REVISED RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

APR1400 Design Certification

Korea Electric Power Corporation / Korea Hydro & Nuclear Power Co., LTD

Docket No. 52-046

RAI No.:	**328-8422**
SRP Section:	**4.4 – Thermal and Hydraulic Design**
Application Section:	**4.4.6.1, also in 4.3 and 7.2**
Date of RAI Issue:	**12/07/2015**

Question No. 04.04-8

Title 10 of the Code of Federal Regulations, Part 52.47, "Contents of Applications; technical information" requires, in part, that the application must contain a level of design information sufficient to enable the Commission to judge the applicant's proposed means of assuring that construction conforms to the design and to reach a final conclusion on all safety questions associated with the design before the certification is granted. The information submitted for a design certification must include performance requirements and design information sufficiently detailed to permit the preparation of acceptance and inspection requirements by the NRC.

Topical Report CENPD-170, "CPC, Assessment of the Accuracy of PWR Safety System Actuation as Performed by the Core Protection Calculators," was provided to the staff for audit. The staff considers this report to contain design information necessary for the Commission to reach a safety finding on the design. This document contains necessary details on the intended function and operation of the Core Protection calculator, including processes used to develop and adjust system constants, which cannot be found in the DCD or the associated technical report APR1400-F-C-NR-14003-P, "Functional Design Requirements for a Core Protection Calculator System for APR1400." Staff seeks to clarify the regulatory standing of this document and ensure design basis commitments are clearly established. Please provide an explanation of the intended use of this document with respect to establishing the design basis of the plant and indicate where the relevant design information is contained in the DCD or reference the document appropriately to establish a clear design basis for the APR1400 Core Protection Calculator System.

Response

After the first publication of CENPD-170 in 1975, CPC Improvement Program has been developed and implemented by the CPC Oversight Committee, consisting of Arizona Nuclear Power Project(ANPP), Arkansas Power and Light Company(AP&L), Louisiana Power and Light Company(LP&L) and Southern California Edison(SCE), with Combustion Engineering as its technical consultants. The CPC Improvement Program is a program of CPC modifications and methodology improvements to reduce future reload efforts and to reduce unnecessary plant trips. The CPC and methodology changes for the CPC Improvement Program were approved by USNRC [Reference 1], and have implemented into several US nuclear plants such as SONGS-2 and PVNGS, and so on. The CPCS design for APR1400 comes from those of PVNGS and thus its functional design is based on those described in Reference 1.

In addition, since there were many modifications by CPC Improvement Program, several parts of CENPD-170 are not appropriate to refer for CPC design for APR1400. CENPD-170 is composed mainly of 3 major sections which are Power Distribution Synthesis, Thermal Margin Technique, and CPC Uncertainty Analysis. These 3 major sections were partly modified by the CPC Improvement Program, and those for APR1400 are described in DCD References as shown below.

CENPD-170 sections	DCD References
Power distribution synthesis	Section 4.3 'POWER Distribution Algorithm' in CPC FDR
Thermal margin technique	Section 4.4 'STATIC DNBR and POWER DENSITY' in CPC FDR
CPC uncertainty analysis	CPC Setpoint methodology TeR (APR1400-F-C-NR-14001-P)

The Technical Report of CPC Setpoint analysis methodology for APR1400 (APR1400-F-C-NR-14001-P) will be added to Reference of DCD.

Reference 1 "CPC and Methodology Changes for the CPC Improvement Program," CEN-310-P-A, April, 1986.

Supplemental Response – (Rev.1)

KHNP is submitting the supplemental response to RAI 328-8422 in order to provide the response to the CPCS audit issues held on January 18~22, 2016 as Attachment 1. In addition, APR1400-F-C-NR-14001-P, Rev.0, "CPC Setpoint Analysis Methodology for APR1400" will be revised to add the above content regarding DNBR uncertainty for mixed cores according to the response to item 3(Core Protection Calculator(CPCS) Issues) of the NRC Audit for DCD Chapter 4. This information is included in Attachment 2.

Impact on DCD

There is no impact on DCD.

Impact on PRA

There is no impact on PRA.

Impact on Technical Specifications

There is no impact on Technical Specifications.

Impact on Technical/Topical/Environmental Reports

Technical Report (APR1400-F-C-NR-14001) will be revised as shown in Attachment 2.

