ON THE ASPECT OF EVALUATION OF CRITICAL CHANNEL POWER AND ASSOCIATED UNCERTAINTY IN CANDU SLOW LOSS OF REGULATION EVENT ANALYSIS

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ABSTRACT

Critical channel power is an important figure-of-merit parameter in safety analysis of CANDU® slow loss of regulation events. Evaluation of critical channel power and associated uncertainty for such events is complex; it has entailed significant analytical and experimental effort and numerous discussions within the CANDU Industry. Critical heat flux and dryout power are two other important parameters that are closely related to critical channel power and to each other. Unfortunately, in many occasions, particularly in the context of uncertainty analysis, the relationship of the critical heat flux and dryout power with critical channel power is very much confused. This paper clearly defines the terms of critical heat flux, dryout power, and critical channel power that are frequently referenced in CANDU slow loss of regulation event analysis, and clarifies their inter-relationship and the associated proper uncertainty propagation formulation. This information should provide better understanding of the sensitivity of critical channel power to the critical heat flux parameter, and better understanding on how to properly quantify the uncertainty in critical channel power due to uncertainty in critical heat flux.

KEYWORDS

CANDU; Critical Heat Flux; Critical Channel Power; Uncertainty

1. BACKGROUND

While CANDU reactors (see Figure 1) share some similarities with other water-cooled reactors in heat transport system (HTS) design, their core configuration is unique. Unlike other water-cooled reactors, in which the reactor fuel assemblies and the coolant/moderator are contained in a high-pressure reactor pressure vessel, the CANDU reactor core is structured as an assembly of multiple horizontal parallel channels (pressure tubes inside calandria tubes) that are immersed in the moderator housed in a low-pressure vessel (calandria). These channels are each fuelled with 12 or 13 fuel bundles and cooled with high-pressure coolant flowing inside the pressure tubes (e.g., the fuel channels). A typical CANDU reactor core is comprised of approximately 300 to 500 fuel channels, depending on the reactor power output. These horizontal parallel fuel channels are connected, through feeder pipes, to inlet and outlet headers which are connected to pumps or steam generators (see Figure 2). As these fuel channels are part of the HTS pressure boundary and also one of the multiple physical barriers to fission product release, their integrity must be ensured during normal operation, anticipated operational occurrences (AOOs) and design basis accidents (DBAs).

During normal operation and AOOs, reactor power and power distribution are regulated, through various control rods and liquid zone controllers that are strategically spatially installed in the core to achieve desired power level and power profile. In an event of loss of regulation (LOR) the reactor power may drift and/or the power profile may distort, possibly causing challenges to fuel and fuel channel integrity. If the LOR event is fast and causes an appreciable change in global power, the neutron power detection system and/or the process trip parameters, such as HTS high pressure, would initiate a trip to shut down the reactor using two independent CANDU shutdown systems (SDSs). However, if the LOR event is slow and local in the core, the above-mentioned trips may not initiate or may not initiate timely. This kind of LOR event could cause local distortions to the power profile, resulting in a large number of possible axial and radial power distributions that may not be symmetrical, that may pose challenges to some affected fuel bundles and channels. To terminate such slow LOR event, zone detectors are installed strategically across the core to detect regional power excursions through detection of high neutron flux. The detectors would actuate the reactor safety system if the regional power approaches pre-defined power limits. These pre-defined power limits are called the regional overpower protection (ROP) or neutron overpower protection (NOP) trip setpoints. These trip setpoints are determined by a complex analysis of stylized slow LOR accidents – ROP/NOP analysis.

The safety acceptance criterion used to determine the ROP/NOP trip setpoints is “trip before fuel sheath dryout” with prescribed probability and confidence level. Fuel sheath dryout is evaluated through critical channel power (CCP) analysis. Therefore, appropriate evaluation of CCP and associated uncertainty is essential to ROP/NOP trip setpoint calculations.

