# Thermal Design Methodology

**Revision 2**

**Non-Proprietary**

**May 2017**

# REVISION HISTORY

| Revision | Date | Page | Description |
|---|---|---|---|
| 0 | July 2012 | All | First Issue |
| 1 | November 2014 | All | Overall Missing or misused articles and prepositions<br>Detailed uncertainties for SCU<br>Additional analysis method for thermal margin evaluation |
| 2 | May 2017 | 8, 19 | Consideration of the approved CHF Topical report (APR1400-F-C-TR-12002-P-A) from USNRC |

# ABSTRACT

This technical report is prepared to summarize the applicable design computer codes and methodologies for the APR1400 DCD analyses. It could be submitted to the US NRC as a supplementary document to review the core thermal design methodology applied to the APR1400.

The Minimum DNBR (MDNBR) which measures the core thermal margin is predicted for the APR1400 core by the design computer codes. The TORC and CETOP codes are used to analyze the core thermal margin.

The TORC code is used in thermal design and safety analyses to perform detailed modeling of the core and the hot assembly and to determine MDNBR in the hot assembly. The CETOP code is a fast running tool used in thermal design and safety analyses to calculate MDNBR in the hot subchannel. While the TORC code can be applied directly to the thermal analysis and safety analyses, typically the TORC code is used to benchmark the MDNBR results of the CETOP code so that the CETOP results are conservative relative to those of TORC code.

The KCE-1 CHF correlation was developed by a non-linear multiple-regression analysis for the measured CHF data with local fluid conditions calculated by using the subchannel analysis code, TORC, for PLUS7 fuel. The KCE-1 CHF correlation is implemented to both TORC and CETOP codes. The application of the KCE-1 CHF correlation with TORC and CETOP codes is in full compliance with the conditions of the SER on the codes and modeling.

The TORC DNBR analysis is performed for the thermal design including the generation of DNBR SAFDL and the thermal margin model. DNBR SAFDL is determined statistically by combining uncertainties of system parameters at least a 95-percent probability with 95-percent confidence. The thermal margin analysis model is generated by adjusting the difference of the calculated DNBRs from the TORC and CETOP codes.

The design methodology and computer codes applied to the APR1400 DCD analyses are in full compliance with the conditions of the SER.

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# ACRONYMS AND ABBREVIATIONS

| | |
|---|---|
| AOO | anticipated operational occurrence |
| APR1400 | Advanced Power Reactor 1400 |
| CETOP | C-E thermal on-line program |
| CFR | Code of Federal Regulations |
| CHF | critical heat flux |
| COLSS | core operating limit supervisory system |
| CPCS | core protection calculator system |
| DCD | design control document |
| DNBR SAFDL | departure from nucleate boiling ratio specified acceptable fuel design limit |
| HTRF | heat transfer research facility |
| KEPCO NF | Korea Electric Power Corporation Nuclear Fuel company, Ltd. |
| KHNP | Korea Hydro & Nuclear Power Co., Ltd. |
| M/P | measured-to-predicted CHF ratio |
| PDF | probability distribution function |
| PWR | pressurized water reactor |
| SCU | statistical combination of uncertainties |
| SER | safety evaluation report |
| TORC | thermal hydraulic of a reactor core |
| US NRC | U.S. Nuclear Regulatory Commission |
| WEC | Westinghouse Electric Company |

# 1 INTRODUCTION

The purpose of this technical report is to present a comprehensive thermal design methodology utilized in the APR1400 DCD analyses.

The major design criterion for PWR core thermal analysis is to prevent a CHF or DNB from happening in the core during normal reactor operation and AOOs.

This technical report outlines the overall process used to perform the core thermal analysis and the features of the computational tools, TORC code (References 1 & 2) and CETOP code (Reference 3), which include the KCE-1 CHF correlation (Reference 4) that is used for the APR1400 core design and safety analysis.

Chapter 2 describes the design bases that are used for the core thermal analysis involved in the KEPCO NF methodology. Chapter 3 delineates the features of the TORC and CETOP computer codes approved by NRC and core modeling using these codes. Chapter 4 elaborates on the KCE-1 CHF correlation implemented in the TORC and CETOP codes for the purpose of DNBR calculation. Chapter 5 describes the SCU analysis method (Reference 5) used to determine DNBR SAFDL at least a 95-percent probability with 95-percent confidence. Chapter 6 describes the generation method of generic thermal margin analysis model for the thermal margin evaluation.

## 2 DESIGN BASES

The major design criterion for the thermal design is established to provide assurance that there be at least a 95-percent probability with 95-percent confidence that the hot fuel rod in the core does not experience a CHF during normal operation or AOOs. The limitation on CHF is expressed in the DNBR SAFDL. According to the NRC Regulations, Title 10, Chapter I, of CFR Appendix A to Part 50, AOOs mean those conditions of normal operation which are expected to occur one or more times during the life of the nuclear power unit.