Response to NRC AUDIT for DCD Chapter 4.4
January 18~22, 2016

Question

Core Protection Calculator (CPCS) Issues:
The staff requests a summary presentation on the entire process, from initial cycle testing to development of the shape annealing and rod shadowing factors, to normal operational experience with the system for similar designs. Detailed issue descriptions are presented below:

1. DCD Section 4.3.3.1.1.4 describes the use of a fixed source MCNP (Monte Carlo, N-particle) code adjoint calculation to determine the shape annealing functions, while CENPD-170-P identifies the DOT/DORT codes for this purpose. What methodology is to be used for the APR1400 to synthesize the power shapes, and what deviations are taken from the CENPD-170-P methodology?

Response
APR1400 uses MCNP code instead of DOT/DORT for SAF (Shape Annealing Function) calculation.

DOT is a 2-dimentional radiation transport calculation code and MCNP is a 3-dimentional transport calculation code. For the SAF calculation, DOT code simulates the ex-core detector configuration as a simple annulus in the R-Z coordinates system while MCNP code simulates realistic geometry of the ex-core detector.

2. CENPD-170-P specifies a number of analytical methods and computer codes, so will these same codes be utilized for the APR1400, and if so, have these codes been reviewed and approved by the NRC?

Response

The codes described in Chapter 3 of CENPD-170 are not used in APR1400.

After the first publication of CENPD-170 in 1975, CPC Improvement Program has been developed and implemented by the CPC Oversight Committee, consisting of Arizona Nuclear Power Project (ANPP), Arkansas Power and Light Company(AP&L), Louisiana Power and Light Company(LP&L) and Southern California Edison(SCE), with Combustion Engineering as its technical consultants. The CPC Improvement Program is a program of CPC modifications and methodology improvements to reduce future reload efforts and to reduce unnecessary plant trips. The CPC and methodology changes for the CPC Improvement Program were approved by USNRC [Reference 1], and have implemented into several US nuclear plants such as SONGS-2 and PVNGS, and so on. The CPCS design for APR1400 comes from those of PVNGS and thus its functional design is based on those described in Reference 1.

In addition, since there were many modifications by CPC Improvement Program, several parts of CENPD-170 are not appropriate to refer for CPC design for APR1400. CENPD-170 is composed mainly of 3 major sections which are Power Distribution Synthesis, Thermal Margin Technique, and CPC Uncertainty Analysis. These 3 major sections were partly modified by the CPC Improvement Program, and those for APR1400 are described in DCD References as shown below.

CENPD-170 sections	DCD References
Power distribution synthesis	Section 4.3 ‘POWER Distribution Algorithm’ in CPC FDR
Thermal margin technique	Section 4.4 ‘STATIC DNBR and POWER DENSITY’ in CPC FDR
CPC uncertainty analysis	CPC Setpoint methodology TeR (APR1400-F-C-NR-14001-P)

The Technical Report of CPC Setpoint analysis methodology for APR1400 (APR1400-F-CNR-14001-P. ML15009A195) will be added to Reference of DCD.

Reference 1 “CPC and Methodology Changes for the CPC Improvement Program,” CEN-310-P-A, April, 1986.

Also see the response to RAI 328-8422 Q04.04-8.

3. APR1400-F-C-NR-14001-P, Rev.0, "CPC Setpoint Analysis Methodology for APR1400" states that the CPC calculations are verified using a large number of power distributions at BOC, MOC, and EOC, but it does not describe how the CPC constants (e.g., Fxy and Fq) are calculated, especially when new fuels are introduced (i.e., mixed cores). For mixed cores, how will multiple DNBR uncertainties be implemented in the CPC algorithms?

Response

The CPCS constant Fxy is installed as measured value after reactor startup test. The procedure to install Fxy in CPCS is described in the answer to Question 7. The Fq is calculated as the product of installed Fxy by measured Fz. The measured Fz is determined based on the on-line ex-core detector signals.

In the mixed cores, the DNBR uncertainty is calculated for two fuel types respectively. The larger value between the two DNBR uncertainties is selected and installed in CPC for conservatism. In addition, the DNBR penalty factor for the mixed cores resulting from the type of CHF correlation is added to the DNBR uncertainty. Because, in the CPC algorithm, there is only one type of CHF correlation, i.e. CE-1 type correlation, the correlation coefficient of the new fuel is installed in CPC in the mixed cores. Therefore, the DNBR thermal margin decrement of existing fuel compared to new fuel is considered as the DNBR penalty factor for the mixed cores. DNBR penalty factor is defined as the following equation in case the existing fuel reaches to DNBR limit faster than the new fuel.