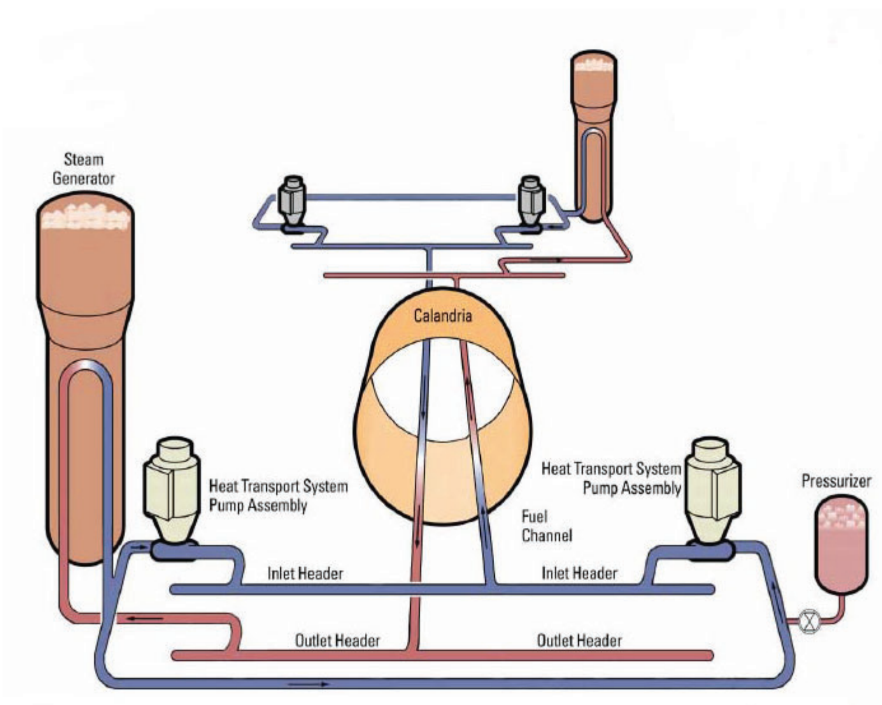

Figure 1. CANDU Reactor Primary Heat Transport System

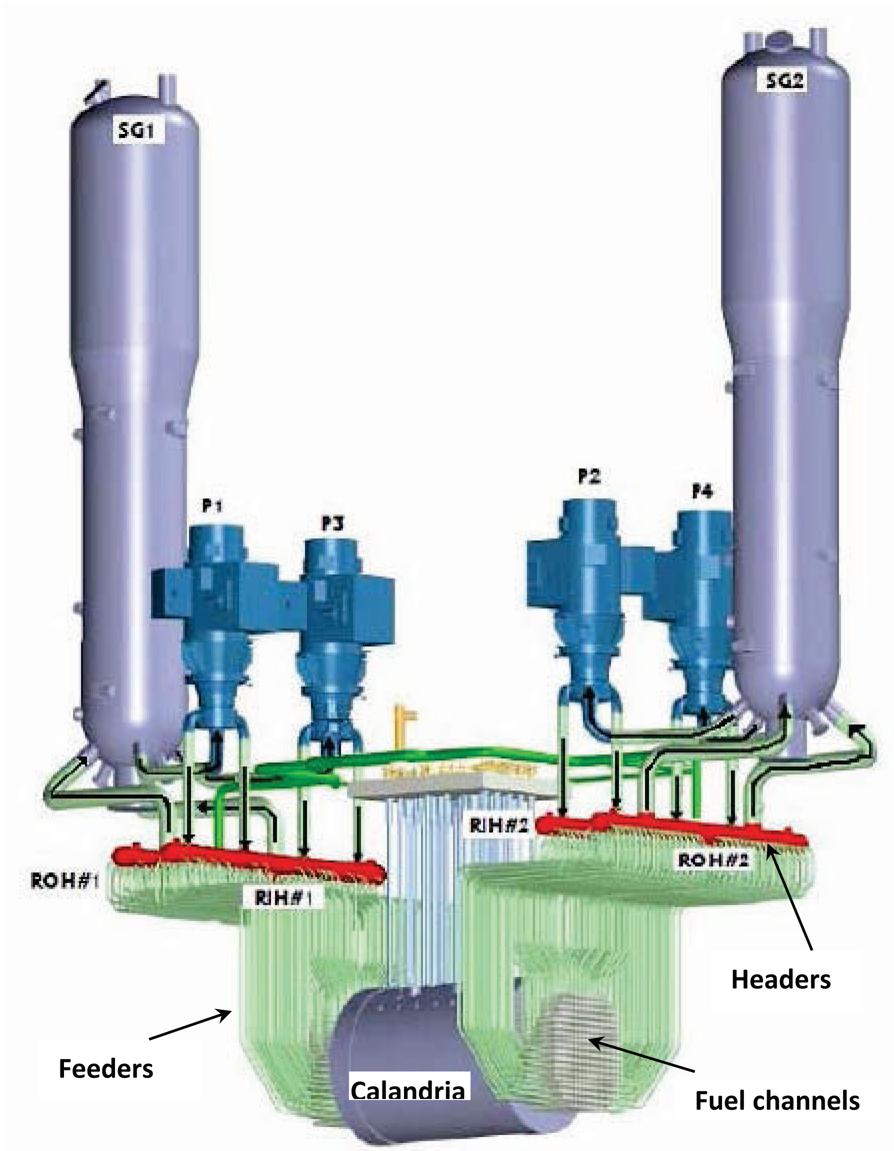

Figure 2. Illustration of CANDU Core Design Containing Multiple Parallel Fuel Channels

2. CRITICAL HEAT FLUX, DRYOUT POWER, CRITICAL CHANNEL POWER AND THEIR APPLICATIONS IN ROP/NOP ANALYSIS

The terms of critical heat flux (CHF), dryout power and CCP are frequently used in evaluation of CCP and associated uncertainty. While these terms are related to each other, they should be carefully distinguished in the context of evaluation of CCP and associated uncertainty.

Fuel Sheath Dryout Phenomenon

Prior to defining these terms, the fuel sheath dryout phenomenon in CANDU fuels is briefly described. During normal operation of a CANDU reactor, the fuel bundles are cooled effectively with single-phase liquid water thus the fuel sheath temperatures remain low. In an event leading to an increase in channel power, vapour generation starts, leading to the formation of bubbly flow, and with a further increase in power, the transition to annular flow regime. If the channel power continues to rise, CHF will eventually occur as a result of dryout of the liquid film on the sheath or when the liquid film flow approaches zero. This type of CHF is in general moderate, and poses no challenge to the sheath integrity. However, its accurate determination is essential, since it marks the start of the post-CHF heat transfer regime, where heated sheath temperatures can rise to unacceptable levels. Therefore, fuel sheath dryout is used as a demarcation point on which fuel safety margin can be based.

Critical Heat Flux

