ADVANCED TOPICAL REPORT SAFETY EVALUATION (ATRSE)

BY THE OFFICE OF NEW REACTORS OF

TOPICAL REPORT APR1400-F-C-TR-12002-P, REVISION 0,

"KCE-1 CRITICAL HEAT FLUX CORRELATION FOR PLUS7 THERMAL DESIGN"

KOREA HYDRO & NUCLEAR POWER CO., LTD.

PROJECT NO. PROJ0782

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1.0 INTRODUCTION

By letter dated January 7, 2013 (Ref. 1), Korea Hydro and Nuclear Power Co., Ltd. (KHNP) in conjunction with its affiliate company Korea Electric Power Corporation (KEPCO), submitted Topical Report (TR) Advanced Power Reactor 1400 (APR1400)-F-C-TR-12002-P, Revision 0, "KCE-1 Critical Heat Flux Correlation for Plus7 Thermal Design," (Ref. 2) to the U.S. Nuclear Regulatory Commission (NRC) for review and approval, in support of its application for design certification of the APR1400 reactor design. The purpose of this topical report is to justify the use of the KCE-1 Critical Heat Flux (CHF) correlation for PLUS7 fuel design for the pressurized water reactor (PWR) application. The TR presented the test data analysis and results for the KCE-1 CHF correlation development, and a description of the CHF test facility and test procedures. The KCE-1 CHF correlation can be applied to the thermal design and plant safety analyses for the PLUS7 fuel design within the approved range of operating parameters. The applicant applied the KCE-1 CHF correlation with the Thermal Hydraulics of a Reactor Core (TORC) subchannel computer code to perform the analyses and results presented in the TR.

The NRC staff started the review of the TR in August 2013, and issued a non-public proprietary request for additional information (RAI), RAI 3-7443 (Ref. 3), with 18 questions regarding the applicant's analyses, computer codes, test procedures, assumptions, and uncertainties to support its safety review of the TR. A follow-up public meeting was held with the applicant on May 1, 2014 (Ref. 4), at the NRC offices in Rockville, MD, to discuss various proprietary and non-proprietary issues raised by the questions in RAI 3-7443. The non-proprietary meeting presentation made by the applicant is publicly available in the Agencywide Documents Access and Management System (ADAMS) (Ref. 5). The applicant submitted its responses to various RAI 3-7443 questions as Reference 6 (Questions 2, 3, and 5); Reference 7 (Questions 4, 10, 11, 12, and 18); and Reference 8 (Questions 1, 6, 7, 8, 9, 13, 14, 15, 16, and 17). The staff held numerous clarification public teleconferences throughout this process, but there were several issues that could not be resolved. In January 2015, the staff conducted a two-day regulatory audit (Ref. 9) to resolve the outstanding issues regarding the applicant's RAI responses, and to establish the qualification status of Columbia University's Heat Transfer Research Facility (HTRF), where the CHF tests were conducted. The audit allowed the staff to review the applicant's data, calculations, and supporting documents to gain an in-depth understanding of the TR as well as the applicant's responses to RAI 3-7443. The staff summarized the overall audit findings in an audit report (Ref. 10). As one of the fundamental outcomes of the audit, the applicant updated and resubmitted its response to RAI 3-7443 (Ref. 11) in March 2015, which did not address several concerns iterated by the staff during the audit. For reasons explained later, the applicant's response to RAI 3-7443, Questions 6, 7, 9, and 17, were still not acceptable to the staff. The staff therefore conducted another public meeting with the applicant on September 3, 2015 (Ref. 12), where the applicant presented supplemental information regarding the KCE-1 CHF correlation to resolve the outstanding technical issues related to these four RAIs. Following the commitments made in the September 3, 2015, public meeting, the applicant revised and resubmitted its responses to the four open RAI questions (RAI 3-7443 Questions 6, 7, 9, and 17) (Ref. 13) in October 2015, to incorporate the supplemental information discussed during the meeting.

This safety evaluation report (SER) documents the NRC staff's review and findings regarding the TR, APR1400-F-C-TR-12002-P, Revision 0. Where appropriate, the staff discussed the response to the RAI 3-7443 questions in this SER. The aspects of the responses to the RAI questions not discussed in this SER were for the NRC staff's information or clarification and were found adequate. Based on its review of the TR, the NRC staff finds that the use of the KCE-1 CHF correlation is acceptable in calculating the CHF for the PLUS7 fuel design, provided that the conditions and limitations specified in Section 5.0 of this SER are met.

2.0 SUMMARY OF THE TOPICAL REPORT

The applicant had the CHF tests conducted for PLUS7 fuel at the HTRF at Columbia University in New York, NY. The purpose of the testing was to collect data to develop an applicable CHF correlation for the PLUS7 fuel design. The PLUS7 fuel incorporates an advanced "R" mixing vane grid design. The split mixing vanes attached to the top of the grid strap are intended to improve the heat transfer between the coolant and fuel rods. The PLUS7 fuel CHF tests were performed with two test sections simulating configurations with and without a guide thimble tube. The TR uses the "TS101" designation for the thimble subchannel test section and "TS102" for the matrix subchannel test section without a guide thimble tube, as illustrated by Figures 2-3, "Mid-Grid of Thimble Subchannel Test Section TS101," and 2-4, "Mid-Grid of Matrix Subchannel Test Section TS102," in the TR, respectively. Each test section was composed of a 6×6 heater rod bundle with a fixed heated length of 381 cm (150 in.) and a fixed grid span of 39.9 cm (15.7 in.), which simulated the PLUS7 fuel geometry. All tests were performed with a non-uniform (chopped cosine) axial power distribution and a radial power split of approximately [ ]PROP between hot and cold rods.

The TR describes the CHF tests that the HTRF conducted to support the KCE-1 CHF correlation development. The functional formula of the KCE-1 CHF correlation is identical to the Westinghouse CE-1 CHF correlation. The coefficients of the KCE-1 CHF correlation were determined by a non-linear multiple-regression analysis of the measured CHF data with local fluid conditions in the test sections calculated by using the Westinghouse subchannel thermal-hydraulic analysis code TORC (Ref. 14). The KCE-1 CHF correlation departure from nucleate boiling ratio (DNBR) limit was determined with a 95 percent probability and at a 95 percent confidence level (95/95 DNBR limit). The CHF test data and the statistical methods applied to the correlation development and validation are described in appendices to the TR. The KCE-1 CHF correlation can be applied to the thermal design and plant safety analyses involving PLUS7 fuel.

3.0 REGULATORY BASIS

General Design Criterion (GDC) 10, "Reactor Design," in Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, Appendix A "General Design Criteria for Nuclear Power Plants" (Ref. 15), requires that the reactor core and associated coolant, control, and protection systems shall be designed with an appropriate margin to ensure that the specified acceptable fuel design limits (SAFDLs) are not exceeded during any condition of normal operation, including

anticipated operational occurrences (AOOs). GDC 10 is relevant to the CHF correlation, as it is used to establish safety-related margins for the fuel and cladding integrity. To ensure compliance with GDC 10, the staff confirmed that the thermal-hydraulic design of the core and the reactor coolant system was accomplished using acceptable analytical methods; is equivalent to or is a justified extrapolation from proven designs; provides adequate margins of safety from conditions that would lead to fuel damage during normal reactor operation and AOOs; and is not susceptible to thermal-hydraulic instability.

NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition" (Ref. 16) (henceforth, "SRP"), Section 4.4, "Thermal and Hydraulic Design," describes the staff's review process for thermal and hydraulic design applications. One of the acceptance criteria specified in SRP Section 4.4 for the evaluation of fuel design limits ensures that the hot fuel rod in the core does not experience departure from nucleate boiling (DNB) during normal operation or AOOs. This requires addressing the uncertainties in the values of process parameters, core design parameters, calculation methods, and instrumentation in the assessment of thermal margin with at least a 95 percent probability at a 95 percent confidence level. The origin of each uncertainty, such as fabrication uncertainty, computational uncertainty, and measurement uncertainty should be identified. According to Appendix B in SRP Section 4.2, "Fuel System Design," fuel cladding failure is presumed if local heat flux exceeds the thermal design limits.

The regulations in 10 CFR 50.34, "Contents of Applications; Technical Information," require that safety analysis reports (SARs) be submitted that analyze the design and performance of structures, systems, and components provided for the prevention of accidents and mitigation of consequences of accidents. As part of the core reload design process, licensees are responsible for reload safety evaluations to ensure that their safety analyses remain bounding for the design cycle. To confirm that the analyses are bounding, licensees confirm that those key inputs to the safety analyses (e.g., CHF) are conservative with respect to the design cycle.

