Thermal Design Methodology

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ABSTRACT

This technical report is prepared to summarize the applicable design computer codes and methodologies for Advanced Power Reactor 1400 (APR 1400) DCD analyses. It could be submitted to United States Nuclear Regulatory Commission (USNRC) as a supplementary document to review the core thermal design methodology applied to APR1400.

The minimum value for the departure from nucleate boiling ratio (MDNBR) which measure the core thermal margin, is predicted for an APR1400 core by the design computer codes. The TORC (Thermal hydraulics Of a Reactor Core) code and the CETOP code 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 hot assembly and to determine MDNBR in the hot assembly. The CETOP code is a fast running tool which is used in thermal design and safety analyses to calculate MDNBR in the hot subchannel. While the TORC code can be applied directly in the thermal analysis and safety analyses, typically the TORC code is used to benchmark the MDNBR results of CETOP code such that the CETOP results are conservative relative to those of TORC code.

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 Safety Evaluation Report (SER) on the codes and modeling.

TORC DNBR analysis is performed for the thermal design including the generations of Departure from Nuclear Boiling Ratio Specified Acceptable Fuel Design Limit (DNBR SAFDL) and thermal margin model. DNBR SAFDL is determined statistically combining uncertainties of system parameter at least a 95-percent probability with a 95-percent confidence level. Thermal margin model is generated adjusting the difference of calculated DNBRs from TORC and CETOP codes.

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

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1. INTRODUCTION

The objective of this topical report is to present a comprehensive thermal design methodology utilized in Advanced Power Reactor 1400 (APR1400) DCD analyses.

The major design criterion for Pressurized Water Reactor (PWR) core thermal analysis is to prevent a Critical Heat Flux (CHF or DNB, Departure from Nucleate Boiling) from happening in the core during normal reactor operation and Anticipated Operational Occurrences (AOOs).

This topical report will outline the overall process used to perform the core thermal analysis and the features of the computer tools, TORC code (Ref. 1 & 2) and CETOP code (Ref. 3), which includes the KCE-1 CHF correlation (Ref. 4) that will be used for APR1400 core design and safety analysis.

Chapter 2 of the report describes the design bases that are used for the core thermal analyses process 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 DNBR calculation purpose. Chapter 5 describes Statistical Combination of Uncertainties (SCU) analysis method (Ref. 5) used to determine Specified Acceptable Fuel Design Limit (SAFDL) of DNBR (DNBR limit) at least a 95-percent probability with a 95-percent confidence level. Chapter 6 describes the method of generation of generic thermal margin model for the thermal margin evaluation.

2. DESIGN BASES

The major design criterion for the thermal design was established to provide assurance that there be at least a 95-percent probability at 95-percent confidence level 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 Code of Federal Regulations (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 minimum value for the departure from nucleate boiling ratio (MDNBR) which serves as a measure for the core thermal margin, is predicted for an APR1400 reactor by the core thermal hydraulic design codes. The TORC (Thermal hydraulics Of a Reactor Core) code and the CETOP code are used to analyze the core thermal margin. Using the codes and core analysis models based on the core thermal design methodology, the minimum DNBR 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[1]. A brief description of this code and its use is given here.

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 were necessary to enable a realistic analysis of thermal-hydraulic conditions in all geometries. The principal revisions to arrive at 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 when 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.

Details of the lateral momentum equations and the empirical correlations used in the TORC code are given in CENPD-161-P-A[1].

Verification of the TORC code is reported in CENPD-161-P-A[1] and CENPD-206-P-A[2]. The verification of TORC code includes a comparison of subchannel coolant temperature rise and overall pressure drop for CHF test bundles, and full size open core effects of actual operating reactor data.

The TORC code had been approved by USNRC for use in 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 TORC, is streamlined for use in thermal margin analysis. The CETOP code, a variant of the TORC code, is used as a design code for APR1400 thermal margin analyses. A complete description of CETOP is provided in CEN-214(A)-P[3]. A brief description of this code and its use is given here.

- The CETOP has the same theoretical bases as TORC, 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[3].

- 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 CETOP relative to TORC is assured by benchmarking analyses which demonstrate that CETOP yields accurate or conservative DNBR results relative to TORC.