# 3 DESIGN CODES AND MODELING

The MDNBR which serves as a measure for the core thermal margin, is predicted for the APR1400 by the core thermal hydraulic design codes. The TORC and CETOP codes are used to analyze the core thermal margin. Using the codes and core analysis models based on the core thermal design methodology, the MDNBR is predicted and the thermal margin is evaluated and confirmed.

## 3.1 Design Codes

### 3.1.1 TORC Code

The thermal margin analysis activity for the reactor core is performed using the TORC code, which is using an open channel analytical method based on the COBRA-IIIC code. A complete description of the TORC code and the application of the code for the detailed core thermal margin analyses are contained in CENPD-161-P-A (Reference 1). A brief description of this code and its use is given below.

The COBRA-IIIC code solves the conservation equations for mass, axial and lateral momentum, and energy for a collection of parallel flow channels that are hydraulically open to each other. Since the size of a channel in design varies from the size of a fuel assembly or more to the size of a subchannel within a fuel assembly, certain modifications are necessary to enable a realistic analysis of thermal-hydraulic conditions in all geometries. The principal revisions of the TORC code, which leave the basic structure of COBRA-IIIC unaltered, are in the following areas:

- Modification of the lateral momentum equation for core wide calculations where the smallest channel size is typically that of a fuel assembly.
- Addition of the capability for handling non-zero lateral boundary conditions on the periphery of a collection of parallel flow channels. This capability is particularly important in case of analyzing a group of subchannels within the hot fuel assembly.
- Addition of the capability to handle non-uniform core exit pressure distributions
- Insertion of standard ABB-CE empirical correlations and the ASME fluid property relationships

The details of the lateral momentum equations and the empirical correlations used in the TORC code are given in CENPD-161-P-A.

The verification of the TORC code is reported in CENPD-161-P-A and CENPD-206-P-A (Reference 2). The verification of the TORC code includes a comparison of subchannel coolant temperature rise and overall pressure drop for CHF test bundles, and actual operating core data.

The TORC code had been approved by the US NRC for use in a licensing application of reactor core analyses for steady-state calculations. The application should be limited to conditions of single phase flow or homogeneous two-phase flow (such as bubbly flow). When used for the analysis of flow blockage conditions, the blockage must be assumed to occur in the high power fuel assembly.

### 3.1.2 CETOP Code

The CETOP code, derived from the same theoretical bases as the TORC code, is streamlined for use in thermal margin analysis. The CETOP code, a variant of the TORC code, is used as a design code for the

APR1400 thermal margin analyses. A complete description of the CETOP code is provided in CEN-214(A)-P (Reference 3). A brief description of this code and its use is given below.

- The CETOP code has the same theoretical bases as the TORC code, but has been improved to reduce execution time.
- The CETOP code uses the transport coefficients to obtain accurate determination of diversion crossflow and turbulent mixing between adjoining channels with a less detailed calculation model. Details of the conservation equations and the empirical correlations used in the CETOP code are given in CEN-214(A)-P.
- Furthermore, a prediction-correction method is used to solve the conservation equations, replacing the iterative method used in the TORC code, and thereby reduce execution time.
- The conservatism of the CETOP code relative to the TORC code is assured by benchmarking analyses which demonstrate that the CETOP code yields accurate or conservative DNBR results relative to the TORC code.

The detailed core thermal margin calculations by the TORC code are used primarily to support the simplified design core thermal margin calculation code, CETOP. Verification of the CETOP code is reported in CEN-214(A)-P.

US NRC staff had reviewed the CETOP topical report. The review included the conservation equations, constitutive equations, transport coefficients, method of solutions, and the benchmark results compared to the TORC. The licensee had provided comparison between the CETOP and TORC results over the whole spectrum of operating conditions for ANO-2, Calvert Cliff Units 1 and 2, and San Onofre 2 and 3. In all cases, the CETOP calculated MDNBR lower than the TORC. Since the TORC code had been approved for use in CE thermal margin design, the staff concluded, based on the conservatism of the CETOP relative to the TORC, that the CETOP code was acceptable for ANO-2 thermal margin calculations. Based on their review, the acceptance of the CETOP carried the condition that the conservative hot assembly inlet flow factor was used for ANO-2 Cycle 2.

## 3.2 Core Modeling

### 3.2.1 TORC Modeling

The TORC code is an adaptation of the COBRA-IIIC code with modifications including an improved lateral momentum equation and lateral boundary condition capability to simulate actual core behaviors for all flow channels in the core. It is a detailed model code predicting the steady-state thermal hydraulic characteristics of nuclear reactor cores. The TORC code divides the core into a series of control volumes and solves three-dimensional conservation equations for each control volume. Thereby, it predicts fluid thermal hydraulic local conditions at every position in the core.

