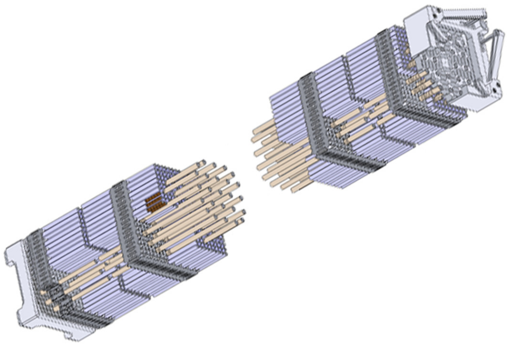
A New Concept for Critical Heat Flux Correlations for Safety Analysis of Pressurized-water Reactors



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The most important priority of nuclear power plant design is "safety." The accident at Fukushima Dai-ichi Nuclear Power Plant following the Great East Japan Earthquake has re-emphasized the importance of nuclear safety, requiring us to make a strong effort to improve safety. Mitsubishi Heavy Industries, Ltd. (MHI) has been working on improving the accuracy and reliability of safety design evaluations to ensure safer and more flexible operations of pressurized-water reactors (PWRs). In this context, MHI has developed and utilized a new concept for critical heat flux correlations.

1. Critical heat flux

A pressurized-water reactor (PWR) fuel consists of many fuel rods containing sintered uranium dioxide pellets sealed in zirconium alloy cladding. The fuel cladding provides an important barrier to prevent fission products in a fuel rod from being released to the outside. The fuel is cooled by high-pressure coolant (light water) at a temperature sufficiently lower than the boiling point. Therefore, it is unlikely that the fuel cladding will fail due to overheating unless a severe accident, such as a loss-of-coolant accident, occurs and results in core exposure. However, even with a sufficient amount of coolant in the reactor core, if a significant increase of local power occurs, a steam film can be formed by rapid steam generation on the fuel surface, which may cause fuel failure due to overheating. Such a phenomenon is known as "departure from nucleate boiling (DNB)." One of the most important objectives of PWR safety analysis is to prevent DNB.

Critical heat flux (CHF) is the local thermal power where DNB occurs. The value of the CHF depends on the pressure, temperature (*i.e.*, enthalpy), flow velocity of the coolant and fuel specification, especially the geometry and layout of the grid spacers, which hold the fuel rods in a square array. For the safety design of the nuclear plant, CHF is measured against various coolant conditions using the electrically heated test equipment that precisely simulates the actual fuel. Based on a database obtained from the test, empirical DNB correlations are derived and are utilized to predict CHF in the safety analysis.

2. Background and characteristics of new correlations

Mitsubishi Heavy Industries, Ltd. (MHI) has been working on improving the analysis technology for the core thermal-hydraulic behavior including DNB correlations. The enhancement of its accuracy revealed the margins involved in DNB predictions, and this has contributed to safely increasing the core operating flexibility, such as extensions of fuel burnup and/or the operating cycle length, as well as to improving the fuel economy. However, it also required the safety analysis method to accommodate a wider range of applications.

Safety analysis technologies have been improved based on those originally developed in the U.S. in 1970s and 1980s. MHI is replacing these conventional technologies including DNB correlations with independently developed expertise with higher level applicability based on the analysis experience and knowledge accumulated in MHI. These efforts are expected to contribute to the further development of safety analysis technologies.

(1) Structure of the new correlation system

The new correlation system developed by MHI consists of multiple correlations according to the fuel type and applicable conditions. In this stage, MHI has completed the development of the two basic correlations, the Mitsubishi generalized correlation for a standard grid (MG-S) and the Mitsubishi generalized correlation for a non-vane grid (MG-NV), while further development of the new correlations continues and will enhance the applicability of the system.

MG-S is the correlation applicable to MHI's standard design fuel. The allowable range of reactor operating conditions is defined using this correlation, and in the case of an abnormal event, the reactor will be shut down by the safety-protection system before it exceeds this predetermined operating limit. This correlation is also used in the safety analysis to confirm the validity of such safety systems.

