UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, DC 20555 - 0001

February 23, 2017

Mr. Victor M. McCree
Executive Director for Operations
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

SUBJECT: SAFETY EVALUATION OF APR1400 TOPICAL REPORT, "KCE-1 CRITICAL HEAT FLUX CORRELATION FOR PLUS7 THERMAL DESIGN"

Dear Mr. McCree:

During the 640th meeting of the Advisory Committee on Reactor Safeguards, February 9-11, 2017, we met with representatives of the NRC staff, Korea Electric Power Corporation (KEPCO), and Korea Hydro & Nuclear Power Company, Ltd., (KHNP) to review the advanced topical report safety evaluation (ATRSE) of topical report, "KCE-1 Critical Heat Flux Correlation for PLUS7 Thermal Design." Our APR1400 Subcommittee reviewed these and the other referenced documents during a meeting on December 14, 2016. This topical report was submitted by KHNP in conjunction with its affiliate company, KEPCO, in support of APR1400 design certification.

CONCLUSION

There is reasonable assurance that the use of the KCE-1 critical heat flux correlation is acceptable in calculating the critical heat flux for the PLUS7 fuel design, provided the conditions and limitations identified by the staff are met.

BACKGROUND

The topical report was submitted to justify the use of the KCE-1 critical heat flux (CHF) correlation for PLUS7 fuel in pressurized water reactor (PWR) applications. The PLUS7 fuel, which was jointly developed by Westinghouse Electric Corporation and KEPCO, includes features to promote mixing and improve heat transfer between the fuel and the coolant, increasing the thermal margin for departure from nucleate boiling or CHF, in a PWR.

DISCUSSION

In the ATRSE, the staff concludes that there is reasonable assurance that the use of the KCE-1 correlation is acceptable for calculating CHF for the PLUS7 fuel design, provided several conditions and limitations are met. The staff developed these limitations based on their review of data from CHF testing, the correlation development, and the computer code for applying this correlation. We concur with the staff assessment as discussed below.

CHF Testing

CHF tests were conducted in 2001 at the Columbia University Heat Transfer Research Facility (HTRF) to verify the thermal performance of the PLUS7 fuel. From the 1950's through 2003, when it was closed, the HTRF was used to develop extensive CHF data relevant to nuclear fuel designs. Since the CHF tests were conducted for safety-related design purposes, the HTRF staff qualified the engineering design and materials supplied to ensure that the CHF data conformed to applicable requirements of 10 CFR Part 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants."

Two 6 x 6 electrically-heated prototypic rod array test sections were fabricated to simulate PLUS7 fuel for the CHF test program: one included a central control rod guide tube and the other did not. The axial power distribution of the heated rods was a fixed, symmetric cosine shape with prototypical radial power peaking conditions. Prototypic split type mixing vane spacer grids at scale axial spacing were used to simulate actual fuel rod bundle fluidic conditions important to developing the CHF correlation.

CHF test data points were collected for the two test sections. The applicable range of parameters tested for development of the correlation was defined to include the expected range of normal operation of APR1400, including transients, in terms of pressure, local mass flux, and local steam quality. The NRC-approved TORC thermal-hydraulic computer code was used to calculate local mass flux and quality for the test conditions. Pressure and temperature measurements were made at the beginning and end of the heated rod length. The CHF point was confirmed to occur during testing when incrementally increasing power led to sudden temperature excursions of 10 to 30°F, as measured by thermocouples internal to the heater rods (located axially just below the top seven mixing vane grids). All CHF locations measured were detected by thermocouples located in the upper half of the rod bundles above the middle of the heated length.

The HTRF data analysis includes a heat loss correction based on experimental data because heat losses from the test sections would artificially increase the estimated CHF. Additional conservatisms were added in the data analysis to generate the KCE-1 correlation parameters. The staff reviewed these added conservatisms and found them acceptable.

KCE-1 CHF Correlation and Departure from Nucleate Boiling Ratio Limit Development

The KCE-1 correlation is an empirical fit to measured CHF data as a function of local fluid properties. The KCE-1 functional form is based on the approved CE-1 correlation, which has been used to analyze U.S. power reactors since 1979. The KCE-1 parameters are applicable to the PLUS7 fuel design. The staff reviewed KHNP's data analysis and found it acceptable.

