COMPUTER CODES PROGRAMS AS PART OF THE APR1400 DESIGN CONTROL DOCUMENT AUDIT REPORT

Korea Hydro and Nuclear Power Co., Ltd. (KHNP) and Korea Electric Power Corporation (KEPCO)

APR1400 DESIGN CERTIFICATION
Docket No. 52-046

NRC Audit Team:

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1.0 SUMMARY

On June 29, 2015, through July 2, 2015, the U.S. Nuclear Regulatory Commission (NRC) conducted an audit at the Westinghouse Office in Rockville, Maryland. The audit was to confirm that the design calculations performed in support of the Advanced Power Reactor 1400 (APR1400) design certification (DC) application are consistent with the descriptions of the computer codes in the design control document (DCD) Tier 2, Section 3.9.1, "Special Topics for Mechanical Components."

2.0 BASIS

To support the NRC staff's review of the information in the DCD, as described in Title 10 of the *Code of Federal Regulations* (10 CFR) 52.47, the NRC staff audited detailed verification and validation (V&V) reports associated with the computer programs used for APR1400 DC, as listed in DCD Tier 2, Subsections 3.9.1.2.1, including the pre- and post-processors used for the analyses. The review was to verify that design analyses follow the requirements of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (BPV) Code, as required by 10 CFR 50.55(a), and are consistent with the descriptions in the DCD.

3.0 Objectives:

The objectives of this audit were to:

- Review individual program documents that describe author, source code, executable file(s), input deck(s), dated version, and user's manual and theoretical formulations.
- Review individual program verification reports.
- Review individual program flow chart logic.
- Review individual program V&V benchmark packages.
- Review individual program input and output data and program limitations.

- Review quality assurance (QA) procedures for control and maintenance of program.

4.0 Observations and Results:

Computer Codes Programs

The NRC staff reviewed the V&V packages of the applicant's in-house and commercial codes listed in DCD Tier 2, Subsections 3.9.1.2.1 that were used as part of the APR1400 design process.

Documentation for in-house and commercial codes identified in the audit plan was available and reviewed, as following:

- Individual program documentation.
- Individual program flow charts.
- Individual program verification reports.
- Individual program V&V benchmark packages.
- Individual program input and output data and program limitations.
- QA procedure for control and maintenance of computer programs.

1. Doc No. 00000-AM-VV-031, "ASHSD Software Verification and Validation Report," Revision 00 dated December 31, 2012:

   The ASHSD computer code is used in the structural modal analysis to obtain the dynamic response of the core support barrel under normal operating conditions and a loss of coolant accident. The program yields the dynamic shell and beam mode response of the structural system. The ASHSD computer code was migrated from the UNIX operating system to the Windows operating system. ASHSD V&V calculation was documented in Doc. No. 0000-D1-VV-028, "Software Verification and Validation Report – ASHSD Version 1 Modification 4," Revision 00, dated August 20, 2007. The NRC staff noted that a record of revisions was not included in Doc No. 00000-AM-VV-031, "ASHSD Software Verification and Validation Report," Revision 00. The applicant was requested to include the record of revisions into the Doc No. 00000-AM-VV-031, "ASHSD Software Verification and Validation Report," Revision 00. Subsequently, the applicant will update Doc No. 00000-AM-VV-031, ASHSD "Software Verification and Validation Report," Revision 00 and provide the updated documents for the staff's review in the follow-up audit. The NRC staff found that the applicant's plan to provide updated V&V documents is acceptable.

2. Doc No. 00000-AM-VV-034, "CESHOCK Software Verification and Validation Report," Revision 00 dated December 31, 2012:

   The CESHOCK program is used to obtain the transient response of the reactor internals and fuel assemblies due to pipe break and seismic loads. The V&V calculations were documented in V&V report, Doc. No. 3L186-ME-CA240-26, "Vertical Pipe Break Analysis of Reactor Internal," Revision 02, dated May 17, 2011, and Doc. No. 0000-VA-VV-027, "Software Verification and Validation Report – CESHOCK Version 6

Modification 5" Revision 00, dated January 2012. The applicant was requested to have these V&V calculation packages to be translated into the English language for NRC staff's reviews. Subsequently, the applicant provided these V&V reports in an English version for the staff's review. The NRC staff found that a record of revisions was not included in Doc No. 00000-AM-VV-034, "CESHOCK Software Verification and Validation Report" Revision 00, dated December 31, 2012. The applicant committed to include the record of revisions into the Doc No. 00000-AM-VV-034, "CESHOCK Software Verification and Validation Report," Revision 00 and will provide the updated documents for the staff's review in the follow-up audit. The NRC staff found that the applicant's plan to provide the updated V&V documents is acceptable.

