KHNP APR1400 DESIGN CERTIFICATION APPLICATION DEVELOPMENT AUDIT REPORT

Introduction

On March 23, 2009, Korea Electric Power Corporation (KEPCO) initially notified The Nuclear Regulatory Commission (NRC) of its interest to begin pre-application discussions and its intent to submit an application under Title of the Code of Federal 10 (CFR) Part 52 for a Design Certification (DC) for its Advanced Power Reactor (APR1400) design. This letter is available in the Agencywide Document Access and Management System (ADAMS) under accession number ML090890521. Since 2009, KHNP has kept NRC informed of its progress toward submitting the APR1400 DC application. At the time this audit was planned, KHNP planned to submit its DC application in July 2013 (ML12311A267). On August 29, 2011, Korea Hydro and Nuclear Power Co., Ltd. (KHNP) informed NRC that the Korean government reorganized the nuclear industry in Korea and moved the nuclear research and development function of KEPCO to KHNP. KEPCO and KHNP reached an agreement that both companies would work together to submit the DC application with KHNP taking the lead (ML13168A311). On July 25, 2013, KHNP notified NRC that KHNP and KEPCO will submit the DC application as co-applicants (ML13206A309). On August 22, 2013, KHNP notified the NRC (ML13239A122) that the DC application would be submitted at the end of September 2013 to allow time to improve the completeness and level of detail of the DCD

Since 2009, NRC has held twelve pre-application meetings with KHNP on the APR1400 design. The purpose of these meetings was for KHNP to inform NRC staff of the basics of the design and to obtain generic feedback from NRC staff to develop a complete and technically sufficient application.

During this period, KHNP also submitted its Quality Assurance Program Description (QAPD) for review and approval, along with three other Topical Reports: Fluidic Device Design; Realistic Evaluation Methodology for Large Break Loss-of-Coolant Accident; and KCE-1 Critical Heat Flux correlation for PLUS7 Thermal Design. NRC completed its review of the QAPD and will soon issue its Safety Evaluation. NRC accepted the three Topical Reports for review.

As part of its policy to effectively and efficiently plan review work, the NRC staff has audited other DC applications prior to their formal submission for staff review. To continue implementing this policy, the NRC staff conducted a pre-application audit of the KHNP APR1400 Design Certification Document (DCD) and supporting technical reports and calculations. The audit focused on six major areas identified from previous DC applications to have required significantly more staff interactions with the applicants to address unresolved design issues and therefore have caused significant schedule challenges.

NRC Audit Team:

The following NRC staff members from The Office of New Reactors (NRO) participated in the audit (see Enclosure 2 for a complete list of audit participants):

- William Ward (Senior Project Manager)
- Samuel Lee (Branch Chief)
- Bret Tegeler (Audit Team Lead for Seismic Analysis)
- Clinton Ashley (Audit Team Lead for Long Term Cooling, ex-vessel)
- Shanlai Lu (Audit Team Lead for Long Term Cooling, in-vessel)
- Deanna Zhang (Audit Team Lead for Instrumentation and Controls
- Michelle Hart (Audit Team Lead for Radiation Protection)
- James Bongarra (Audit Team Lead for Human Factors Engineering)
- Todd Hilsmeier (Audit Team Lead for Probabilistic Risk Analysis)

The following key individuals from Korea Hydro and Nuclear Power Co. Ltd. (KHNP), their partners and consultants participated in the audit (see Enclosure 2 for a complete list of audit participants):

- Myung Ki Kim, DCD Project Manager
- Seung Jong Oh, Advisor
- Yun Ho Kim, Director, KHNP Washington Office
- Harry (Hyun Seung) Chang, DCD Licensing Engineer

1.0 SUMMARY

The audit was conducted at the Hilton Hotel and Executive Meeting Center in Rockville Maryland, from June 3 to June 7, 2013. The NRC staff conducted the audit in accordance with the NRO Office Instruction NRO-REG-108, “regulatory Audit”. The plan for this audit was issued on May 29, 2013, and is available in ADAMS under accession number ML13143A529. During the audit, the NRC team and KHNP met daily to discuss issues identified by the NRC team.

The audit focused on the following major areas that have been identified from previous design certification applications that required the greatest staff effort to review and/or caused the greatest impacts to review schedules:

- Seismic Analysis (Chapter 3 and Section 19.1.5 of the DCD).
- Long-term cooling and Generic Safety Issue 191 (GSI-191) (Chapters 4 and 6).
- Instrumentation and Controls (Chapter 7).
- Radiation Protection (Chapters 11, 12, and 15, and Section 3.11).
- Severe Accident Analysis and Probabilistic Risk Assessment (Chapter 19).
- Human Factors Engineering (Chapter 18).

The KHNP provided multiple copies of the APR1400 DCD, the Topical Reports, and major supporting technical reports. Examples of important calculations were also provided. KHNP set up laptop computer workstations connected to a printer and a collection of compact disks to provide information not provided in hard copy.

The audit commenced with an entrance meeting. At this meeting, KHNP and NRC discussed the schedule of activities for the audit, initial documents for review, and introduced their key staff. Daily briefings were held by the NRC audit team to discuss observations. The audit and the briefings were attended by those identified in Enclosure 2. At the final exit briefing, the NRC audit team stated that all of its objectives as stated in the audit plan had been met. However, NRC staff also identified a number of gaps between staff's expectations for a complete and technically sufficient DCD and the DCD that KHNP provided for the audit. The gaps are discussed in Sections 3.1 through 3.8 this report.