The application of the TORC code for detailed core thermal margin calculations typically involves two or three stages.

The first stage consists of calculating coolant conditions throughout the core on a coarse mesh basis. The core is modeled such that the smallest unit represented by a flow channel is a single fuel assembly. The three-dimensional power distribution in the core is superimposed on the core coolant inlet flow and temperature distributions. The core inlet flow and core exit static pressure distributions are obtained from flow model tests, and the inlet temperature for normal four-loop operation is assumed to be uniform. The axial distributions of flow and enthalpy in each fuel assembly are then calculated on the basis that the fuel assemblies are hydraulically open to each other. Also the transport quantities of mass, momentum and energy which cross the lateral boundaries of each flow channel are determined in this stage.

In the second stage, typically the hot assembly and adjoining fuel assemblies are modeled with a coarse mesh. The hot assembly is typically divided into four partial assembly regions. One of these regions is centered on the subchannels adjacent to the rod having the MDNBR. The three-dimensional power distribution is superimposed on the core coolant inlet flow and temperature distributions. The lateral transport of mass, momentum, and energy from the results of the first stage is imposed on the peripheral boundary enclosing the hot assembly and its neighbors. The axial distributions of flow and enthalpy in each channel are calculated as well as the transport quantities of mass, momentum, and energy which cross the lateral boundary of each flow channel. In some cases, the hot assembly detail normally included in the second stage is included in the first stage, thereby eliminating the need for this intermediate stage. In these cases, the second stage is the subchannel model discussed below.

The third stage involves a fine mesh modeling of the partial-assembly region which centers on the subchannels adjacent to the rod having the minimum DNBR. All of the flow channels used in this stage are hydraulically open to their neighbors. The output from the results of the second stage, in terms of the lateral transport of mass, momentum, and energy is imposed on the lateral boundaries of the partial assembly region in this stage. The engineering factors are applied to the adjacent rods of subchannel with the MDNBR to account for uncertainties on the enthalpy rise and heat flux due to manufacturing tolerances. The local coolant conditions are calculated for each flow channel. These coolant conditions are then input to the DNB correlation and the MDNBR is determined.

A more detailed description of this procedure with an example is contained in CENPD-161-P-A. This procedure is used to analyze in detail any specific three-dimensional power distribution superimposed on an explicit core inlet flow distribution. The detailed core thermal margin calculations are used primarily to support the simplified design core thermal margin calculation code, CETOP, discussed below.

### 3.2.2 CETOP Modeling

The CETOP code developed based on the TORC code is used as a design code for the APR1400 thermal margin analyses. The conservatism of the CETOP model relative to the TORC model is assured by benchmarking analyses which demonstrate that the CETOP model yields accurate or conservative DNBR results relative to the TORC model.

The conservation equations for mass, momentum and energy are derived in a control volume representing a flow channel of finite axial length. Two types of flow channels are considered in the representation of a reactor core: (1) averaged channels, characterized by averaged coolant conditions, and (2) lumped channels, in which boundary subchannels, contained within the main body of the channel, are used in the calculation of interactions with neighboring flow channels. An averaged channel is generally made of a large size and is located far from the subchannel at which the MDNBR occurs. With the help of boundary subchannels, a lumped channel describes in more detail the flow conditions near the MDNBR location, and is made of a small flow area.

The CETOP design model has a total of four thermal-hydraulic channels to model the open-core fluid phenomena. Channel 2 is a quadrant of the hottest assembly in the core and Channel 1 is an assembly which represents the remaining portion of the core. The boundary between Channels 1 and 2 is open for crossflow, but there is no turbulent mixing across the boundary. Turbulent mixing is only allowed within Channel 2. The outer boundaries of the total geometry are assumed to be impermeable and adiabatic. The lumped Channel 2 includes Channels 3 and 4. Channel 3 lumps the adjacent subchannels to Channel 4 at which the MDNBR occurs. The location of the MDNBR channel is determined from a detailed TORC analysis of a core. Table 3-1 shows the thermal hydraulic models for the CETOP code as well as the TORC code

Table 3-1. Thermal Hydraulic Models for TORC and CETOP Codes

| T-H Model or Correlation | Correlation or Constant |
|---|---|
| Single phase friction factor | TS |
| Two-phase friction factor multiplier | |
| Void Model | |
| Forced flow diversion | |
| Axial power distribution | |
| Crossflow resistance relationship | |
| Diversion crossflow resistance factor ($K_{ij}$) | |
| Turbulent momentum factor | |
| Traverse momentum parameter ($s/l$) | |
| Number of axial nodes | |
| Thermal conduction in the coolant | |
| Turbulent diffusion coefficient ($1/Pe^{(*)}$) | 0.013 (0.0035) for non-mixing vane grid,<br>0.038 (0.0101) for mixing vane grid |
| Two-phase flow model | TS |
| Inlet flow option | |

(*) Note : Pe = Pecklet Number

# 4 CHF CORRELATION

The KCE-1 CHF correlation (Reference 4) was developed by a non-linear multiple-regression analysis for the measured CHF data with local fluid conditions calculated by using the subchannel analysis code, TORC, for the PLUS7 fuel.