3. Doc. No. ND-G-CV-033, "HeadPR Version 1," Revision. 2, October 2014:

The HeadPR computer code calculates the available reinforcement and the reinforcement that is needed for penetrations in the hemispherical heads. The calculations included:

- Calculates penetration reinforcement for spherical heads.
- Calculates minimum area necessary to provide sufficient reinforcement.
- Calculates maximum area of reinforcement using ASME Section III requirements.

The HeadPR computer code is verified by compared with the results obtained using the personal computer program Mathcad and Ulchin 5/6 calculation number UE-111CN-110 "Tentative Sizing of Bottom Head and Closure Head," Revision 1. The staff found the verification for HeadPR computer code is consistent with ASME BVP Section III, "Rules for Construction of Nuclear Facility Components-Division 3-Containments for Transportation & Storage of Spent Nuclear Fuel & High Level Radioactive Material & Waste," requirements.

4. Doc. No. "Verification of NOZPROG (Version 1)" Revision 10:

The NOZPROG computer code calculates the maximum stress intensities developed in nozzles and nozzle-vessel intersection due to combinations of various external pipe loads under pressure. Nozzle general membrane stresses are also calculated with the equation for a beam with hollow cylinder cross section. The stress results of NOZPROG computer code are verified by compared with the stress results obtained by hand calculations. The staff found that the verification for NOZPROG met the ASME BPV Section III requirements and is acceptable.

5. Doc. No. ND-G-CV-019, AFP2D "Verification of AFP2D (Version 3)" Revision 8, dated December 2014:

The AFP2D computer code is used to obtain the fatigue evaluation for pressure vessels in accordance with ASME Code, Section III. The program uses 2-D axisymmetric models to perform fatigue calculations in accordance with ASME Subsection NB-3222.4 (e) and to evaluate simplified elastic-plastic analysis as stipulated in ASME Subsection NB-3228.5 for nozzles and shell structures. The flow chart of AFP2D computer code included:

1. Read linearized stress produced by “PRSECT” commend of ANSYS.
2. Calculated the range of primary plus secondary stress intensity.
3. Provides the stress table of primary plus secondary stress.
4. Calculates the range of total stress intensity.
5. Provides the stress table of total stress.
6. Perform fatigue analysis in accordance with ASME Subsection NB-3222.4.

The program was verified by comparisons of program results and hand-calculated solutions. In review of V&V reports, the staff observed that the presented cut stress plane and mesh density size of finite element analysis (FEA) models did not accurately define the cut stress plane. In general, the cut stress plane is performed at the location where the maximum stress intensity occurred in the FEA models to calculate the maximum stress intensity ranges for the fatigue evaluation. The cut stress plane of presented FEA model in V&V report appeared to be arbitrary cuts, where maximum stress intensity may not occur at this location. To validate the cut stress plane and its location, sensitivity sturdies of the cut stress plane and mesh density size of the FEA model should be performed to ensure that the accurate solutions will be obtained. As a result of this observation, the applicant committed the sensitivity reports pertaining to the cut stress plane for fatigue calculation and mesh density size modeling for the staff’s review in the follow-up audit. The NRC staff found that the applicant’s proposed response to the audit observation is acceptable.

6. Doc. No. ND-G-CV-027, “Verification of AFPOST (Version 2),” Revision 7:

The AFPOST computer code is used to obtain the fatigue evaluation for pressure vessels in accordance with ASME Code, Section III, Subsection NB. The AFPOST computer code combines thermal stresses resulting from ANSYS run, calculates primary plus secondary stresses, total maximum stress intensity and their stress ranges, performs fatigue evaluation in accordance with ASME Section III, Subsection NB 3222.4 (2). The AFPOST computer code has the capabilities to perform fatigue calculation for 2-D axisymmetric models and 3-D models.

The program was verified by comparisons of program results and hand-calculated solutions. Similarly to the staff’s observation related to the AFPOST computer code, the staff found that the presented cut stress plane and mesh density size in the FEA model in V&V report were not fully validated. To validate the cut stress plane and its location, sensitivity sturdies of the cut stress plane and mesh density size of the FEA model should be performed to ensure that the accurate solutions will be obtained. The applicant will provide the sensitivity reports pertaining to the cut stress plane for fatigue calculation and mesh density size modeling for the staff’s review in the follow-up audit. The NRC staff found that the applicant’s proposed response is acceptable.

