APR1400 NRC DC PRA SUMMARY

Young H. In[1], Han-gon Kim[2], Myungro Kim[3]

[1] *Enercon Services, Inc.: 3846 Gladstone Drive, Naperville, IL 60565, USA, yhin@enercon.com*
[2] *KHNP-CRI: 70, 1312-gil, Yuseong-daero, Yuseong-gu, Daejeon, 34101, Korea, kimhg1108@khnp.co.kr*
[3] *KEPCO E&C: 269 Hyeoksin-ro, Gimcheon-si, Gyeongsangbuk-do, 39660, Korea, mrkim@kepco-enc.com*

The APR1400 Design Certification Application (DCA) was submitted to the U.S. Nuclear Regulatory Commission (NRC) for review in December 2014, and the application was accepted for review by the NRC in March 2015. The NRC requires a DCA to include a description of the design-specific PRA and its results per 10 CFR 52.47 where NRC Regulatory Guide (RG) 1.206 and Standard Review Plan (SRP) Chapter 19 provide additional guidance.

A comprehensive PRA was performed for the APR1400 DCA that includes applicable internal and external initiating events and all plant operating modes. Some initiating events are screened from detailed analysis based on their applicability to the design phase while others are treated qualitatively (e.g., other external events). The approach used for risk evaluation of seismic events includes a PRA-based seismic margins assessment (SMA) rather than a seismic PRA. The APR1400 PRA is based on basic elements and approaches given in the ASME PRA Standard (ASME/ANS RA-Sa-2009), as endorsed by NRC RG 1.200, and using the methodological guidance of NUREG/CR-2300 and NUREG-1150. The PRA results and insights are documented in the APR1400 design control document (DCD) Chapter 19.

The PRA results and risk insights confirmed that the APR1400 design incorporates features to reduce the overall risk compared to operating plants, and demonstrates how the risk associated with the design compares against the NRC goals of less than 1×10^{-4}/year for core damage frequency (CDF) and less than 1×10^{-6}/year for large release frequency (LRF). This paper summarizes the results and insights from the APR1400 PRA, as documented in the DCD Chapter 19.

I. INTRODUCTION

The APR1400 is an Advanced Pressurized Water Reactor (ALWR) with a core thermal output rating of 1,400 MWe designed by Korea Electric Power Company (KEPCO) and Korea Hydro and Nuclear Power (KHNP) company. The history of the APR1400 design goes back to the Combustion Engineering (CE) System 80+, which is owned by Westinghouse. The original System 80+ design was improved by with advanced design features to enhance the safety and operational flexibility based on the plant design and operating experiences in Korea. Major design concepts and methodologies remain same as System 80+: RCS configuration and major components, four train safety injection system with direct vessel injection, CE safety analysis codes and methodologies, and other design features. The major design features that are different from System 80+ are as follows: pre-stressed concrete cylindrical containment, SIT fluidic device, improved digital I&C, advanced control room design, Plus 7TM fuel, passive autocatalytic recombiners (PAR) in addition to hydrogen igniters, other severe accident designed features such as emergency containment spray backup system (ECSBS) and external reactor vessel cooling (ERVC). Shin Kori unit 3, the first unit of the APR1400, started commercial operation in 2015, and unit 4 which is same design is expected to start operating in early 2017. Two more APR1400 unit are under construction as Shin Hanul units 1 and 2 in South Korea, which are expected to be completed in 2017 and 2018, respectively. Four more APR1400 units are planned for both the Shin Kori and Shin Hanul sites in Korea. Four more APR1400 units are under construction at Barakah in the United Arab Emirates, which are scheduled to be completed by 2020.

This paper summarizes information related to the probabilistic risk assessment (PRA) performed to support design certification (DC) of the APR1400 submitted to NRC in 2014 where the DC application was accepted in 2015. The APR1400 DC application is currently under review by NRC.