4.0 TECHNICAL EVALUATION

4.1 Background Information

4.1.1 Departure from Nucleate Boiling (DNB) and Critical Heat Flux (CHF) Correlation

Departure from Nucleate Boiling (DNB) occurs when heat flux on a fuel rod surface is increased to the extent that the boiling water flowing past the fuel rod transitions from nucleate boiling to film boiling. This phenomenon causes a dramatic decrease in the heat transfer rate because of the generation of a vapor film on the fuel rod surface when the bubbles coalesce and prevent the water from reaching the surface of the fuel rod. The deterioration in heat transfer because of the DNB forces the fuel rod surface temperature to rise sharply, which may lead to fuel damage. The heat flux which causes the transition from nucleate boiling to film boiling at DNB is known as the Critical Heat Flux (CHF).

In PWRs, DNB is primarily a local phenomenon caused by the bubble crowding on the fuel rod surface. Fuel damage because of DNB is prevented by using rigorous correlations that conservatively predict the CHF to ensure that the peak heat flux in the core during normal reactor operation or an AOO will always remain below the predicted CHF. DNB is determined by CHF correlations that use the local fluid conditions as input. To prevent DNB at a location along the fuel rod, the departure from nucleate boiling ratio (DNBR) is used, which is the ratio of the calculated local CHF to the actual local heat flux during operation under the same fluid conditions, as defined below.

$$DNBR = \frac{q''_{\text{CHF}\,|\,\text{Same Location and Coolant Conditions}}}{q''_{\text{Location}}}$$

The DNBR is a measure of how close the actual heat flux is to the calculated CHF. For conservatism, the DNBR should always be greater than 1.0, so that the local heat flux is less than the CHF and the specific fuel rod will not undergo DNB. If the DNBR is greater than 1.0 for all locations in the core, DNB will not occur on any fuel rods, which provides assurance that there will be no fuel failure. If the DNBR is less than or equal to 1.0, the local heat flux is greater than or equal to the CHF, and the specific fuel rod will likely go through DNB. Because of the associated high surface temperatures, it is possible that the fuel rod experiencing the DNB may fail. Therefore, to produce a conservative safety analysis, any fuel rod that experiences DNB is assumed to have failed. When the fuel rod fails, the cladding ruptures and the first fission product barrier is breached. The radioactive nuclides, which were being contained by the cladding, will escape from the fuel rod and will be released into the reactor coolant system. Although fuel failures are undesirable, it is not possible to preclude all failures and therefore nuclear power plants have a cleanup system which can process a limited number of fuel failures.

To avoid the DNB occurrence in the bundle and to ensure that the number of fuel failures will be extremely small and limited to the “clean up” capability of the plant, a minimum DNBR value greater than 1.0 needs to be calculated by accounting for uncertainties and non-conservatisms in the empirical CHF correlation, plant parameters, and AOOs. To account for any uncertainties and non-conservatisms in the empirical CHF correlation, a one-sided 95/95 DNBR limit is used to bound the correlation’s prediction. The TR presents a statistical analysis to support a 95/95 DNBR limit of 1.124 for the PLUS7 fuel design.

4.1.2 KCE-1 CHF Correlation for PLUS7 Fuel Design

Many parameters can affect DNB or CHF such as pressure, mass flux, quality, heated length, heat flux distribution, rod bundle shape, grid spacers, wall superheat, flow memory, flow pattern, bubble size/population, bubble layer thickness, and flow instability (Ref. 17). Because of the complex nature of the DNB phenomenon, CHF correlations have empirical functional forms and are based on experimentally measured values of the CHF and CHF parameters for the specific fuel design. The functional formula of the KCE-1 CHF correlation for the PLUS7 fuel design is identical to the CE-1 CHF correlation. As presented by the applicant in the TR, the KCE-1 CHF

correlation includes the following parameters: pressure, local mass flux, local quality, heated hydraulic diameter ratio of the subchannel to the subchannel matrix, latent heat of vaporization, and the Tong factor to account for the non-uniform axial power distribution. The application of the Tong factor additionally requires the knowledge of non-uniform axial heat flux distribution and heated length from the section inlet to the CHF location. The applicant determined the eight empirical coefficients in the KCE-1 CHF correlation by a non-linear multiple-regression analysis of the measured CHF data with local fluid conditions calculated by using the subchannel analysis code TORC.

4.2 <u>Critical Heat Flux Test Program and Procedures</u>

The CHF tests for the PLUS7 fuel geometry were conducted at Columbia University's HTRF in New York, NY, which was in operation from 1951 to 2003, and collected an extensive amount of DNB data relevant to nuclear reactor fuel design. Over the years, several applicants used data from the HTRF facility and subjected the data to quality assurance (QA) reviews, and the NRC staff reviewed and certified the resulting CHF correlations. During the January 2015, audit (Ref. 10), the applicant provided a description of the QA Program (QAP) used at the HTRF test facility for the CHF tests for the PLUS7 fuel design. The documents (Refs. 18, 19, 20, and 21) furnished during the audit showed details of the [ ]^{TS}. Reference 18 described [ ]^{TS}. It confirmed that the facility had mandated a QAP in conformance with applicable requirements of the latest edition of ANSI/ASME NQA-1, "Quality Assurance Program Requirements for Nuclear Facilities" with addenda, which was documented in the HTRF/QAP, Revision 5, issued March 1998. As the CHF tests were safety related, the HTRF qualified the engineering design and materials supplied to meet the necessary QA requirements to ensure that the CHF data conform to the applicable requirements of 10 CFR Part 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Reprocessing Plants." Reference 19 refers to a [ ]^{TS} (Ref. 20) that documents an audit of the HTRF performed by [ ]^{TS}, 10 CFR Part 50, Appendix B, and 10 CFR Part 21, "Reporting of Defects and Noncompliance." During the January 21–22, 2015 NRC audit, the staff was also informed that, because the Department of Energy (DOE) sponsored several HTRF research programs, DOE also conducted an audit of the facility on an annual basis.

The applicant did not describe in the TR whether or how frequently the instrumentation calibrations were performed at this test facility. In RAI 3-7443, Question 4, the staff asked the applicant about the test section flow measuring instrumentation details and calibration. Considering the significance of accurately measuring the mass flow rate and inlet/outlet conditions in the overall computation of the CHF and its subsequent design implications, it is important to establish that the calibration of related instruments were performed following an approved test procedure and QAP. The staff noted that Reference 18 mentions that [

]TS; water temperature measurements at both the inlet and outlet of the test sections by using calibrated platinum resistance temperature detectors (RTD) and calibrated iron-constantan thermocouple (Type J); and pressure measurements made at the beginning and the end of the heated length. No void fraction and quality measurements were made during the PLUS7 CHF tests. The reported local quality is based on the lumped parameter calculations using TORC. The applicant also provided a detailed description of a typical testing day that stressed the necessity to achieve steady state before the CHF data point was taken. The applicant explained how the water layer between outside the channels and the acrylic glass wall would help achieve steady state and curb the heat losses to the surroundings. Repeatability of the data was assured at the beginning and end of every day. In its updated response to RAI 3-7443, Question 4, (Ref. 11), along with the QA documents reviewed by the staff during the January 21–22, 2015 NRC audit, the applicant explained the instrumentation, redundancy and diversity applied to measurements. During the audit, the staff examined Reference 19, which showed that [

]TS. Instrumentation and measuring devices were calibrated against devices traceable to the National Institute of Standards and Technology (NIST) at frequencies determined by the responsible engineers. The document also included the calibration records and calibration certificates of the measuring instruments used in the CHF test loop. Therefore, the staff concludes that **RAI 3-7443, Question 4, is resolved and closed**.

During the audit, the applicant explained that measurements of the thermocouples attached to the rods were only used to identify CHF occurrence qualitatively and were not used for any other purpose. As reported in the TR, the CHF point was confirmed to occur during the testing when incrementally increasing the total power led to a sudden temperature excursion of 5 to 15 ^{o}C (10 to 30 ^{o}F) inside the heater rods. When the temperature excursion was minimal, the HTRF obtained additional confirmation of the validity of the CHF point by observing a characteristic temperature decay with power reduction, as the CHF zone was rewetted. In its response to RAI 3-7443, Question 2 (Ref. 6), the applicant cited Reference 23 to corroborate the CHF point confirmation criteria. Based on the technical discussion during the audit, the information provided in the RAI response, and the supporting reference, the staff concluded that the applicant justified the CHF identification criteria, and considers **RAI 3-7443, Question 2, resolved and closed**. The applicant's response to RAI 3-7443, Question 3 (Ref. 6), clarified that the asymmetrical configuration of the seven thermocouples installed axially to measure the wall temperature along the 381 cm (150 in.) long heated length between the beginning of heated length (BOHL) and end of heated length (EOHL), posed no challenges to CHF identification. The applicant clarified that with a symmetric cosine axial power distribution applied to PLUS7 CHF tests, all CHF locations are downstream of the peak power for the test section with uniformly arranged spacer grids. That is why more thermocouples were installed in the upper part of the heated length. Figure 3-1, "As-measured CHF elevations for PLUS7 CHF test," in the RAI response shows that all CHF locations for PLUS7 fuel are at [
]TS , i.e., past the middle of the heated length. Since the CHF locations were at the thermocouples located past the middle of heated length, the asymmetrical configuration was

appropriate as it gathered more information where CHF was expected. Therefore, the staff considers **RAI 3-7443, Question 3, resolved and closed**.