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APR1400-F-C-NR-14001-P, Rev.0, "CPC Setpoint Analysis Methodology for APR1400" will be revised to add the above content regarding DNBR uncertainty for mixed cores as the Attachment 2.

4. The CPC power distribution uncertainties are evaluated based on core simulator 3D power distributions for a variety of conditions. Are design basis AOO events included in the database of power shapes?

Response

For the evaluation of power distribution uncertainties, about 4,800 power shapes are generated by the combination of changes in the burnups, power levels, CEA positions, and Xenon conditions, as follows.

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Although the power shapes are not generated for the specific design bases AOO events, the power shapes generated at these core conditions include all the possible shapes within the analysis range of CPC hot pin ASI (-0.6 ~ +0.6), where the highly peaked shapes could be obtained in cases of Xenon transient simulations. Therefore, it is judged that about 4,800 power shapes includes the power shapes which can occur during the AOO events.

5. Describe the reload process for setpoint calculations. Define what calculations are performed generically for each fuel type, and which are cycle-specific. Identify the codes used and provide a reference to the approval SER.

Response

1) Reload process for setpoint calculations.

The reload analyses of CPCS design are as follows;

- Determination and/or verification of CPCS database & Reload Data Block (RDB) constants
- Determination of CPCS overall uncertainty factors and related addressable constants
- Determination of other CPCS addressable constants
- CPCS RDB implementation and testing (if required)

CPCS analyses for each cycle are divided into two parts; non as-built and as-built analyses.

a. Non As-built Analyses
    - Digital Setpoint Core Simulation Analysis (DSCSA)
    - CEAC PF Multiplier and COOS DNBR Limit Lines Analysis
    - CPCS Database Analysis
    - CPCS Overall Uncertainty Analysis (OUA)
    - CPCS Addressable Constants Analysis (ACA)
b. As-built Analyses
    - CPCS Startup Test Data Analysis
    - CEFAST Database Constants Analysis

The purpose of DSCSA is to generate neutronics and thermal hydraulics information to be used in CPCS database preparation and CPCS uncertainty analyses.

CPCS database analysis is performed to revise or verify the implemented DNBR penalty factors, CEAC inoperable penalty factors and RDB constants.

CPCS overall uncertainty analysis (OUA) is performed to provide CPCS overall uncertainty constants. This analysis is to determine appropriate values for the DNBR and LPD uncertainties (BERRi values). In order to ensure that the DNBR and LPD calculations are conservative (at least 95/95 probability and confidence level) when all pertinent sources of inaccuracy are taken into account, five constants (BERR0~BERR4) are included in the CPC software to adjust certain outputs from essentially "best estimate" to "95/95" values.

CPCS addressable constant analysis (ACA) is performed to determine CPCS addressable constants, requirements, and notification to be transmitted to utility.

CPCS startup test data analysis is performed to provide the Fxy verification data to be used in CEFAST database analysis.

CEFAST database constants analysis is to generate CEFAST database file to be transmitted to the site for the Fast Power Ascension (FPA). The CEFAST (CE FAst STartup) code provides the capability to perform the rapid analysis of NSSS data obtained during the startup tests to generate and validate CPC constants.

2) Calculations performed for each fuel type, and for each cycle

All the calculations are performed every reload cycle, and these calculations are all cycle-specific. The non as-built analyses are performed for each fuel type in reload core.

3) Codes used, and their reference to the approval SER.

The ROCS code is used for reactor core simulator neutronics calculation for the CPCS Uncertainty Analysis. ROCS code was approved by USNRC in Reference 1.

The CETOP code calculates minimum DNBR (MDNBR) for a region of the core where MDNBR is likely to occur. CPCS uses a simplified CETOP model called CETOP-2. CETOP-2 is tuned to CETOP by adjusting the enthalpy coefficient. CETOP code was approved by USNRC in Reference 2.

The CPCSIM code is used to calculate the CPCS overall uncertainties for DNBR and LPD calculations. The CPCSIM code is based on CPC Functional Design Requirement and Modified Statistical Combination of Uncertainties (Reference 3) approved by USNRC.

Reference 1. CENPD-266-P-A (Proprietary), “The ROCS and DIT Computer Codes for Nuclear Design,” Combustion Engineering, Inc., April 1983.