## 4.1 KCE-1 CHF Correlation

To verify the thermal performance of the PLUS7 fuel, CHF tests were conducted at Columbia University's HTRF in New York and the KCE-1 CHF correlation was developed by using the measured CHF data. Two types of test sections, which simulate the PLUS7 fuel, were fabricated to conduct the CHF tests. Test section 101 was the thimble subchannel test section simulating the guide thimble tube and the flow channels around it. Test section 102 was the matrix subchannel test sections simulating a square flow channel surrounded by only four heater rods. The radial and axial power distributions for each test section had a non-uniform and a cosine shape with a peak of 1.475, respectively. Table 4-1 represents the characteristics of geometrical configuration of the CHF tests.

The local fluid conditions for the CHF test data were computed using the subchannel analysis code, TORC. The functional formula of the KCE-1 CHF correlation is identical to the CE-1 CHF correlation, and is given as follows:

$$q''_{CHF} = \frac{B_1 (d/d_m)^{B_2} \left[ (B_3 + B_4 P)(G/10^6)^{(B_5 + B_6 P)} - (G/10^6)\,\chi\, h_{fg} \right]}{(G/10^6)^{(B_7 P + B_8 (G/10^6))}} \cdot \left( \frac{1}{F_C} \right)$$

where,

| | |
|---|---|
| $q''_{CHF}$ | Predicted CHF, MBtu/hr-ft$^2$ |
| $P$ | Pressure, psia |
| $d$ | Equivalent heated diameter of subchannel of interest, inch |
| $d_m$ | Equivalent heated diameter of matrix subchannel, inch |
| $G$ | Local mass flux at the CHF location, lbm / hr-ft$^2$ |
| $x$ | Local quality at the CHF location |
| $h_{fg}$ | Latent heat of vaporization, Btu/lbm |
| $F_C$ | Tong's non-uniform axial power distribution correction factor |

The applicable ranges of parameters for KCE-1 CHF correlation are shown in Table 4-2.

Based on the Reference 4, the DNBR limit of KCE-1 CHF correlation was determined to be 1.124 using the statistical analysis at least a 95-percent probability with 95-percent confidence. The mean and the standard deviation of M/P were [ ]$^{TS}$ and [ ]$^{TS}$, respectively.

## 4.2 Applicability to Design Analysis

The KCE-1 CHF correlation applies to the evaluation of the thermal design for the PLUS7 fuel assembly of the APR1400 in accordance with the CHF or DNB acceptance criteria defined in Chapter 2. The KCE-1 CHF correlation is implemented to both the TORC and CETOP codes. The TORC code is used in thermal design and safety analyses to perform detailed modeling of the core and hot assembly and to determine the MDNBR in the hot assembly. The CETOP code is a fast running tool which is used in thermal design and safety analyses to calculate the minimum DNBR in the hot subchannel. Thus, the evaluation of the thermal design is performed by predicting DNBR with the TORC and CETOP codes.

Consequently, the TORC and CETOP codes are valid with the application of the KCE-1 CHF correlation, and the application of the KCE-1 CHF correlation to the CETOP code for the APR1400 is equivalent to its application to the TORC code.

Table 4-1. Geometrical Configurations of the CHF Tests

| Test Section No. | Bundle Array | Rod Diameter [*in*] | Rod Pitch [*in*] | Heated Length [*in*] | Grid Spacing [*in*] | Guide Thimble | Guide Thimble Diameter [*in*] | Axial Power Distribution |
|---|---|---|---|---|---|---|---|---|
| 101 | 6x6 | 0.374 | 0.506 | 150.0 | 15.7 | Yes | 0.980 | 1.475 cosine |
| 102 | 6x6 | 0.374 | 0.506 | 150.0 | 15.7 | No | N/A | 1.475 cosine |

Table 4-2. Applicable Ranges of Parameters for KCE-1 CHF Correlation

| Parameter | Range |
|---|---|
| System Pressure | 1750 ~ 2415 *psia* |
| Local Mass Flux | 0.85 ~ 3.15 *Mlbm/hr-ft*$^2$ |
| Local Quality | -0.150 ~ 0.275 |

# 5 STATISTICAL THERMAL MARGIN METHODOLOGY

The statistical thermal margin describes the method used to statistically combine the system parameter uncertainties in the thermal margin analyses for the APR1400. A detailed description of the uncertainty probability distributions and response surface techniques is presented in this chapter. Also, this chapter demonstrates that there is at least a 95-percent probability with 95-percent confidence that the limiting fuel pin will avoid DNB so long as the MDNBR found with the best estimate design thermal margin analysis model remains at or above DNBR SAFDL.

