U.S. NUCLEAR REGULATORY COMMISSION REGULATORY AUDIT OF CHAPTER 4 (REACTOR) OF THE APR1400 DESIGN CONTROL DOCUMENT

APR1400 DESIGN CERTIFICATION
Docket No. 52-046

AUDIT PLAN

APPLICANT:	Korea Hydro and Nuclear Power Co., Ltd. (KHNP) and Korea Electric Power Corporation (KEPCO)
APPLICANT CONTACTS:	James Ross, KHNP Andy Jiyong Oh, KHNP
DURATION:	An audit will be conducted from January 20, 2016 through January 21, 2016, at the Westinghouse Electric Co. facilities in Rockville, MD. Follow-up audit activities at NRC Headquarters via KHNP's electronic reading room (or at KHNP's facilities in Vienna, VA) may be necessary at various times.
LOCATION:	WEC 11333 Woodglen Drive Rockville, MD 20852 NRC Headquarters Two White Flint North 11545 Rockville Pike Rockville, MD 20852-2738 KHNP Washington DC Center 8100 Boone Blvd. Suite 620 Vienna, VA 22182

I. BACKGROUND

On March 4, 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. (KHNP) (Reference 1). The staff initiated Phase 1 of the application design certification review on March 9, 2015. The staff determined that efficiency gains would be realized by auditing the documents supporting the calculations presented in the design control document (DCD) in lieu of requests for additional information (RAIs) asking the applicant to docket the calculation files. The purpose of this audit is to allow the NRC technical staff to gain an understanding of the supporting calculations to better focus staff inquiries to the applicant. During the audit and other interactions with the applicant, detailed RAIs may be developed, which would be part of future formal correspondence.

Enclosure

II. REGULATORY AUDIT BASIS

The purposes of this audit is for the staff to: (1) gain an understanding of APR1400 supporting calculations and analyses to reach a reasonable assurance finding and (2) review related documentation and non-docketed information to evaluate conformance with the Standard Review Plan (SRP) or technical guidance.

According to Title 10 of the *Code of Federal Regulations* (10 CFR) Part 52.47(a)(3)(i) a design certification application must contain a final safety analysis report (FSAR) that includes a description of principal design criteria for the facility. An audit is needed to evaluate the safety conclusions that need to be made regarding Chapter 4 of the APR1400 DCD and identify detailed information related to the applicant's principal design criteria.

The NRC staff must have sufficient information to ensure that acceptable risk and adequate assurance of safety can be documented in the NRC staff's safety evaluation report (SER).

This regulatory audit is based on the following:

- 10 CFR Part 50 Appendix A, General Design Criteria (GDC):
    - GDC 10, "Reactor design," The reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences.
    - GDC 12, "Suppression of reactor power oscillations," The reactor core and associated coolant, control, and protection systems shall be designed to assure that power oscillations which can result in conditions exceeding specified acceptable fuel design limits are not possible or can be reliably and readily detected and suppressed.
    - GDC 13, "Instrumentation and Control," as it relates to assuring instrumentation is provided to monitor variables and systems over their anticipated ranges for normal operation, for anticipated operational occurrences, and for accident conditions as appropriate to assure adequate safety, including those variables and systems that can affect the fission process, the integrity of the reactor core, the reactor coolant pressure boundary, and the containment and its associated systems.
    - GDC 20, "Protection System Functions," as it relates to the protection system to be designed: (1) to initiate automatically the operation of appropriate systems including the reactivity control systems, to assure that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences and (2) to sense accident conditions and to initiate the operation of systems and components important to safety.

- GDC 21, “Protection System Reliability and Testability,” as it relates to assuring the protection system is designed for high functional reliability and inservice testability commensurate with the safety functions to be performed as well as redundancy and independence sufficient to assure that: (1) no single failure results in loss of the protection function and (2) removal from service of any component or channel does not result in loss of the required minimum redundancy.

- GDC 22, “Protection System Independence,” as it relates to the design of the protection system to assure that the effects of natural phenomena, and of normal operating, maintenance, testing, and postulated accident conditions on redundant channels do not result in loss of the protection function.

- GDC 23, “Protection System Failure Modes,” as it relates to assuring the protection system is designed to fail into a safe state or into a state demonstrated to be acceptable on some other defined basis if conditions such as disconnection of the system, loss of energy, or postulated adverse environments are experienced.

- GDC 24, “Separation of Protection and Control Systems,” as it relates to assuring the protection system is separated from control systems to the extent that failure of any single control system component or channel, or failure or removal from service of any single protection system component or channel which is common to the control and protection systems leaves intact a system satisfying all reliability, redundancy, and independence requirements of the protection system as well as assuring that interconnection of the protection and control systems is limited to assure that safety is not significantly impaired.