II. REGULATORY REQUIREMENTS

The primary objectives of the PRA during the design phase are as follows:

a. Identify and address potential design features and plant operational vulnerabilities, where a small number of failures could lead to core damage, containment failure, or large releases.

b. Reduce or eliminate the significant risk contributors of existing operating plants that are applicable to the new design by introducing appropriate features and requirements.

c. Select among alternative features, operational strategies, and design options to demonstrate that the design poses an acceptably low risk of severe accidents.

The PRA also identifies risk-informed safety insights based on systematic evaluations of the risk associated with the design, construction, and operation of the plant such that the following can be identified and described from Regulatory Guide 1.206, "Combined License Applications for Nuclear Power Plants (LWR) Edition," U.S. Nuclear Regulatory Commission, Rev. 0, June 2007:

a. Describe the design robustness, levels of defense-in-depth, and tolerance of severe accidents initiated by either internal or external events.

b. Describe the risk significance of specific human errors associated with the design, including a characterization of the significant human errors that may be used as an input to operator training programs and procedure refinement.

c. Demonstrate how the risk associated with the design compares against the NRC's goals of less than 1×10^{-4}/year for core damage frequency (CDF) and less than 1×10^{-6}/year for large release frequency (LRF). In addition, compare the design against the NRC's approved use of a containment performance goal, which includes: (1) a deterministic goal that containment integrity be maintained for approximately 24 hours following the onset of core damage for the more likely severe accident challenges and (2) a probabilistic goal that the conditional containment failure probability be less than approximately 0.1 for the composite of all core damage sequences assessed in the PRA.

d. Assess the balance of preventive and mitigative features of the design, including consistency with the NRC's guidance in SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor Designs," U.S. Nuclear Regulatory Commission, Washington, DC, letter issued April 2, 1993 and Staff Requirements Memoranda issued July 21, 1993, and the associated Staff Requirements Memorandum (SRM).

e. Demonstrate whether the plant design, including the impact of site-specific characteristics, represents a reduction in risk compared to existing operating plants.

f. Demonstrate that the design addresses known issues related to the reliability of core and containment heat removal systems at some operating plants (i.e., the additional TMI-related requirements in 10 CFR 50.34(f), "Contents of Applications; Technical Information – Additional TMI-related requirements," U.S. Nuclear Regulatory Commission, June 2009).

The results and insights of the PRA are used to support other programs as follows:

a. Support regulatory oversight processes such as the Mitigating Systems Performance Index (MSPI) and the significance determination process (SDP), and other programs that are associated with plant operations (e.g., Technical Specifications, reliability assurance, human factors, Maintenance Rule implementation).

b. Identify and support the development of specifications and performance objectives for the plant design, construction, inspection, and operation, such as the inspections, tests, analyses, and acceptance criteria (ITAAC); the reliability assurance program (RAP); Technical Specifications (TS); and combined license (COL) action items and interface requirements.

A COL applicant that references the APR1400 design certification is to confirm that the PRA in the design certification bounds the site-specific design information and any design changes or departures, or update the PRA to reflect the site-specific design information and any design changes or departures. The primary requirements, guidance, policies, and standards utilized to complete the PRA are specified in Reference 1 through 14:

PRA results and insights are addressed, including internal and external event evaluation during full-power operations and during low power and shutdown operations. External events that are evaluated include seismic, internal fire, and internal flood. Level 1 and Level 2 results are reported. This report also describes the uses and applications of the PRA, PRA quality, design, and operational features that are intended to improve plant safety, and PRA input to design programs and processes. Chapter 19 of the APR1400 Design Control Document (DCD)[15] describes these elements.

III. ANALYSES AND METHODS

The scope of the APR1400 PRA includes a Level 1 and Level 2 PRA for internal and external events (including internal flooding and internal fire) at full-power, as well as low power and shutdown (LPSD) conditions. The design changes resulting from the additional requirements established to manage and mitigate beyond design basis external events (BDBEEs) as a result of the Fukushima Dai-Ichi event are not included in the PRA.