## 5.1 Concept of SCU

SCU methods address uncertainties in both system and state parameters. The system parameters describe the physical system such as the reactor geometry, pin-by-pin radial power distribution, inlet and exit flow boundary condition, etc. These are not monitored in detail during reactor operation. The state parameters are monitored while the reactor is in operation and include the core average inlet temperature, primary loop flow rate, primary loop pressure, etc. The system parameter uncertainties are combined to yield DNBR SAFDL. The use of DNBR SAFDL with the best estimate thermal margin model provides at least a 95% probability with 95% confidence that the limiting fuel pin would not experience DNB. The state parameters are used to generate the COLSS/CPCS addressable constants mentioned in the COLSS/CPCS design.

## 5.2 Sources of Uncertainties

The following types of uncertainties are identified in the MDNBR predictions from the TORC code :

- Inlet flow factor for the limiting assembly
- Inlet flow factors for the limiting assembly's adjacent three assemblies
- Clad outer diameter
- Clad pitch
- Engineering heat flux factor
- Engineering enthalpy rise factor
- KCE-1 CHF correlation
- TORC code

Inlet flow factors are used in detailed TORC analysis. Ratios of the local to the core average mass velocities are input for every flow channel in the core-wide analysis. But, only inlet flows to the limiting assembly and those assemblies which are immediately adjacent to it are included in this method because MDNBR in the limiting assembly is unaffected by changes in the inlet flow of assemblies which are diagonally adjacent to the limiting assembly.

The means and standard deviations of the inlet flow factors for the limiting assembly and its adjacent assemblies can be calculated at the 95 % confidence level by the following equations.

$$\mu_{1-\alpha} = \bar{x} - t_{f,\alpha} \cdot \frac{S_{test}}{\sqrt{N}}$$

$$\sigma_{1-\alpha} = \sqrt{\sigma^2_{test,1-\sigma}} = \sqrt{\frac{f \cdot S^2_{test}}{\chi^2_{f,1-\alpha}}}$$

where,

- $f$ : degree of freedom
- $\alpha$ : significance level
- $\bar{x}$ : sample mean
- $S_{test}$: sample standard deviation

Variations in the clad diameter and pitch change subchannel flow area and also change the local heat flux. Manufacturing tolerances on the fuel clad diameter and pitch allow for the possibility that they can systematically above nominal throughout the entire fuel assembly. Therefore, the standard deviation of the mean is used in SCU analysis to characterize the systematic clad outer diameter and pitch.

The uncertainties of the clad diameter and pitch are determined based on [ ]$^{TS}$.

The standard deviation of the mean is used in SCU analysis to characterize the systematic pitch and clad diameter uncertainties. The standard deviation of the mean is given by $\sigma/\sqrt{N}$, where N is the number of measurement.

To calculate the standard deviations of the means for the clad diameter and pitch, the tolerance limits of the [ ]$^{TS}$ are assumed to be [ ]$^{TS}$. And, the number of measurement is assumed as that of each parameter. Thus, the standard deviations of the mean for these parameters are determined as fol

[ ]$^{TS}$

The engineering heat flux factor is used to take into account the effect on local heat flux of the deviation from nominal design and specifications that occur in fabrication of the fuel.

The factors related to fabrication uncertainty for the calculation of the engineering heat flux factor are as follows;

- Pellet Outer Diameter, $D_p$
- $UO_2$ Density, $\rho_p$
- Enrichment, $\eta_p$
- Fuel rod outer diameter, $D_r$

The heat flux on the surface of fuel rod can be expressed as follows;

$$q" = f(D_p, \rho_p, \eta_p, D_r) = C \cdot \frac{D_p^2 \cdot \rho_p \cdot \eta_p}{D_r}$$

where, $C$ : constant

The standard deviations of the engineering heat flux factor can be expressed as follows;

$$\sigma_{heat\ flux\ factor} = \sqrt{4\left(\frac{\sigma_{D_p}}{D_p}\right)^2 + \left(\frac{\sigma_{\rho_p}}{\rho_p}\right)^2 + \left(\frac{\sigma_{\eta_p}}{\eta_p}\right)^2 + \left(\frac{\sigma_{D_r}}{D_r}\right)^2}$$

where, $\sigma_{D_r}$ : standard deviation of the fuel rod outer diameter

The engineering enthalpy rise factor accounts for the effects of manufacturing deviations in fuel fabrication from nominal dimensions and specifications on the enthalpy rise in the subchannel adjacent to the rod with the minimum DNBR.

