CCP Assessment under CANDU Normal Operating Condition with Various Reactor Power

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1. Introduction

CANDU (CANadian Deuterium Uranium) reactor is widely used for energy generation in a number of countries such as Canada, China and Republic of Korea [1]. In the present, three CANDU reactors, Wolsong-2, 3 and 4, are operating and supply not a few portions of electric power in Korea. Thus, safe and robust CANDU operation is essential.

Thermalhydraulics is the most important part for safe CANDU Nuclear Power Plant (NPP) operation and reactor core management for evading fuel failure. For CANDU, Critical Heat Flux (CHF) and Critical Channel Power (CCP) should be accounted for the thermalhydraulic integrity of fuel channel. CHF is the heat flux at the moment when dryout occurs, in which the nucleate boiling on the surface of high-temperature fuel sheath is converted into the film boiling. Especially, in CANDU reactor core, the power of the fuel channel where CHF occurs is defined as CCP.

Meanwhile, flexible operation or load following operation of NPP (Nuclear Power Plant) is one of the interested issue due to the energy mix policy and plan for the sustainability of electric power supply in Korea [2]. However, any research has not been found to analyze CCP sensitivity by changing reactor power and thermalhydraulic conditions derived from reactor power variation under the CANDU flexible operation conditions. In order to achieve the stable and safe flexible operation of CANDU, it is meaningful to assess CCP with various CANDU reactor powers for providing the basic data to compute Regional Overpower Protection (ROP) trip setpoints.

In this paper, we estimated CCP as a function of power under normal operating condition for the generic CANDU reactor by using NUCIRC (NUclear heat transport CIRcuit thermo hydraulics analysis Code).

2. Methodology and Assessments

This paper introduces calculation processes for CCP of generic CANDU NPP using NUCIRC. First, basic thermalhydraulic conditions and assumptions, vital to make a CCP calculation, are stated. And then, it is explained how build models and assess the CCP tendencies depending on reactor power. In this study, all data was produced by NUCIRC version 2.3.5 (NUCIRC 2.3.5).

2.1 Basic Conditions of CANDU Reactor

All input factors for the NUCIRC model were measured by a variety of instrumentation on site. The thermalhydraulic input data applied to this analysis from the site measurement includes the inlet and outlet header temperature, outlet header pressure, pressure drop between inlet and outlet headers [3]. Other system dimensions stated in the CANDU6 design specifications are feeder geometry, orifice and venturi data, feeder loss coefficients, fuel channel and bundle burnup distribution, and etc [4]. By modeling based on input data, NUCIRC develops a primary heat transport system (PHTS) model and figures out CCP by reflecting aging effects as roughness of feeders and orifice degradation coefficients.

2.2 CHF and CCP

In PWRs, CHF is mainly regarded as Departure from Nucleate Boiling (DNB) phenomena. But, slightly differed from PWR, CCP is deemed instead of CHF in CANDU, so NUCIRC assesses CCP using CHF correlation and other relevant correlations such as Onset of Significant Void (OSV) and Two-Phase Frictional Multiplier (TPFM). CCP was calculated for each fuel channel.

2.3 Application for NUCIRC Modeling

NUCIRC is the thermalhydraulics computational analysis code and usually utilized for establishing model and calculating thermalhydraulic conditions and CCP of CANDU NPP [3]. And, NUPREP is the input file pre-processor for the NUCIRC computation [4]. When calculating CCP, NUCIRC reflects the design dimension of primary heat transport system and measured flow properties by a number of instruments. NUCIRC 2.3.5 envelops the range of pressure tube diametral creep up to 6.8%. Differed from the previous version, the correlations and parameters related CHF, OSV, TPFM and correction factor can be applied over 5.1% creep. Therefore, CCP evaluation by NUCIRC 2.3.5 in high aging cases is applicable.

2.4 Assumption and Dataset

In this assessment, it was assumed that the generic CANDU NPP has loaded the modified 37-fuel element (37M) and all the coolant, systems and components in PHTS have behaved in accordance with the latest updated CHF-related correlations.