Reference 2. CEN 214(A)-NP (Non-Proprietary), “CETOP-D Code Structure and Modelling Methods for Arkansas Nuclear One Unit 2,” Combustion Engineering, Inc., July 1982.

Reference 3. CEN-356(V)-P-A, “Modified Statistical Combination of Uncertainties,” Rev. 01-P-A, Combustion Engineering, Inc., May, 1988.

6. For the core average axial power distribution, CENPD-170 discusses conversion of ex-core detector responses to peripheral core power at three core rings and then to 20-node axial shape using up to eight algorithm constants (α1 through α8). These are apparently pre-calculated to represent flat-, saddle-, top-, or bottom-peaked axial shapes for beginning-of-life (BOL) and end-of-life (EOL) conditions. The CPC uses some degree of pattern recognition on the 3-ring axial power distribution to determine which of the four power shape types are present and then uses a cubic spline fit to data. The applicant should explain how the constants are developed, whether they are cycle-dependent or burnup-dependent within one cycle, and how they will be verified against plant data.

Response

It is written in the CENPD-170 that depending upon the time in life (BOL or EOL) or power shape type, CPC algorithm automatically selects one of two boundary point power correlation coefficients (BPPCC). One set of the BPPCC constants (α1-α4) is used in case of BOL chopped cosine shapes, and the other (α5-α8) is used in case of MOL flat shapes~~,~~ and EOL or saddle shapes.

However, only one set of BPPCC constants is used for the actual plants. CPC uses the measured BPPCC constants which are determined during the startup test at the site. The BPPCC constants are determined by using the least square fitting of startup test data, and are used for the whole cycle.

Therefore, the BPPCC constants are cycle-dependent, but they are not burnup-dependent. They are installed using the plant measured data, and thus, further verification is not needed.

7. For the radial power distribution, planar radial peaking factors and axial augmentation factors are used to define the hot pin power as a function of the CEA configuration (both normal and abnormal). No detail is provided on the method used to define these. The applicant should explain how the planar radial peaking factors will be calculated, whether cycle- or burnup dependent, what codes are used in the calculations, and whether the codes have been approved.

Response

The planar radial peaking factors (Fxy) installed in CPCS is the maximum Fxy which could be happened during one cycle. It means CPCS always use the most conservative Fxy for LPD and DNBR calculation during the cycle.

In order to estimate this maximum Fxy, the nuclear design code, ROCS is used to analyze one cycle's Fxy variation with burnup (refer to figure 1). The maximum calculated Fxy for the cycle (= Cycle Maximum $Fxy_{,BE}$ in figure 1) is initially installed in CPCS with additional penalty for reactor startup test. During the physics test at startup, the Fxy is measured at a specific burnup point, and this measured Fxy (=Fxy^{M} in figure 1) is compared to the calculated Fxy at the burnup ($Fxy_{,BE}$ in figure 1). If Fxy^{M} is greater than $Fxy_{,BE}$, the Cycle Maximum $Fxy_{,BE}$ adjusted by the ratio of $Fxy^{M}/Fxy_{,BE}$ is finally installed in CPCS, and otherwise the Cycle Maximum $Fxy_{,BE}$ is finally installed in CPCS

Therefore, the planar radial peaking factors installed in CPCS is cycle dependent, but it is independent of burnup. ROCS code was approved by USNRC in Reference 1.

Reference 1. CENPD-266-P-A (Proprietary), "The ROCS and DIT Computer Codes for Nuclear Design," Combustion Engineering, Inc., April 1983.

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Figure 1. Typical Fxy Variation for Reload Cores

8. Explain how the axial augmentation factors are calculated, whether cycle- or burnup-dependent, what codes are used, and whether the codes have been approved.

Response

In the current CPC algorithms, the axial augmentation factors described in the CENPD-170 were deleted because the 3-D power peaking factor increase due to fuel densification was demonstrated to be insignificant and USNRC approved this conclusion in Reference 1.

Reference 1. “CPC and Methodology Changes for the CPC Improvement Program,” CEN-310-P-A, April, 1986.

9. Explain whether the CPC constants will be adjusted during a cycle to reduce conservatism, and if so, how the changes are verified.

