Audit Summary
Non-LOCA Chapter 15 Audit of the APR1400 Design Certification Document

A. Background

On March 5, 2015, the U.S. Nuclear Regulatory Commission (NRC) accepted the design certification application for docketing for the Advanced Power Reactor 1400 (APR1400) submitted by Korea Hydro and Nuclear Power Co., Ltd. (KHNP) (Reference 1). The staff initiated Phase 1 of the application design certification review on March 9, 2015. The purpose of the audit was to gain a better understanding of the calculations supporting the safety analyses presented in Chapter 15 of the APR1400 Design Certification Document (DCD). This audit summary is accompanied by an audit plan (Reference 2).

B. Regulatory Audit Bases

This regulatory audit was based on the following:

- APR1400 DCD Chapter 15
- NUREG-0800, Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition, Chapter 15

C. Logistics

The audit was conducted from NRC headquarters via KHNP's electronic reading room. The electronic reading room was maintained by CERTREC.

Date: June 10, 2015 – June 16, 2016

Location: NRC Headquarters
Two White Flint North
11545 Rockville Pike
Rockville, MD 20852-2738

D. Audit Team Members

The following NRC staff members participated in substantive discussions during the audit:

Don Carlson, Technical Reviewer
Matt Thomas, Technical Reviewer
Timothy Drzewiecki, Technical Reviewer
Jim Steckel, Project Manager

The NRC staff was augmented with contractors. These additional participants include the following:

Information Systems Laboratories, Inc.
Doug Barber
James Servatius

E. Applicant and Industry Staff Participants

KHNP
Jiyong Oh

F. Documents Audited

The list of documents audited is organized by relevant DCD sections below:

DCD Section 15.0.2

- CCVR-TH-02-02, Revision 0, "Computer Code Verification Report of CETOP," January 30, 2002.
- NPSD-150-P, Revision 4, "CETOP: Thermal Margin Model Development," April 2003.
- 00000-SS-VV-030, Revision 00, "Software Verification and Validation Report of CESEC-III 89300 MOD5CS," June 14, 2015.
- VV-FE-0416, "Software Verification and Validation Report – HERMITE Rev 1.6 Mod 0," February 2, 1998.

DCD Sections 15.1.1-15.1.4

- 3L186-SA-CA070-018, Revision 0, "Inadvertent Opening of Steam Generator Atmospheric Dump Valve Event Analysis for SKN 3&4 FSAR," January 26, 2011.

DCD Section 15.4.1

- APR1400-F-A-TM-12037-P, Revision 0, "Bank CEA Withdrawal from Subcritical or Low Power Analysis for US-APR1400," July 15, 2012.

DCD Section 15.4.2

- APR1400-F-A-TM-12036-P, Revision 0, "Bank CEA Withdrawal at Power Analysis for US-APR1400", July 15, 2012.

DCD Section 15.4.3

- APR1400-F-A-TM-12004-P, Revision 0, "CEA Drop Analysis for USAPR1400," August 31, 2012.

DCD Section 15.4.4

- APR1400-F-A-TM-12035-P, "Startup of Inactive Reactor Coolant Pump (RCP) for the US-APR1400," June 29, 2012.

DCD Section 15.4.8

- APR1400-F-A-TM-12009-P, Revision 0, "APR1400 NRC DC – Barrier Performance Analysis for CEA Ejection Accident," September 2, 2012.
- APR1400 -F-A-TM-12009-P, Revision 1, "APR1400 NRC DC – Barrier Performance Analysis for CEA Ejection Accident," June 4, 2013.
- APR1400 -F-A-TM-12011-P, Revision 0, "APR1400 NRC DC – CEA Ejection Accident Analysis for Fast Trip DNBR Case," July 9, 2012.
- APR1400 -F-A-TM-12011-P, Revision 1, "APR1400 NRC DC – CEA Ejection Accident Analysis for Fast Trip DNBR Case," June 19, 2013.
- APR1400 -F-A-TM-12012-P, Revision 0, "APR1400 NRC DC – Enthalpy Case Analysis for CEA Ejection Event," November 17, 2012.
- APR1400 -F-A-TM-12012-P, Revision 1, "APR1400 NRC DC – Enthalpy Case Analysis for CEA Ejection Event," May 22, 2013.

DCD Section 15.6.2

- 3L186-SA-CA070-023, Revision 0, "Double-Ended Break of a Letdown Line Outside Containment Upstream of the Letdown Line Isolation Valve for SKN 3&4 FSAR," January 31, 2011.

DCD Section 15.6.3

- 3L186-SA-CA070-024, Revision 0, "Steam Generator Tube Rupture with and without a Loss of Offsite Power Analysis for SKN 3&4," January 14, 2011.

