The Thermal-hydraulic Analysis for the Aging Effect of the Component in CANDU-6 Reactor

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1. Introduction

CANDU reactor consists of a lot of components, including pressure tube, reactor pump, steam generator, feeder pipe, and so on. These components become to have the aging characteristics as the reactor operates for a long time. The aging phenomena of these components lead to the change of operating parameters, and it finally results to the decrease of the operating safety margin. Actually, due to the aging characteristics of components, CANDU reactor power plant has the operating license for the duration of 30 years and the plant regularly check the plant operating state in the overhaul period. As the reactor experiences the aging, the reactor operators should reduce the reactor power level in order to keep the minimum safety margin, and it results to the deficit of economical profit [1]. Therefore, in order to establish the safety margin for the aged reactor, the aging characteristics for components should be analyzed and the effect of aging of components on the operating parameter should be studied.

In this study, the aging characteristics of components are analyzed and revealed how the aging of components affects to the operating parameter by using NUCIRC code. Finally, by scrutinizing the effect of operating parameter on the operating safety margin, the effect of aging of components on the safety margin has been revealed.

2. Numerical Methods

The thermal-hydraulic analysis of a CANDU-6 reactor fuel channel was performed with an inlet header temperature of 262°C, an outlet header pressure of 9.99 MPa, and a header-to-header pressure drop of 1282 kPa by using NUCIRC code [2]. In the present calculation, the circuit analysis has been conducted by using the ITYPE 6 among NUCIRC calculation options. The detailed calculation procedure for the circuit analysis is well documented in the reference [3].

The main aging components are firstly defined and the effect on the operating parameter, such as inlet header temperature and pressure, has been analyzed. The values of aging parameter for the 15, 30 EFPY (Effective Full Power Year) is predicted from the historical operating data of Wolsung units. In order to reveal the independent effect of the aging parameter, the reference values has been used for the plant operating condition except the considered aging parameter.

3. Results

In this study, the aging effect has been analyzed for the main aging components of pressure tube, feeder pipe, steam generator, orifice degradation factor. The feeder pipe roughness for 15, 30 EFPY is predicted from the operating data of Wolsung unit 4 in 11 EFPY by extrapolating from the present data sets. The values of feeder pipe roughness are listed in Table.1 and the other aging parameter is secured by same method, which is not shown in this paper.

Table 1. The feeder pipe roughness with EFPY

EFPD	EFPY	Pass 23	Pass 41	Pass 67	Pass 85
0	0	0.00030	0.00030	0.00030	0.00030
2086	5.7	0.00045	0.00029	0.00014	0.00007
2986	8.2	0.00143	0.00094	0.00039	0.00033
3873	10.6	0.00215	0.00221	0.00048	0.00041
5475	15	0.00346	0.00451	0.00064	0.00054
10950	30	0.00792	0.01236	0.00118	0.00101

Figure 1 and 2 shows the effect of feeder pipe roughness on the inlet header temperature and pressure, respectively. It is revealed from the figure that the feeder pipe roughness tends to increase the inlet header pressure, while it has small effect on the inlet header temperature. The inlet header temperature tends to increase with the creep ratio of pressure tube, while the inlet header pressure decrease with the creep ratio because the pressure drop in the reactor core decrease with the creep ratio and the outlet header pressure is kept to a constant value by the pressurizer.

Figure 3 shows the change of boiler flow rate with respect to the feeder pipe roughness. As the feeder pipe roughness increases, the flow resistance in the RCS is increased, which results to decrease the flow rate in RCS. Hence, the decreased flow rate in the RCS makes the pressure drop in the core to decrease, which results in the increased inlet header pressure.

Figure 4 and 5 shows the effect of steam generator fouling factor on the inlet header temperature and pressure, respectively. It is revealed from the figure that the steam generator fouling tends to increase the inlet header temperature, while it has small effect on the inlet header pressure. As the steam generator fouling increases, the heat transfer between primary and secondary RCS is decreased due to the low heat conductivity in steam generator, which results to increase the inlet header temperature. Since the steam generator fouling factor is only related to the heat transfer rate, the inlet header pressure and RCS flow rate is nearly unaffected by the steam generator fouling factor.



Fig.1 The effect of feeder pipe roughness on the inlet header temperature.



Fig.2 The effect of feeder pipe roughness on the inlet header pressure.



Feeder Pide Roughness

Fig.3 The effect of feeder pipe roughness on the boiler flow rate



Fig.4 The effect of steam generator fouling on the inlet header temperature



Fig.5 The effect of steam generator fouling on the inlet header pressure

Figure 6 shows the effect of orifice degradation on the inlet header pressure. Since orifice degradation tends to increase the core flow rate due to the decreased flow resistance, the inlet header pressure is shown to increase with the orifice degradation factor. Though it is not shown here, the orifice degradation has a negligible effect on the inlet header temperature.



Fig.6 The effect of orifice degradation on the inlet header pressure

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REFERENCES

[1] W. H. HARTMANN, H. CHOW, V. CAXAJ, "ROP Trip Setpoint Aging Trends for Wolsong-1,", AECL 59-03330-AR-001, Rev 0 (2004)

[2] M. F. LIGHTSTONE, "NUCIRC-MOD1.505 Users Manual," TTR-516, Atomic Energy of Canada Limited (1993)

[3] J. H. Bae et al., "The Channel and Circuit Analysis Methodology for the Thermal-Hydraulic Analysis of CANDU-6 Reactor, KAERI/TR-4901/2012 (2012)