In CANDU design and safety analysis, CHF is the heat flux at which fuel sheath surface starts experiencing intermittent dryout. CHF is a local parameter and is influenced primarily by local thermalhydraulic parameters. However, fuel channel axial power profile (APP), or axial flux distribution (AFD), can have an important effect on its location and magnitude. It should be noted that departure from nucleate boiling (DNB) type CHF, which is typical of PWR designs, is highly unlikely to occur in CANDU reactors due to the CANDU fuel bundle design and operating conditions, notably the lower subcooling in CANDU fuel channels.

Dryout Power

Dryout power is the channel power at which the fuel bundle string starts experiencing fuel sheath dryout, where dryout is initiated anywhere in the fuel bundle string, for given thermalhydraulic conditions, such as pressure, differential pressure, flow, inlet temperature. Dryout powers are experimentally measured with full-scale bundle string simulators, normally with controlled inlet coolant temperature, outlet pressure, and coolant flow. In addition to these thermalhydraulic conditions, dryout powers are measured for different pre-designed AFDs along the fuel string simulator and for different diametral creep levels of the flow channel, as dryout powers are a strong function of AFD and channel creep. AFD affects dryout power in two ways: through the thermalhydraulic conditions along the channel, and through the CHF, as mentioned in the previous paragraph. Channel diametral creep results in a flow by-pass over the fuel string, reducing significantly the cooling effectiveness of the fuel bundles.

