UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

August 21, 2018

MEMORANDUM TO: Michael D. McCoppin, Chief
Licensing Branch 2
Division of Licensing, Siting,
and Environmental Analysis
Office of New Reactors

FROM: George F. Wunder, Senior Project Manager ***/RA/***
Licensing Branch 2
Division of Licensing, Siting,
and Environmental Analysis
Office of New Reactors

SUBJECT: AUDIT SUMMARY FOR THE KOREA HYDRO AND NUCLEAR POWER CO., LTD. ADVANCED POWER REACTOR 1400 CHAPTERS 4, 5, AND 9 SYSTEMS RELATED TO THE REACTOR, REACTOR COOLANT SYSTEM AND CONNECTING SYSTEMS

The purpose of this memorandum is to document for the Office of New Reactors, Division of Licensing, Siting, and Environmental Analysis, Licensing Branch 2, the summary of the Regulatory Audit conducted from July 29, 2015, through April 27, 2017. During this period a team from the U.S. Nuclear Regulatory Commission's (NRC) Office of New Reactors performed an audit of Korea Hydro and Nuclear Power Co., Ltd. (KHNP), Advanced Power Reactor 1400 (APR1400) information related to Chapters 4, 5, and 9 of the KHNP standard design certification application. The purpose of the audit was to gain a better understanding of the calculations that support, and the documentation related to, the reactor, reactor coolant system and connecting systems, and auxiliary systems in Chapters 4, 5, and 9 of the APR1400 Design Control Document. The information contained in the APR1400 electronic reading room was reviewed, as well as evaluation of detailed information submitted by KHNP as part of responses to requests for additional information.

The enclosed audit summary provides the observations made by NRC staff during the audit, along with clarifications discussed with KHNP during the audit. A list of audit team members is provided in the audit summary, along with the list of documents audited by NRC staff.

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301-415-1494

The plan for this audit, dated June 6, 2015, is documented and can be found in Agencywide Documents Access and Management System under Accession No. ML15159B162.

Docket No. PROJ0782

Enclosure:
Audit Summary

cc: See next page

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AUDIT SUMMARY

CHAPTERS 4, 5, AND 9 (NON-CHAPTER 15)

AUDIT OF THE ADVANCED POWER REACTOR 1400

DESIGN CONTROL DOCUMENT

1.0 BACKGROUND

On March 5, 2015, the U.S. Nuclear Regulatory Commission (NRC) accepted the design certification application for docketing the Advanced Power Reactor 1400 (APR1400) submitted by Korea Hydro and Nuclear Power Co., Ltd. (KHNP) (Reference 1). The NRC staff initiated Phase 1 of the design certification application review on March 9, 2015.

The purpose of the audit was to gain a better understanding of the calculations supporting, and to review documentation, related to the reactor, reactor coolant system and connecting systems, and auxiliary systems presented in Chapters 4, 5, and 9 of the APR1400 Design Control Document (DCD). This audit summary is accompanied by an audit plan (Reference 2).

2.0 REGULATORY AUDIT BASES

This regulatory audit is based on the following:

- Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, Appendix A, "General Design Criteria" (GDC):
    - GDC 10, "Reactor Design"
    - GDC 11, "Reactor Inherent Protection"
    - GDC 20, "Protection System Functions"
    - GDC 25, "Protection System Requirements for Reactivity Control Malfunctions"
    - GDC 26, "Reactivity Control System Redundancy and Capability"
    - GDC 27, "Combined Reactivity Control Systems Capability"
    - GDC 28, "Reactivity Limits"
    - GDC 29, "Protection against Anticipated Operational Occurrences"
    - GDC 62, "Prevention of Criticality in Fuel Storage and Handling"
- 10 CFR 50.61, "Fracture Toughness Requirements for Protection against Pressurized Thermal Shock Events"
- 10 CFR 50.68, "Criticality Accident Requirements"
- NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition"
    - Section 4.3, "Nuclear Design"
    - Section 4.6, "Functional Design of Control Rod Drive System"
    - Section 5.4, "Reactor Coolant System Component and Subsystem Design"
    - Section 9.1.1, "Criticality Safety of Fresh and Spent Fuel Storage and Handling"
    - Section 15.0, "Introduction—Transient and Accident Analysis"

- Regulatory Guide 1.206, “Combined License Applications for Nuclear Power Plants”
  - Section C.I.4.6, “Functional Design of Reactivity Control Systems”
  - Section C.I.5.4.1, “Reactor Coolant Pumps”

3.0 LOGISTICS

The audit was conducted from the NRC headquarters via KHNP’s electronic reading room and from Westinghouse Electric Company’s, Rockville, Maryland office.

