TRANSIENT AND ACCIDENT ANALYSES AS PART OF THE APR1400 DESIGN CONTROL DOCUMENT AUDIT

JULY 14, 2015 – SEPTEMBER 30, 2015

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

APR1400 DESIGN CERTIFICATION
Docket No. 52-046

Location: 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

Purpose:

The purpose of this audit is for the staff to: (1) gain an understanding of the Advanced Power Reactor 1400 (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.

Background:

On March 5, 2015, the U.S. Nuclear Regulatory Commission (NRC) accepted the design certification application for docketing for the 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 NRC 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 (RAI) asking that the applicant 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 there may be detailed NRC requests for information developed, which would be part of future formal correspondence.

Regulatory Audit Basis:

Title 10 of the *Code of Federal Regulations* (10 CFR) 52.47(a)(3)(i), states that a design certification application must contain a final safety analysis report (FSAR) that includes a description of the principal design criteria for the facility. An audit is needed to evaluate the safety conclusions that need to be made regarding Chapter 15 of the APR1400 DCD and to identify detailed information related to the applicant's principal design criteria.

Enclosure

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:

- Regulatory Guide 1.157, "Careful consideration should be given to the range of applicability of a model when used in a best estimate code."

- SRP 15.0, p. 15.0.2-7, "For changes to previously approved models, the reviewers can limit their review to the new material if they determine that there is nothing new that will invalidate the previous approval, including the range of applicability for the analysis method."

- SRP 15.0, p. 15.6.5-7. "Regulatory Guide 1.157, 'Best Estimate Calculations of Emergency Core Cooling System Performance,' and Appendix K to 10 CFR Part 50, provide guidance and requirements on evaluation models needed to demonstrate compliance with the acceptance criteria. Appendix K also specifies documentation required for evaluation models."

- 10 CFR Part 50, Appendix K. II, Required Documentation, 1.a. "A description of each evaluation model shall be furnished. The description shall be sufficiently complete to permit technical review of the analytical approach including the equations used, their approximations in different form, the assumptions made, and the values of all parameters or the procedure for their selection, as for example, in accordance with a specified physical law or empirical correlation."

- SRP 15.0, p. 15.0.2-7, "For changes to previously approved models, the reviewers can limit their review to the new material if they determine that there is nothing new that will invalidate the previous approval, including the range of applicability for the analysis method."

- 10 CFR 50 Appendix A, General Design Criteria (GDC) 10, "Reactor Design."

- SRP 15.4.6, "Inadvertent Decrease in Boron Concentration in the Reactor Coolant System (PWR)."

- 10 CFR 100, "Reactor Siting."

- SRP 15.4.7, "Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position."

- GDC 27, "Combined Reactivity Control Systems Capability," requires the reactivity control systems to be designed to have a combined capability, in conjunction with poison addition by the emergency core cooling system, of reliably controlling reactivity changes to assure that, under postulated accident conditions and with appropriate margin for stuck rods, the capability to cool the core is maintained.

- GDC 28, “Reactivity Limits,” requires the reactivity control systems to be designed with appropriate limits on the potential amount and rate of reactivity increase to assure that the effects of postulated reactivity accidents can neither: (1) result in damage to the reactor coolant pressure boundary greater than limited local yielding nor (2) sufficiently disturb the core, its support structures or other reactor pressure vessel internals to impair significantly the capability to cool the core.

- 10 CFR Part 50 Appendix A, GDC

  - GDC 10, requires that specified acceptable fuel design limits (SAFDL) are not to be exceeded during normal operation, including the effects of anticipated operational occurrences (AOOs).

  - GDC 13, requires the availability of instrumentation to monitor variables and systems over their anticipated ranges to assure adequate safety, and of appropriate controls to maintain these variables and systems within prescribed operating ranges.

  - GDC 17, requires provision of an onsite electric power system and an offsite electric power system to permit functioning of structures, systems, and components important to safety.

  - GDC 20, requires that the protection system initiate automatically appropriate systems to assure that SAFDLs are not exceeded as a result of AOOs.

  - GDC 25, requires that the reactor protection system be designed to assure that SAFDLs are not exceeded in the event of a single malfunction of the reactivity control system.

Regulatory Audit Scope:

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 the transient and accident analyses described in APR1400 DCD Tier 2 Chapter 15, specifically Subsections 15.2.2, 15.4.6, 15.4.7, 15.4.1, 15.4.2, 15.4.3, 15.4.8 and 15.8. The following are areas which the NRC staff intends to review on this audit:

- The staff needs information confirming the applicability of the evaluation model used for inadvertent opening of a steam generator atmospheric dump valve (IOSGADV) event described in Section 15.1.4.3.1 to analyze the decrease in feed water temperature event described in Section 15.1.1, and the increased in feed water flow event described in Section 15.1.2.