- GDC 29, “Protection Against Anticipated Operational Occurrences,” as it relates to protection and reactivity control systems to be designed to assure an extremely high probability of accomplishing their safety functions in the event of anticipated operational occurrences.

- 10 CFR 50.55a(h), “Protection and Safety Systems,” as it relates to compliance with Institute of Electrical and Electronics Engineers (IEEE) Standard (Std) 603-1991, “IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations,” and the January 30, 1995, correction sheet.

III. REGULATORY AUDIT SCOPE AND METHODOLOGY

The NRC staff will conduct this audit in accordance with the guidance provided in NRO-REG-108, Regulatory Audits” (Reference 2). The staff intends to review information, documents and supporting calculations related to APR1400 DCD Tier 2 Chapter 4. The following are areas that the NRC staff intends to review during this audit:

Core Design Issues:

1. Since the analysis method for large break LOCA differs from that approved for the System 80+ design (due to the use of RELAP5/MOD3.3), the staff would like to understand if the core design process for determining limiting pressure drops and component loads will differ from that described in DCD Section 4.4 (i.e., using only the TORC and CETOP codes).

2. The staff notes that the source of the gadolinium isotopic data used by DIT/ROCS is not clearly identified in the approved code topical reports. To address this concern, the audit will seek to identify the source of the gadolinium isotopic data used by DIT/ROCS. If the source cannot be identified from existing documentation, this should be clearly stated and the DIT multigroup library data for the Gd isotopes should then be examined for consistency with data from later evaluated nuclear data files such as ENDF/B-V.

3. The applicant presented axial core power distributions at various core depletion states assuming only unrodded operation. However, the APR1400 is designed to use regulating rods to control power shapes. The audit will seek clarification on whether the presented unrodded axial power shapes are meant to result in peak core power densities that bound those for cases where regulating rods are partially inserted in the core.

4. Demonstration of detection of CEA misalignment or drop; the applicant stated: "Since the plant protection system (CPCs and CEACs) detects the CEA positions by means of two independent sets of reed switches and uses this information in determining margin to trip, it is not necessary to rely on in-core or ex-core nuclear instrumentation to detect control element misalignment or drop. Thus, this testing is not performed." The audit is to examine if the applicant's justifications are acceptable for not performing: (1) demonstrated CEA positions and misalignment tests and (2) demonstration of the capability of the incore neutron flux instrumentation to detect rod misalignment equal to or less than the Technical Specifications limits for control rod misalignment.

Core Protection Calculator (CPCS) Issues:

The staff requests a summary presentation on the entire process, from initial cycle testing to development of the shape annealing and rod shadowing factors, to normal operational experience with the system for similar designs. Detailed issue descriptions are presented below:

1. DCD Section 4.3.3.1.1.4 describes the use of a fixed source MCNP (Monte Carlo, N-particle) code adjoint calculation to determine the shape annealing functions, while CENPD-170-P identifies the DOT/DORT codes for this purpose. What methodology is to be used for the APR1400 to synthesize the power shapes, and what deviations are taken from the CENPD-170-P methodology?

2. CENPD-170-P specifies a number of analytical methods and computer codes, so will these same codes be utilized for the APR1400, and if so, have these codes been reviewed and approved by the NRC?

3. APR1400-F-C-NR-14001-P, Revision 0, “CPC Setpoint Analysis Methodology for APR1400” states that the CPC calculations are verified using a large number of power distributions at BOC, MOC, and EOC, but it does not describe how the CPC constants (e.g., Fxy and Fq) are calculated, especially when new fuels are introduced (i.e., mixed cores). For mixed cores, how will multiple DNBR uncertainties be implemented in the CPC algorithms?

4. The CPC power distribution uncertainties are evaluated based on core simulator 3D power distributions for a variety of conditions. Are design basis AOO events included in the database of power shapes?

5. Describe the setpoint calculation process for reloads. Define what calculations are performed generically for each fuel type, and which are cycle-specific. Identify the codes used and provide a reference to the approval SER.

6. For the core average axial power distribution, CENPD-170 discusses conversion of ex-core detector responses to peripheral core power at three core rings and then to 20-node axial shape using up to eight algorithm constants (α1 through α8). These are apparently pre-calculated to represent flat-, saddle-, top-, or bottom-peaked axial shapes for beginning-of-life (BOL) and end-of-life (EOL) conditions. The CPC uses some degree of pattern recognition on the 3-ring axial power distribution to determine which of the four power shape types are present and then uses a cubic spline fit to data. The applicant should explain how the constants are developed, whether they are cycle-dependent or burnup-dependent within one cycle, and how they will be verified against plant data.