Response

The CPC constants related to power distribution are planar radial peaking factors (Fxy), rod shadowing factor (RSF), Shape annealing matrix (SAM), and boundary point power correlation constants (BPPCC); and the constants should be installed using the measured value during reactor startup test. For the reactor startup, these constants are pre-calculated and loaded into the CPC prior to the test. These pre-calculated values must include additional penalty to assure that the CPC remain conservative prior to the installation of the measured values. These additional penalties in the CPC constants are deleted after startup test by installing of the measured Fxy, RSF, SAM and BPPCC into the CPC.

The CPC constants related to power distribution are adjusted to reduce conservatism only after startup test, and they are not changed during a cycle.

10. Explain where in the safety analysis is the time delay that the core protection calculator (CPC) will utilize before reverting to the “predetermined penalty factor” (PF), as discussed in the response to RAI # 274-8277 (ML15363A340; Question# 07.01-37), that will be large value to ensure a core protection calculator (CPC) initiated departure from nucleate boiling ratio (DNBR) reactor trip and/or a local power density (LPD) reactor trip, explained, defined, and captured, within the design basis event’s “sequence of events.”

Response

The time delay is not considered in the DCD Ch. 15 safety analysis.

There is no design basis event in DCD Ch. 15 that assumes the occurrence of CEA misoperation concurrent with the both CEAC failure.

In a view point of safety analysis, the probability of CEA misoperation during the time delay (30 seconds) is rare and the resulting consequence of fuel failure and radiological dose could be limited to other postulated accidents in the DCD Ch. 15.

11. Based on the definition of a "pre-determined penalty factor" (PF) added to the APR1400 FSAR, Tier 2, Table 7.2-7, mark-ups (FSAR mark-up page# 7.2-71), as provided in the response to RAI #274-8277 (ML15363A340; Question# 07.01-38), explain how the predetermined PF will be guaranteed to be large enough during the entire fuel lifecycle to ensure that a CPC initiated reactor trip is reached.

Response

There were some inconsistencies in the terms.

	FDR	DCD Tier 2, TeR
DNBR and LPD penalty factors selected when both CEACs become inoperable due to CEACs failure or in-test	Pre-selected PF	Pre-assigned PF (DCD Tier 2) Pre-determined PF (TeR)

For consistency, DCD Tier 2 and the Safety I&C System Technical Report will be revised as follows:

- "Pre-assigned PF" and "Pre-determined PF" will be replaced with "Pre-selected PF."

The maximum PF for DNBR and LPD (PFMAXD & PFMAXL) are determined by the maximum value of static PF and Xenon PF transmitted from CEAC to CPC.

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The minimum power level used by the CPC is about 20%, thus, the effect of applying a PF of 8.0 is equivalent to a power level of 160%. The value of 8.0 is large enough to initiate a plant trip as described in Section 2.6.4 of CEN-310-P-A, "CPC and Methodology Changes for the CPC Improvement Program", approved by NRC as follows:

"… For all outward deviations, a penalty of 8.0 will be used if the deviation exceeds a preset maximum value. This will cause an immediate CPC based plant trip for any actual deviation not stopped by the CWP (CEA Withdrawal Prohibit). …"

Refer to responses of I&C Public meeting held on May 2 and 3, 2016.

Core Operating Limit Supervisory System (COLSS) Issues:

1. The staff SER for the System 80+ refers to topical reports CENPD-169 and CEN-312 for the detailed description of the COLSS. The staff reviewers would like to understand if these documents could be considered part of the design basis for the APR1400. If so, please make these documents available in the Electronic Reading Room.

Response

After the first publication of CENPD-169 in 1975, COLSS has been improved and changed by Combustion Engineering. For example, the DNBR calculation algorithm was "BULL" in CENPD-169, but it was changed to "CETOP" as in the current COLSS. Since there were many modifications, several parts of CENPD-169 are not appropriate to refer for COLSS design for APR1400.

The COLSS design for APR1400 comes from those of PVNGS and thus its functional design is based on those described in CEN-312.

CEN-312 and the COLSS functional design requirements (APR1400-F-C-NR-14002-P. ML15009A191) that provides the design bases and the detailed algorithm for COLSS is sufficient for reference. CEN-312 will be available in the Electronic Reading Room.

Core Design Issues:

1. Since the analysis method for large break LOCA differs from that approved for the System 80+ design (due to the use of RELAP5/MOD3.3), the staff would like to understand if the core design process for determining limiting pressure drops and component loads will differ from that described in DCD Section 4.4 (i.e., using only the TORC and CETOP codes).