Critical Channel Power

Critical channel power is a specific channel dryout power, which is the result of a given reactor core condition and configuration where multiple fuel channels are connected in parallel to inlet and outlet headers. As reactor power changes (i.e., increases) during transients or accidents, flow re-distribution occurs resulting in thermalhydraulic conditions that favour a lower dryout power in a specific fuel channel. In CANDU reactor operation, channel flow is not a controlled parameter. During transients or accidents, channel flow varies with changes of reactor and channel powers (as illustrated in Figure 3), and with thermalhydraulic conditions at inlet and outlet headers. CCPs are simulated using sophisticated computer codes at the reactor core conditions, accounting for the interdependence of power and thermalhydraulic parameters. While the terms of dryout power and CCP have different connotations in general, they may be interchangeable under some specific situations, provided the associated thermalhydraulic conditions are clearly specified.

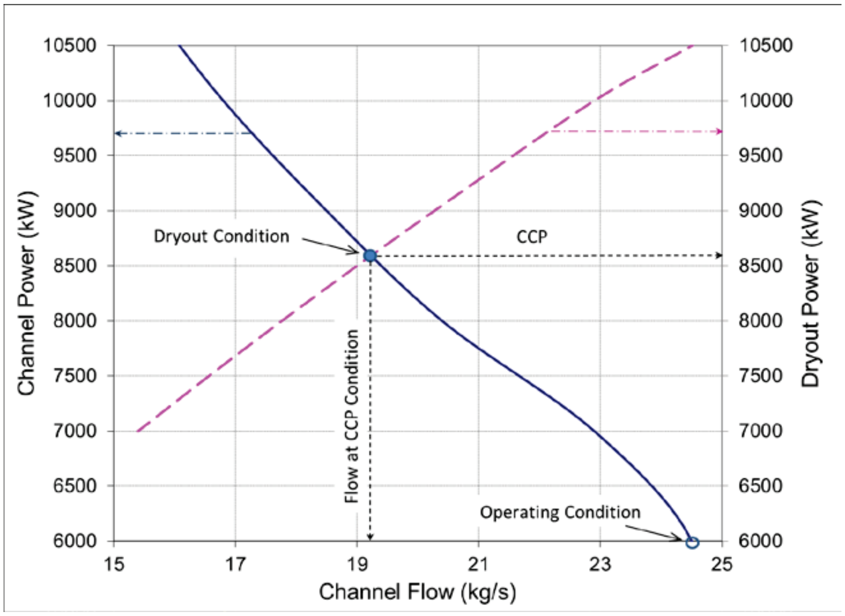

Figure 3. Relationship between Channel Power, Channel Dryout Power and Channel Flow

Figure 4 illustrates the process by which CCPs are simulated. First, the measured dryout powers, together with the thermalhydraulic conditions, creep information, AFD, and detected dryout location, are processed to obtain CHF data. Subsequently, an empirical CHF correlation, usually as a function of local pressure, mass flux, thermodynamic quality, and channel creep, and accounting for AFD effect, is developed based on a best-fit to the CHF data. This CHF correlation is assumed to be applicable at all thermalhydraulic conditions, channel creeps, and AFDs that may be encountered during the reactor transients/accidents. The correlation is then implemented in a thermalhydraulic computer code, to simulate CCPs for various AFDs, various creeps and various thermalhydraulic conditions of interest.

Typical CCPs are simulated in two steps.

- Simulation of system thermal hydraulic of the complete heat transport system circuit to determine thermalhydraulic conditions at the reactor headers and the neutronic reactor core response to map the three-dimensional power transients. In this step, fuel channels are grouped into various categories (depending on the expected similarity of their hydraulic conditions), where channel thermalhydraulic conditions are averaged for each group. The simulation is coupled with physics code simulation which calculates the three-dimensional power transient.
- Simulation of CCPs based on the predicted header conditions and channel power profiles. This uses an iteration process, where channel flow is first calculated for a given combination of header conditions, channel power profile, and an assumed channel power. Dryout power is then calculated based on the predicted channel flow. This iteration continues until convergence is achieved for channel flow and dryout power. The converged dryout powers are the CCPs (refer to Figure 3).