2.0 REGULATORY BASIS

- Title 10 of the *Code of Federal Regulations (CFR)*, Part 52 – Licenses, Certifications, and Approvals for Nuclear Power Plants, provides the requirements regarding an application for a new reactor design certification. Subpart B – Standard Design Certifications, section 52.46 – Contents of applications; general information, states, "The application must contain all of the information required by 10 CFR 50.33(a) through (c) and (j)."

- Title 10 of the *Code of Federal Regulations (CFR)*, Part 52 – Licenses, Certifications, and Approvals for Nuclear Power Plants, provides the requirements regarding an application for a new reactor design certification. Subpart B – Standard Design Certifications, section 52.47 – Contents of applications; technical information, states, "The application must contain a level of design information sufficient to enable the Commission to judge the applicant's proposed means of assuring that construction conforms to the design and to reach a final conclusion on all safety questions associated with the design before the certification is granted. The information submitted for a design certification must include performance requirements and design information sufficiently detailed to permit the preparation of acceptance and inspection requirements by the NRC, and procurement specifications and construction and installation specifications by an applicant."

- Title 10 of the *Code of Federal Regulations (CFR)*, Part 52 – Licenses, Certifications, and Approvals for Nuclear Power Plants, provides the requirements regarding an application for a new reactor design certification. Subpart B – Standard Design Certifications, section 52.47(a), states, "The application must contain a final safety analysis report (FSAR) that describes the facility, presents the design bases and the limits on its operation, and presents a safety analysis of the structures, systems, and components and of the facility as a whole, and must include the following information:…"

- Regulatory Guide (RG) 1.206, "Combined License Applications for Nuclear Power Plants (LWR Edition)," Parts I and III, provide additional guidance regarding the information to be provided by the DC applicant to support a future Combined License Applicant.

- Design Certification Applications are evaluated following the guidance provided in NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," (SRP).

3.0 OBSERVATIONS AND RESULTS

The KHNP made available key members of their technical staff to answer NRC staff questions and discuss any topics that arose. A large amount of information was made available in a well-organized manner. The audit team also found the logistical arrangements to be very well thought out and planned by KHNP. This allowed NRC staff to focus on the audit in quiet rooms with all the necessary documents. Additional breakout rooms were available for discussions.

In summary, the audit accomplished all of the intended objectives. The staff had the opportunity to review a large number of documents and discuss the documents with KHNP technical staff. All of the audit areas outlined in the audit plan were evaluated.

The observations and findings made during this audit constitute NRC staff's feedback to KHNP on areas of the APR1400 DCD that should be improved prior to the application submission for NRC review. These should be treated by KHNP as recommendations, which, if implemented, will increase the likelihood of acceptance when the application is submitted. To be clear, the staff has performed an audit and not an acceptance review of the APR1400 DCD and is not making an acceptance decision at this time. Staff will not follow up on these findings.

The NRC staff made the following observations:

3.1. Seismic Analysis

This portion of the audit focused on DCD sections pertaining to seismic analysis, structural design, wind/hurricane, flooding, barrier design, seismic margin, and containment performance. The guidance and acceptance criteria used by the staff were from RG 1.206 and the relevant SRP sections. Further, the audit considered lessons learned from the recent reviews of other new reactor design certification applications.

The KHNP provided a significant amount of supporting documentation (e.g., design calculations), in addition to the DCD, to help NRC staff in understanding the technical basis for the APR1400 design. Staff found this material useful and relevant to the review and believes that KHNP should consider incorporating some of this material into DCD sections that require additional information based on the findings of this audit.

The seismic team identified review areas where necessary DCD information was missing. Some of the missing information is considered to be significant (i.e., potential for impacting overall structural design evaluation) while other missing information is considered to have a relatively lower impact on the design evaluation. In addition, the team identified areas of the DCD with low levels of quality (i.e., information was insufficient or not clear enough to perform a substantive review). The DCD must provide information that is clear and complete for the staff to perform the safety review in an effective and timely manner. Without the identified information, particularly the information pertaining to the facility's structural design description, getting past the acceptance review would be difficult. The key audit findings and observations are provided below, followed by a more detailed description.

Key Findings and Observations

Staff identified the following examples of important missing information in the KHNP DCD. Staff notes that past and present DC applicants have performed significant re-design and/or analysis efforts in the three subject areas identified below. Therefore, it is important to provide full descriptions of these areas in the application so that any issues can be identified and addressed early and negative schedule impacts are avoided.

1. An aircraft impact assessment (AIA) summary and results were not provided. Staff is aware that this is dependent on the availability of other information and that its development is in progress.
2. Foundation stability analysis results for the nuclear island were not provided.
3. Analysis of Central and Eastern United States (CEUS) hard-rock high frequency (HRHF) ground motions was not provided.

Staff identified the following examples of missing information that is necessary for staff review, but is less likely to have a significant impact on the facility structural design, based on staff's judgment of the APR1400 design.

1. Fuel rack seismic analysis was not provided.
2. Description, analysis, and details of large containment penetration designs were not provided.
3. Seismic fragility values, defined as high confidence in the low probability of failure (HCLPF), for significant structures, systems, and components (SSCs) were not provided.
4. Description and analysis of the seismic interaction of Seismic Category II SSCs over Seismic Category I SSCs (Category II/I) were not provided.
5. Description of the construction sequence of the reactor building complex was not provided.

Examples of the lack of sufficient information in the DCD included:

1. Minimal description of containment internal structures (Seismic Category I structure) design.
2. Minimal description of the analysis and design of seismic subsystems.
3. Insufficient number, description, method for seismic detailing, and design of critical sections.