Dates: June 29, 2015 – April 27, 2017

Location: NRC Headquarters
Two White Flint North
11545 Rockville Pike
Rockville, Maryland 20852

Westinghouse Electric Company
11333 Woodglen Drive, Suite 202
Rockville, Maryland 20852

4.0 AUDIT TEAM MEMBERS

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

Alexandra L. Burja, NRC Technical Reviewer
Donald R. Carlson, NRC Technical Reviewer
Hien M. Le, NRC Technical Reviewer
Zhian Li, NRC Technical Reviewer
Shanlai Lu, NRC Technical Reviewer
Jim A. Steckel, NRC Project Manager
Alex Velazquez-Lozada, NRC Technical Reviewer

5.0 APPLICANT AND INDUSTRY STAFF PARTICIPANTS

Woochong Chon, KEPCO Nuclear Fuel
Jiyong Oh, KHNP
Christopher Tyree, AECOM

6.0 DOCUMENTS AUDITED

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

<u>DCD Section 4.3</u>

- GM-FE-0005, Revision 0, “Calculation of Reactivity Biases and Uncertainties,” November 1992
- GM-FE-0021, Revision 1, “Determination of Modified Fuel Temperature Correlation,” May 1997
- GM-FE-0021, Revision 2, “Determination of Modified Fuel Temperature Correlation,” October 2001

- GM-FE-0040, Revision 0, “Bias and Uncertainties for Single Spectrum Geometry DIT Tablesets,” January 1996
- GM-FE-0008, Revision 0, “Calculation of Temperature Coefficient, Power Coefficient, Rod Worth, Inverse Boron Worth and Reactivity Biases and Uncertainties,” December 1992
- NE-86-033, “Standardization of the Methodology for Generating TORC Physics Data,” January 1986
- PHA-81-092, “Burnup Uncertainties Between ROCS and CECOR,” May 1981
- CE-CES-129, Revision 9-P, “Physics Biases and Uncertainties,” March 2003
- Y34-FE-0042, “BOC-1 Reactivity and Power Coefficient Biases,” June 1993
- Y34-FE-0068, “Summary of Biases for YGN 3 Cycle 1 Design,” May 1994
- CENPD-275-P, Revision 1-P-A, “CE Methodology for Core Design Containing Gadolinia-Urania Burnable Absorbers,” May 1988
- 10587-NE-050, Revision 0, “Physics Data Generation of CEA Mechanical Design Justification for YGN-3&4 Initial Core Final Design”
- 10587-NE-058, Revision 0, “CEA Depletion and Tip Effects in YGN-3&4 Initial Core Final Design,” March 1992

DCD Section 4.6

- 11A60-IC-DS630, Revision 00, “Design Specification for Digital Rod Control System Power Cabinet”
- “Information for NRC Audit Section 4.6,” 2015
- WNA-CN-00204-GEN-P, Rev. 1, “Failure Modes and Effects Analysis of the Four Coil Digital Rod Control System,” January 2016

DCD Section 5.4.1

- Flow diagram for hydrodynamic shaft seal system
- Pump and motor bearings drawings and explanatory information
- LTR-APR-15-09-P, Revision 0, “Justification that the APR1400 RCP [Reactor Coolant Pump] Seals will not Exceed the Exit Temperature Specification Limit,” November 2015
- Information related to anti-rotation devices
- LTR-APR-15-08-P, Revision 0, “Hydrodynamic Shaft Seal SBO [Station Black-out] Qualification Test Procedure, HDD 254 Type F (Preliminary),” April 2015
- LTR-APR-15-03-P, Revision 0, “APR1400 ELAP Analysis Results Evaluation and Test Matrix Construction for Integral KSB Type F RCP Seal Test Program,” September 2015
- 11A60-FS-DS485, Revision 3, “Design Specification for Reactor Coolant Pump Motors,” March 2015
- 11A60-FS-DS480, Revision 3, “Design Specification for Reactor Coolant Pumps,” March 2015
- APR1400-A-M-NR-14002-P, Revision 1, “Extended Loss of AC Power Capability for APR1400 KSB RCP Seals,” May 2016

DCD Section 9.1.1

- WCAP-17889-P, Revision 0, "Validation of SCALE 6.1.2 with 238-Group ENDF/B-VII.0 Cross Section Library for APR1400 Design Certification," June 2014, and related calculation note