The detailed core thermal margin calculations by TORC 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[3].

USNRC staff had reviewed the CETOP topical report. The review included the conservation equations, constitutive equations, transport coefficients, method of solutions, and the benchmark result compared to 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 minimum DNBR lower than the TORC calculations. Since the TORC code had been approved for use in CE thermal margin design, the staff concludes, based on the conservatism of CETOP relative to TORC, that the CETOP code was acceptable for ANO-2 thermal margin calculations. Based on their review, the acceptance of CETOP carried the condition that the conservative hot assembly inlet flow factor or a smaller value be 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 the nuclear reactor cores. The TORC code divides the core into a series of control volumes and solves 3-dimensional conservation equations for each control volume thereby predicting 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 discussed, and the inlet temperature for normal four-loop operation is assumed 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 determined during this stage are the transport quantities of mass, momentum and energy which cross the lateral boundaries of each flow channel.

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 to five partial assembly regions. One of these regions is centered on the subchannels adjacent to the rod having the minimum DNBR. 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 stage one calculations 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 the 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 stage two calculations, in terms of the lateral transport of mass, momentum, and energy is imposed on the lateral boundaries of the stage three

partial assembly region. Engineering factors are applied to the minimum DNBR rod and subchannel 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 minimum value of DNBR in the core is determined.

A more detailed description of this procedure with example is contained in CENPD-161-P-A[1]. 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 APR1400 thermal margin analyses. The conservatism of CETOP model relative to TORC model is assured by benchmarking analyses which demonstrate that CETOP model yields accurate or conservative DNBR results relative to 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 of relatively large size and is located far from the location at which MDNBR occurs. With the help of boundary sub-channels, a lumped channel describes in more detail the flow conditions near the MDNBR location, and is of relatively 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 average coolant conditions for 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 al1owed 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 subchannels adjacent to the MDNBR hot Channel 4. The location of the MDNBR channel is determined from a Detailed TORC analysis of a core.

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})	
The turbulent momentum factor	
The traverse momentum parameter (s/l)	
The number of axial nodes	
Thermal conduction in the coolant	
The turbulent diffusion coefficient (1/Pe)	0.013 (0.0035) for non-mixing vane grid, 0.038 (0.0101) for mixing vane grid
Two-phase flow model	TS
Inlet flow option	

(*) Note: Pe = Pecklet Number

4. CHF CORRELATION

KCE-1 CHF correlation (Ref. 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 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 Heat Transfer Research Facility (HTRF) in New York and the KCE-1 CHF correlation[4] 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. TS 101 was the thimble subchannel test section simulating the guide thimble tube and the flow channels around it. TS102 was the matrix subchannel test sections simulating a square flow channel surrounded by four heater rods only. The radial and axial power distributions for each test section were non-uniform and a cosine shape with a peak of 1.475. The characteristics of geometrical configuration of CHF test are as follows;

(unit : inch)

Test Section No.	Bundle Array	Rod Diameter	Rod Pitch	Heated Length	Grid Spacing	Guide Thimble	Guide Thimble Diameter	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

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

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

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 as follows;

Parameter	British Unit
System Pressure	1395 ~ 2415 *psia*
Local Mass Flux	0.85 ~ 3.15 *Mlbm/hr-ft*^{2}
Local Quality	-0.150 ~ 0.275

The KCE-1 CHF correlation DNBR limit was determined to 1.124 using the statistical analysis at least a 95-percent probability at 95-percent confidence level. The mean and the standard deviation of M/P are 0.9866 and 0.05304, 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 and the reactor core of APR1400 in accordance with the CHF or DNB acceptance criteria defined in Chapter 2. The KCE-1 CHF correlation is implemented to both 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 MDNBR in the hot assembly. The CETOP code is a fast running tool which is used in thermal design and safety analyses to calculate MDNBR in the hot subchannel. Thus, the evaluation of the thermal design is performed by predicting DNBR with TORC and CETOP codes.

The topical reports for both TORC code and CETOP code will remain valid with the application of KCE-1 CHF correlation, therefore, the application of KCE-1 CHF correlation with CETOP code for APR1400 is equivalent to its application with TORC code.