G. Description of Audit Activities and Summary of Observations

Audit activities and observations are organized by the applicable section of the DCD and are summarized below.

DCD Section 15.0.2

The audit team examined CCVR-TH-02-02, Revision 0 and made the following observation. This document contains source code that implements the KCE-1 critical heat flux (CHF) correlation into CETOP. The CE-1 and KCE-1 CHF correlation equation, definition of variables, and values of coefficients are provided in the document. The form of the CHF correlation did not change; only the values for the coefficients changed. The table below shows the coefficients for the older CE-1 CHF correlation and the new KCE-1 CHF correlation. The source code also showed an additional change to the thermal diffusion coefficient for PLUS7 fuel with mixing vane grids from 0.0035 to 0.0101. The document was examined to confirm that the CE-1 and KCE-1 correlations were correctly programmed into CETOP. The staff confirmed that

the KCE-1 equation and coefficients in the source code agree with the equation and values of coefficients on page 4-10 of topical report APR1400-F-C-TR-12002-P, Revision 0.

The audit team examined a software verification and validation report for the CESEC-III code. The document addressed changes to CESEC-III to model pilot operated safety relief valve (POSRV) and centrifugal charging pumps (CCP). This document was examined to confirm that KHNP properly verified and validated the changes to CESEC-III to model these new components. For APR1400, the charging pump of the chemical and volume control system has been changed from a positive displacement pump to a CCP and the pressurizer safety valve on the OPR1000 plant is replaced with a pressurizer POSRV. The input required to activate the new models were examined along with the required input to specify the characteristics of the POSRVs and CCPs such as back pressure, flow versus backpressure, valve stroke time, valve flow area versus time, and valve lift and reseat pressures. The testing of the programming changes to CESEC-III was also examined to ensure the models were working as programmed.

The audit team examined a software verification and validation report for the HERMITE code. This document addressed changes that allow for the modeling of transient system pressure. The addition of the transient pressure option required changes to HERMITE to allow for changing water properties as a function of timestep. This document was examined to confirm that KHNP properly verified and validated the changes to HERMITE to model the new transient pressure option. The input required to activate the new model was examined along with the testing of the programming changes to ensure the transient pressure option was working as programmed.

<u>DCD Sections 15.1.1-15.1.4</u>

NRC staff examined the calculation supporting DCD Section 15.1.4, “Inadvertent Opening of a Steam Generator Relief for Safety Valve.” In examining the calculation, NRC staff noted calculation input parameters, assumptions, and justifications. The NRC staff noted that the limiting case was determined by a parametric study. In particular, the NRC staff noted that the plant initial conditions of reactor coolant core inlet temperature, pressurizer level, and reactor coolant flow rate were varied over a suitable range. Additionally, the NRC staff noted that the

single-failure consideration of excessive feedwater flow had a negligible impact on the minimum departure from nucleate boiling ratio (DNBR).

DCD Section 15.4.1

The NRC staff examined the calculation supporting DCD Section 15.4.1, "Uncontrolled Control Rod Assembly Withdrawal from a Subcritical or Low Power Startup Condition." In examining the calculation, the NRC staff noted key calculation inputs and their sources. In particular, the NRC staff noted:

1. The total delayed neutron fraction was taken from the cycle minimum value, occurring at end of cycle (EOC) conditions. The assumed neutron lifetime and the average decay constants were likewise taken from EOC conditions. These parameters were obtained from an engineering design data document.
2. The most positive moderator temperature coefficient (MTC) and least negative fuel temperature coefficient (FTC) were utilized. These parameters were obtained from an engineering design data document.
3. A bottom peaked axial power shape is used for the scram reactivity as obtained from an engineering design data document.
4. Reactivity for the withdrawn bank is inserted over a period of 0.71 seconds.
5. The analog variable overpower trip (VOPT) step setpoint is used for this event, but conservatively chosen to be 25 percent instead of 14 percent.
6. A conservative scram worth was chosen consistent with the minimum worth at low power.
7. Once the control element assembly (CEA) reinsertion begins as a result of the reactor scram, the reactivity for the withdrawn bank is removed over a period of 0.71 seconds, similar to its insertion rate.
8. The average linear heat generation rate (LHGR) was obtained from an engineering design document. This LHGR was multiplied by the three-dimensional peaking factor and the peak power during the transient to obtain the peak LHGR (PLHGR).
9. The position of highest pressure in the reactor coolant system (RCS) is at the RCP discharge. As such, the peak pressure is calculated by adding the RCS pressure (pressurizer) and the pressure drop for the pump (between cold leg at safety injection and surge line).