- The staff needs information confirming the applicability of the evaluation model used for a loss of condenser vacuum (LOCV) event described in Section 15.2.3.3.1, to analyze the loss of external load event described in Section 15.2.1,

the turbine trip event described in Section 15.2.2, the main steam isolation valve closure event described in Section 15.2.4, the loss of non-emergency alternating current power to the stations auxiliaries event described in Section 15.2.6, and the loss of normal feedwater flow event described in Section 15.2.7.

- The staff will review the change from CE-1 to KCE-1 critical heat flux (CHF) correlation used on CETOP and STRIKIN-II.

- In Section 15.4.6.3.1, the staff needs information to clarify: (1) why the complete boron mixing model yields conservative times to criticality for Modes 4 and 5 while assuming complete reactor coolant system (RCS) boron mixing; (2) any tests, experiments or analyses that have been performed, which demonstrate that adequate boron mixing exists and that an undiluted area of the lower plenum/core would not cause an early criticality; (3) Boron Dilution Alarm System (BDAS) alarm set points, such that the operator actions times are still preserved based on the existence of any boron diluted volume in the core and the neutron source range detectors sensing it and (4) conservatisms related to BDAS alarm set points which account for any boron diluted in the lower plenum section of the core volume.

- In Section 15.4.7.2, the staff needs clarification on whether symmetric control element assembly (CEA) worths are compared to each other during the beginning of cycle (BOC) or startup physics testing. In addition, the staff needs information about whether symmetric CEA bank worths are compared to predicted or measured values.

- The staff needs clarification on the location of the limiting misloaded assembly described in Sections 15.4.7.2 and 15.4.7.3.2, and what limiting case is described in the DCD.

- The staff needs information that clarifies if the 15 percent maximum F_{xy} increase given in Section 15.4.7.3.3, "Results," is the maximum change in F_{xy} at any time during the cycle between the reference (as-designed) loading pattern and the misloaded pattern or the maximum change between in-core flux measurements.

- In Section 15.4.7.3.3, the staff needs clarification on the reason why the measurement uncertainty is included, as these are undetectable misloadings. In addition, the staff needs information on why the uncertainty wouldn't be the upper tolerance limit applied to the values calculated by ROCS computer code and will review the comparison between the increase in $F_{xy,}$ with the increase in $F_{\Delta H}$. The staff needs information about the accuracy of the CESEC-III input parameters to the consequence analysis mode, the validity of the key assumptions used, the technical bases for the determination of the sequence of events, and that important parameters are in the valid ranges for the computer codes described in Section 15.4.1, 15.4.2, 15.4.3 and 15.4.8.

- The staff needs information about the assumed power changes under the reactivity insertion and the timing of each of the states as listed in Table 15.4.8-1.

The staff will also review the Doppler Effect, moderator temperature coefficients, effective delayed neutron fraction, β_{eff} (at BOC or EOC?), effective delayed neutron precursor decay constant, λ_{eff}, average prompt neutron generation life time, lp, and other parameters in Table 15.4.8-2.

- The staff needs clarification on why the feedwater line break, letdown line break, and CEA ejection events described in Chapter 15 do not mention the use of safety injection system (SIS), while in Table 4.6-1, in Chapter 4 indicates that the SIS is required to mitigate those events.

- The staff needs information to clarify the technical basis for why RCS depressurization and leakage were not accounted for during the analysis of the control element assembly ejection (CEAE) accident described in Section 15.4.8 and why Section 15.4.8 does not mention the use of the SIS during the CEAE accident while Section 6.3 indicates that the SIS provides capability to mitigate the CEAE accident.

- The staff will review the applicability of the critical heat flux (CHF) correlation used in HRISE to calculate departure from nucleate boiling ratio (DNBR) for Plus 7 fuel as described in Section 15.1.5.3.1.

The NRC staff also intends to review information, documents and supporting calculations related to the transient and accident analyses described in the Technical Report, ," APR1400-Z-A-NR-14006-P, Rev. 0, "Non-LOCA Safety Analysis Methodology." The following are areas that the NRC staff intends to review:

- The staff will review updates of previously NRC approved code for non-loss of coolant accident (LOCA) events (e.g.). Some of those changes are:

    - o change in CESECIII computer code from Safety Relief Valve to Pilot Operated Safety Relief Valve and related delay times; change from positive displacement pump to centrifugal displacement pump in the chemical and control volume system (CVCS) and related discharge pressure; and changes and modifications to model anticipated transient without scram (ATWS);

    - o initialization method and boundary conditions of COAST computer code for the coastdown of the reactor coolant pumps; and

    - o change from CE-1 to KCE-1 CHF correlation employed in CETOP and STRIKIN-II computer codes to calculate thermal margin and minimum DNBR.

- The staff needs information regarding the assumption that, neglecting to not model cross flow in CESEC-III during highly non-symmetrical events, is acceptable. The staff will review the accuracy of CESEC-III to predict the pressurizer pressure during a turbine trip as presented in Figure 3.1-10 and related uncertainty of the model.

- The staff needs information to confirm the acceptability of COAST to predict conservatively the mass flow rate of the RCP during the first 5 to 10 seconds of a coastdown event and not during the complete period of the transient.