The MG-NV correlation is used in the safety analysis of the special events, in which the reactor inadvertently restarts under a low-pressure and low-flow-rate condition after shutdown. The MG-NV was developed based on the low-pressure and low-flow-rate data, to which the MG-S correlation is not applicable. Since the data simulating PWR fuel under low-pressure and low-flow-rate conditions are limited, a conservative correlation was developed by referring to the test data for the fuel geometries using grid spacers without mixing vanes, in which DNB is more likely to occur than standard fuel using grid spacers with mixing vanes.

The new correlation set covers the entire range of coolant conditions assumed in the present PWR safety analyses. In addition, it provides reliable analysis even for low-pressure and low-flow-rate conditions, where only a highly conservative evaluation was applicable since the conventional correlations are not (**Figure 1**).

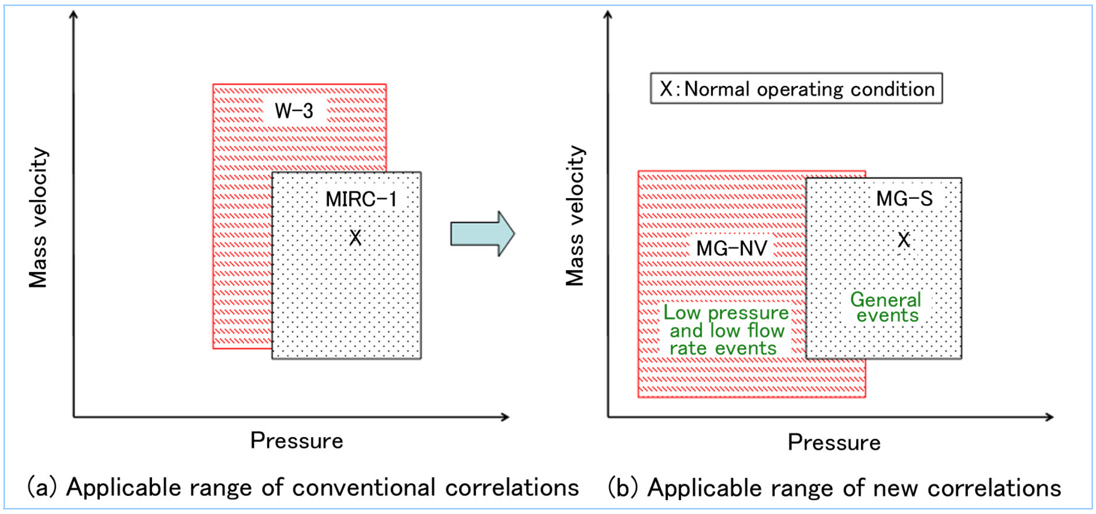

Figure 1 Applicable range of CHF prediction correlations
The current applicable range of the MG-S correlation is the same as that of the MIRC-1 correlation. The applicable range of the MG-NV correlation covers the region corresponding to the conventional W-3 correlation, while excluding unnecessary high pressures and high flow rates and including additional low pressures and low flow rates.

(2) Characteristics of the new correlations

The CHF generally decreases as the coolant enthalpy increases. Most of the existing correlations are based on a simple linear relationship between CHF and coolant thermal equilibrium quality (i.e., the non-dimensional expression of enthalpy) due to the insufficient data analysis technique at their origination in the 1970s and 1980s. Therefore, the applicability of these correlations is limited to a narrow range of thermal equilibrium quality. In creating the MG-S and MG-NV correlations, MHI analyzed the dependency of the CHF on the coolant conditions in detail, and developed a technique to simulate the CHF characteristics against thermal equilibrium quality with an appropriate curve (**Figure 2**).

Accordingly, the MG-S correlation is expected to be suitable for extending the database in the future. The MG-S correlation can flexibly accommodate fuel design improvements that are intended to enhance fuel performance to prevent DNB. Besides the flexibility for extending the database range, the MG-S correlation has a high degree of accuracy (i.e., 17% uncertainty), equivalent to conventional correlations (**Figure 3**). In addition, the MG-NV correlation covers low-flow-rate conditions, for which conventional correlations are not applicable due to their limited thermal equilibrium quality range. The MG-NV correlation has significantly improved the CHF prediction accuracy for low-pressure and low-flow-rate conditions (uncertainty: 30% => 19%) (**Figure 4**). These two new correlations are expected to make large contributions towards improving the reliability of the safety analysis.

