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Aging Effects of CANDU on CCP with Various Reactor Power under Normal Condition

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1. Introduction

Thermalhydraulic analysis is one of the most important safety analyses for Nuclear Power Plant (NPP) operation and reactor management for avoiding fuel failure. In CANDU (CANadian Deuterium Uranium) reactor, Critical Channel Power (CCP) should be estimated for thermalhydraulic safety of CANDU fuel channel. CCP is defined as the power of the fuel channel where Critical Heat Flux (CHF) occurs and can be regarded instead of CHF in CANDU [1].

Flexible or load-following operation is emerging as a technical issue for NPPs in recent years. Its availability has the potential to serve as advantages for NPPs worldwide, allowing NPPs to operate more safely and sustainably for an extended period [2]. Especially, flexible operation of NPP is highlighted issue for the energy-mix policy and stable power supply in Korea.

For this reason, evaluating the impact of various aging condition and reactor power on CCPs is essential for the thermalhydraulic safety analysis of CANDU reactor and fuel. Also, this assessment should be conducted progressively and systematically. Therefore, CCPs were evaluated under normal operating condition with various reactor power and Effective Full Power Days (EFPD) to find out the aging effect on CCP. In this study, NUCIRC was utilized to compute CCPs.

2. Methodology and Assessments

NUCIRC, thermalhydraulic computational analysis code, is used for making model and calculating conditions and CCP of CANDU. In this assessment, all data were assessed by NUCIRC 2.3.5 version. NUCIRC 2.3.5 is the latest version that includes the range of pressure tube diametral creep over 5.1% and related CHF, Onset of Significant Void (OSV), Two-Phase Frictional Multiplier (TPFM) correlations and correction factors. This paper verifies the CCP trend with various reactor power and aging effects of the generic CANDU on CCP under normal condition.

2.1 Thermalhydraulic Conditions

In the case of the specific reactor power levels which the generic CANDU reactor remained for a period of time, all input factors including thermalhydraulic conditions such as reactor inlet header temperature, outlet header pressure and header-to-header pressure drop were measured by a variety of instrumentation on site [3]. However, for the CCP estimation for the other various reactor power, thermalhydraulic conditions were assumed to vary based on reactor power, considering thermalhydraulic conditions under two reference reactor power levels. It was because it is practically difficult to remain the specific reactor power of generic CANDU reactor for a long time under operating conditions. Additionally, since it is impossible to measure the thermalhydraulic conditions in operational aging that have not actually occurred, thermalhydraulic conditions for aging effects were supposed based on the recent measured data.

2.2 Reactor Power Variation

In this research, all models were made considering the continuous reactor power variation during flexible operation of CANDU reactor, where the reactor power is going to change gradually and scenarios involve operating at the specific lower power levels such as 40%, 60% and so on for a certain period. As described in section 2.1, thermalhydraulic parameters in models for the long-term remaining reactor power were applied using on-site measurement values. For other reactor power, models were composed of estimated value of thermalhydraulic parameters based on the measurement values at the reference reactor power.

2.3 Aging Effect

Thermalhydraulic modeling reflecting aging effects were established and CCP computation were conducted at intervals of 1,000 EFPD from the reference aging to 3,000 EFPD after then. Main dimensions of feeders and fuel channel influenced by aging were feeder roughness, orifice degradation and creep, and etc. [4]. These factors have an effect on NUCIRC-calculated flow for each fuel channel and CCPs. Therefore, by incorporating the variations of these factors due to aging effects into the thermalhydraulic modeling, the aging effects can be assessed.

2.4 CCP Assessments

CCPs were calculated by using NUCIRC 2.3.5 based on the latest updated CHF-related correlations and correction factors as stated above. The CCP assessment results varied due to various model parameters such as reactor power, thermalhydraulic conditions and feeder roughness, orifice degradation, creep from aging effects. In this assessments, Fig. 1 to 4 show the CCP profiles of all fuel channels at the reference operation aging, 1,000, 2,000 and 3,000 EFPD after reference aging under normal condition with low reactor power level.

Fig. 5 is the comprehensive CCP assessments results, including Fig. 1 to 4, from reference aging to 3,000 EFPD after reference aging with various reactor power. It is noted that overall CCPs tend to decline almost linearly and consistently as aging increases. Although this figure depicts CCPs with low reactor power level far from the occurrence of film boiling, estimated CCP decreases about 6 % (450 kW) as the aging increases 3,000 EFPD, indicating that CCP is significantly affected by aging.