Since the enthalpy rise factor is an average of the hot channel rods along their entire length, the factor cannot exceed the local linear heat rate factor. The standard deviation of the local linear heat rate factor is [ ]TS. Because the local linear heat rate factor is the upper limit of the enthalpy rise factor, the uncertainty of the engineering enthalpy rise factor is [ ]TS.

The KCE-1 CHF correlation is used in the TORC code to determine whether DNB occurs or not. This correlation is based on [ ]TS from 6x6 rod bundle with [ ]TS test sections. The mean and standard deviation of M/P of the KCE-1 CHF correlation are [ ]TS, respectively. Therefore, the CHF correlation uncertainty may be characterized by a normal distribution with a mean of [ ]TS and standard deviation of [ ]TS, yields the DNBR limit of 1.124, as described in Section 4.1.

The uncertainty of the KCE-1 CHF correlation for the PLUS7 fuel is used together. The mean and standard deviation ($S_D$) of the KCE-1 correlation are as follows;

[ ]TS

$$\sigma_{1-\alpha} = \sqrt{\frac{f \cdot S^2}{\chi^2_{f,1-\alpha}}}$$

where, $\sigma_{1-\alpha}$ : standard deviation of parent population at $\alpha$ significant level

$f$ : degree of freedom, $f = n - 1$

$\chi^2_{f,1-\alpha}$ : chi square distribution, $x^2_{f,1-\alpha} = f \times \left(1 - \frac{2}{9f} + z_p\sqrt{\frac{2}{9f}}\right)^3$

The standard deviation at 95% confidence is calculated as follows;

$\alpha$ = 0.05

$z_p$ = -1.645 (f > 100)

[ ]TS

The TORC code represents an approximate solution to the conservation equations of mass, momentum, and energy. Simplifying assumptions are made, and the experimental correlations are used to arrive at the algorithms contained in the TORC code. Hence, the code associated with an inherent calculation uncertainty is combined statistically with the standard deviation of the response surface to assess the effect of code uncertainties on the DNBR limit.

The uncertainties of the system parameters for the most limiting assembly and code/CHF correlation are calculated based on [ ]TS and the generic values of PLUS7 fuel and the CHF test results as shown in Table 5-1.

## 5.3 MDNBR Response Surface

A response surface is a functional relationship which involves several independent variables and one dependent variable. The surface is created by fitting the constant of an assumed functional relationship to data obtained from “experiments”. The response surface provides a convenient means by which accurate estimates of a complex or unknown function’s response can be obtained. Since the response surface is a relatively simple expression, it can be applied to the analytic techniques where more complex functions would make an analytic solution intractable.

In the present application, a single detailed TORC analysis is treated as an “experiment”. A carefully selected set of detailed TORC “experiment” is conducted, and a functional relationship is fitted to the minimum DNBR results. An orthogonal central composite experiment design is used to generate the response surface. The total number of experiments to generate the response surface using this experiment design is

$$\textit{No. of TORC cases} = 2^k + 2k + 1$$

where, k is the number of independent variables considered. The results of these experiments may then be manipulated by means of the least square estimator

$$\bar{b} = (\eta' \eta)^{-1} (\eta')\bar{z}$$

where, $\bar{z}$ is the vector of experimental results to yield the coefficients, which defines the response surface

$$\begin{aligned} z &= MDNBR_{RS} \\ &= b_0 + \sum_{i=1}^{k} b_i \eta_i + \sum_{i=1}^{k} b_{ii} (\eta_i^2 - c) + \sum_{i=1}^{k} \sum_{j=1(i<j)}^{k} b_{ij} \eta_i \eta_j \end{aligned}$$

where,

$MDNBR_{RS}$ MDNBR calculated by the response surface model

$\eta_i = (x_i - \alpha_i)/\beta_i$ , coded value of the system parameter $(x_i)$ to be treated in the response surface

$\alpha_i$ is chosen such that $\eta_i = 0$ at nominal conditions

$\beta_i$ is chosen such that the range of the response surface includes $2\sigma$ range of each of the system parameters

$b_i$ response surface coefficient found from TORC results by means of the least square estimator

$c$ a constant determined from the number of experiments conducted ($c = (2^k + 2\alpha^2)/(2^k + 2k + 1)$)

## 5.4 Combination of PDFs

The response surface described in Section 5.3 is used in conjunction with Monte Carlo techniques to combine PDFs for each independent variable into a resultant DNBR PDF.