Detailed Findings and Observations (by DCD Section)

DCD Tier 1 Information

Staff review of KHNP DCD Tier 1 information identified missing information in the site parameters table (Table 2.1-1) such as the dynamic bearing capacity, minimum soil angle of internal friction, and criteria for allowable settlement between adjacent buildings. These engineering parameters are important for ensuring reactor building stability, and function of building interfaces (e.g., piping, conduits, tunnels, etc.) and for verification of the suitability of the standard design at a given site.

Staff review of Tier 1, Subsection 2.2, found that the functional description of site-specific buildings/structures (essential service water supply structure, component cooling water heat exchanger building, essential service water conduits, and class IE electrical duct runs) was not provided. In addition, staff noted that some building general arrangement figures (Subsection 2.2.1) were missing key dimensions and column line designations. Staff also noted that Tables 2.2.1 and 2.2.2 included thickness of wall/floor member thicknesses, but did not provide construction tolerances.

Staff review noted the certified seismic design response spectra (CSDRS) are based on an enhanced RG 1.60 design spectra that does not envelop the CEUS seismic ground motion designated here as HRHF ground motion. The DCD (Tier 2, Subsection 3.7) refers to the analysis of effects of HRHF, but is not clear whether HRHF is considered as an additional CSDRS for rock sites. If HRHF is to be considered as a CSDRS, it will need to be included in Tier 1. It should also be noted that while Subsection 3.7.1.1.3 discusses the analysis of CEUS HRHF, and references Appendix 3.7B, the Appendix 3.7B contains no information.

DCD Tier 2 Information

Staff review of DCD Tier 2, Subsection 1.2, found that there was no site layout provided to show the proposed site configuration, to indicate the scope of the certified design, and to differentiate what is considered as site-specific structures.

Staff review of DCD Tier 2, Subsections 3.3.1 and 3.3.2 pertaining to wind loading and extreme winds respectively, found these sections to address the relevant SRP criteria. Staff's review of DCD Tier 2, Subsection 3.4.2, relating to internal/external flooding identified that a foundation waterproofing system was not identified. For DCD Subsection 3.5.3, Barrier Design Procedures, the method for analyzing the effects of concrete spalling was not provided.

Staff review of Subsection 3.7.1, "Seismic Design Parameters", found that there is insufficient discussion of how the standard design satisfies the minimum design ground motion of 0.1 g peak ground acceleration (PGA) at the foundation elevation in the free-field (Appendix S to 10 CFR Part 50) using the CSDRS and the design basis soil profiles.

Staff review of Subsection 3.7.2, "Seismic System Analysis", found that the results of the assessment of the standard design for the effects of seismic interaction, or Category II/I, is not provided. Guidance for assessing the effects of Category II/I is provided in SRP Section 3.7.2.8.

As mentioned earlier, staff review notes that Appendix 3.7B, pertaining to HRHF effects, is empty. Staff notes that if incoherent ground motion is considered, then the application should provide comparisons of incoherent and coherent in-structure response spectra as described in SRP Section 3.7.2.

Staff review also noted that Section 3.7.2 is not clear on where the various analysis methods and software are used (e.g., direct integration time-history versus SASSI). In addition, the application should clearly reflect whether the SASSI subtraction or direct method was utilized for the seismic analysis. Further, the staff found that there is insufficient discussion on how floor loads, live loads, and major equipment are represented in the dynamic models.

Staff review of Subsection 3.7.2 found that the DCD does not describe sensitivity studies for addressing uncertainties in seismic analysis (ref. SRP Section 3.7.2). Sensitivity studies are typically performed to address modeling assumptions pertaining to water table effects, foundation uplift, location of model boundaries, element discretization, etc.

Staff review found that the DCD is missing a description of the method for considering concrete cracking in seismic analysis (ref. SRP 3.7.2). Staff noted that the DCD mentions that the method for developing member forces for structural design is based on equivalent static methods, but found no justification or discussion of the level of conservatism of this approach, as described in SRP 3.7.2.

Staff review of Subsection 3.7.2 found no description of the method for a Combined License (COL) applicant to compare site-specific parameters (shear wave velocity profiles, foundation input spectra) to the assumptions used in the DCD seismic analysis. In addition, staff noted that key building locations should be identified (with enveloped and broadened in-structure response spectra [ISRS]) so that a COL can compare ISRS when site-specific analysis is performed for the standard plant to justify the standard plant suitability at a given site.

Staff review of Subsection 3.7.3, “Seismic Subsystem Analysis”, found that the DCD is missing information pertaining to a number of SRP review areas. For example, the determination of number of earthquake cycles, criteria used to separate fundamental frequencies of components and equipment from the forcing frequencies of the supporting structure, interaction of other systems with Seismic Category I structures, multiple-supported equipment and components with distinct input, and the torsion effects of eccentric masses are missing.

Staff review of DCD Section 3.8.1, “Concrete Containment”, found that the section does not provide a value for the containment ultimate pressure capacity as discussed in SRP Section 3.8.1. Regulatory criteria require a determination of the internal pressure capacity for containment structures as a measure of safety margin above the design basis accident pressure. In addition, loads from external pressures, explosions, and aircraft impact were not clearly addressed.

Staff review of this subsection also found that the containment design has minimal description of key critical section details and corresponding descriptions. For example, the design of the connection of the containment shell to the basemat was not provided. In addition, the reinforcement design near large penetrations was not provided. These locations are critical areas where past containment testing has shown structural failures to initiate. It should be noted that information regarding these, and other sections considered to be critical for the design of the reactor containment building, should be designated in the application as Tier 2*. The DCD did not provide description, analysis, and details of large containment penetrations such as the equipment hatch and personnel access airlocks. While detailed design is not expected for these steel components at the DC stage, a design description and general layout design drawings should be provided in addition to an analysis to justify the containment capacity with presence of the large penetrations.