7.0 DESCRIPTION OF AUDIT ACTIVITIES AND SUMMARY OF OBSERVATIONS

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

DCD Section 4.3

The staff evaluated the following documents related to benchmarking the nuclear design codes DIT and ROCS with the ENDF/B-IV cross section library and with PLUS7 fuel containing gadolinium: A-GM-FE-0005, Revision 0; A-GM-FE-0021, Revision 1; A-GM-FE-0021, Revision 2; A-GM-FE-0040, Revision 0; A-GM-FE-0008, Revision 0; NE-86-033; PHA-81-092; CE-CES-129, Revision 9-P; Y34-FE-0042; Y34-FE-0068; and CENPD-275-P, Revision 1-P-A. CENPD-275-P, Revision 1-P-A described the Combustion Engineering (C-E) methodology for core designs containing gadolinia-urania fuel. The multiple calculation notes showed how biases and uncertainties are calculated for the APR1400; measurements from eight United States C-E plants are compared to DIT/ROCS code predictions, and the deviations are used to statistically calculate bias and uncertainty. CE-CES-129, Revision 9-P contains the bias and uncertainty values applied to the APR1400. Overall, the information clarified how biases and uncertainties are determined for the APR1400.

The staff also reviewed documentation related to control rod depletion and the approach to estimate the change in differential control rod worth in documents 10587-NE-050, Revision 0 and 10587-NE-058, Revision 0. Due to the staff's concerns that effects of B-10 burnout on control element assembly worth could become unacceptable without imposing service limits, the staff issued request for additional information (RAI) 293-8332, Question 04.03-5.

Although the staff requested documentation related to the reactor fluence vessel calculation as part of the audit, the documentation provided for the audit had already been submitted on the docket. Therefore, the staff reviewed the document and issued RAIs outside of the audit.

DCD Section 4.6

NRC staff reviewed the information related to the control rod drive system (CRDS). The applicant's high-level summary of the information requested described in detail the isolation of the essential trip function and non-essential control function, and the design specification for the digital rod control system (DRCS) demonstrated this in detail. In addition, the failure modes and effects analysis of the DRCS showed that no single failure in the DRCS could either (1) cause a reactivity transient through the dropping or uncontrolled stepping of control rods or (2) prevent a reactor trip.

The applicant also provided some clarification regarding the combined performance of the reactivity control systems, including transients for which use of the CRDS and the safety injection system is required. However, not all cases were described in adequate detail, so NRC staff issued RAI 136-8081, Question 04.06-6.

DCD Section 5.4.1

NRC staff reviewed the flow diagram for the hydrodynamic shaft seal system and noted that it shows additional details of the seal injection flow paths within the pump seal package and out to the high-pressure seal water cooler and of the jet pump that were not included in DCD Figure 5.1.2-2, "Reactor Coolant Pump Flow Diagram." Therefore, NRC staff issued RAI 307-7835, Question 5.4-1, Parts 1 and 4, requesting the applicant to add these details to the DCD. NRC staff reviewed the pump and motor bearings drawings and explanatory information and noted that the clarifying details provided by the applicant could be used to enhance the description of these components in DCD Subsection 5.4.1.2, so NRC staff issued RAI 307-7835, Question 5.4-1, Part 8, requesting the applicant to add these details to the DCD.

NRC staff reviewed Technical Report (TeR) LTR-APR-15-09-P, Revision 0, "Justification that the APR1400 RCP Seals will not Exceed the Exit Temperature Specification Limit," November 2015, and noted that the provided information adequately justifies the 22 degrees Fahrenheit (°F) margin discussed in DCD Section 5.4.1.2 for loss of component cooling water event. This resolved the NRC staff's question raised in RAI 307-7835, Question 5.4-1, Part 3.

NRC staff reviewed the information related to anti-rotation devices and noted that the provided details could be used for the description of these devices in DCD Subsection 5.4.1.2, so NRC staff issued RAI 307-7835, Question 5.4-1, Part 6, requesting the applicant to add these details to the DCD.

NRC staff reviewed TeR LTR-APR-15-08-P, Revision 0, "Hydrodynamic Shaft Seal SBO Qualification Test Procedure, HDD 254 Type F (Preliminary)," April 2015 and TeR LTR-APR-15-03-P, Revision 0, "APR1400 ELAP Analysis Results Evaluation and Test Matrix Construction for Integral KSB Type F RCP Seal Test Program," September 2015, and noted that the provided seal testing program could be used to obtain test results that address the NRC staff's concern regarding seal leakage during an SBO event. The staff raised this concern in RAI 307-7835, Question 5.4-1, Part 7.

NRC staff reviewed documents 11A60-FS-DS485, Revision 3, "Design Specification for Reactor Coolant Pump Motors," March 2015, and 11A60-FS-DS480, Revision 3, "Design Specification for Reactor Coolant Pumps," March 2015, to ensure that the vendor shop test verifies integrity of the assembled flywheel at 125 percent of operating speed as specified in inspection, test, analysis, and acceptance criteria Item 2.4.1.9.b.