Response

Refer to Response of RAI 326-8408 Q04.04-5

2. The staff notes that the source of the gadolinium isotopic data used by DIT/ROCS is not clearly identified in the approved code topical reports. To address this concern, the audit will seek to identify the source of the gadolinium isotopic data used by DIT/ROCS. If the source cannot be identified from existing documentation, this should be clearly stated and the DIT multigroup library data for the Gd isotopes should then be examined for consistency with data from later evaluated nuclear data files such as ENDF/B-V.

Response

The ENDF/B-IV data tapes, which were used for the nuclides on the DIT design library, including gadolinium isotopes, were originally received from National Nuclear Data Center, BNL and the DIT data library were used by the DIT/ROCS codes to perform reactor nuclear design for APR1400 DCD. Please refer Revised Response (Rev.1) to RAI 419-8517, Question 04.03-8.

3. The applicant presents axial core power distributions at various core depletion states assuming only unrodded operation. However, the APR1400 is designed to use regulating rods to control power shapes. The audit will seek clarification on whether the presented unrodded axial power shapes are meant to result in peak core power densities that bound those for cases where regulating rods are partially inserted in the core.

Response

The axial core power shapes presented in DCD are typical axial power distributions expected during normal unrodded base-loaded operation of APR1400 initial core. Depending on the operating margin to operation limits, the regulating rods could be used to control the axial power distributions. The peak core power density will be monitored by COLSS during operation.

4. Demonstration of detection of CEA misalignment or drop; the applicant states: “Since the plant protection system (CPCs and CEACs) detects the CEA positions by means of two independent sets of reed switches and uses this information in determining margin to trip, it is not necessary to rely on in-core or ex-core nuclear instrumentation to detect control element misalignment or drop. Thus, this testing is not performed.” The audit is to examine if the applicant’s justifications are acceptable for not performing
(1) demonstrated CEA positions and misalignment tests and (2) demonstration of the capability of the incore neutron flux instrumentation to detect rod misalignment equal to or less than the Technical Specifications limits for control rod misalignment.

Response

CPCS must use safety grade inputs since CPCS is a protection system. But in-core detector is not a safety grade instrumentation. So, CPCS uses RSPT (Reed Switch Position Transmitter, DCD 7.2.1.1) to detect CEA misalignment. Incore neutron flux instrumentation is not used to detect CEA misalignment.

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The core power measurement uncertainty factor for the LPD calculation (BERR4) is obtained by selecting the largest of the CPC thermal power error ($X_{CA}+X_{TF}$) or the CPC neutron flux power errors ($X_{CA}+X_{NF}+X_{TF}$) over the core power range from 0-130% full power. For the LPD calculation, the CPC selects the largest of the thermal power or the neutron flux power. Next, the uncertainty constant (BERR4) and the power level dependent error (X_{SC}) are applied as an additive bias to the selected power.

2.2.4 Axial Shape Index Uncertainty

The axial shape index (ASI) for the core average and the hot-pin power distributions is computed from the power generated in the lower and upper halves of the core:

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The ASI error is defined by:

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The core average and hot-pin ASI uncertainty analyses are performed by comparing the CPC synthesized ASI and the reactor core simulator ASI. The resulting error distributions are analyzed to obtain the upper and lower 95/95 tolerance limits. The hot-pin ASI and the core average ASI uncertainties performed for SKN Unit 3&4 cycle 1 are presented in Tables 2-3 and 2-4.

2.2.5 DNBR Uncertainty for Mixed Cores

In the mixed cores, the DNBR uncertainty is calculated for two fuel types respectively. The larger value between the two DNBR uncertainties is selected and installed in CPC for conservatism. In addition, the DNBR penalty factor for the mixed cores resulting from the type of CHF correlation is added to the DNBR uncertainty. Because, in the CPC algorithm, there is only one type of CHF correlation, i.e. CE-1 type correlation, the correlation coefficient of the new fuel is installed in CPC in the mixed cores. Therefore, the DNBR thermal margin decrement of existing fuel compared to new fuel is considered as the DNBR penalty factor for the mixed cores. DNBR penalty factor is defined as the following equation in case the existing fuel reaches to DNBR limit faster than the new fuel.

$$\text{DNBR Penalty Factor} = (\text{Limiting Heat Flux})_{\text{existing fuel}} / (\text{Limiting Heat Flux})_{\text{new fuel}}$$

where, Limiting Heat Flux is the heat flux corresponding to DNBR limit.