Staff performed a review of Subsection 3.8.3, “Concrete and Steel Internal Structures of Concrete or Steel Containments”, and found the DCD to have insufficient general arrangement drawings and design description. In addition, the DCD did not contain critical sections such as the reactor supports, steam generator supports, pressurizer supports, connections to the basemat, and other internal structures identified in Subsection 3.8.3.

The design and physical configuration of these sections should be clearly described in figures and text. Staff review of this Subsection noted that while America Concrete Institute standard number 349 (ACI 349) is referenced for design of the structures, the Subsection did not discuss methods for performing seismic detailing (e.g., Chapter 21 to ACI 349) to ensure ductility under beyond-design basis events. In addition, no specific version of the ACI 349 code, or any of the codes and standards applicable to Section 3.8, was provided.

Staff performed a review of DCD Subsection 3.8.4, “Other Seismic Category I Structures”, and similarly found that there is minimal description of critical sections (including lack of figures) for the auxiliary building and diesel generator buildings. The application should describe the method for identifying critical sections, clearly describe the method of design, and clearly indicate essential seismic detailing features such as the use of standard hooks and stirrups. Further, staff review of Subsection 3.8.4 found that the DCD contained no discussion on spent fuel rack (and assemblies) seismic analysis. An acceptable method of evaluating spent fuel rack assemblies is described in SRP Section 3.8.4.

Lastly, staff review of DCD Subsection 3.8.4 found no reference to inspection and/or maintenance criteria for any structures, other than the reactor containment building, after construction. The Maintenance Rule (10 CFR 50.65) requires maintenance programs for both Seismic Category I and II structures.

Staff performed a review of DCD Subsection 3.8.5, “Foundation Design”, and found that there is minimal description of critical sections (including lack of figures) for the reactor building basemat and connection of the exterior foundation walls to the basemat. In addition, there was no description of the use of a waterproofing membrane (mentioned above) and the use of foundation basemat mudmat, if applicable. The DCD also did not contain a description of concrete placement and sequence of construction as discussed in SRP 3.8.5. The staff review also noted that the limited number of figures provided of the nuclear island foundation were not consistent with the DCD text description. Similarly, information describing the foundations of the Emergency Diesel Generator Buildings was limited in both text and figures.

Staff reviewed Subsection 3.8.6, “Combined License (COL) Information Items”, and found that they lack specificity in technical areas relating to design and analysis of safety related structures. The information items also lacked specific criteria for the COL applicant to address information items.

Staff performed a review of DCD, Subsection 19.1.5, “Seismic Margin”, and found that the DCD section does not contain a Seismic Equipment List indicating HCLPF values for significant SSCs. In addition, this section does not contain a COL action item for confirming that the as-built plant has adequate seismic margin (ref. DC/COL-Interim Staff Guidance [ISG]-20).

The design methodology and structural analysis of the sump strainer were not included in the technical reports. Section 3.2 discusses this information gap further.

3.2. Long Term Cooling and Generic Safety Issue 191 (GSI-191)

This portion of the audit report provides the staff's observations related to Generic Safety Issue 191 (GSI-191). GSI-191 was identified from NRC's experience with previous design certification application reviews to have required additional staff interactions with the applicants and/or caused significant delays to schedules. The staff reviewed the APR1400 DCD Chapter 4 "Reactor", Chapter 6 "Engineered Safety Features", and Chapter 9 "Auxiliary Systems", the GSI-191 Technical Report, and long term cooling related calculations.

Detailed Findings and Observations

Long Term Core Cooling

Staff findings and observations are detailed in Enclosure 4, Table 1, "APR1400 Pre-Application Audit – General Observations on Generic Safety Issue -191," and Enclosure 5, Table 2, "APR1400 Pre-Application Audit – Observations on DCD Tier 2, Chapters 6 and 9, and Tier 1, section 2.7 (By Design Control Document (DCD) Section)."

The following is a summary of the details found in Enclosures 4 and 5.

- Documenting the design basis for the long term cooling analysis (e.g., strainer and fuel):

  The staff understands that KHNP evaluated strainer and fuel performance using staff guidance (SECY-12-0093, Option 1 and associated clean plant criteria) and did not conduct design-specific testing. However, the staff views that KHNP did not adequately demonstrate that the conditions and assumptions used to develop the staff's guidance were applicable to the APR1400 design. The staff believes that KHNP should document/reference how existing tests, which helped form the basis for the staff's guidance, are applicable to APR1400 design in order to demonstrate that no additional design-specific testing is required. If the available (existing) tests do not fully address KHNPs design and operating conditions, then the staff would expect KHNP to conduct design-specific testing. All current PWR design certification applications developed and support their design/licensing basis using the results of the design-specific testing of strainers and fuel assemblies. These results are provided in the DCD or supporting technical reports. Staff believes that a similar approach is needed for the APR1400.

- Content of the documents reviewed:

  The staff observed apparent inconsistencies, lack of clarity, missing information, and some non-conservative assumptions in the documentation KHNP provided to the staff. During the audit, staff shared examples with KHNP in areas such as net positive suction head (NPSH), coatings, debris characteristics, debris transport assumptions for strainer and fuel, strainer performance, water holdup, structural analysis of the debris screens, and fuel performance.

- KHNP design changes:

  The staff acknowledges the positive developments that KHNP has made related to the debris source term for the APR1400 design since it was originally proposed roughly two years ago. Specifically, KHNP redesigned the insulation types used on components and piping in containment, such that the fiber debris source term from insulation debris is eliminated and a relatively small amount of fiber debris is present in the form of latent debris.