The Level 1 and 2 evaluations of internal events at full-power conditions are based on the basic elements and approaches given in ASME/ANS RA-Sa-2009[14], as endorsed by U.S. Nuclear Regulatory Commission (NRC) Regulatory Guide (RG) 1.200[7], and using the methodological guidance of NUREG/CR-2300[16] and NUREG-1150[17]. Level 1 PRA evaluation is composed of the following technical elements:

a. Initiating event analysis
b. Accident sequence analysis
c. Success criteria analysis
d. System analysis (including system dependencies)
e. Data analysis and common cause analysis
f. Human reliability analysis (HRA)
g. Quantification

The Level 2 PRA results are produced in terms of large release frequency (LRF) for internal events at full power and the evaluation involves the following:

a. Plant damage state (PDS) analysis
b. Containment response analysis
c. Accident progression analysis
d. Quantification

The evaluation of internal fire events is based on the basic methodology and approach given in NUREG/CR-6850[18]. A qualitative evaluation identifies fire compartments and susceptible components and a quantitative analysis evaluates initiating events and fire scenarios. The evaluation of internal flooding is based on the basic methodology and approach given in the ASME/ANS PRA Standard. A qualitative evaluation identifies flood areas and sources and a quantitative analysis evaluates initiating events and flood scenarios. The Level 2 evaluation of the flooding and fire events at full-power conditions is based on the same approach as for internal events. Fault trees are modified to take into account flood/fire-induced failures of severe accident mitigation features and these fault trees are mapped into the internal events through the associated PDSs.

The evaluation of a seismic external event is based on the seismic margin analysis (SMA) guidance in the ASME/ANS PRA Standard[14]. The PRA-based SMA model is based on the internal events of the PRA model expanded to account for structural dependencies. Other external events (i.e., high winds and tornadoes, external floods, transportation accidents, nearby facility accidents, etc.) are subject to screening criteria consistent with the ASME/ANS PRA Standard.

The evaluation of internal events, internal fire and internal flooding for low power and shutdown (LPSD) operations, uses the same basic methods as the evaluations for operations at power. A representative set of initiating events is chosen and modeled for a set of plant operational states (POSs).

The APR1400 PRA is developed from the available design information. If sufficient design information is not available, then design information from the reference plants is used. The reference plants are Shin Kori units 3 and 4. Reference plant design information used includes, but is not limited to, System Descriptions, System Design Requirements, cable routing database, component fragilities, etc.

To be effective in supporting the design process and to provide meaningful results with regard to judging the overall risk posed by the design, the PRA reflects a level of detail limited by the following:

a. The availability of certain design details, operating procedures, and other information
b. The level at which usable reliability data are available
c. At present, elements of the detailed design and other supporting information that are not available to support the PRA include the following:
d. The specific routing of piping – relevant to an assessment of internal flooding events
e. The routing of control and power cables – relevant to an assessment of internal fire events
f. The specific location of key equipment within the rooms – relevant to assessments of internal flooding and fire events
g. Emergency and other operating procedures – relevant to human reliability analysis
h. The conceptual design information (CDI), which is not finalized during the design phase

PRA was performed that is consistent with the level of design detail available. In the case of internal fire events, the frequencies and the evaluation of equipment that could be affected reflect bounding assumptions. These assumptions have been refined, within the context of the available information, to avoid masking risk contributors from other sources due to overly conservative treatment.

A COL applicant that references the APR1400 design certification is required to review as-designed and as-built information and conduct walkdowns as necessary to confirm that the assumptions used in the PRA, including PRA inputs to reliability assurance program (RAP) and severe accident mitigation design alternatives (SAMDA), remain valid with respect to internal events, internal flooding and fire events (routings and locations of pipe, cable and conduit), and HRA (i.e., development of normal operating procedures, emergency operating procedures and training), external events (including PRA-based SMA based upon high confidence, low probability of failure (HCLPF) seismic fragilities), and the shutdown procedures.

The level of detail in the APR1400 PRA is commensurate with the guidance set forth in the ASME/ANS PRA Standard and NRC RG 1.200. Where detailed design information is not available, appropriate bounding assumptions are used consistent with the guidelines in the ASME/ANS PRA Standard and NRC RG 1.200.