Except for the system dimensions mentioned in Section 2.1, the major parameters affecting CCP are pressure tube diametral creep and thermalhydraulic conditions of the PHTS coolant and system. Creep can be calculated based on the effective full power days (EFPD) of CANDU reactor, as well as the diameter measurement of the pressure tube and code prediction data on creep. The aging condition of generic CANDU reactor are assumed for the two aging points and all input factors for the NUCIRC model, including creep and thermalhydrualic conditions, were hypothesized at the specifically assumed aging conditions. Furthermore, almost all of the thermalhydraulic conditions were evaluated to vary according to the reactor power and applied to the input data in this research. In summary, all the calculation is carried out based on the generic CANDU reactor condition.

2.5 CCP with Reactor Power

Fig. 1 and 2 show the CCP profiles of all the fuel channels at 40% and 100% reactor power computed in this study under normal operating condition for one of the assumed aging points. In general, CCPs of the central fuel channels are higher than those of the outer channels, and as the reactor power increases, overall CCPs tend to decline slightly.

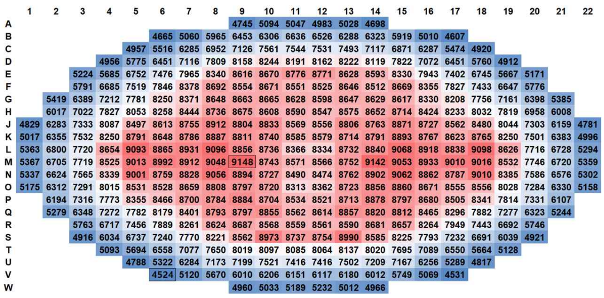

Fig. 1. CCP profile at 40% power under normal condition.

Fig. 2. CCP profile at 100% power under normal condition.

Fig. 3 illustrates the NUCIRC CCP computational results with various reactor power. In this figure, CCP decreases only about 1% (65kW) as the reactor power increases from 0% to 100%, indicating that CCP is not quite affected by the reactor power-related factors. It also shows that CCP increased linearly and quite consistently with the decrease of reactor power up to 80% (black dots and linear line), but beyond 80% (red dots and line), it decreased sharply, unlike the previous trend. Additionally, almost all of the CCPs have this tendency in general. It is expected that primary coolant near the fuel channel in PHTS might be boiling over 80% reactor power and the coolant is changed from liquid single-phase to liquid-vapor two-phase. And so, thermalhydraulic properties of coolant could be dramatically changed. Meanwhile, it can be hypothesized that the trend shown in the figure is more attributed to the thermalhydraulic conditions that are influenced by reactor power rather than reactor power itself.

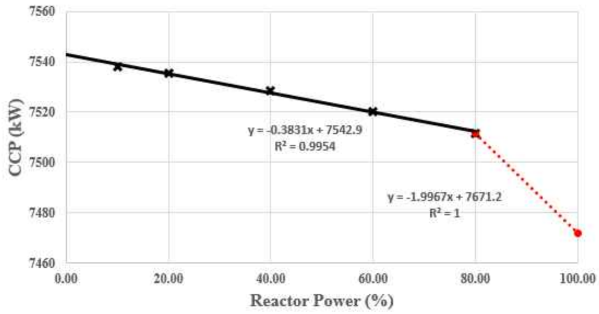

Fig. 3. Average CCP of all fuel channels as a function of reactor power for the generic CANDU6.

3. Conclusions

From the result of the NUCIRC computation in this investigation, CCP of the generic CANDU reactor decreased as the reactor power increased. In addition, at low power conditions, CCP is higher and more favorable in terms of fuel thermalhydraulic safety analysis. However, the results of this research are based on the hypothesized measurement data and parameters at the specific aging. In order to obtain more practical and reliable data for flexible operation, continuous estimation results will be derived using realistic and real-time collected on-site measurement data. Expanding and improving this estimation, it is important to assess CCP for more continuous and minutely reactor power intervals to understand the real CCP fluctuation with actual reactor power variations. And also, we plan to investigation how CCP changes due to the aging effects during flexible operation in the future. Finally, the calculated CCP is used for core safety design such as thermal margin and ROP trip setpoint evaluation for continuous flexible operation.

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