DCD Section 15.4.2

The NRC staff examined the calculation supporting DCD Section 15.4.2, "Uncontrolled Control Element Assembly Withdrawal at Power." In examining the calculation, the NRC staff noted key calculation inputs and their sources. In particular, the NRC staff noted:

1. The total delayed neutron fraction was taken from the cycle minimum value, occurring at EOC conditions. The assumed neutron lifetime and the average decay constants were likewise taken from EOC conditions. These parameters were obtained from an engineering design data document.
2. The most positive MTC and least negative FTC were utilized. These parameters were obtained from an engineering design data document.

3. The average LHGR was obtained from an engineering design document. This LHGR was multiplied by the three-dimensional peaking factor and the peak power during the transient to obtain the PLHGR.
4. The position of highest pressure in the RCS is at the RCP discharge. As such, the peak pressure is calculated by adding the RCS pressure (pressurizer) and the pressure drop for the pump (between cold leg at safety injection and surge line).

<u>DCD Section 15.4.3</u>
The NRC staff examined the calculation supporting DCD Section 15.4.3, "Control Rod Misoperation (System Malfunction or Operator Error)." In examining the calculation, the NRC staff noted key calculation inputs and their sources. In particular, NRC staff noted:

1. The 12-finger CEA drop analysis is to be included in future design analyses for the core operating limit supervisory system and the CPCS. It is noted that if a 12-finger CEA drops, the CPCS will appropriately generate a trip if it is necessary because the CPCS conservatively calculates DNBR every 0.05 seconds.
2. The most negative MTC was used which maximizes the return to power. This parameter was obtained from an engineering design data document.
3. The most negative FTC was utilized.
4. The EOC kinetics parameters, including the minimum delayed neutron fraction, are used to maximize the heat flux increase.
5. The dropped rod was inserted over 2 seconds instead of 4 seconds.
6. The distortion factor is multiplied to the integrated radial peaking factor associated with the given rod configuration to determine the peak radial power distortion.
7. The distortion factor is multiplied to the initial integrated radial peaking factor in order to obtain the PLHGR.

<u>DCD Section 15.4.4</u>
The NRC staff examined the calculation supporting DCD Section 15.4.4, "Startup of an Inactive Reactor Coolant Pump." In examining the calculation, NRC staff noted key calculation inputs and their sources. In particular, the NRC staff noted:

1. The maximum temperature difference between the RCS and steam generators was obtained from an engineering design data document.
2. The applicant referenced specific values from Technical Specifications (TS) LCO 3.1.1, 3.1.2, and 3.9.1 for shutdown margin required in Modes 3 through 6. However, Technical Specifications for the APR1400 does not provide any specific values for shutdown margin, but references the Core Operating Limits Report. To clarify this discrepancy, NRC staff issued request for additional information (RAI), RAI 217-8217, Question 15.4.4-1.
3. The isothermal temperature coefficients were obtained from the safety analysis input manual. Using calculation inputs obtained from the audit, NRC staff performed confirmatory calculations and obtained results that were consistent with the applicant's results.

4. The audited material, however, did not establish the basis for the input values or the required shutdown margin. Therefore, NRC staff issued RAI 217-8217, Question 15.4.4-1.

<u>DCD Section 15.4.8</u>

The NRC staff examined the calculation supporting DCD Section 15.4.8, “Spectrum of Control Element Assembly Ejection Accidents.” In reviewing the calculation, NRC staff noted key calculation inputs and their sources. In particular, the NRC staff noted:

1. The applicant’s peak pressure analysis used a maximum gap conductance to maximize heat transfer from the fuel rod to the reactor coolant. The applicant confirmed the conservatism of this by performing a sensitivity study utilizing STRIKIN-II.
2. The applicant performed a sensitivity study to investigate the impact of initial core power on peak pressure. The calculations demonstrated that an analysis initiated from hot full power conditions results in bounding values for peak pressure in the RCS and main steam system.
3. Analyses use a minimum, N-2 scram, worth of 5 %$\Delta\rho$ to minimize the decay of heat flux after reactor trip.
4. The DNBR analyses utilize top peak axial power shapes to minimize DNBR.
5. The Hot Channel Flow Factor (HCFF) is set in STRIKIN-II in order to make the minimum DNBR calculated in STRIKIN-II conservative with respect to the value calculated by

6. The CEA ejection time is determined by assuming a 2500 psid pressure differential across the pressure boundary and no viscous or drag forces on the ejected CEA. An ejection time of 0.05 seconds was used in this analysis.
7. The DNBR analysis for hot full power was performed at 95 percent rated thermal power. The NRC staff issued RAI-8395, Question 15.4.8-3, to clarify the basis for this value.
8. The DNBR analysis for the CEAE event uses the post-ejected axial power shape in the hot channel and the pre-ejected axial power shape in the average channel. However, the fuel enthalpy analysis for the CEAE event uses the pre-ejected axial power shape in both the hot and average channels. NRC staff issued RAI 340-8395, Question 15.4.8-4 to clarify.
9. The most positive MTC was used to maximize the positive feedback and heat flux. The following MTC values were used for each power level.