IV. RESULTS

The phrase “risk insights” refers to the results and findings that come from PRA. Specifically, risk insights include information about:

a. Design features that are the highly effective in reducing risk with respect to operating plants
b. Major contributors to risk, including equipment failures and operator actions
c. Major contributors to the uncertainty associated with the risk results

Risk insights from each hazard evaluated for different operational modes (shown below) are shown below as described in the DCD:

- Internal Events PRA for Operations at Power
- Internal Fire Risk Evaluation
- Internal Flooding Risk Evaluation
- Level 1 Internal Events PRA for LPSD Operations

- Level 2 Internal Events PRA for LPSD Operations
- Internal Fire PRA for LPSD Operations
- Internal Flooding PRA for LPSD Operations
- Seismic Risk Evaluation (qualitative assessment)

Operation Modes	Hazards	Level 1 (CDF)	Level 2 (LRF)
Full Power	Internal Events	1.3E-06	1.2E-07
	Internal Fire	1.9E-06	1.6E-07
	Internal Flood	2.2E-07	1.7E-08
LPSD	Internal Events	2.8E-06	1.2E-07
	Internal Fire	1.7E-06	1.3E-07
	Internal Flood	1.8E-08	1.8E-08

The total CDF from all internal events, external events, at-power and shutdown analyses is less than 8E-6/yr. The total LRF from all contributors is about 5E-7/yr where the units for CDF and LRF are expressed in terms of "reactor calendar year" (shortened to "/yr"). Level 2 portion of the internal flood for LPSD modes were bounded by CDF.

A list of significant PRA insights and assumptions regarding how the design and operational features affect the plant risk, and how uncertainties affect the PRA models in representing the plant risk is described in the DCD. To provide reasonable assurance that this information is incorporated into the design process, the PRA insights and assumptions are categorized as follows:

a. Design Requirement: A design feature that requires specific design details be preserved to maintain its validity. If a future design change affects a design requirement, the PRA model needs to be reanalyzed to determine the significance of the change. Each design requirement is referenced to applicable subsection(s) in the DCD.

b. Operational Program: An operational feature that requires specific operational programs, such as procedures or training, to be preserved to maintain its validity. Development of operating and maintenance procedures is the responsibility of the COL applicants. Other operational programs that address PRA insights and assumptions are the Maintenance Rule, Technical Specifications, and development of the baseline PRA model. Each operational program is referenced to applicable COL item(s).

c. PRA Model Insight: An assumption that provides significant information about the PRA model or its results, but does not require design details or operational programs to maintain its applicability. PRA model insights should be maintained in the Baseline PRA model development and should be considered when making risk-informed decisions.

There are total of 64 specific items identified as key risk insights where 40 items are related to the base design features (including severe accident design features) and the operational programs that are important in PRA, and 24 items are directly associated with PRA assumptions, models, results and importance.

V. CONCLUSIONS

The APR1400 design has evolved from current PWR technology that incorporates features intended to make the plant safer and easier to operate as compared to currently operating plants. The PRA results and risk insights should confirm that the design incorporates features to reduce overall risk compared to operating plants. The PRA, as evaluated and documented in the DCD, has been used to achieve the following:

a. Identify and address potential design and operational vulnerabilities (i.e., failures or combinations of failures that are significant risk contributors that could drive the risk to unacceptable levels with respect to NRC safety goals).

b. Reduce or eliminate known weaknesses of existing operating plants that are applicable to the new design, by introducing appropriate features and requirements.

c. Select among alternative features, operational strategies, and design options.

d. Develop an in-depth understanding of the design's robustness and tolerance of severe accidents initiated by either internal or external events.

e. Examine the risk significance of specific human errors associated with the design, and characterize the significant human errors in preparation for better training and more refined procedures.

f. Determine how the risk associated with the design compares against the NRC safety goals of less than 1E-4/year for core damage frequency (CDF) and less than 1E-6/year for large release frequency (LRF).