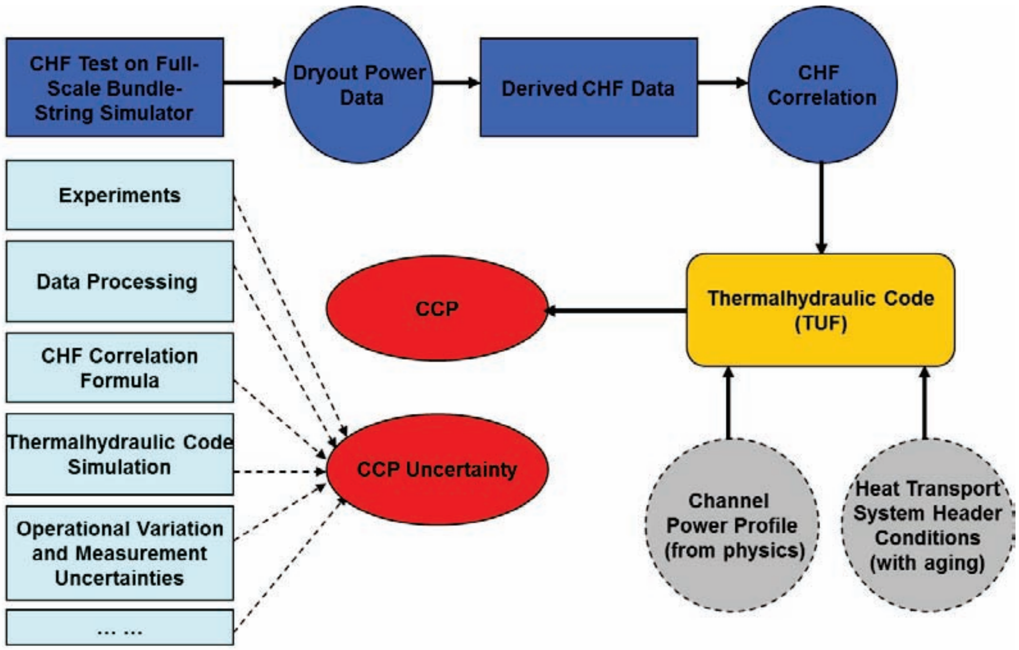

Figure 4. Process of CCP Simulations

In CANDU reactor design, hundreds of fuel channels are connected in parallel to the inlet and outlet headers. As a result, the header conditions are dominated by the system thermalhydraulics and the thermalhydraulics of all these fuel channels collectively. For a slow LOR event where regional power excursion is encountered, the overall header conditions are not expected to change appreciably during the event. Therefore, CCPs are practically calculated based on constant header conditions, i.e., constant inlet header temperature, outlet header pressure and header-to-header differential pressure. Small changes in header conditions due to the integral reactor power variation are considered through a power at trip (PAT) correction.

Simulation of CCPs is accompanied by uncertainty analysis. Among the various uncertainty sources, uncertainty in CHF is one of the major and direct contributors. Therefore, sensitivity of CCP to the change in CHF is always a point of interest.

3. EVALUATION OF CRITICAL CHANNEL POWER AND ASSOCIATED UNCERTAINTY

For a fuel channel with an AFD of $q(z)$, the channel power may be expressed as

$$Channel\ Power = P_h \int_0^L q(z)dz \tag{1}$$

where, P_h is the heated perimeter of fuel bundles and L is the channel heated length.

For the same fuel channel, CHF varies along the channel length, due to changes in the local thermalhydraulic parameters that affect it. When channel power changes, both $q(z)$ and $CHF(z)$ change their values correspondingly. While the shape of the $q(z)$ curve is preserved, its amplitude varies proportionally with the channel power. The channel power, at which the $q(z)$ curve becomes tangent to the CHF curve, is by definition the channel dryout power, DOP, which may be expressed as

$$DOP = P_h \int_0^L q^0(z)dz \tag{2}$$

where, $q^0(z) = q(z),\ when\ \min\left(\frac{CHF(z)}{q(z)}_{z=0\sim L}\right) = 1$, at a specific local z along the length of the channel.

Figure 5 illustrates the relation of $q(z)$ and $CHF(z)$ at initial dryout conditions. An iteration process is necessary to determine dryout power for a given set of thermalhydraulic boundary conditions, channel creep profile, and channel nominal AFD.