4.3 <u>Test Section Heat Losses</u>

In RAI 3-7443, Question 1, the staff asked the applicant to demonstrate that the heat losses from the CHF test section were duly accounted for in its CHF test data for the entire range of the tested bundle power. The applicant was expected to offer conservative estimates of the loss of generated heat that would fail to reflect in the local fluid conditions because of convection to the ambient or through axial conduction to the rod's end. Ignoring the heat losses from the test section would be non-conservative, as it would make CHF look higher than it actually is. The RAI response described, and the staff confirmed during the audit (Ref. 10), that a [ ]^{TS} heat balance acceptance criterion was followed based on the [ ]^{TS} (Ref. 18). However, this did not address the staff's concern that the overall heat balance was not performed for each CHF data point and was rather tested only under subcooled conditions at [

]^{TS} conditions. The staff expected that these test conditions would involve much smaller heat losses than in the CHF test range used for the KCE-1 correlation development that involves bundle powers up to an order magnitude higher, inlet temperatures up to [ ]^{TS}, flow rates up to [ ]^{TS}, and pressures up to [ ]^{TS}.

During the audit, the applicant presented a bounding heat loss analysis that showed that even though the bundle power increased from [ ]^{TS} in the PLUS7 test range, the inlet water temperature increased from [ ]^{TS}. Assuming a [ ]^{TS}, the resulting temperature difference between the section inlet flow temperature and the ambient temperature increased by [ ]^{TS}. The applicant stated that as the heat loss would conservatively be proportional to the temperature difference between the water flowing through the heated section and the ambient, the rise in input electrical power from the minimum to the maximum ([ ]^{TS}) during testing outpaces the corresponding rise in heat losses. The staff concludes that this shows that as the electrical power input increases, even though the heat loss from the test section would increase, its proportion relative to the input power would actually decrease, which makes the [ ]^{TS} heat loss acceptance criterion observed at the minimum bundle power to be bounding for the entire domain of KCE-1 CHF test conditions. The staff further notes that the maximum heat losses from the measured electrical power at the direct current (DC) generator terminals (Bbpwr) and at the test section inlet and outlet bus (Tspwr) were [

]^{TS}. The mean values were [ ]^{TS}, respectively. The difference between the two heat losses corresponds to the loop components that are not a part of the heated test section. The applicant conservatively factored the heat loss from the test section into the CHF data reduction through a heat-input correction factor that was a function of the inlet water temperature, which allowed the measured CHF values to be based on the rise in fluid enthalpy through the test section. This eliminated the heat-loss related bias from the CHF

test data. The applicant considered the deviation between the correction factor and the measured value in estimating the overall CHF measurement uncertainty, as addressed in its response to RAI 3-7443, Question 14 (Ref. 8). The applicant resubmitted the RAI response (Ref. 11) to justify the applicability of the [ ]^{TS} heat loss acceptance criterion and to show that the heat losses measured at the lowest input power are bounding. Accordingly, the staff considers **RAI 3-7443, Question 1, to be resolved and closed**.

The staff issued RAI 3-7443, Question 5, to ask the applicant to describe how the overall bundle power was determined, as it was not clear from the TR. In its RAI 3-7443, Question 5, responses (Refs. 6 and 11), the applicant described that the HTRF obtained the bundle power (Tspwr) from the measured bus-to-bus power (Bbpwr) using a voltage correction factor that was derived from the measured bus-to-bus voltage and the test section voltage. Current metering/readouts for protection/control/test operation were provided by switchboard shunts and were recorded by the data acquisition system. Measurements of voltages were made between the two ends of heater rod (bus-to-bus voltage), and between the bottom end of the copper end pieces and the tip of the top nickel piece (connecting through top nickel plate) between the test section inlet and outlet (test section voltage). These voltages are conditioned through precision resistor divider networks and amplifiers for entry into the data acquisition system and readout in the control room. The HTRF maximum value of overall bundle power was 12 MW (240 Volt*50,000 Ampere). Based on the description provided, the staff considers **RAI 3-7443, Question 5, to be resolved and closed.**

4.4 Spacer and Part-Length Heated Rod Effects

The HTRF conducted all CHF tests with a constant heated length of the rod (381 cm (150 in.)) and a constant grid spacing (39.9 cm (15.7 in.)). In RAI 3-7443, Question 12, the staff asked the applicant to explain whether the lack of heated length and grid spacing parameters in the KCE-1 correlation would affect its applicability to the actual PLUS7 fuel bundle safety analyses. During the testing conducted for some other CHF correlations, heated length and grid spacing were also varied and duly accounted for in the resulting CHF correlation using additional terms. In its response to RAI 3-7443, Question 12 (Ref. 7), the applicant responded that the KCE-1 correlation was developed by using the CHF data from test sections with the same axial geometry as PLUS7 fuel as described in Table 2-1, “Characteristics of Geometrical Configuration of CHF Test Sections,” and illustrated in Figure 2-9, “Axial Geometrical Configuration of Test Section,” of the TR. This means that the effects of grid spacing and heated length on CHF were inherently included in the measured data, and thus were captured in the KCE-1 correlation. In addition, the KCE-1 correlation is limited to the PLUS7 geometry with fixed grid spacing and heated length. Based on these facts, there is no need for heated length and grid spacing parameters in the KCE-1 correlation, and the staff concludes that the response is acceptable and **RAI 3-7443, Question 12, is resolved and closed**.

4.5 Axial Power Profile and Tong Factor

The CHF test data for the PLUS7 fuel geometry were obtained by using a non-uniform axial power distribution (i.e., a symmetric chopped cosine power profile with a peak of 1.475 at the

middle of the heated length). The test sections were also designed with a varying radial power distribution such that the highest power rods were in the middle of the bundle. In RAI 3-7443, Question 6, the staff asked the applicant to explain the appropriateness of testing a single axial profile and why the inlet/bottom or outlet/top peaked power profiles were not included in the test matrix. In its response to RAI 3-7443, Question 6 (Ref. 8), the applicant described the tested symmetric cosine axial power distribution as the typical axial power profile resulting from the two-dimensional neutron diffusion equation for the finite cylinder geometry (Ref. 24) representing the PLUS7 fuel design. The response additionally cited that the [

]^{TS}. The response also clarified the application of the Tong factor (Fc), defined by the following equation, [

]^{TS}.

$$F_c = \frac{q_{\mathrm{CHF,EU}}}{q_{\mathrm{CHF,NU,measured}}}$$

The Tong factor is meant to account for different axial power shapes. Table A-3, "KCE-1 CHF Correlation Database," in the updated RAI 3-7443, Question 6, response (Ref. 11), shows that [

]^{TS}, which is consistent with the response to RAI 3-7443, Question 3 (Ref. 6), which shows that all CHF points were observed at thermocouples that are downstream of the axial flux peak location. As no testing of the PLUS7 fuel geometry was conducted with a uniform axial power distribution, no Tong factor could be customized by the applicant for the KCE-1 CHF correlation. Such an optimization of the Tong factor for the PLUS7 fuel split vane mixing grid geometries would require testing both uniform and non-uniform axial power distributions with and without the guide thimble tube, but it was not done for the tested PLUS7 fuel geometry. In RAI 3-7443, Question 7, the staff also asked the applicant to justify using the standard Tong factor with the KCE-1 correlation to predict the CHF for the PLUS7 fuel geometry. In its response to RAI 3-7443, Question 7 (Ref. 11), the applicant stated that the standard Tong factor was applicable to design and safety analyses of the PLUS7 core with the KCE-1 correlation based on its non-dependency on fuel design. The Tong factor does not have any terms related to fuel geometry and solely depends upon the axial flux distribution and the resulting local quality at the CHF location for the given mass flux. The applicant also provided citations (Refs. 26 and 27) to demonstrate that the standard Tong factor used by the KCE-1 correlation has been shown in previous CHF test programs to conservatively apply to various fuel designs and the corresponding CHF correlations under similar application environments as that of the KCE-1 CHF correlation. References 26 and 27 show that applying the standard Tong factor with the CE-1 CHF correlation had predicted CHF conservatively in several axially non-uniformly heated rod bundles. The staff, nevertheless, asked the applicant to qualify the statements by demonstrating additional conservatism in the KCE-1 correlation for the typical non-tested heat flux profiles to cover the actual axial power distribution experienced during the operation of PLUS7 fuel cores. In its final response to Question 6 (Ref. 13), the applicant stated that the [

][TS]. The range of Tong factor ([ ][TS]) in the KCE-1 CHF correlation application using the PLUS7 tested cosine shape is shown in Figure 6-2, “Distribution of M/P versus Tong Factor Fc (Application Database).” The Tong factor range from the KCE-1 cosine tests is [ ][PROP] the range covered in previous CE-1 CHF tests with several different axial power shapes (Ref. 27).