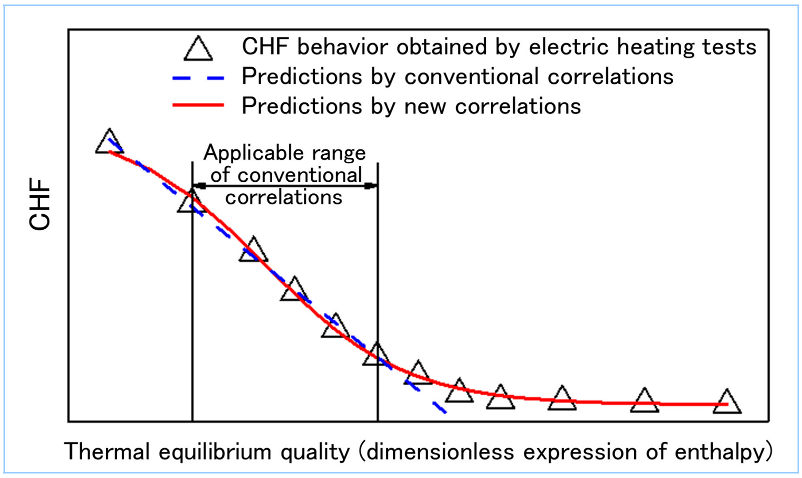

Figure 2 Characteristics of new DNB correlations
Conventional DNB correlations predict CHF only for the limited thermal equilibrium quality range. The new DNB correlations have been developed based on a flexible curve fitting technology to represent the CHF behavior against thermal equilibrium quality in a more appropriate manner.

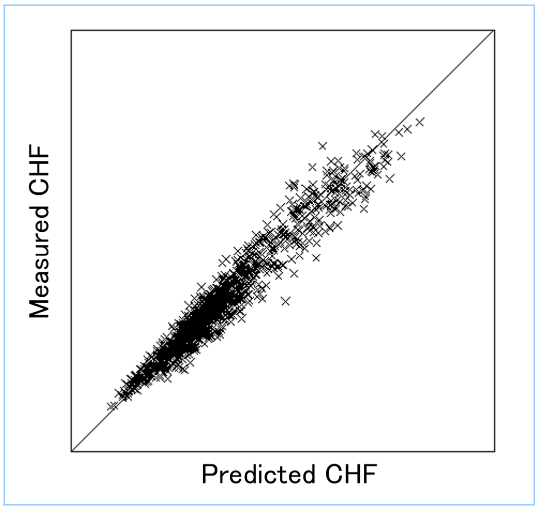

Figure 3 Predictions by the MG-S correlation
The predictions by the MG-S correlation agree well with test results. The uncertainties remain unchanged from those of conventional correlations.

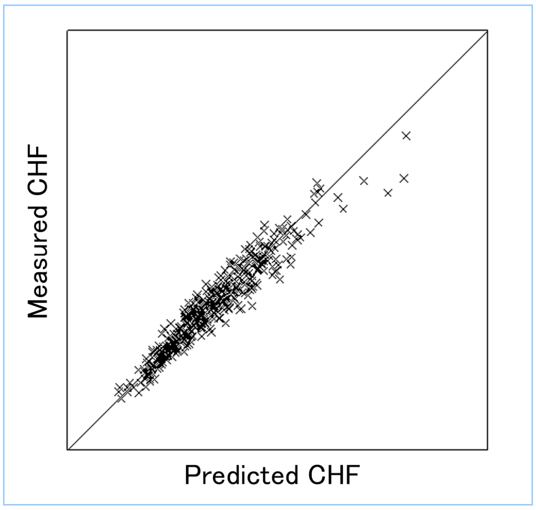

Figure 4 Predictions by the MG-NV correlation
The predictions by the MG-S correlation agree well with test results including low-pressure and low-flow-rate conditions. Uncertainties are significantly reduced in comparison with conventional correlations.

3. Future prospects

MHI is continuously conducting research and development to enhance the safety and the reliability of nuclear power plants through improved safety analysis. The CHF data acquired in this research will be incorporated into the database of the DNB correlation system. It is expected that the further extended database will result in a more reliable correlation system with a wider range of applicability.