CCP is a specific channel dryout power for given reactor core configuration and thermalhydraulic boundary conditions. As illustrated in Figure 4, CCPs of CANDU fuel channels are simulated using complex computer codes at given boundary conditions at reactor inlet and outlet headers and for various axial power profiles considered in slow LOR events. The simulation of CCPs is subject to uncertainties of various sources, notably, the uncertainty in CHF correlation, the uncertainty in computer code simulation of channel flow, local pressure, and local quality (enthalpy), the uncertainty in pressure tube diametral creep, and the uncertainty in boundary condition at reactor inlet and outlet headers. This paper discusses only the contribution of uncertainty in CHF correlation to the uncertainty in CCP.

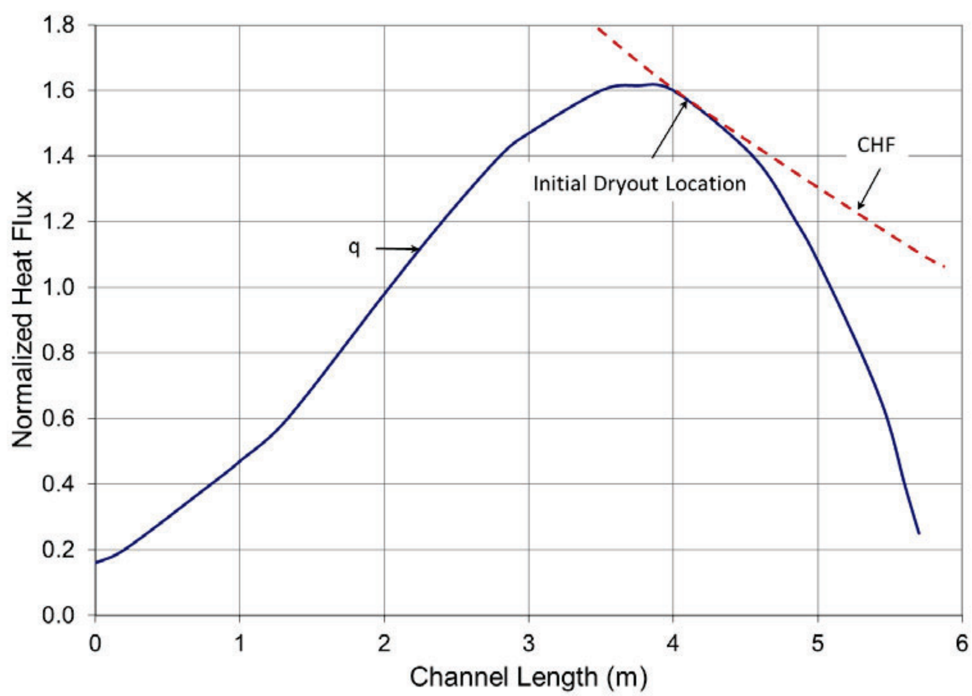

Figure 5. Relation of Heat Flux Distribution and CHF at Initial Dryout Conditions.

CHF Correlation Uncertainty

In CANDU fuel design and safety analysis, bundle-specific empirical CHF correlations are applied. These CHF correlations, normally expressed as a function of fuel channel flow, local pressure (averaged over the cross section), local quality (or enthalpy, averaged over the cross section), and pressure tube diametral creep, are simply a best fit to the CHF values derived from the dryout power data obtained with full-scale bundle-string simulators. As a result, the uncertainty in CHF correlation stems primarily from the uncertainty in dryout power measurement (as shown in Figure 4), the uncertainty in CHF correlation expression and regression, and the uncertainty in correlation interpolation. It can be quantified through experimental uncertainty analysis, statistical propagation, and engineering judgment, if necessary. This uncertainty term is independent of the safety analysis code used or the nuclear power plant to be analyzed. It should be noted that the CHF correlation is developed for use in one-dimensional computer codes, and is based on out-core CHF tests conducted at given cross-section-averaged thermalhydraulic conditions in a single fuel channel. No sub-channel code is used in the CHF correlation development.

Using experimental data to perform analyses and develop empirical models is a routine task in many engineering practices and scientific research activities. However, presenting results of measurements or results of predictions from empirically based models is not always properly done, due to the lack of or the incorrect statement of uncertainty that must accompany the results.