Customarily, correlations are developed using a training dataset, and are validated against a separate dataset, which is preferably a completely independent set of measurements. For the KCE-1 development, only one dataset was generated and all the experimental data were used to define the correlation parameters. Statisticians often refer to this process as over-fitting. KHNP and the staff independently evaluated the impact of over-fitting on the KCE-1 correlation by randomly removing data subsets from the training dataset and using them for validation. The staff determined that over-fitting would account for less than a 5% non-conservative decrease in the CHF ratio, which is smaller than their estimate of the conservatism introduced by other assumptions. Therefore, the staff found the KCE-1 development process acceptable.

Data for generation of the KCE-1 correlation were collected at discrete pressure values. The correlation shows good agreement at high pressures, but a non-linear drift is apparent at lower pressures. The staff was concerned that the non-linear drift in the correlation results could underpredict CHF in the lower pressure range. Therefore, the staff imposed a limitation of applicability of the KCE-1 correlation so that it may not be used at pressures lower than 1750 psia. The staff performed a comprehensive analysis of correlation trends and drift for other parameters (specifically local mass flux and local quality) and found no discernible trends. However, the staff imposed a limitation to the range of applicability to remain within the bounds for which experimental data were obtained.

KCE-1 Correlation Use

The KCE-1 correlation is to be applied with the TORC code in thermal design and safety analyses of a PWR core containing PLUS7 fuel. TORC was used to calculate the coefficients for this CHF correlation. This code is used to perform detailed modeling of a PWR core and the hottest assembly and to determine the minimum departure from nucleate boiling ratio in the hottest assembly. Because other calculational tools may have different transport properties and numerical schemes, the staff imposed a limitation that the KCE-1 correlation shall be used with TORC using the models and parameters specified in the topical report.

We note that numerous incremental changes, such as programming-error corrections or adjustments for operating system changes, could have been implemented during the thirty years since the TORC code was approved. In their review for the APR1400 effort, the staff completed an audit that resolved their concerns about the effects of incremental changes in TORC.

SUMMARY

We concur with the staff conclusion that there is reasonable assurance that the use of the KCE-1 CHF correlation is acceptable in calculating the CHF for the PLUS7 fuel design, provided the conditions and limitations identified by the staff are met.

Sincerely,

/RA/

Dennis C. Bley
Chairman

REFERENCES

1. U.S. Nuclear Regulatory Commission, "Advanced Power Reactor 1400, Advanced Topical Report Safety Evaluations for the 'KCE-1 Critical Heat Flux Correlation for Plus7 Thermal Design' and the 'Fluidic Device Design for the APR1400'," October 25, 2016 (ML16230A224).

2. U. S. Nuclear Regulatory Commission, "Advanced Topical Report Safety Evaluation (ATRSE) by the Office of New Reactors of Topical Report APR1400-FC-TR-12002-P, Revision 0, 'KCE-1 Critical Heat Flux Correlation for PLUS7 Thermal Design' Korea Hydro & Nuclear Power Co., Ltd. Project No. PROJ0782," August 12, 2016 (ML16288A758).

3. Korea Electric Power Corporation and Korea Hydro & Nuclear Power Company, Ltd., Topical Report APR1400-F-C-TR-12002-P, "KCE-1 Critical Heat Flux Correlation for PLUS7 Thermal Design," November 2012 (ML13018A158).

4. Korea Electric Power Corporation and Korea Hydro & Nuclear Power Company, Ltd., Topical Report APR1400-F-C-TR-12002-NP, "KCE-1 Critical Heat Flux Correlation for PLUS7 Thermal Design," November 2012 (ML13018A147).

5. Korea Electric Power Corporation and Korea Hydro & Nuclear Power Company, Ltd., Technical Report APR1400-F-C-NR-12001-NP, "Thermal Design Methodology," Revision 1, November 2014 (ML15009A123).

6. Korea Electric Power Corporation and Korea Hydro & Nuclear Power Company, Ltd., "Korea Electric Power Corporation and Korea Hydro & Nuclear Power Co., Ltd Application for Design Certification of the APR1400 Standard Design," December 23, 2014 (ML15006A098).

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1. U.S. Nuclear Regulatory Commission, "Advanced Power Reactor 1400, Advanced Topical Report Safety Evaluations for the 'KCE-1 Critical Heat Flux Correlation for Plus7 Thermal Design' and the 'Fluidic Device Design for the APR1400'," October 25, 2016 (ML16230A224).

2. U. S. Nuclear Regulatory Commission, "Advanced Topical Report Safety Evaluation (ATRSE) by the Office of New Reactors of Topical Report APR1400-FC-TR-12002-P, Revision 0, 'KCE-1 Critical Heat Flux Correlation for PLUS7 Thermal Design' Korea Hydro & Nuclear Power Co., Ltd. Project No. PROJ0782," August 12, 2016 (ML16288A758).

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