NRC staff reviewed the draft version of TeR APR1400-A-M-NR-14002-P, Revision 1, "Extended Loss of AC Power Capability for APR1400 KSB RCP Seals," May 2016, that was developed to document the completed seal test results. This was used to address RAI 443-8555, Question 05.04-2, Part 2, which requested KHNP to include a summary of the test result in the DCD.

DCD Section 9.1.1

NRC staff reviewed the code validation report WCAP-17889-P to ascertain proper benchmarking and implementation of bias and bias uncertainty in the new and spent fuel storage rack criticality analyses. The report describes validation using three subsets of experiments: fresh fuel without absorbers, fresh fuel with absorbers, and fresh and spent fuel with absorbers. The bias and bias uncertainty calculated for these subsets are applied to the new fuel storage racks, spent fuel storage rack Region I, and spent fuel storage rack Region II, respectively. Although the report generally follows the methodology described in NUREG/CR-6698, "Guide for Validation of Nuclear Criticality Safety Calculational Methodology," WCAP-17889-P does not list the calculated effective multiplication factor (k_{eff}) values and uncertainty for each validation case, as NUREG/CR-6698 recommends. In addition, WCAP-17889-P does not identify which experiments were used for which validation subset but rather provides a total number of experiments in each subset. The staff tallied up the number of experiments it expected to be in each subset based on the experiment descriptions provided in WCAP-17889-P and the sources of the experiments, but the staff's numbers did not match the totals in WCAP-17889-P.

Therefore, the staff audited the supporting calculation notes to see the list of k_{eff} and uncertainty values and to verify the categorization of experiments. The staff observed that seven experiments without strong absorbers (LEU-COMP-THERM-013, Case 1, and LEU-COMP-THERM-037, Cases 1-6) seemed to be incorrectly included in the "with absorbers" subsets, which the staff noted could impact the bias and bias uncertainty values for the spent fuel pool (SFP). As a result, the staff requested audit phone calls to understand reasons for the categorization and the effects if the experiments were categorized incorrectly.

Through two calls, one on November 30, 2016, and one on April 27, 2017, the staff came to understand the logic behind the categorization of six of the experiments. The six cases selected from the benchmark experiment LEU-COMP-THERM-037 include concrete as a reflecting material and were selected for the "with absorbers" categories so the impact of concrete could be included in the validation suites for the SFP. The staff notes that some of the SFP models do include concrete and understands the intent to incorporate concrete in the validation, even if strong absorbers are not present in those six cases. In addition, the staff estimated during the audit of the calculated k_{eff} values that the net effect of those cases being in the "with absorbers" subsets would be on the order of the Monte Carlo uncertainty in the criticality calculations and therefore negligible. Although LEU-COMP-THERM-013, Case 1 still appears to be categorized incorrectly, the staff notes that the impact on the bias and bias uncertainty values from that single case would also be negligible. Furthermore, the applicant doubles the value of the code bias uncertainty for use in the criticality analysis, which is more than enough to cover any impacts from how the experiments were categorized. For these reasons, the staff concluded there is no safety issue, and the audit resolved the staff's concerns regarding experiment categorization.

8.0 RAIS RESULTING FROM AUDIT

NRC staff issued several RAIs based on the audit material. These RAIs are available in the Agencywide Documents Access and Management System (ADAMS). ADAMS accession nos. are provided in the table below.

RAIs Resulting from Audit

DCD Section	RAI Number	ADAMS Accession Number
4.3	293-8332	ML15314A018
4.6	136-8081	ML15227A013
5.4.1	307-7835	ML15320A152

9.0 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 Reactor, Reactor Coolant System and Connecting Systems, and Auxiliary Systems as Part of the Review of the APR1400 Design Control Document," June 15, 2015 (ADAMS Accession No. ML15159B162 (NON-PUBLIC)).

3. 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," and 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants."

4. NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," Section 4.3, "Nuclear Design," Revision 3, March 2007; Section 4.6, "Functional Design of Control Rod Drive System," Revision 2, March 2007; Section 5.4, "Reactor Coolant System Component and Subsystem Design," Revision 2, March 2007; Section 9.1.1, "Criticality Safety of Fresh and Spent Fuel Storage and Handling," Revision 3, March 2007; and Section 15.0, "Introduction—Transient and Accident Analysis," Revision 3, March 2007 (ADAMS Accession No. ML070660036).

5. Regulatory Guide 1.206, Sections C.I.4.6, "Functional Design of Reactivity Control Systems"; and C.I.5.4.1, "Reactor Coolant Pumps," June 2007 (ADAMS Accession No. ML15233A056).

6. KHNP APR1400 DCD, Tier 2, Chapter 4, "Reactor"; Chapter 5, "Reactor Coolant System and Connecting Systems"; and Chapter 9, "Auxiliary Systems," December 23, 2014 (ADAMS Accession No. ML15006A059).