The applicant further stated that while other shapes may produce more limiting DNBR, the Tong factor of the tested cosine shape is more limiting. The applicant demonstrated this by considering four classic power shapes: top-peaked, double-humped, symmetric cosine, and bottom-peaked. Figures 6-3, “Axial Behavior of KCE-1 DNBR for Each Power Shape,”, and 6-4, “Axial Behavior of Fc for Each Power Shape,” in the response demonstrate that the double-humped and top-peaked shape could be more limiting for DNBR, but its Tong factors are [ ][TS]. While the bottom-peaked shape has higher Tong factors than that of the cosine shape within the test range, the staff understands that it is rarely limiting as there is a high degree of sub-cooling and low quality which generally preclude CHF. Thus, the minimum DNBR from bottom-peaked shapes is typically much higher (i.e., more conservatively predicted than that from the top-peaked or cosine shapes). The staff recognizes that the likelihood of having a CHF at the start of the bundle is remote, and further believes that the KCE-1 correlation would conservatively predict the CHF in that region. Based on the information provided by the applicant, the staff concluded that using a symmetric cosine profile as the non-uniform axial power distribution of the PLUS7 fuel geometry was adequate for its CHF testing. Therefore, the applicant’s response is acceptable, and **RAI 3-7443, Question 6, is resolved and closed**.

4.6 Uncertainties in CHF Measurements

SRP Section 4.4, Acceptance Criterion 1, deals with various uncertainties involved in the CHF measurement and correlation development, such as fabrication uncertainty, computational uncertainty, and measurement uncertainty of instrumentation. As the applicant supplied no discussion about these uncertainties in the TR, in RAI 3-7443, Question 14, the staff asked the applicant to provide the information. In its response to RAI 3-7443, Question 14 (Ref. 8), the applicant provided a detailed listing of the measurement uncertainties of the instrumentation employed in the CHF tests in Table 14-1, “Measurement Uncertainties for PLUS7 CHF Tests,” extracted from Reference 28. However, the applicant did not provide any information about any computational uncertainties or the overall uncertainty in the measured CHF. During the audit, the applicant offered a detailed overview of the uncertainties involved in the CHF measurements. The applicant also explained that [ ][TS]. During the audit, the applicant also explained that they also had accounted for an “operational” uncertainty in CHF measurement by keeping the maximum incremental rise in heat flux at [ ][TS]. The applicant explained that as the directly

measured values of the electrical power input to the test section were used to normalize the heat flux profile and subchannel code TORC was not used for this purpose. The staff accepted that there are no computational uncertainties involved in the overall CHF measurement and data reduction.

In its updated response to RAI 3-7443, Question 14 (Ref. 11), the applicant also supplied an overall uncertainty analysis of the CHF measurement that accounted for the uncertainties involved in power measurement, temperature-dependent voltage correction, incremental heat flux stepping to approach CHF, tube wall thickness, and surface area. The applicant determined a maximum of [ ]^{TS} uncertainty in the measured CHF value based on [

]^{TS}. In its updated RAI response, the applicant also stated that the overall uncertainty in the measured CHF data is inherently captured in the 95/95 DNBR limit, which is determined by the measured-to-predicted (M/P) CHF values statistics. The staff concluded that the applicant provided sufficient information about the experimental uncertainties involved and established the overall uncertainty in CHF measurement, as requested by the staff. The staff found the applicant's treatment of the uncertainties, and the updated response to RAI 3-7443, Question 14 (Ref. 11), acceptable. The staff considers **RAI 3-7443, Question 14, to be resolved and closed**.

Using a CHF correlation for reactor core analysis requires nodalizing the core geometry to solve mass, momentum, and energy conservations across the core for the given heat flux profile and the inlet flow rate while accounting for the local transport properties of the coolant. This is done by using a subchannel analysis computer code such as TORC. Given the initial and boundary conditions of a transient from the system's code, the subchannel code can calculate the local fluid conditions in the core to use with the correlation to calculate the CHF at those conditions. During reactor operation, the calculated CHF is divided by the local heat flux to calculate the operating DNBR value for technical specification monitoring purposes. The applicant determined the coefficients of the KCE-1 CHF correlation by a non-linear multiple regression analysis of the measured CHF data along with the local fluid conditions calculated by using TORC. The main input data used for TORC are summarized in Table 4-1, "Main Input Data of the TORC Model for CHF Test Data Analysis," of the TR but no discussion of the selection of inputs was provided by the applicant. In RAI 3-7443, Question 15, the staff asked the applicant to provide justifications and sources for its TORC input selections. In its final response to RAI 3-7443, Question 15 (Ref. 11), the applicant stated that the TORC input parameters given in Table 4-1 of the TR were [
]^{TS}. The response described various TORC input parameters and its adjustments to reflect the design characteristics of PLUS7 fuel. Table 15-1, "TORC Input Data Consistency with Design Constitutive Relations," in the response supplied justifications for using various TORC input parameters and its consistency with the design constitutive relations used in TORC to model various single-phase and two-phase heat transfer and fluid flow characteristics. Table 15-2, "TORC Design Constitutive Relations and Applicable Ranges," showed that the CHF data are taken at conditions that fall within the applicable range of the TORC design constitutive relations. Accordingly, the staff accepts the applicant's use of the TORC input data summarized in Table 4-1 of the TR. Therefore, **RAI 3-7443, Question 15, is resolved and closed**.

TORC is the only subchannel code that was used for the KCE-1 CHF correlation development. However, the TR mentioned that the KCE-1 CHF correlation can also be used with a different subchannel code, CETOP-D. The staff issued RAI 3-7443, Question 16, to inquire about the differences between the two codes, especially how they would calculate the local fluid conditions in the subchannels. According to the initial RAI 3-7443, Question 16, response (Ref. 8), the CETOP-D subchannel code used a different model to calculate the transport properties and a different numerical scheme to solve the conservation equations than the TORC code. During the audit, the staff asked for the justification of using the KCE-1 CHF correlation with CETOP-D code, as its different transport properties module and different numerical scheme may entail computational uncertainties potentially warranting additional non-conservatism in the 95/95 DNBR limit. The staff stressed that an application of the CETOP-D code for the design and safety analyses with the KCE-1 CHF correlation would require an assurance that the MDNBR calculated by CETOP-D shall always be bounded by the MDNBR calculated by TORC at the same boundary conditions. In its updated response to RAI 3-7443, Question 16 (Ref. 11), the applicant agreed to delete all references to CETOP-D from the TR and limit the application of the KCE-1 CHF correlation to TORC. As discussed during the audit and reflected by the updated response to RAI 3-7443, Question 16, the application of the KCE-1 CHF correlation is limited to the PLUS7 fuel geometry with the TORC subchannel computer code. The staff included a limitation in Section 5.0 of this safety evaluation to clarify that the use of the KCE-1 correlation with any other subchannel code will require additional review by the NRC. The staff considers **RAI 3-7443, Question 16, to be resolved and closed**.

4.7 Statistical Evaluation of 95/95 DNBR Limit

The applicant considered the following topics for the individual and collective statistical treatment of the M/P CHF ratios for various test data groups within and across TS101 (thimble subchannel test section) and TS102 (matrix subchannel test section) datasets: data groups comparison, treatment of outliers, normal distribution, homogeneity of variance and means, and the 95/95 DNBR limit. The applicant used standard statistical tests, including the D' Normality test at the 95-percent confidence level for groups with more than 50 data points. The applicant performed the Bartlett test (homogeneity of variance), and Unpaired *t* test (homogeneity of means) to test whether the data groups used for the correlation development could be pooled. The applicant performed the non-parametric Wilcoxon-Mann-Whitney test to test the null hypothesis that all the data of test sections TS101 and TS102 were sampled out of the same population. The results of these tests were provided in Tables 5-1 and 5-2, and the applicant provided brief descriptions of the statistical tests in TR Appendix B.

The DNBR limit which meets the 95/95 acceptance criterion, was determined by using Owen's one-sided tolerance limit method (Ref. 29). Use of this method has been previously approved by the NRC (Ref. 30). The general equation for Owen's method is as follows:

$$Limit_{95/95\_DNBR} = \frac{1}{\overline{\frac{M}{P}} - K_{95/95} \cdot \sigma}$$

Where

$\overline{\frac{M}{P}}$ is the test population mean of the measured-to-predicted CHF ratios.