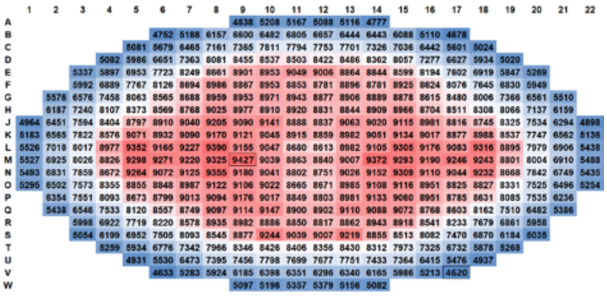

Fig. 1. CCPs with low power level at reference aging.

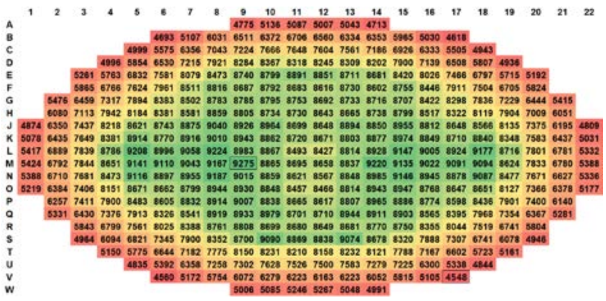

Fig. 2. CCPs with low power level at 1,000 EFPD after reference aging.

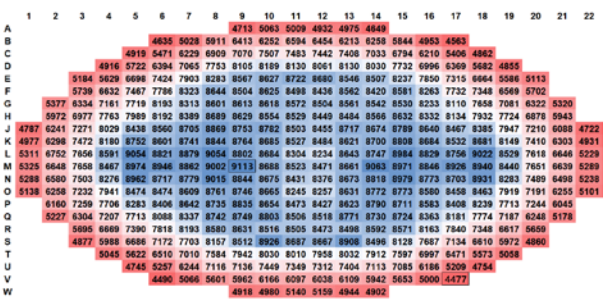

Fig. 3. CCPs with low power level at 2,000 EFPD after reference aging.

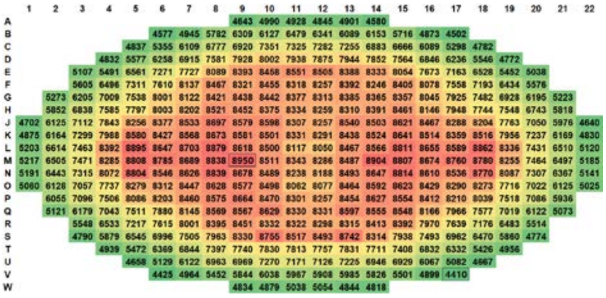

Fig. 4. CCPs with low power level at 3,000 EFPD after reference aging.

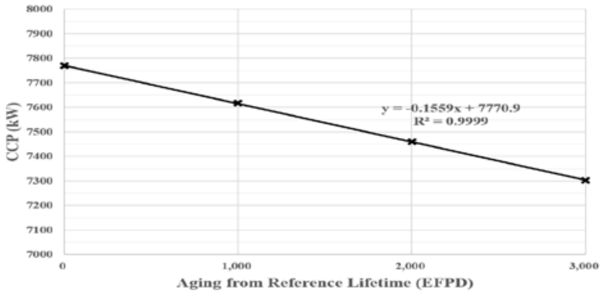

Fig. 5. Aging effects on average CCP for the generic CANDU.

3. Conclusions

From the assessment result, it is mainly noticed that the decreasing tendency of CCP is caused by increasing of creep and adverse changes of thermalhydraulic conditions while increasing operation aging. But, the additional investigation is still needed for identifying the quantitative effect of each factor. Also, estimation results should be derived using real-time collected on-site measurement to reflect more realistic data for flexible operation of the generic CANDU reactor. Lastly, it is indispensable to figure out CCPs under the most limiting operating case for the generic CANDU reactor to identify the regional overpower protection (ROP) trip setpoints from the viewpoint of thermalhydraulic safety analysis. Eventually, it is expected that the CCP calculation results are going to be used as the input data for evaluation of ROP trip setpoints with various reactor power and aging in the future.

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