Conclusion:

The staff concluded that DCD Tier 2, Chapters 6 and 9 of the KHNP APR1400 DC application are consistent with the format and content prescribed in RG 1.206, with the exceptions noted in the Tables 1 and 2 (Enclosures 4 and 5). The staff concluded that the GSI-191 Technical Report and associated long term cooling calculations contained areas for improvement in order to be consistent with the content prescribed by RG 1.82 (RG 1.82 is referenced by RG 1.206). The staff believes that these exceptions and areas for improvement, if not appropriately addressed, would pose a risk to the acceptance review phase of the APR1400 Design Certification application.

GSI-191 In-vessel Downstream Effects

Sump Design Technical report APR-1400-E-A-T-13001-P describes KHNP's evaluation of the in-vessel downstream effects. KHNP appears to adopt the clean plant option described in SECY-12-0093, "Closure Option for Generic Safety Issue – 191, Assessment of Debris Accumulation on Pressurized-Water Reactor Sump Performance."

The APR1400 plant uses reflective metal insulation for the components and generates no fibrous debris within the zone of influence. KHNP assumes 150 pounds of latent debris in the containment with 15 percent, or 22.5 pounds, of fibrous debris. Assuming 75 percent debris transport to the IRWST and 45 percent of sump strainer bypass, the fibrous debris reaching the reactor core is 14.3 grams per fuel assembly based on 241 fuel assemblies in the core. This appears to satisfy the clean plant criterion of 15 grams per fuel assembly (FA) fibrous debris described in the letter from William Ruland of NRC to John Butler of NEI, "NRC Review of Nuclear Energy Institute Clean Plant Acceptance Criteria for Emergency Core Cooling Systems," dated May 2, 2012. The assumption of 75 percent debris transport and 45 percent strainer bypass of fiber appear to be obtained from the Ruland letter, which were based on operating plants data. The applicability of these assumptions to the APR1400 plant and sump strainer design needs to be justified with debris bypass tests for the specific strainer design.

The technical report provides a comparison of the debris capture characteristics of the PLUS7 fuel assembly used in the APR1400 plant and Combustion Engineering's (CE) Guardian fuel assembly, which was evaluated in WCAP-16793, Revision 2. KHNP concluded that since there is only a small difference in the flow areas between the PLUS7 and CE Guardian fuel, the head loss that could occur with PLUS7 FA is bounded by that of the standard P-grid since the head loss in Guardian grid is bounded by the standard P-grid.

The technical report also provides hand calculations of the available driving head for the hot leg break and cold leg break conditions, respectively. The calculations are based on some assumptions that may not be conservative.

The calculations conclude that the available driving heads are slightly higher than that described in WCAP-16793. With the available driving head and the debris capture characteristics of the PLUS7 fuel assembly, KHNP concludes that APR1400 meets the clean plant criterion and therefore does not propose to conduct fuel assembly head loss tests to demonstrate long-term core cooling capability with debris blockage. However, since APR1400 is a new plant design, and consistent with other new plant designs with similar amount of fibrous debris per FA, the applicant must provide test information applicable to their fuel design to demonstrate that the debris induced flow blockage would not impede long-term core cooling.

Another comment: The available driving head calculations were performed for the hot leg break and cold leg break conditions with the safety injection into the direct vessel injection lines. Since there is an operator action to switch the safety injection to simultaneous hot-leg and direct vessel injection about 90 minutes after the event initiation, the available driving heads for this safety injection configuration need to be evaluated by the applicant.

DCD Section 4.2 Fuel Design and Fuel Design Topical Report

APR1400 fuel design information is mainly summarized in the Section 4.2 of its pre-submittal DCD and the detailed information will be provided in the planned re-submittal of the fuel design topical report. The initial screening of the report and the review of information collected from KHNP through three meetings on this subject identified the need of new information in the following areas:

- The fuel performance code package used to analyze the fuel behavior under various operating and accident conditions: It is not clear to the staff whether these methods were approved before or not. Even if they are approved, it is not clear whether the original approval is still applicable to the PLUS7 fuel design of APR1400.

- The fuel testing and post-irradiation data associated with the PLUS7 fuel are not part of the DCD, or part of the original fuel design topical report. These test data are the key to demonstrate that the fuel design has been successfully used in operating PWRs and the analysis package has been validated using these data.

- The fuel seismic analysis under safe-shutdown earthquake (SSE) and loss-of-coolant accident (LOCA) loads: No information is currently available in DCD Section 4.2 or in its fuel design topical report although staff was shown an internal analysis report on this subject.

- The post-irradiation and poolside testing plan. This is needed as part of the design review submittal to allow future verification of the fuel design once the COL issued.

After the identification of the above information through the fuel design topical report review and the DCD pre-submittal audit, KHNP discussed a plan to submit the required information and shared an internal fuel seismic analysis report for staff to audit. According to KHNP's plan, the fuel performance codes and methods of discussed in the first bullet above and PLUS7 fuel testing data will be added into the new fuel design topical report when the revised version is submitted in fall of 2013. The fuel seismic analysis under SSE and LOCA loads will be submitted to NRC as a technical report based on the internal analysis report shared with the staff during the audit. The post-irradiation and poolside-testing plan is briefly mentioned in the DCD. Additional enhancement will be provided through the RAI process.

The KHNP fuel seismic analysis report titled “Structural Analysis of RVI [reactor vessel internals] and Fuel Assemblies for Seismic and Branch Line Pipe Break Loads” June 2013, KNF-TR-IMD-13001 Rev. 00 was audited by the staff. Staff met with KHNP experts twice and discussed the following aspects of the report:

- Component stress analysis;
- Flow induced damping during LOCA;
- CE-SHOCK code applicability to PLUS7 fuel design and the approval history;
- Irradiation assisted growth;
- Pluck test vs. forced vibration test; and
- None-linear effect with large deflection;

Throughout the interaction, staff provided the detailed feedback on each aspect and expressed the expectations of the new submittal contents and the level of details.