σ is the effective standard deviation of all the M/P data.

$K_{95/95}$ is a tolerance multiplier which provides the 95/95 probability/confidence level, and is a function of the effective degrees of freedom in the test series.

4.8 Challenges to the Statistical Evaluation of 95/95 DNBR Limit

During the review, the NRC staff identified multiple challenges to the applicant's proposed 95/95 DNBR limit of 1.124. These challenges included a non-conservative data trend at low pressures; the existence of a non-conservative sub-region around 12.07 MPa (1,750 psia); and the generation of the 95/95 DNBR limit using only training data but no validation data. The applicant had not accounted for these three non-conservatisms in the development of the KCE-1 CHF correlation. In addition, the applicant could not quantify the magnitude of the conservatism gained by its specific use of the Tong factor to possibly cover the three non-conservatisms. These staff concerns were expressed in RAI 3-7443, Questions 6, 7, 9, and 17 (Ref. 3). The issues could not be resolved by the earliest RAI responses (Ref. 8), audit (Ref. 10), and the post-audit response update (Ref. 11). A discussion to resolve these took place in another public meeting with the applicant (Ref. 12). The issues were resolved with a subsequent RAI response revision (Ref. 13), as described in the following sections.

4.8.1 Non-Conservative Data Trend

SRP Section 4.4 outlines the DNB acceptance criterion to provide assurance that there is at least a 95-percent probability at a 95-percent confidence level that the hot fuel rod in the core does not experience a DNB or transition condition during normal operation or AOOs. The calculation of a single 95/95 DNBR limit to bound the overall uncertainty of a CHF correlation is predicated on three statistical assumptions: (1) normality, (2) homoscedasticity, and (3) independence. Of these three assumptions, the staff considered the assumption of independence the most important. Statistical independence implies that random sampling can represent the underlying population that comprises mutually independent and identically distributed random variables that have the same probability distribution. This means that an element in the sequence is independent of the random variables that came before it. In general, when the probability distribution of one observation is affected by the level of another, the observations are said to be statistically dependent (Ref. 31). Based on Figure 5-3, "95/95 DNBR Limits for Each Data Group," of the TR, the staff concluded that the assumption of independence may not hold as the KCE-1 CHF correlation does not behave consistently

throughout the applied pressure range of 9.62–16.65 MPa (1,395–2,415 psia), and its uncertainty is sensitive to system pressure. The staff noted that the five pressure datasets in Figure 5-3 used in the correlation development seem to be from four different populations, and there is a distinct non-conservative trend of decreasing predictive capability with pressures from 15.17 MPa (2,200 psia) to 12.07 MPa (1,750 psia). While the trend in the M/P values has clearly reversed by the low pressures around 9.62 MPa (1,395 psia), it is not apparent how far the trend continued in the empty region between 9.62 to 12.07 MPa (1,395–1,750 psia) before reversing. The staff also noted that no data are available within the 9.62–12.07 MPa (1,395–1,750 psia) range to evaluate the magnitude of the non-conservatism associated with the non-conservative trend in the data at low pressures. In RAI 3-7443, Question 8, the staff asked the applicant to justify the use of a single statistical 95/95 DNBR limit to bound the KCE-1 CHF correlation over its entire application domain, primarily focusing on the non-conservative data trend between 9.62 and 12.07 MPa (1,395 and 1,750 psia). In its earlier response to RAI 3-7443, Question 8 (Ref. 8), the applicant did not offer any justification for the use of KCE-1 correlation in that range. The issue was discussed in detail at the audit and the applicant informed the staff that it planned to address the non-conservative data trend by excluding the low pressure region of 9.62–12.07 MPa (1395–1750 psia) from the applicable range of the KCE-1 CHF correlation and limiting the applicable range to 12.07–16.65 MPa (1750–2415 psia). The NRC staff would find this resolution acceptable, however, the applicant's post-audit response to RAI 3-7443, Question 8 (Ref. 11), did not reflect the commitment for reduced applicable pressure range. Table 18-1, "Range of AOO Design Analysis for APR1400," in the response to RAI 3-7443, Question 18 (Ref. 11), still showed the KCE-1 CHF correlation applicable pressure range to be 9.62–16.65 MPa (1,395–2,415 psia). Therefore, the NRC staff has formalized a limitation on the use of the KCE-1 CHF correlation that includes the modified pressure range of 12.07–16.65 MPa (1,750–2,415 psia), as documented in Section 5.0 of the present SER. The staff also concludes that there were no other significant non-conservative trends in the M/P CHF data as a function of the KCE-1 correlation variables. The reduced pressure range has allowed the staff to consider **RAI 3-7443, Question 8, to be resolved and closed**.

The M/P CHF ratios of all test data were plotted for system pressure, local mass flux, local quality, and equivalent heated diameter ratio in Figure 5-3, "Distribution of M/P versus System Pressure," through Figure 5-6, "Distribution of M/P versus Equivalent Heated Diameter Ratio," of the TR, respectively. Figure 4-1, "System Pressure versus Average Heat Flux of Test Section," through Figure 4-3, "Inlet Mass Flux versus Average Heat Flux of Test Section," provide similar plots for bundle average heat flux data. However, the TR does not provide the corresponding plots of the measured CHF data. In RAI 3-7443, Question 11, the staff asked the applicant to supply the corresponding plots to identify any adverse and non-linear trends in the measured CHF data. In its response to RAI 3-7443, Question 11 (Ref. 11), the applicant provided the corresponding plots for pressure, local mass flux, and equivalent heated diameter ratio. The applicant supplied the plot for local quality as a part of its response to RAI 3-7443, Question 10. The staff did not observe any adverse or non-linear trends for these plots of the measured CHF data. Therefore, the staff considers **RAI 3-7443, Question 10 and Question 11, to be resolved and closed**.

4.8.2 Non-Conservative Test Data Sub-Region

One important assumption commonly made for CHF correlations is that the predictive behavior is consistent over the entire application domain when they are used in reactor safety analysis. The very notion of 95/95 statistics presumes that any error associated with the prediction of a CHF correlation must be random and uniformly distributed over the entire application domain of the correlation, which is defined by the limited ranges of their input predictor variables. The staff tested the validity of the assumption for the KCE-1 CHF correlation by identifying any non-conservative sub-regions in the application domain. As the correlation's predictive capability would be degraded in non-conservative sub-regions, its existence may impair the reactor safety analysis. The NRC staff used the method proposed by Kaizer (Ref. 32), and identified a non-conservative sub-region at pressures near 12.07 MPa (1,750 psia), qualities near 0.1, and local mass fluxes near 2 $Mlb_m/hr\text{-}ft^2$), and this was the technical basis for RAI 3-7443, Question 9. The staff's method to identify the sub-region is a multidimensional approach capable of determining whether the CHF correlation's predictive behavior is likely to be because of random effects or because of degraded predictive capability. Because of a higher concentration of the non-conservative M/P CHF data points in the identified sub-region, mainly clustered around 12.07 MPa (1,750 psia), the KCE-1 correlation's predictive capability was degraded below what would have been justified by the 95/95 DNBR limit.

During the audit, the staff asked the applicant to quantify the margin in the 95/95 DNBR limit required to accommodate the non-conservative sub-region, so that the staff could understand how much of the margin gained in the DNBR limit by the Tong factor usage was consumed by the non-conservative sub-region. The non-conservative sub-region identified by the staff contains a higher than expected number of M/P points that fell below the 95/95 DNBR limit of 1.124 than can be explained by random chance. In Table A-3, "KCE-1 CHF Correlation Database," of the TR, seven points fall below M/P of [ ]TS that corresponds to DNBR of 1.124. Six out of those seven points correspond to 12.07 MPa (1,750 psia) pressure. In its updated response to RAI 3-7443, Question 9 (Ref. 11), the applicant did not articulate a justification for the use of the KCE-1 correlation in this sub-region. However, during the September 3, 2015, public meeting (Ref. 12) and through the subsequent RAI response (Ref. 13), the applicant emphasized that the number of M/P values below the 95/95 DNBR limit in the identified non-conservative sub-region near 12.07 MPa (1,750 psia) was based on the correlation development database that [ ]TS. The upper part of Table 5-4, "KCE-1 CHF Correlation Statistical Data per Local Fluid Condition Extraction Method," of the TR shows the DNBR limits of 1.124 and [ ]TS for the TS101 and TS102 datasets, respectively. Further, the staff noted that the number of M/P values below the 95/95 DNBR limit is reduced and the DNBR limit is reduced to [ ]TS when [ ]TS, as shown in the lower part of Table 5-4 of the TR. This shows that [ ]TS as factored in the DNBR limit of 1.124 is conservative. Table 5-4 also shows that the DNBR limit of 1.124 for the TS101 dataset for thimble subchannel test section is at least [ ]TS more conservative than the DNBR limit of [ ]TS for the TS102 for the matrix subchannel test section. Because of the

considerations noted above, the staff concludes that the effect of the thimble channel guide tube is conservative and has been factored into the KCE-1 CHF correlation.