g. Determine containment performance against the NRC containment performance goal, which includes a deterministic goal that containment integrity be maintained for approximately 24 hours following the onset of core damage for the more likely severe accident challenges and a probabilistic goal that the conditional containment failure probability (CCFP) be less than approximately 0.1 for the composite of core damage sequences assessed in the PRA.

h. Assess the balance of preventive and mitigate features of the design, including consistency with guidance in SECY-93-087 and the associated staff requirements memoranda.

i. Demonstrate that the plant design represents a reduction in risk compared to existing operating plants.

j. Demonstrate that the design addresses known issues related to the reliability of core and containment heat removal systems at some operating plants.

k. Support regulatory oversight processes and programs that are associated with plant operations (e.g., Technical Specifications, reliability assurance, Maintenance Rule, etc.).

l. Identify and support the development of design requirements, such as inspection, tests, analysis, and acceptance criteria (ITAAC), reliability assurance program (RAP), Technical Specifications, and combined license (COL) action items and interface requirements.

ACKNOWLEDGMENTS

This work was performed under the sponsorship of KEPCO and KHNP, as a part of the APR1400 design certification submittal to NRC.

REFERENCES

1. NRC Policy Statement 50 FR 32138, "Severe Reactor Accidents Regarding Future Designs and Existing Plants," U.S. Nuclear Regulatory Commission, August 1985.
2. NRC Policy Statement 51 FR 28044, "Safety Goals for the Operations of Nuclear Power Plants," U.S. Nuclear Regulatory Commission, August 1986.
3. NRC Policy Statement 52 FR 34884, "Nuclear Power Plant Standardization," U.S. Nuclear Regulatory Commission, September 1987.
4. NRC Policy Statement 59 FR 35461, "Regulation of Advanced Nuclear Power Plants," U.S. Nuclear Regulatory Commission, July 1994.
5. NRC Policy Statement 60 FR 42622, "The Use of Probabilistic Risk Assessment Methods in Nuclear Regulatory Activities," U.S. Nuclear Regulatory Commission, August 1995.
6. 10 CFR 50.34, "Contents of Applications; Technical Information," U.S. Nuclear Regulatory Commission, June 2009.
7. Regulatory Guide 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," Rev. 2, U.S. Nuclear Regulatory Commission, March 2009.

8. Regulatory Guide 1.206, “Combined License Applications for Nuclear Power Plants (LWR) Edition,” U.S. Nuclear Regulatory Commission, Rev. 0, June 2007.
9. SECY-90-016, “Evolutionary Light-Water Reactor (LWR) Certification Issues and Their Relationship to Current Regulatory Requirements,” U.S. Nuclear Regulatory Commission, Letter issued January 12, 1990 and Staff Requirements Memoranda issued June 26, 1990.
10. SECY-93-087, “Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor Designs,” U.S. Nuclear Regulatory Commission, Washington, DC, letter issued April 2, 1993 and Staff Requirements Memoranda issued July 21, 1993.
11. SECY-06-0220, “Licenses, Certifications, and Approvals for Nuclear Power Plants,” Final Rule to update 10 CFR Part 52, (RIN AG24), U.S. Nuclear Regulatory Commission, Letter issued October 31, 2006.
12. NUREG-0800, “Probabilistic Risk Assessment and Severe Accident Evaluation for New Reactors,” Section 19.0, Rev. 2, U.S. Nuclear Regulatory Commission, June 2007.
13. ASME/ANS RA-S-2008, “Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications,” (Revision 1 RA-S-2002), American Society of Mechanical Engineers, April 2008.
14. ASME/ANS RA-Sa-2009, Addenda to ASME/ANS RA-S-2008, American Society of Mechanical Engineers, February 2009.
15. APR1400-K-X-FS-14002-P, “APR1400 Design Control Document Tier 2”, KEPCO & KHNP, Revision 0.
16. NUREG/CR-2300, “PRA Procedures Guide,” U.S. Nuclear Regulatory Commission, January 1983.
17. NUREG/CR-1150, “Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants,” U.S. Nuclear Regulatory Commission, December 1990.
18. NUREG/CR-6850, “EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities,” U.S. Nuclear Regulatory Commission, September 2005.