The effect of system parameter uncertainties on DNBR is combined with the effect of uncertainty in the CHF correlation by computing a $\Delta\mathrm{DNBR}$ caused by deviation of the system parameter from nominal:

$$\Delta DNBR = DNBR_{R.S.} - DNBR_{NOM}$$

where, $DNBR_{R.S.}$ is a calculated DNBR by the response surface model and $DNBR_{NOM}$ is the DNBR calculated on the nominal system parameters by the TORC code. A point is randomly chosen from the CHF correlation PDF and combined with the $\Delta DNBR$ to yield a DNBR value :

$$DNBR = DNBR_{CHF} + \Delta DNBR$$

This process is repeated for 20,000 randomly selected sets of the system parameters and points from the CHF correlation PDF, and then a resultant DNBR PDF is generated.

## 5.5 Application

The statistically derived DNBR PDF is applied to design analyses. Since the mean and standard deviation of the DNBR PDF are calculated from the finite number, they are converted to the mean and standard deviation for an infinite number.

The statistical DNBR limit derived in Section 5.4 contains no allowance for the adverse impact on DNBR of fuel rod bowing and any penalty, if it is needed. Finally, the DNBR SAFDL is determined at least a 95-percent probability with 95-percent confidence by considering fuel rod bowing and any penalty as follows;

$$\text{DNBR SAFDL} = \mu + 1.645\ \sigma$$

Thus, the DNBR SAFDL contains allowance for uncertainties in the CHF correlation and system parameters as well as a rod bow and any penalties and is generated to 1.29 for the APR1400 thermal margin and safety analysis.

This methodology using the statistical combination of uncertainties is same to the methodology (Reference 5) of WEC which is approved by the NRC. Therefore, it is applicable to the APR1400 thermal design.

Table 5-1. Uncertainties of Parameters

| Parameters | Mean (μ) @ 95% Confidence Level | Standard Deviation (σ) @ 95% Confidence Level | |
|---|---|---|---|
| Hot channel inlet flow | | | TS |
| Flow adj. to hot channel (A ass'y) | | | |
| Flow adj. to hot channel (B ass'y) | | | |
| Flow adj. to hot channel (lumped of C ass'y+D ass'y) | | | |
| Clad outer diameter [*inch*] | | | |
| Clad pitch [*inch*] | | | |
| Engineering enthalpy rise factor | | | |
| Engineering heat flux factor | | | |

# 6 THERMAL MARGIN ANALYSIS MODEL

## 6.1 Concept of Thermal Margin Analysis Model

The thermal margin analysis model (CETOP model) calculates the DNBRs at the operating condition obtained by the on-line monitoring system. Generally, the CETOP model is benchmarked to calculate the DNBR conservatively, by tuning the inlet flow factor of the CETOP code input data, because the CETOP code calculates DNBR faster than the TORC code, but less accurately. This inlet flow factor accounts for the deviations in minimum DNBR due to model simplification. Both the TORC and CETOP codes utilize the same conservation equations of mass, momentum and energy. But, the CETOP code uses transport coefficients, serving as weighting factors, for the faster-running and accurate prediction in treatments of crossflow and turbulent mixing between adjoining channels. And, the CETOP code retains the empirical correlations and the steam table which are included in the TORC code to maintain the consistency of the analysis. In the CETOP code, the following correlations are used;

- Fluid properties in the ASME steam table
- Heat transfer coefficient correlation of Dittus-Boelter and Jens-Lottes
- Single-phase friction factor given by Blasius form
- Sher-Green and Modified Martinelli-Nelson correlations for two-phase friction factor multiplier
- Modified Martinelli-nelson correlation for calculating void fraction
- Spacer grid loss coefficient correlation
- Correlation for turbulent interchange
- Hetsroni cross flow correlation
- KCE-1 CHF correlation developed with the CHF test

## 6.2 Comparison Between TORC and CETOP Predictions

The limiting in the CETOP model means the lowest delta DNBR, defined by the following equation,

$$Delta\ DNBR = DNBR_{TORC} - DNBR_{CETOP}$$

where, $DNBR_{TORC}$ and $DNBR_{CETOP}$ are calculated, by using the TORC and the CETOP codes respectively, at the same heat flux. The value of the delta DNBR should be more than ‘zero’ at the most limiting condition. If the value is less than ‘zero’, then, it should be converted into an overpower penalty as follows;

$$Overpower\ \ penalty = \frac{q''_{TORC}}{q''_{CETOP}}\ at\ the\ same\ DNBR$$

### 6.3 Generic CETOP Model

For the generation of the generic CETOP model, the operating condition sets are selected within the applicable operating range. The input base decks of the TORC and the CETOP codes are made for the limiting assembly. If the location of the MDNBR calculated by TORC is around the center guide tube, the CETOP model is generated on the bases of pin power distribution of an 8x8 array including center guide tube. And if the location of the minimum DNBR calculated by TORC is around corner guide tube, the CETOP model is generated on the bases of the pin power distribution of an 8x8 array including the corner guide tube. The CETOP and TORC codes are run at all operating condition sets. The CETOP code searches the heat flux at the DNBR SAFDL. The TORC code calculates DNBRs with the corresponding heat flux. The inlet flow factor in the CETOP input base deck is adjusted so that the delta DNBRs are slightly greater than zero (not negative) at all operating conditions for the limiting assembly.