A proper uncertainty analysis must account for random, systematic and correlated uncertainties. Since measured variables can share identical systematic effects, correlated uncertainties can occur and must be considered in analyses. This is especially the case when the measurements of two or more variables are determined simultaneously in an experiment. The key to the proper estimation of the uncertainty associated with regression is a comprehensive accounting of all significant systematic effects and correlated systematic effects [1].

Systematic effects can be quantified and a correction or correction factor can be applied to compensate for the effect. It is assumed that, after correction, the expected value of the error arising from a systematic effect is zero. But the uncertainty of a correction applied to a measurement result to compensate for a systematic effect is *not* the systematic error in the measurement. It is instead a measure of the uncertainty of the result due to incomplete knowledge of the required value of the correction [2].

To determine the exact systematic effect in a measurement, it would be necessary to compare the true value and measurements. This is not always possible. However, every effort must be made to identify and account for all systematic effects. If there is no source of data to estimate systematic effects, the estimate must be based on judgement or experience.

In most cases a desired quantity Y is not measured directly, but is determined from N other quantities X_1, X_2,…, X_N through a functional form f:

$$Y = f(X_1, X_2, \dots X_N) \tag{3}$$

The input quantities X_1, X_2,…, X_N may themselves be measured and may depend on other quantities, including corrections and correction factors for systematic effects.

The law of propagation of uncertainty is based on first-order Taylor series approximation of a function f [2] [3]:

$$u_c^2(y) = \sum_{i=1}^{N} \left(\frac{\partial f}{\partial x_i}\right)^2 u^2(x_i) + 2\sum_{i=1}^{N-1} \sum_{j=i+1}^{N} \frac{\partial f}{\partial x_i}\frac{\partial f}{x_j} u\left(x_i, x_j\right) \tag{4}$$

where u_c is the combined standard uncertainty, $u(x_i)$ the standard uncertainty of variable x_i, and the variables x_i and x_j are the estimates of X_i and X_j, and $u(x_i,x_j)$ is the estimated covariance associated with x_i and x_j.

The degree of correlation between x_i and x_j is characterized by estimated correlation coefficient

$$r\left(x_i, x_j\right) = \frac{u\left(x_i, x_j\right)}{u(x_i)u\left(x_j\right)} \tag{5}$$

Contribution of CHF Correlation Uncertainty to CCP Uncertainty

According to Equation (2) and Figure 5, *DOP* for a given channel creep profile and nominal AFD is seen as a function of CHF and q:

$$DOP = f(CHF, q) \tag{6}$$

where, $CHF = CHF(P, G, x)$, where P, G, x are local pressure, mass flux and thermodynamic quality, respectively.

Assuming all source uncertainties are free from known biases, the uncertainty in dryout power can be written as follows:

$$U_{DOP}^2 = \left(\frac{\partial DOP}{\partial q}\right)^2 U_q^2 + \left(\frac{\partial DOP}{\partial CHF}\right)^2 U_{CHF}^2 + \left(\frac{\partial DOP}{\partial P}\right)^2 U_P^2 + \left(\frac{\partial DOP}{\partial G}\right)^2 U_G^2 + \left(\frac{\partial DOP}{\partial x}\right)^2 U_x^2 \tag{7}$$

where all parameters are at dryout conditions. $U_q, U_{CHF}, U_P, U_G, and\ U_x$ are the standard uncertainties in local heat flux, CHF, pressure, mass flux and thermodynamic quality, respectively.

While, $U_q, U_P, U_G, and\ U_x$ are plant, event, and computer code specific, U_{CHF} is associated solely with CHF correlation, which is independent of dryout-power simulation process.