- The staff needs information to support the assumption on page 61 that a gap exist between the fuel pellet and the cladding when the radial gap is greater than 1.5 times the sum of the pellet plus the clad surface roughness.

- The staff will review HERMITE Version 1.6 and any changes from the previously approved version by the NRC.

Documents and Information Necessary for the Audit:

The following documents are to be made available to the NRC staff, either at the KHNP Washington, DC Center, or in the electronic reading room. This is not a comprehensive list of documents that the staff will be reviewing as part of the audit, as there may be a need to review additional data and calculations supporting the basis for these documents. Appropriate handling and protection of proprietary information shall be acknowledged and observed throughout the audit.

- Calculations and documentation related to changes to CESEC-III for POSRV, Centrifugal Pumps and ATWS.

- Calculations and documentation related to initialization and boundary conditions imposed on COAST computer code for the coastdown of the reactor coolant pumps.

- Calculations and documentation related to the use of KCE-1 in CETOP and non-LOCA version of STRIKIN-II.

- Calculations and documentation related to HERMITE Version 1.6 and any changes from the previously NRC approved version.

- NRC approval letters for computer codes used in APR1400 Tier 2 DCD Chapter 15 and technical reports incorporated by reference, including but not limited to COAST, HRISE, STRIKIN-II, PARCH, HERMITE Version 1.6, CELDA, BORON, CEPAC, NAFLOW, FATES and FIESTA.

- Calculations and documentation related to the barrier performance of the event described on Section 15.4.1 of APR1400 DCD Tier 2 and evidence that it is bounded by uncontrolled CEA withdrawal at power event described in Subsection 15.4.2.4.

- Calculations and documentations related to the barrier performance of the event described in Section15.4.1 and information to confirm that it is bounded by uncontrolled CEA withdrawal at power event described in Subsection 15.4.2.4.

- CESEC-III computer code user's manual. Input deck of the CESEC-III code for uncontrolled control element assembly withdrawal from a subcritical or low-power startup condition. Output file of the analyses.

- CETOP code computer code user's manual with the KCE-1 CHF correlation. Input deck of the CETOP code for control rod ejection. Output file of the analyses.

- Calculation documentation and supporting references for the Large Break LOCA analysis.

This list may need to be revised, as there may be a need to review additional data and calculations supporting the basis for these documents.

Audit Team:

Shanlai Lu (Team Lead), NRO, Senior Reactor Systems Engineer
Alexander Velazquez-Lozada, NMSS, Thermal Engineer
Jeff Schmidt, NRO, Senior Reactor Systems Engineer
James Gilmer, NRO, Reactor Systems Engineer
Alexandra Burja, NRO, General Engineer
Matt Thomas, NRO, General Engineer
Donald Carlson, NRO, Senior Project Manager
James Steckel, NRO, Project Manager

Team Assignments:

The following table identifies the NRC staff and their review areas:

Table-1. NRC On-site Review Team

Review Area	**NRC Staff Name**
Transient and Accident Analyses	S. Lu
Computer Codes	A. Velazquez
RCS Boron Concentration and Fuel Assemblies Position	J. Schmidt
Criticality	D. Carlson/A. Burja
ATWS	J. Gilmer
Control Element Assembly Ejection Accidents	M. Thomas
Steam System Piping Failure	T. Drzewiecki
Project \Support	J. Steckel

Applicant Contacts:

Christopher Tyree, KHNP
Harry Chang, KHNP

Logistics:

The NRC staff and the applicant have agreed that the audit will be conducted via an electronic reading room and in the KHNP office in Vienna, VA. In support of this approach, the applicant has agreed to make knowledgeable staff available, along with relevant documentation, to support the staff's review and discussion of the material.

The audit is scheduled for a period of three months and onsite audits will be scheduled as needed. The staff requests that all document titles identified by the NRC staff, be available at the beginning of the audit in the electronic reading room and prior to any scheduled onsite audit. 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.

Special Requests:

The NRC staff requests that KHNP provide:

- Document titles responsive to the audit areas listed in Section III of this audit plan.

- Searchable electronic copies of the documents listed above.

- KHNP personnel to provide any necessary overviews of the APR1400 reactor; reactor coolant system and connecting systems; auxiliary systems; and transient and accident analyses DCD information and related documents.

Audit Activities and Deliverables:

The NRC audit team review will cover the technical areas identified in the Documents and Information Necessary for the Audit section 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 any 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 Chapters 6 and 19 and any related information provided in other chapters, in preparation of the NRC staff's SER.

An audit will be conducted for approximately three months from NRC Headquarters via KHNP's electronic reading room; however the audit may also be carried out at KHNP's facilities in Vienna, VA, if the technical information is only retained in hard copy.

Follow up audits at the NRC Headquarters via KHNP's electronic reading room (or at KHNP's facilities in Vienna, VA), may be necessary at various times.

If necessary, any circumstances related to the conductance of the audit will be communicated to James Steckel (NRC) at 301-415-1026 or James.Steckel@nrc.gov.

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

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

3. APR Design Control Document, Revision 0, issued December 2014.