In its revised response to RAI 3-7443 (Ref. 13), the applicant demonstrated that when a calculated Fc is used in the KCE-1 CHF predictions, the M/P versus pressure plot in Figure 9-1, "M/P versus Pressure with Tong factor Fc," shows no M/P data point below the M/P value associated with the DNBR limit of 1.124. Figure 9-1 in the RAI response, which is the plot of the M/P application database ([ ]TS), shows the lowest M/P value of [ ]TS corresponding to a DNBR of [ ]TS. This point belongs to the TS101 dataset, which is more conservative than the TS102 dataset whose lowest M/P value of [ ]TS corresponds to a DNBR of [ ]TS. The 95/95 DNBR of the entire correlation application database is [ ]TS, which corresponds to an M/P value of [ ]TS. The applicant thus demonstrated that the proposed DNBR limit of 1.124 has about [ ]TS conservative margin compared to the lowest M/P data point with a DNBR of [ ]TS, and a [ ]TS margin compared with the 95/95 DNBR of [ ]TS of the entire application database. The staff finds that these margins and statistics appropriately accommodate the non-conservative sub-region; therefore, **RAI 3-7443, Question 9, is resolved and closed**.

4.8.3 Non-Conservative Overfitting of the Test Data

Best practices in fitting CHF correlations gathered from CHF topical reports reviewed by the NRC staff have suggested that a given CHF test database should be divided into a training dataset and a validation dataset (Ref. 33). Then, the applicant should fit correlation coefficients using the larger training dataset and independently validate it against the smaller validation dataset to ensure a consistent behavior of the correlation. This process helps the applicant assess whether the correlation lacks in predictive capability on data not used in the development of the correlation. In RAI 3-7443, Question 17, the staff inquired whether some test data were initially excluded from the KCE-1 correlation coefficient generation and were later used for independent correlation validation. The staff was concerned about the potential for "overfitting," which in this instance means that all available CHF data points in the database were used in the regression analysis to optimize the KCE-1 CHF correlation coefficients and no points were set aside to perform an independent validation of the correlation. The TR did not report any validation of the resulting correlation with an independent data set, so the correlation would likely be slightly non-conservative when applied at conditions for which it was not tested. As all the same CHF data that were used to generate the correlation were also used to evaluate its 95/95 DNBR limit, the staff determined there was a need to quantify the inherent non-conservatism. In its first response to RAI 3-7443 (Ref. 8), the applicant stated that the potential for overfitting is not expected in the KCE-1 CHF correlation, but did not elaborate.

During the audit, the applicant was asked again by the staff to address the potential for a decrease in the KCE-1 correlation's predictive capability because of overfitting. The staff asked the applicant to estimate the non-conservatism in the 95/95 DNBR limit because of overfitting, by running a random-sub-samples analysis of the CHF database with a larger training (around 80 percent) and smaller validation (around 20 percent) datasets, and demonstrate that the conservative use of the Tong factor more than compensates for this. In its updated response to

RAI 3-7443, Question 17 (Ref. 11), the applicant submitted results with a discrete k-folds cross-validation analysis of the correlation development database, which the staff expected to be less conservative than a continuous random-sub-samples analysis. The applicant's 5-fold analysis shows a maximum 95/95 DNBR limit of [ ]TS for an 80 percent training-20 percent validation database distribution. Table 17-4, "M/P Statistics for Cases with Max. MAPE for Each k-folds," showed an increasing trend in the 95/95 DNBR limit from 1.124 to [ ]TS as the number of k-folds increased from 2 to 5, which suggested an about [ ]TS non-conservatism in the DNBR limit because of overfitting. The NRC staff's own random sub-samples confirmatory analysis of the applicant's data, which is equivalent to repeating the applicant's 5-folds analysis 10,000 times and generating a continuous probability density histogram, shows approximately a [ ]TS non-conservatism in the DNBR limit. After the September 3, 2015, public meeting, the applicant submitted a revised response to RAI 3-7443, Question 17 (Ref. 13), that showed a random sub-samples analysis that the applicant performed to quantify the non-conservatism because of using all available data as training data and leaving none for independent validation. Figure 17-4 shows the applicant's results of a continuous probability density distribution of the M/P values for 1,000 runs. This analysis is consistent with the NRC staff's own sub-sampling analysis that the correlation has a lower predictive capability on data that were not used in generating the correlation, and the NRC staff's expectations based on previous experience. The information provided in various updates of the RAI 3-7443, Question 17, response, shows approximately a [ ]TS non-conservatism because of overfitting, which is consistent with the staff's similar analysis. Thus, the demonstrated conservatism ([ ]TS, as explained in Section 4.8.4, more than accounts for the non-conservatism because of overfitting. The staff therefore considers **RAI 3-7443, Question 17, to be resolved and closed**.

4.8.4 Quantification of the Tong Factor Conservatism

In RAI 3-7443, Questions 6 and 7, the staff inquired about the use of the Tong factor and the symmetric cosine axial power distribution with the KCE-1 correlation development and application. In its response to RAI 3-7443, Questions 6 and 7 (Ref. 8), the applicant provided more details of its treatment of the Tong factor in the KCE-1 correlation's development and subsequent use. The applicant stated during the first public meeting and through RAI responses as well as the audit, that it had treated the "Tong Factor" in a very conservative manner that would reduce the correlation's predicted CHF value. The applicant emphasized that the KCE-1 CHF correlation was developed based on [

]TS. An application of [

]TS. The applicant further explained that the CHF predicted by the resulting KCE-1 correlation is treated as if it was [

]TS. The staff recognized that the conservatism at the correlation application stage is fundamentally caused [

]^{TS} that appears in the denominator of the KCE-1 correlation.

The staff accepted the applicant's logic that such a treatment of [

]^{TS}. However, the applicant could neither quantify the conservatism inherent in the Tong factor treatment with the KCE-1 correlation nor demonstrate that it more than made up for the non-conservatisms that were identified by the staff. The staff was unable to understand the applicant's method for quantifying the Tong factor conservatism in the earlier RAI response (Ref. 8), and requested data in tables and plots during the audit. In its updated response to RAI 3-7443, Question 6 (Ref. 11), the applicant provided data, resubmitted after the audit, which was found to be inconsistent with the narrative, and did not reconcile with a Tong factor conservatism. The applicant stated in its response that [ ]^{TS} the effect of axial power distribution on CHF through its Tong factor method has built about [ ]^{TS} conservatism in the 95/95 DNBR limit (1.124) for its application in the design and safety analyses. However, when the staff more closely examined the eight figures and Table A-3, "KCE-1 CHF Correlation Database," in the RAI 3-7443, Question 6, response, the staff noted inconsistencies in the data that suggested that the actual margin created by the Tong factor conservatism was uncertain and may be much less than [ ]^{TS}.

The submitted information showed two different sets of Tong factors. The set that is documented in the modified Table A-3 and Figure 6-7, "Distribution of M/P versus Tong Factor Fc at MDNBR location without Fc," of the response, is not the same as the one that has [ ]^{TS} margin and is depicted in the remaining seven figures of the RAI response. The Tong factors' range documented in Table A-3 is [ ]^{TS}, which corresponds with Figure 6-7. Figure 6-8, "Distribution of M/P versus Tong Factor Fc at MDNBR location with Fc," showing a plot of M/P values vs. Tong factor depicts much larger Tong factors, i.e., about [ ]^{TS}. Though Figure 6-8 is consistent with Figures 6-1, "KCE-1 CHF Correlation Predicted CHF vs. Measured CHF," through 6-6, "Distribution of M/P versus Tong Factor Fc at indicated CHF location," whose purpose is to demonstrate [ ]^{TS} conservatism, the Tong factors used in these figures are much larger than the ones that are documented in Table A-3 and are likely to give a smaller conservatism. Essentially, the applicant could not demonstrate the magnitude of the conservatism because of [

]^{TS}.