Case	MTC Value (pcm/ºF)
95 percent power	0.25
50 percent power	2.5
20 percent power	4.0
Hot zero power	5.0

10. CESEC-III cannot model the VOPT, only the high power trip. Additional conservatism is included in the high power trip to account for the uncertainty associated with deficiencies in excore detector performance (an 11% excore power penalty)
    a. $Setpoint = \frac{[CEILING]+[Excore\ Power\ Penalty]}{Initial\ Power\ Level}\ , Power\ \geq 95\%$
    b. $Setpoint = \frac{[Initial\ Power\ Level\ ]+[STEP]+\ [Excore\ Power\ Penalty]}{Initial\ Power\ Level}\ , Power < 95\%$
    c. CEILING = 116.5%, STEP = 14%
    d. VOPT setpoint of 1.275 hot full power delays the reactor trip to maximize fuel enthalpy
11. KHNP credits a 3-second delay before the effects of a loss of offsite power impact the analysis.
12. The peak cladding temperature remains below 1200 ºF and the hot channel gas plenum pressure remains well below the system pressure. The cladding temperature and gas plenum pressure are not sufficient to cause the fuel rod ballooning or fuel rupture.
13. All fuel failures are attributed to departure from nucleate boiling (DNB). The percentage of fuel to undergo DNB is calculated by the convolution method and is provided below:

<u>DCD Section 15.6.2</u>
The NRC staff observed in calculation note 3L186-SA-CA070-023, Revision 0, “Double Ended Break of a Letdown Line Outside Containment Upstream of the Letdown Line Isolation Valve for SKN 3&4 FSAR,” that the CESEC-III calculation is supported by several hand calculations including: (1) letdown line flow resistance, (2) overall heat transfer coefficient for the regenerative heat exchanger, and (3) enthalpy of the letdown outlet fluid.

<u>DCD Section 15.6.3</u>
The NRC staff examined the calculation supporting DCD Section 15.6.3, “Steam Generator Tube Rupture with and without a Loss of Offsite Power Analysis for SKN 3&4.” In examining the calculation, the NRC staff noted key calculation inputs and their sources. In particular, the NRC staff noted:

1. The initial conditions that represent the limiting cases, for both dose consequences and thermal-margin, were determined by performing a sensitivity study.
2. Thermal-margin analyses that utilized a reactor overpower margin (ROPM) of 18 percent produced values for the minimum DNBR that were below the safety limit, which indicates fuel failure. The analyses presented in the DCD show no fuel failure for the steam generator tube rupture event. This caused the staff to question if the analysis presented in the DCD represented the limiting case. Accordingly, the NRC staff issued RAI 370-8450, Question 15.6.3-4.

3. A sensitivity study was performed to determine the impact of single failures on the event. The applicant considered the failure of one emergency diesel generator (effects two safety injection pumps), the failure of a safety injection pump, and a failure of any one auxiliary feedwater pump to start or auxiliary feedwater valve to function.

H. Requests for Additional Information Resulting from Audit

The NRC staff issued several RAIs based on the audit material. These RAIs are available in ADAMS. The ADAMS accession numbers are provided in the table below.

RAIs Resulting from Audit

DCD Section	RAI Number	ADAMS Accession Number
15.0.2	387-8485, Question 15.0.2-1	ML16032A111
15.4.4	217-8217, Question 15.4.4-1	ML15295A510
15.4.8	340-8395, Question 15.4.8-3	ML15351A301
15.4.8	340-8395, Question 15.4.8-4	ML15351A301
15.6.3	370-8450, Question 15.6.3-4	ML16019A276

I. References

1. "Letter to Korea Hydro and Nuclear Power Co., Ltd., and Korea Electric Power Corporation – Acceptance of the Application for Standard Design Certification of the Advanced Power Reactor 1400," March 4, 2015 (ADAMS Accession No. ML15041A455).
2. "Staff Regulatory Audit Plan Regarding Transient and Accident Analyses as Part of the Review of the APR1400 Design Control Document," June 10, 2015 (ADAMS Accession No. ML15147A597).