Fuel Pellet Thermal-Conductivity Degradation Issue

NRC issued Information Notice (IN) 2009-23 on October 8, 2009, on “NUCLEAR FUEL THERMAL CONDUCTIVITY DEGRADATION”. IN 2009-23 informed licenses and new reactor design applicants the irradiation assisted fuel pellet thermal conductivity degradation and possible impact on fuel design and fuel related primary loop and containment accident and transient analysis. This IN has been addressed by many new reactor DC applicants. During this audit, staff learned that KHNP has not yet addressed this issue with the planned submittal. Two meetings were held between KHNP management and NRC staff on this issue. It was concluded and agreed that this issue is expected to alter significantly the results of the fuel performance analysis, accidents and transient analysis of reactor primary system, and containment system. In addition, it may result in required codes and methods methodology upgrades, which are subject to staff review and approval. Therefore, this finding is considered significant. The impact on the submittal and schedule is yet to be determined.

Considering the potential large amount of work to update the codes, methods and the analysis results, KHNP proposed to submit the DCD as it is and immediately launch the effort to prepare a technical report documenting all the necessary changes associated with the correction of the thermal-conductivity model. It is expected that it may not only affect the DCD but also topical reports submitted so far. During the meeting with KHNP, staff pointed out that this effort might be extensive and the review effort and schedule may be impacted.

Post-LOCA Long-Term Cooling Analysis

During the audit, staff met with KHNP experts and discussed the long term cooling for a cold leg slot break and the potential loop seal clearing. It is unknown at this point what the elevation difference between the top of active fuel (TAF) and the bottom of the loop seal. According to KHNP, this issue may not have been examined and that the detailed geometry information will be provided later.

GSI-191 Ex-Vessel Downstream Effect

Staff reviewed Section 4.1, "Ex-Vessel Downstream Effect," of Technical Report APR1400-E-A(NR)-13001-P Revision A for any issues that could affect the completeness or accuracy of the information the APR1400 DC application. Staff guidance in RG 1.82 states that debris may be carried downstream of the emergency core cooling system (ECCS) strainer, thus causing downstream blockage or wear and abrasion. The areas of concern identified for downstream ex-vessel components include (1) blockage of system flowpaths at narrow flow passages (e.g., containment spray nozzles, some pump internal flow passages, and tight-clearance valves), and (2) wear and abrasion of surfaces (e.g., pump running surfaces) and heat exchanger tubes and orifices. Staff review indicated that further information is needed in the technical report to describe the characteristics of the post-LOCA debris that will bypass the sump strainers and enter the ECCS. The report should also provide more detail in describing the methodology used to demonstrate component and system operation with post-LOCA fluids for the required mission time. Examples of specific areas where additional information may be needed are listed below:

- Identify all ex-vessel components in the downstream ex-vessel evaluation.
- Specify amount debris that will bypass the sump strainer and enter the ECCS (type, quantity, size, density, concentration in ppm, etc.). Describe basis for the debris bypass.
- Specify bypass debris concentration (ppm) based on volume of recirculation fluid.
- Pumps (including mechanical seals) and valves are to be qualified in accordance with QME-1-2007 as endorsed by RG 1.100 to operate with the post-LOCA fluids for the required mission time of 30 days. ITAAC are needed to verify pump and valve qualification with the post-LOCA fluids for the required mission time.
- Describe the heat exchanger evaluations for plugging, wear, fouling and thermal performance to ensure heat exchangers will operate as designed with the post-LOCA fluids for the required mission time. ITAAC are needed to verify heat exchanger performance with the post-LOCA fluids for the required mission time.
- System performance evaluations should conclude that system flow parameters are acceptable for the 30-day mission time when components and piping reach their maximum wear. ITAAC verification may be needed.
- Provide an evaluation of chemical effects on downstream ex-vessel system operation and components.
- Further describe the evaluation methodology used to determine that debris will not block instrument tubing.
-

3.3. Instrumentation and Controls (I&C)

During the pre-application audit for the APR1400 design certification application, the NRC Instrumentation and Controls (I&C) staff reviewed the APR1400 DCD, Tier 2, Chapter 7 and the APR1400 DCD, Tier 1, Chapter 2.5 along with supporting technical and topical reports.

Staff used the guidance provided in NUREG-0800, the Standard Review Plan, Chapter 7, "Instrumentations and Controls," to support the audit activities. Staff finds that the I&C information presented in these documents needs significant revision prior to submission of the APR1400 DC application in order to reduce the risk the current information would present during an acceptance review. Below is a summary of staff's observations.

Observations

- Staff found that the lack of functional and design information as well as the development process of I&C systems would make it difficult for the staff to make an acceptability finding. Additional information is required. Examples of information not provided, but necessary for a successful acceptance review include:

    - Detailed functional descriptions of the Plant Protection System (PPS) and the Engineered Safety Feature-Component Control System (ESF-CCS), including details on the implementation of reactor trip functions, ESF actuation functions, and ESF control functions, along with detailed functional diagrams from sensor to actuation device.

    - Detailed descriptions of certain functions performed by the safety I&C systems such as priority functions and permissive functions. This includes descriptions of how the priority scheme is implemented in the ESF-CCS and in the component interface module (CIM) to address the guidance of Digital I&C ISG-04, Section 2. A description of how the CIM will be 100 percent tested was also not provided.

    - The APR1400 DCD stated that commercial grade items will be dedicated for use for certain PPS and ESF-CCS components, but it did not list the components that will be commercially grade dedicated and it did not provide the critical characteristics of these components for the dedication process.