These difficulties in resolving the Tong factor conservatism and other outstanding issues despite the staff's repeated efforts through the audit and several rounds of the RAI response updates, led to the public meeting with the applicant at the NRC offices on September 3, 2015 (Ref. 12). During the meeting, the applicant agreed with the three non-conservatisms identified by the NRC staff and addressed them. They acknowledged that the conservative treatment of the Tong factor could not be demonstrated based on the information submitted thus far, and explained that the KCE-1 CHF database submitted as Table A-3 in the TR was the intermediate "correlation development database," and not the final "correlation application database." The applicant also provided, in the meeting package, the application version of the CHF database, correlation application results, revised statistics and figures that were previously not presented

in the TR or RAI response updates. Contrary to the earlier RAI response (Ref. 11), the staff found the submitted information to be mutually consistent. In its revised response to RAI 3-7443, Question 7 (Ref. 13), the applicant also provided the correlation application database as Table 7-1, "KCE-1 CHF Correlation Database for Application," and explained its differences from the correlation development database. The application database accounted for the application of Tong factor with the KCE-1 correlation on the M/P and DNBR predictions as well as its effect on the subchannel local quantities calculated by using the TORC code. While analyzing the application database, the staff noted that as the minimum DNBR (MDNBR) always corresponds to one of the subchannels that surround the rod undergoing the CHF, applying the Tong factor and rerunning TORC may change the subchannel and node of MDNBR. So, even though all measured quantities and computed local fluid conditions remain the same throughout the TORC subchannel nodalization, the predicted MDNBR location in the application database may vary from the observed one in the development database because of factoring the Tong factor in the CHF prediction by the correlation.

In the TR, the applicant had provided the development database statistics from the comparison of the measured data to the KCE-1 predicted value [ ]^{TS}. In its revised response, the applicant revaluated the statistics for the application database with the realistically calculated Tong factor to demonstrate its conservative margin. As a result, the DNBR limit dropped from 1.124 [ ]^{TS}, which shows that using a [ ]^{TS} at the correlation development stage is equivalent to adding a [ ]^{TS} conservatism to the DNBR. The [ ]^{TS} "Tong factor conservatism" in the proposed DNBR limit of 1.124 is on top of using the most limiting CHF data point (M/P = [ ]^{TS} or DNBR = [ ]^{TS}) in the most bounding TS101 dataset, which is also about [ ]^{TS} more conservative than the actual 95/95 DNBR of [ ]^{TS} of the entire correlation application database. So, the [ ]^{TS} margin built by the Tong factor treatment is in addition to the [ ]^{TS} that already covers the non-conservatism because of the non-conservative sub-region, as explained in Section 4.8.2. The staff concludes that the information presented in the public meeting and the later RAI response has adequately supported the applicant's proposed DNBR limit of 1.124 for the KCE-1 correlation by demonstrating that the KCE-1 correlation has an about [ ]^{TS} conservative bias because of its Tong factor treatment. The staff also concludes that the demonstrated conservatism ([ ]^{TS}) in the use of the Tong factor more than offsets the non-conservatism because of overfitting ([ ]^{TS}). The NRC staff has determined that the KCE-1 CHF correlation will accurately predict DNB occurrence with at least a 95-percent probability at the 95-percent confidence level when applied with the DNBR limit of 1.124. The staff considers **RAI 3-7443, Question 7, to be resolved and closed**. As the applicant relied on the conservatism derived from the application of the KCE-1 CHF correlation with the Tong factor being greater than one for the plant safety analyses, the staff has imposed a limitation in Section 5.0 of this SER, which will mandate the applicant's proposed use of the KCE-1 CHF correlation with [ ]^{TS} as a requirement.

4.8.5 Remaining Uncertainties in the KCE-1 CHF Correlation DNBR Limit

Based on the assessment of the non-conservative test data sub-region and overfitting as described above, the staff concluded that the KCE-1 correlation has at least [ ]^{TS} margin

after accommodating all major non-conservatisms. Not all the following uncertainties are non-conservatisms, and some of them actually are conservatisms.

- The RAI 3-7443, Question 14, response shows a maximum CHF measurement uncertainty of [ ]^{TS}, which was not explicitly factored into their correlation development or the 95/95 DNBR statistical analysis. The staff notes that as the measurement uncertainty is randomly distributed around the CHF data, the maximum non-conservatism it may incur to the 95/95 DNBR limit is about 1.7 percent.

- In Table 9-2, "The Detailed M/P Statistics of Test Section TS101 as the Application of Tong Factor Fc," of its updated response to RAI 3-7443, Question 9 (Ref. 11), the applicant showed that even a different assumption for the subchannel type associated with the CHF could lead to a slightly higher 95/95 DNBR limit of [ ]^{TS}, which exceeded the 1.124 value documented in the TR. So, there is an additional sensitivity to the choice of channel picked for data analysis that was not explored by the staff. Its potential non-conservatism is expected to be about 1 percent.

- As the KCE-1 correlation [ ]^{TS} the data in the guide tube tests (Test 101) because of [ ]^{TS} associated with the CHF, the correlation penalizes itself and provides excessively conservative M/P values. The staff found that each matrix subchannel in Test 101 has a higher predicted CHF than the guide thimble corner subchannel that is farther from CHF. Normally, one could ignore those subchannels and only consider the subchannels that are closest to CHF that have the lowest predicted CHF value. The applicant did not ignore them and rather used some data points, which makes the KCE-1 correlation's predictive capability appear to be worse than it actually is. This conservative bias, which would have increased the [ ]^{TS} demonstrated margin, was not investigated in the staff's review.

- As explained by the applicant in its response to RAI 3-7443, Question 7 (Ref. 13), the original Tong factor used by the KCE-1 correlation was developed based on single tube and annuli data, and has been shown to be excessively conservative in the CE-1 TR (Ref. 27) and in the WNG-1 TR (Ref. 34) for several axially non-uniformly heated rod bundle data. The NRC approved the WNG-1 correlation, which was developed with both uniform and non-uniform data, and its validation database also included data from a PLUS7 test (Test 102). In its response to RAI 3-7443, Question 7, the applicant showed that the KCE-1 correlation with the Tong factor applied is about [ ]^{TS} more conservative than the WNG-1 correlation with a similar correction factor optimized with the WNG-1 uniform/non-uniform data. As no CHF tests with uniform axial power shape were conducted for the PLUS7 fuel design, the standard original Tong factor is applied to the KCE-1 correlation without any adjustment or optimization, which the staff considers to be another unquantified potential conservatism in the use of the KCE-1 correlation.

The staff concludes that the remaining quantified Tong factor conservatism in the 95/95 DNBR limit of 1.124 would more than make up for the above uncertainties associated with CHF measurement, testing, and the subchannel assumption. Therefore, a 95/95 DNBR limit of 1.124 would still be valid.

4.9 Applicability of the KCE-1 CHF Correlation

Section 6 of the TR implied that meeting the 95/95 DNBR limit would also mean meeting the DNB acceptance criterion in SRP Section 4.4 to provide 95/95 assurance that the hot fuel rod in the core would not experience a DNB or transition condition during AOOs. In RAI 3-7443, Question 13, the staff asked the applicant for a justification, emphasizing that the approval of the TR for a given 95/95 DNBR limit would not imply its direct applicability to AOOs that would be separately reviewed under the APR1400 design control document (DCD) review of the thermal design and safety analysis, and that would require additional DNBR margin to cover the limiting transient. In its response to RAI 3-7443, Question 13 (Ref. 8), the applicant committed to modifying the TR to reflect the results of the KCE-1 CHF correlation application to the AOO analysis of APR1400, which will also be included in the corresponding sub-section of the APR1400 DCD Section 4.4. The applicant made an additional commitment to include the supporting information in an updated version of Technical Report APR1400-F-C-NR-12001, “Thermal Design Methodology” (Ref. 35). The applicant maintained the same response in the updated response to RAI 3-7443, Question 13 (Ref. 11). Because the subject review of AOOs is performed under DCD Section 4.4, and because of the above commitments, the staff concludes that the applicant’s response is acceptable, and **RAI 3-7443, Question 13, is resolved and closed**.

As identified in Figure 4-1, “System Pressure versus Average Heat Flux of Test Section,” of the TR, [

][TS] in Figure 4-3, “Inlet Mass Flux versus Average Heat Flux of Test Section,” were also excluded by the applicant for the same reason. The staff considers that this has effectively reduced the application domain of the KCE-1 correlation to a maximum 2,415 psia (16.65 MPa) pressure and 3.15 $Mlb_m/hr\text{-}ft^2$ (4,272 kg/s-m^2) inlet mass flux. In RAI 3-7443, Question 18, the staff asked about the technical bases for selecting the applicable ranges, and whether the applicant envisions the actual PLUS7 design exceeding these ranges in any circumstances. In its response to RAI 3-7443, Question 18 (Ref. 11), the applicant provided the design analysis range of the APR1400 in Table 18-1 along with the applicable range of the KCE-1 correlation. The applicant stated that all APR1400 design/safety analyses are performed within the applicable range of the KCE-1 correlation at the MDNBR location. Among the variables included in Table 18-1, “Range of AOO Design Analysis for APR1400,” the APR1400 design analysis range is within the applicable range of the KCE-1 correlation for pressure and local quality. For inlet mass flux, the upper limit of the KCE-1 correlation range is slightly lower than the APR1400 design analysis range. However, the local

mass flux at the location of MDNBR in actual core analyses is still within the applicable range of the KCE-1 correlation. The higher pressure drop due to higher power and higher quality of the hot subchannel, where calculated DNBR is the minimum, reduces the flow by redistributing it from the hot subchannel to surrounding subchannels. Even though the excluded data were [

]^{TS}, Figures 18-1, "M/P Trend vs. System Pressure (with Excluded Data due to High Pressure and High Mass Flux)," and 18-2, "M/P Trend vs. Local Mass Flux (with Excluded Data due to High Pressure and High Mass Flux)," were reviewed by the staff and clearly show that the M/P behavior of the excluded points is comparable to the data within the applicable range, as shown in for pressure and mass flux, respectively. The staff considers **RAI 3-7443, Question 18, to be resolved and closed**.