The generic CETOP model is determined by cross-checking each CETOP model that described above. For the verification of CETOP models, the TORC input base decks are generated on the bases of the TORC models and the CETOP input base decks are produced to modify ‘address 58’ input value (FR) in the CETOP model generated above. The ‘address 58’ input value is "Maximum rod radial peaking factor wanted for channel 2" and should be changed to be the same with the corresponding TORC input which is based on the maximum pin power in the radial power distribution of nuclear design data. If a CETOP model identifies that its delta DNBR is more than zero for the others, it is determined as a generic CETOP model.

### 6.4 Application

The conservatism of the generic CETOP model should be satisfied about the entire DNBR range with the operating conditions. The range of DNBR analysis is determined as follows ;

$$\mu_{0.95} - 2 \times \sigma_{0.95} \leq DNBR$$

where, $\mu_{0.95}$ : Mean of DNBR SAFDL at 95% confidence

$\sigma_{0.95}$ : Standard deviation of DNBR SAFDL at 95% confidence

The CETOP code searches the heat flux at the low DNBR of the equation above and the TORC code calculates the DNBR with the corresponding heat flux. If the conservatism of the generic CETOP model are violated, the loss of the conservatism can be compensated by using the overpower penalty described in Section 6.2. Thus, the generic CETOP model may be applied to the following cycle if the configuration of fuel assembly and the DNB correlation are not changed.

This methodology for the evaluation of the thermal margin model is same to the methodology (Reference 3) of WEC which is approved by the US NRC.

The application of the CETOP model is evaluated by comparing its MDNBR predictions for the APR1400 reactor core with those obtained from a TORC analysis. A constant inlet flow factor of 0.62 is determined for 4 pumps operation so that MDNBR results predicted by the CETOP model are either conservative or accurate. The used operating conditions and power distributions are same to those of the TORC analysis. To compensate the non-conservatism of the CETOP model, the overpower penalty is applied to the following DNBR range;

| DNBR Range | Overpower penalty |
|---|---|
| DNBR < 0.95 | 0.98 |
| 0.95 ≤ DNBR < 1.22 | 0.99 |
| 1.22 ≤ DNBR | 1.00 |

Therefore, it is applicable to the thermal design for the APR1400 reactor core.

# 7 CONCLUSION

The thermal design methodology is established to provide assurance that there be at least a 95-percent probability with 95-percent confidence that the hot fuel rod in the core does not experience a DNB during normal operation or AOOs.

The TORC and CETOP computer codes approved by the NRC and the core modeling using these codes are used for the APR 1400 core thermal analyses. The TORC code is an open channel analytical method based on the COBRA-IIIC code and the CETOP code is derived from the same theoretical bases as the TORC code.

The KCE-1 CHF correlation was developed by a non-linear multiple-regression analysis for the measured CHF data with local fluid conditions calculated by using the subchannel analysis code, TORC for the PLUS7 fuel. Thus, it is applied to the evaluation of the thermal design for the PLUS7 fuel assembly and the APR1400 and prediction of DNBR.

The statistical thermal margin method is used to statistically combine system parameter uncertainties in the thermal margin analyses. The DNBR SAFDL of 1.29 is determined based on the uncertainty probability distributions of system parameters, response surface techniques, and Monte Carlo techniques at least a 95-percent probability with 95-percent confidence.

The generic thermal margin analysis model (CETOP model) calculating the DNBRs at the operating condition obtained by the on-line monitoring system is generated by tuning the inlet flow factor of the CETOP code input data based on the DNBRs calculated with the TORC and CETOP codes. If the conservatism of the generic CETOP model is violated, the loss of the conservatism can be compensated by using the overpower penalty. The application of the generic CETOP model is confirmed through the thermal margin analysis.

The design methodology and computer codes applied to the APR1400 DCD analyses are in full compliance with the conditions of the SER.

# 8 REFERENCES

1. Combustion Engineering, Inc., CENPD-161-P-A (proprietary), CENPD-161-NP-A (non-proprietary), "TORC Code: A Computer Code for Determining the Thermal Margin of a Reactor Core," April 1986.

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