Since at dryout conditions the q curve becomes tangent to the CHF curve, then

$$U_{DOP}^2 = \left(\frac{\partial DOP}{\partial CHF}\right)^2 \left(U_{CHF}^2 + U_q^2\right) + \left(\frac{\partial DOP}{\partial P}\right)^2 U_P^2 + \left(\frac{\partial DOP}{\partial G}\right)^2 U_G^2 + \left(\frac{\partial DOP}{\partial x}\right)^2 U_x^2 \tag{8}$$

or

$$U_{DOP}^2 = \left(\frac{\partial DOP}{\partial CHF}\right)^2 \left[\left(U_{CHF}^2 + U_q^2\right) + \left(\frac{\partial CHF}{\partial P}\right)^2 U_P^2 + \left(\frac{\partial CHF}{\partial G}\right)^2 U_G^2 + \left(\frac{\partial CHF}{\partial x}\right)^2 U_x^2\right] \tag{9}$$

The sensitivity terms of $\frac{\partial DOP}{\partial CHF}$, $\frac{\partial DOP}{\partial P}$, $\frac{\partial DOP}{\partial G}$ and $\frac{\partial DOP}{\partial x}$ can be determined through sensitivity studies with the aid of safety analysis codes. It should be noted that the value of $\frac{\partial DOP}{\partial CHF}$ differs with different thermal-hydraulic boundary conditions, and also differs with configuration of the channel including feeder pipes. Therefore, a specific value of $\frac{\partial DOP}{\partial CHF}$ should be obtained with specific thermalhydraulic conditions and specific channel configuration considered. As an example, for a specific CANDU channel (and feeder) design and using a one-dimensional computer code, $\frac{\partial DOP}{\partial CHF}$ is simulated about 0.75 (in relative term) if a constant channel flow is assumed, and is about 0.40 (in relative term) if a constant header-to-header differential pressure is assumed. The former resembles the case of computer code verification/validation against the dryout power measurements where channel flows are controlled. The latter is more representative of the CCP simulation, as discussed below. The difference in sensitivity between these two scenarios is due to the difference in thermalhydraulic boundary conditions assumed, which gives rise to different interdependence of channel dryout power on various thermalhydraulic parameters (such as channel flow).

During a plant transient, neither channel flow nor header-to-header differential pressure is constant. The header-to-header differential pressure is usually calculated through a system analysis. In the NOP/ROP-type analysis, an upset in regional channel power is assumed. Because of the regional nature of the event, the header-to-header differential pressure is not expected to vary significantly or is conservatively assumed not to change significantly. Therefore, the assumption of constant header-to-header differential pressure is more appropriate. The residual effect of small header condition change is accounted for through a Power at Trip (PAT) correction.

The above equations exhibit the relation of the uncertainty in dryout power with various uncertainty sources. It may be used to propagate source uncertainties to uncertainty in dryout power or CCP. It should be noted that the quantity of dryout power uncertainty depends on the channel/feeder configuration and the assumed thermalhydraulic boundary conditions. The uncertainty in dryout power prediction determined through code validation (by comparing with dryout power measurements at constant flows) cannot be directly applied to the uncertainty in CCP for a CANDU rector core where multiple channels are connected in parallel through feeders and flows are not controlled parameters.

Other uncertainty terms, U_q, U_p, U_G and U_x, take the sources of computer code simulation uncertainties and uncertainties in initial and boundary conditions such as measurement uncertainties and operational variations, etc., which should be quantified for specific computer code used and specific nuclear power plant analyzed.

4. REMARKS/CONCLUSIONS

CCP is an important figure-of-merit parameter in CANDU slow LOR safety analysis. Evaluation of CCP is a complex process involving experimental measurements, empirical model development and application of complex computer codes, and considering specific core configuration and power profiles. As a result, the evaluation of CCP is subject to uncertainties of various sources which must be properly quantified and accounted for.

Uncertainty in CHF is one of the major uncertainty sources contributing to the uncertainty in CCP. Quantification of uncertainty in CHF requires consolidation of uncertainties from dryout power measurement, empirical CHF correlation regression and interpolation / extrapolation of the correlation, and as well consideration of random variations, systematic biases and correlations between the uncertainty sources. The uncertainty in CHF, once quantified, can be propagated into uncertainty in CCP through sensitivity analysis, as noted in this paper.

The sensitivity of CCP to CHF is dependent on core configuration, power profiles, and thermalhydraulic conditions applied. One should be cautious about the dependency when estimating the CCP uncertainty due to CHF uncertainty. This paper presented one of possible ways to treating CCP uncertainty in CANDU safety analysis.

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