5.0 CONDITIONS AND LIMITATIONS

Based on the foregoing technical and regulatory considerations, the NRC staff concludes that the use of the KCE-1 CHF correlation with a DNBR limit of 1.124 is acceptable for PLUS7 fuel thermal-hydraulic performance and plant safety analyses, provided that the following conditions are met:

1. The KCE-1 CHF correlation shall not be used outside its range of applicability defined by the range of the test data over which it was validated and found to behave in a consistent manner. The approved range for the KCE-1 CHF correlation is defined in the following table:

Parameter	British Units	SI Units
System Pressure	1,750–2,415 (psia)	12.07–16.65 (MPa)
Local Mass Flux	0.85–3.15 ($Mlb_m/hr\text{-}ft^2$)	1,153–4,272 ($kg/s\text{-}m^2$)
Local Quality	-0.15–0.275	

   The staff has modified the applicable pressure range from 9.62–16.65 MPa (1,395–2,415 psia) to 12.07–16.65 MPa (1,750-2,415 psia), because of non-conservative data at lower pressure ranges, as discussed in Section 4.8.1. Further application of any other CHF correlation within or outside the approved range tabulated above for the PLUS7 fuel design would have to be reviewed by the NRC as a part of a DCD or combined license application review, or revision of the TR.

2. The KCE-1 CHF correlation application is approved following the documented specific use of the correlation with Tong factor values that are [ ]^{TS}.

3. The KCE-1 CHF correlation shall be used with the TORC subchannel computer code using the models and parameters specified in the TR. The KCE-1 CHF correlation is dependent on local fluid conditions that shall be calculated by the version of the TORC computer code that was used for the TR. Further application of the KCE-1 CHF

correlation with any other subchannel computer code would require additional NRC review and approval.

4. Modifications to the KCE-1 CHF correlation, its applicability range, or the associated DNBR limit of 1.124 would require additional NRC review and approval.

The NRC staff will require licensees and applicants referencing this TR in licensing applications to document how these conditions are met.

6.0 CONCLUSIONS

Based on its review of Topical Report "KCE-1 Critical Heat Flux Correlation for PLUS7 Thermal Design," APR1400-F-C-TR-12002-P, Revision 0, the NRC staff has reasonable assurance that the use of the KCE-1 CHF correlation is acceptable in calculating the CHF for PLUS7 fuel design, provided that the conditions and limitations specified in Section 5.0 of this SER are met. These conditions and limitations were identified by the staff to address all outstanding technical issues and to document their closure in this SER. Licensees referencing the TR will be required to ensure compliance with these conditions and limitations. Because of the staff's review, all RAI questions are considered closed and resolved. The applicant is expected to update the TR to incorporate the mark-ups of the proposed changes submitted with various RAI responses.

When exercised appropriately, the staff finds the KCE-1 CHF correlation methods described in the TR to be applicable to the PLUS7 fuel thermal-hydraulic performance and plant safety analyses. Considering the overall quality of the data presented and analyses performed, the staff concludes that sufficient inherent conservatism is built into the KCE-1 CHF correlation to more than make up for all the non-conservatisms identified by the staff. The proposed DNBR limit of 1.124 for the KCE-1 CHF correlation provides reasonable assurance that GDC 10 and the SRP Section 4.4 acceptance criterion regarding the evaluation of fuel design limits have been met, and there is at least a 95-percent probability at the 95-percent confidence level that the hot fuel rod in the core would not experience a DNB or transition condition during normal operation.

The NRC staff has reviewed the KCE-1 CHF correlation, and does not intend to review the associated TR when referenced in licensing evaluations. The NRC staff's review was based on the evaluation of the technical merit of the submittal and its compliance with the applicable regulations. If the NRC's regulations or acceptance criteria change such that the conclusions regarding the acceptability of the thermal-hydraulic methods or statistical analyses present in this TR are invalidated, the licensee or applicant referencing the TR will be expected to revise and resubmit its documentation, or submit justification for the continued effective applicability of these methods without revising the respective documentation.

7.0 REFERENCES

1. Submittal letter dated January 7, 2013, "Transmittal of Topical Report APR 1400-F-C-TR- 12002-P/-NP, Rev. 0, 'KCE-1 Critical Heat Flux Correlation for PLUS7 Thermal

Design' for Safety Evaluation, dated November 2012," (Agencywide Documents Access and Management System (ADAMS) Accession No. ML13018A096).

2. Topical Report APR1400-F-C-TR-12002-NP, Revision 0, "KCE-1 Critical Heat Flux Correlation for PLUS7 Thermal Design," November 2012 (ADAMS Accession No. ML13018A147).

3. Non-Public Proprietary Request for Additional Information (RAI) RAI 3-7443 for the APR1400 Topical Report APR1400-F-C-TR-12002-P, "KCE-1 Critical Heat Flux Correlation for PLUS7 Thermal Design," to KHNP, March 25, 2014. Proprietary Information. Not publicly available.

4. "Summary of the April 30, 2014, through May 2, 2014, Public Meetings with Korea Hydro and Nuclear Power Co. Ltd. to Discuss Various Topics Related to the Advanced Power Reactor 1400 Design," June 16, 2014 (ADAMS Accession No. ML14154A030).

5. APR1400-F-C-RA-14001-NP, KEPCO/KHNP Presentation, "KCE-1 Critical Heat Flux Correlation for PLUS7 Thermal Design (APR1400-F-C-RT-12002)," May 1, 2014 (ADAMS Accession No. ML14132A133).

6. Enclosure 1, KHNP Response to RAI 3-7443 on Topical Report, "KCE-1 Critical Heat Flux Correlation for PLUS7 Thermal Design," APR1400-F-C-TR-12002-P, Rev. 0, Questions 2, 3, and 5; April 23, 2014 (ADAMS Accession No. ML14114A562).

7. Enclosure 3; KHNP Response to RAI 3-7443 on Topical Report "KCE-1 Critical Heat Flux Correlation for PLUS7 Thermal Design" APR1400-F-C-TR-12002-P, Rev. 0, Questions 4, 10, 11, 12, and 18, May 23, 2014 (ADAMS Accession No. ML14143A115).

8. Enclosure 3; KHNP Response to RAI 3-7443 on Topical Report "KCE-1 Critical Heat Flux Correlation for PLUS7 Thermal Design" APR1400-F-C-TR-12002-P, Rev. 0; Questions 1, 6, 7, 8, 9, 13, 14, 15, 16, and 17, June 2014 (ADAMS Accession No. ML14176A792).

9. Audit Plan to Review Selected Documents Related to APR1400 KCE-1 Critical Heat Flux Correlation for PLUS7 Thermal Design (APR1400-F-C-TR-12002-P-Rev 0), December 11, 2014 (ADAMS Accession No. ML16228A411).

10. U.S. Nuclear Regulatory Commission's Audit Report for the January 21-22, 2015 Audit to Review Selected Documents Related to KEPCO/KHNP Topical Report on KCE-1 Critical Heat Flux Correlation for PLUS7 Thermal Design (APR1400-F-C-TR-12002-P, Rev 0); issued June 9, 2015 (ADAMS Accession No. ML16228A418).

11. Revised KHNP Response to RAI 3-7443 on Topical Report "KCE-1 Critical Heat Flux Correlation for PLUS7 Thermal Design," APR1400-F-C-TR-12002-P, Rev. 0, March 2, 2015 (ADAMS Accession No. ML15061A056).

12. Summary of Public Meeting With Korea Hydro and Nuclear Power Co. LTD. to Discuss Topics Related to the Advanced Power Reactor 1400 Design (ADAMS Accession No. ML15259A459).

13. Revised Response to RAI 3-7443 on Questions 6, 7, 9, and 17, October 9, 2015 (ADAMS Accession No. ML15282A603).

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15. Title 10 of the *Code of Federal Regulations,* Part 50, "Domestic Licensing of Production and Utilization Facilities."

16. NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants." (ADAMS Accession No. ML070660036)

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35. Korea Electric Power Corporation & Korea Hydro & Nuclear Power Co., Ltd, "Thermal Design Methodology," Technical Report APR1400-F-C-NR-12001-NP, Rev. 1, November 2014 (ADAMS Accession No. ML15009A123).