    - The software development process documented in the Software Program Manual (SPM) does not provide sufficient details of the development process for both safety I&C systems (PPS & ESF-CCS), including the commercial grade dedication process of safety I&C components, and other systems covered by the SPM. Examples include:

        - Section 1.2 of the SPM states, "…specific project plans are not required for each element of this SPM." However, the elements that do not have project plans are not identified in the SPM.

        - In Section 1.2 of the SPM, it is unclear what criteria will be used to determine what requirements, including criteria for documentation, will be developed for specific projects.

        - In Section 1.2 of the SPM, it appears that for specific projects, plans will not be developed at the beginning of the project. This does not demonstrate that adequate pre-planning has been done for these projects.

        - The scope and applicability of the processes described in the SPM has not been clearly defined. For example, in Section 2.1, it is unclear whether the phrase "digital computer-based I&C software" applies to digital logic systems that utilize a programming language during development, but execute no software during the device's operation.

        - Figure 2.1 in Section 2.2.1 does not demonstrate adequate separation of the Design Team from the Verification and Validation (V&V) Team as they report to the same I&C System Engineering Department Manager. As such, this arrangement does not meet Criterion II of 10 CFR Part 50 Appendix B.

        - Section 2.2.2 does not describe the qualifications of the project manager overseeing the project (e.g., technical and organizational capabilities of the project manager should be specified).

        - Use of modified and re-usable software has not been well defined in the SPM, including requirements for quality of such software.

    - The Common Cause Failure (CCF) Coping Analysis Technical Report did not contain sufficient detail to support the conclusions in this report to meet the diversity and defense-in-depth (D3) requirements.

        - Calculations (e.g. hand calculation of Event 5.3.1.1) or the methodology used in the calculations referenced in the CCF Coping Analysis Technical Report in support of conclusions in this technical report need to be included in the application.

        - For event 5.4.2.6.1, the technical report did not demonstrate that for a steam generator tube rupture (SGTR) event, the steam generator (S/G) control system will prevent the S/G from overfilling (i.e. for the listed SGTR event, the S/G control system can prevent S/G level from overfilling given the amount of reactor coolant postulated to flow into S/G.) This technical report did not demonstrate that all operator actions could take place exactly at the 30-minute time period referenced in the report. The technical report did not provide design characteristics to demonstrate that an operator can manually perform several mitigating actions concurrently (i.e., at exactly 30 minutes after accident event initiation). This report did not define what the final stable nuclear plant status is for this postulated accident concurrent with a PPS CCF.

        - For event 5.4.2.7.1, the technical report did not state where the piping break is located (e.g. which reactor vessel piping is the postulated break). The technical report does not provide the detailed design descriptions to demonstrate that this D3 mitigation strategy is acceptable when the result would leave the top portion of the reactor fuel uncovered for 600 seconds.

- Conformance to NRC regulations and guidance needs to be complete and any exemptions to NRC regulations and deviations to NRC guidance should be documented.

- For example, the staff found specific I&C systems listed in DCD Table 7.1-1 that were not identified to comply with several of the regulations or satisfy several of the guidance documents listed in SRP Table 7-1. Examples include:

    - The PPS was not listed as a system that needed to comply with 10 CFR 50.34(f)(2)(v) for bypass and inoperable status.

    - The PPS was not listed as a system that needed to comply with 10 CFR 50.34(f)(xii) Auxiliary Feedwater System Automatic Initiation and Flow Indication.

    - The PPS was not listed as a system that needed to comply with 10 CFR 50.34(f)(2)(xiv) on Containment isolation systems.

    - The Plant Control System (PCS), qualified indication and alarm system-non-safety related (QIAS-N), and Pressurizer Pressure Control System (PPCS) were not listed as systems that needed to comply with 10 CFR Part 50, Appendix A General Design Criteria (GDC) 1, 2, and 4.

    - The PCS was not listed as a system that needed to comply with the GDC 10, Reactor Design.

    - The PPS was not listed as a system that needed to comply with GDC 33, GDC 34, GDC 35, GDC 38, GDC 41, and GDC 44.

    - The diverse actuation system (DAS) was not listed as a system that needed to satisfy RG 1.22, Periodic Testing of Protection System Actuation Functions.

    - The qualified indication and alarm system - post accident monitoring instrumentation (QIAS-P) was not listed as a system that needed to satisfy RG 1.47, "Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems" and RG 1.53, "Application of the Single-Failure Criterion to Nuclear Power Plant Protection Systems."

    - The PPS and ESC-CCS were not listed as systems that needed to satisfy Branch Technical Position (BTP) 7-6, Guidance on Design of Instrumentation and Controls Provided to Accomplish Changeover from Injection to Recirculation Mode.

  In addition, the information presented did not demonstrate how NRC regulations are met, especially the requirements of IEEE Std. 603-1991, as incorporated by in reference in 10 CFR 50.55a(h). For example, no information was presented on how IEEE Std. 603-1991, Clause 5.2 on completion of protective action is met for reactor trip functions and ESF actuation functions in the APR1400 design.

- The clarity of the information presented in the APR1400 DCD and supporting reports lacked clarity and had inconsistent information. Examples include:

    - Section 7.2 of the APR1400 DCD Tier 2 states "RSPT (reed switch position transmitter) signals of all core quadrants are handled using two CPPs (control element assemblies [CEA] position processors) of each channel, and the specified CEA position signals used in core protection calculator (CPC) of the corresponding channel and the other channels are handled." It is not clear to the staff what the term "handled" means and what components are "handling" these signals.

- Section 7.2.1.2 of the APR1400 DCD Tier 2 states, "When two process measurements are available for mitigating a transient, they are assigned to different input modules." It is not clear to the staff what the term "available" means.

