclusion

In this study, it is found that the thermal-hydraulic characteristics due to aging effects varies in accordance with the operation time during steady or transient state in CANDU reactors. Furthermore, the sensitivity analyses are also conducted to identify the effect of ageing elements and components with the operating period.

REFERENCES

[1] IAEA, “Assessment and management of ageing of major nuclear power plant components important to safety: CANDU Reactors assemblies” IAEA-TECDOC-1197, April 2001.