7. For the radial power distribution, planar radial peaking factors and axial augmentation factors are used to define the hot pin power as a function of the CEA configuration (both normal and abnormal). No detail is provided on the method used to define these. The applicant should explain how the planar radial peaking factors will be calculated, whether cycle- or burnup dependent, what codes are used in the calculations, and whether the codes have been approved.

8. Explain how the axial augmentation factors are calculated, whether cycle- or burnup-dependent, what codes are used, and whether the codes have been approved.

9. Explain whether the CPC constants will be adjusted during a cycle to reduce conservatism, and if so, how the changes are verified.

10. Explain where in the safety analysis is the time delay that the core protection calculator (CPC) will utilize before reverting to the “predetermined penalty factor” (PF), as discussed in the response to RAI 274-8277 (ML15363A340; Question 07.01-37), that will be a large value to ensure a core protection calculator (CPC)

initiated departure from nucleate boiling ratio (DNBR) reactor trip and/or a local power density (LPD) reactor trip, explained, defined, and captured, within the design basis event's "sequence of events."

11. Based on the definition of a "pre-determined penalty factor" (PF) added to the APR1400 FSAR, Tier 2, Table 7.2-7, mark-ups (FSAR mark-up page 7.2-71), as provided in the response to RAI 274-8277 (ML15363A340; Question 07.01-38), explain how the predetermined PF will be guaranteed to be large enough during the entire fuel lifecycle to ensure that a CPC initiated reactor trip is reached.

Core Operating Limit Supervisory System (COLSS) Issues:

1. The staff SER for the System 80+ refers to topical reports CENPD-169 and CEN-312 for the detailed description of the COLSS. The staff reviewers would like to understand if these documents could be considered part of the design basis for the APR1400. If so, please make these documents available in the Electronic Reading Room.

IV. INFORMATION AND DOCUMENTS NECESSARY FOR THE AUDIT

The following documents are to be made available to the NRC staff, either at the KHNP Washington D.C. Center, or in the electronic reading room. Appropriate handling and protection of proprietary information shall be acknowledged and observed throughout the audit.

- Any documents requested in Section III of this audit plan, and any documents which have not already been provided to the staff in the electronic reading room which may support the above discussion topics.

V. AUDIT TEAM

The following NRC staff will participate in the audit:

- John Vera, Project Manager
- Shanlai Lu, Sr. Reactor Systems Engineer
- Alexandra Burja, General Engineer
- Kenneth Mott, Electronics Engineer
- James Gilmer, Reactor Systems Engineer
- Donald Carlson, Sr. Project Manager
- Jose March-Leuba, (ORNL Contractor)
- Randy Belles, (ORNL Contractor)

VI. LOGISTICS

The NRC staff and the applicant have agreed that the audit will be conducted from January 20, 2016, through January 21, 2016 at the Westinghouse Electric Co. facilities in Rockville, MD. In support of this approach, the applicant has agreed to make knowledgeable staff available, along with relevant documentation, to support staff review and discussion of the material. The NRC

staff will have internal meetings throughout the audit to discuss preliminary findings. A summary of audit preliminary findings will be provided to the applicant for discussion.

VII. SPECIAL REQUESTS

None.

VIII. AUDIT ACTIVITIES AND DELIVERABLES

The NRC audit team review will cover the technical areas identified in Section III of this audit plan. Depending upon how much effort is needed in a given area, the NRC team members may be reassigned to ensure adequate coverage of important technical elements. The NRC Project Manager will coordinate with KHNP in advance of audit activities to verify specific documents and identify any changes to the audit schedule and requested documents.

The NRC staff acknowledges the proprietary nature of the information requested. It will be handled appropriately throughout the audit. While the NRC staff will take notes, the NRC staff will not remove hard copies or electronic files from the audit site(s).

At the completion of the audit, the audit team will issue an audit summary within 90 days that will be declared and entered as an official agency record in the NRC's Agencywide Documents Access and Management System (ADAMS) records management system. The audit outcome may be used to identify any additional information to be submitted for making regulatory decisions, and it will assist the NRC staff in the issuance of RAIs (if necessary) for the licensing review of APR1400 DCD Chapter 4 and any related information provided in other chapters, in preparation of the NRC staff's Safety Evaluation Report.

If necessary, any circumstances related to the conductance of the audit will be communicated to John Vera, NRC Project Manager at 301-415-5790 or John.Vera@nrc.gov.

IX. 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," issued March 4, 2015, ADAMS Accession Number ML15041A455.

2. NRO-REG-108, "Regulatory Audits," issued April 2, 2009, ADAMS Accession Number ML081910260.