7. Doc. No. ND-G-CV-018, “Verification of TSPOST Code,” Revision 4:

The TSPOST computer code calculates various ASME Section III stress intensities, stress ranges and fatigue usage factors of the perforated tube sheet that are designed to accordance with ASME Section III, Appendix A-8000. The verification by comparing results of TSPOST with those of CE223, as documented in “Computer Code CE223, Tubesheet/Primary Structure Program, CE223 User Information, by N. L. Beard,

Combustion Engineering Inc," dated July 1987. The NRC staff found that the verification for TSPOST met the SRP Section 3.9.1 guidance and is acceptable.

In general, the staff observed that all V&V reports, except ASHSD and CESHOCK V&V reports, did not reference the QA procedure into the reports. The applicant committed to reference the QA procedures into all V&V reports and will provide the updated documents for the staff's review in the follow-up audit.

QA Procedures

The applicant also provided QA procedures for the implementation of ASME Nuclear Quality Assurance (NQA-1) requirements of the DCD of APR1400 computer codes. The NRC staff reviewed the following QA procedures in the audit:

1. KHNP, DC – DG – 03 – 15, APR1400 project internal procedure, Revision 1, dated September 2014.

   The procedure is used to control the internal software requirements for internal software control.

2. KEPCO E&C Design Control, "APR1400 DC Quality Assurance Manual" Section 3, Revision 1, date February 03, 2014.

   The manual section describes the QA requirements and responsibilities to ensure that design activities are adequately controlled in accordance with the requirements of applicable regulations, codes, and standards for the early stages to the final stages of design.

3. KEPCO NSSS, "Engineering Procedure Computer Software," EP 5.13 Revision 14, dated March 31, 2015.

   The procedure specifies the requirements for the development of procedures maintenance use and control for security of computer software utilized in relation to NSSS activities.

4. KEPCO E & C, APR 14 DC Quality Assurance Manual Design Control, SEC 3, Revision 3, dated February 03, 2014.

   The procedure specifies the requirements for the development maintenance use and security of computer software utilized in the design activities of Nuclear Steam Supply System (NSSS), including software requirements software design implementation testing installation and check out, operation and maintenance, retirement phase, software V&V, validation software configuration management, change control, problem reporting and corrective action, access control, procured software, approvals of software, storage and record requirements. The procedure was used in the design and verification of the design inputs, outputs, and methodologies for the ASHSD, and CESHOCK programs.

5. DOOSAN NQCP – 0303, "Nuclear BG Quality Control Procedure NQC P – 300," Revision 10, dated May 20, 2015.

The procedure provides the detailed methods to be implemented and validated the computer codes prior the computer programs are used or changed in the APR1400 design. The procedure also provides the mechanism to control the use of computer software to assure that they are used for appropriate applications.

6. DOOSAN QMBC – 100, "Quality Assurance Manual for a P 1400 Design Certification," Revision 4, dated March 5, 2015.

   The staff observed that the manual covers design responsibilities, control of design inputs, control of design interfaces, control of design processes, control design outputs, design analysis, design verification, independent reviews, alternate calculations, quality testing, design change control, and design documents and records.

7. DOOSAN NQCP – 0301, "Nuclear Quality Control Procedure, Design Control," Revision 22, dated January 22, 2015.

   The staff reviewed sections A6 .0, "Control Design Inputs," A7.0, "Control of Design Process," and A8.0, "Design Verification" and found that the procedure is prepared in accordance with ASME NQA-1 requirements. The staff observed that the procedures also described the verification of computer programs in the control of computer program errors. These procedures were used in conjunction to provide controls for the NOZPROG, TSPOST, Head PR, AFP2D, and AFPOST software inputs, outputs, and methodologies.

Conclusion:

This summary audit report is to be referenced in Section 3.9.1 of the FSER for the AP1400 DCD review. Based on the audit observations, the applicant committed to provide the updated documents for the staff's confirmation in a future follow-up audit:

- The applicant will revise the record of revision for the ASHSD and CESHOCK software V&V reports.
- The applicant will provide two sensitivity reports pertaining to the cut stress plane for fatigue calculation and mesh density size modeling.
- The applicant will provide procedures as references in the calculations and V&V reports.

The NRC staff will conduct a follow-up audit of the revised documents when KHNP notifies the NRC staff that the revised documents are available in the KNHP electronic reading room.

References:

1. 10 CFR Parts 50, "Domestic Licensing of Production and Utilization Facilities," and 10 CFR Part 52 "Licenses, Certifications, and Approvals for Nuclear Power Plants."

2. NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition”, Chapter 3, “Design of Structures, Components, Equipment, and Systems.”

3. KHNP APR1400 Design Control Document – Tier 2 Chapter 03 – Design of Structures, Systems, Components, and Equipment (ML15006A059).