5. STATISTICAL THERMAL MARGIN METHODOLOGY

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

5.1 CONCEPT OF SCU

SCU methods address uncertainties in both system and state parameters. 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. 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. System parameter uncertainties are combined to yield DNBR SAFDL. Use of DNBR SAFDL with the best estimate thermal margin model will provide at least 95% probability with 95% confidence level that the limiting fuel pin would not experience DNB. State parameters are used to generate COLSS/CPCS addressable constants mentioned in COLSS/CPCS design.

5.2 SOURCES OF UNCERTAINTIES

Ten types of uncertainties are identified in 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 enthalpy rise factor
- Engineering heat flux factor
- KCE-1 CHF correlation
- TORC code

Inlet flow factors are used in detailed TORC analysis. Ratios of the local to core average mass velocity 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.

Variations in clad diameter and pitch change subchannel flow area and also change the local heat flux. Manufacturing tolerances on the fuel clad and pitch allow for the possibility that they will be systematically above nominal throughout and 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 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 MDNBR.

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

The KCE-1 CHF correlation is used in the TORC code to determine whether a DNB will occur. This correlation is based on 219 data points from 6x6 rod bundle with 150 inch long test sections. The mean and standard deviation of KCE-1 CHF correlation M/P is 0.9866 and 0.05304, respectively. Therefore, the CHF correlation uncertainty may be characterized by a normal distribution with a mean of 0.9866 and standard deviation of 0.05304, yields the 1.124 DNBR limit, as described in Section 4.1.

The TORC computer code represents an approximate solution to the conservation equations of mass, momentum, and energy. Simplifying assumptions were made, and experimental correlations were used to arrive at the algorithms contained in the TORC code. Hence, the code has 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.

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 may be obtained. Since the response surface is a relatively simple expression, it may be applied in 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 MDNBR 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

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

where, k is 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,

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

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

β_i is chosen such that the range of the response surface will include 2σ 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 probability distribution functions (pdf) for each of the independent variables 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:

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

where, $\mathrm{DNBR}_{\mathrm{R.S.}}$ is a calculated DNBR by the response surface model and $\mathrm{DNBR}_{\mathrm{NOM}}$ 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 $\Delta\mathrm{DNBR}$ to yield a DNBR value :

$$\mathrm{DNBR} = \mathrm{DNBR}_{\mathrm{CHF}} + \Delta\mathrm{DNBR}$$

This process is repeated for 20,000 randomly selected sets of system parameters and points from the CHF correlation pdf, and 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 were calculated from the finite number, they is converted to the mean and standard deviation for 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 needs. Finally DNBR SAFDL is determined at least a 95-percent probability at 95-percent confidence level by considering fuel rod bowing and any penalty as follows;

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

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

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

6. THERMAL MARGIN MODEL

6.1 CONCEPT OF THERMAL MARGIN MODEL

The thermal margin model (CETOP model) calculates the DNBRs at the operating condition obtained by the on-line monitoring system in nuclear power plant. 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.

6.2 COMPARISON BETWEEN TORC AND CETOP PREDICTED RESULTS

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

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

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 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, 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 TORC MDNBR is around center guide tube, the CETOP model will be generated on the bases of pin power distribution of 8x8 array including center guide tube. And if the location of TORC MDNBR is around corner guide tube, the CETOP model will be generated on the bases of pin power distribution of 8x8 array including corner guide tube. The CETOP and the 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 (address 81) 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 limiting assembly.

The generic CETOP model is determined by cross-checking each CETOP model that produced 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 which 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 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 ;

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

where, : Mean of DNBR SAFDL at 95% confidence level

: Standard deviation of DNBR SAFDL at 95% confidence level

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

This evaluation methodology of the thermal margin model is same to methodology[3] of WEC approved by USNRC. Therefore, it is applicable of APR1400 thermal design.

7. CONCLUSION

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

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

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. Thus, it was applied to the evaluation of the thermal design for the PLUS7 fuel assembly and the reactor core of APR1400 and predicted DNBR.

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

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

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

8. REFERENCES

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