- Section 7.4 of the APR1400 DCD Tier 2 states, "Safe shutdown can be taken to the safe shutdown condition in the main control room (MCR) only either in case of onsite or offsite power available. Outside MCR is also considered to bring the reactor in a safe shutdown from the remote shutdown room." The intent of this sentence is unclear to the staff.

- I&C definitions and I&C symbols descriptions were not provided in Chapter 7 of the APR1400 DCD Tier 2.

- Section 7.8.2.1 – States that system testing for diverse protection system (DPS) will be conducted during a shutdown, however, system testing for diverse indication system (DIS) will be performed during an "overhaul." The term overhaul is not defined as it relates to system testing/surveillance periodicity.

- Table 7.8-1 lists the S/G level trip by the DPS at 22.4 percent; however, when reviewing the Setpoint Methodology technical report, the S/G level trip is not listed in Section 2.1 of this report.

- In Section 3.1.2 of the SPM, references to the functional and system specifications and the software requirements specifications are inconsistent.

• The design descriptions and Inspection, Tests, and Acceptance Criteria (ITAACs) presented in the APR1400 DCD Tier 1 did not contain sufficient detail and were inconsistent for the staff to make an acceptability finding. Examples include:

  - No description of how independence is achieved between Class 1E channels and between Class 1E and non-Class 1E Equipment, only that independence is provided.

  - The development process for safety I&C systems and systems of augmented quality did not include sufficient detail. Certain phases of the development process were missing in the design description and corresponding ITAACs.

  - No response time ITAAC was provided for the PPS and ESF-CCS to verify that the response times of the as-built systems are bounded by the assumptions in the safety analysis.

  - Design commitments and acceptance criteria did not match in the ITAAC. An example is ITAAC Item 15 on Table 2.5.1-5 where the design commitment states, "The PPS provides one way communication to non-safety systems and to

- o other channels of the PPS to ensure communication independence." This statement does not adequately match the acceptance criteria "Existence of documentation that concludes the communication between safety and no-safety systems does not prevent the safety function."

Documents Reviewed (see Enclosures 3 and 6 for the full list):

1. APR1400 Tier 1 DCP Rev. F;
2. APR1400 Tier 2 Chapter 7 Rev. H;
3. APR1400 Tier 2 Chapter 15 Rev. H;
4. Setpoint Methodology Technical Report, APR-1400-Z-J-NR-13005-P, Rev. 0
5. Uncertainty Methodology & Application Technical Report, APR1400-Z-J-NR-13004-P, Rev. 0;
6. Safety I&C Topical Report, APR1400-Z-J-EC-13001-P, Rev. D1;
7. Diversity and Defense in Depth Technical Report, APR1400-Z-J-EC-13002-P, Rev 0;
8. CCF Coping Analysis for the APR1400 Technical Report, APR1400-Z-A-NR-13008, Rev 0; and
9. Software Program Manual, APR1400-Z-J-NR-13003-P, Rev 0.

3.4. Radiation Protection

NRC staff audited information in the pre-application APR1400 DCD regarding design basis accident (DBA) radiological consequence analyses and control room radiological habitability. Staff also reviewed the following areas: radiation protection, shielding, radwaste systems, dose analysis, and man-related hazards. NRC staff performed a preliminary evaluation of the information related to these areas against the SRP criteria.

Staff looked at information in APR1400 DCD Chapter 15 (including Appendix 15A), and Sections 6.4 and 6.5. Staff also looked at the following calculations: 1-035-N374-006, Rev. 0 "Dose Cal. (Accident) Large LOCA Offsite Dose"; 1-035-N374-007, Rev. 0 "Dose Cal. (Accident) LOCA-Control Room Dose"; and 1-035-N374-023, Rev. 1 "Dose Cal. (Accident) CR Habitability Analysis – MSLB." In addition, staff reviewed the portions related to radiation protection of DCD Tier 1 and DCD Tier 2 chapters 1, 3, 4, 5, 9, 10, 11, 12, 15, and 16.

The high-level observations were presented to KHNP at the audit exit meeting. Specific details are provided here.

Observations

The level of detail in the DCD Chapter 15 discussion of DBA radiological consequences analyses could be increased to aid in review, but there is no major lack of information. The DCD indentifies appropriate regulatory criteria, regulatory guidance, and computer codes with respect to the DBA and control room habitability dose analyses. The DBAs evaluated in the DCD are consistent with the guidance in RG 1.183 for large light water reactors.

Assumptions, inputs, and methods used in the DBA radiological consequences analyses are given in the DCD discussion. However, more detail of models incorporated in the analyses, and the justification for applicability to the APR1400 design should be given to aid in review. The following are some examples of this need for more detail:

- The LOCA dose calculation uses the Powers model of aerosol removal by natural processes in containment (NUREG/CR-6189). The DCD is not clear that the Powers model is used. The applicability of the model is not discussed in Chapter 15 or Section 6.5.3, and the reference sections in either Chapter 15 or Section 6.5 do not include a reference to the NUREG documentation discussing the model.

- DCD Chapter 15, Appendix 15A states that the computer code ORIGEN-S was used to calculate the core inventory, and there is a pointer to Reference 1 for the appendix. However, the Appendix 15A reference section stated that Reference 1 was removed. Therefore, it is unclear which version of ORIGEN-S was used to calculate the core inventory. Additionally, although some input related to fuel type and burnup was given, no discussion of the applicability of the ORIGEN-S version and chosen cross-section libraries was given for the fuel type and burnup.

- Discussion in DCD Chapter 15 should clearly identify the single failure assumed for each DBA dose analysis and the basis for the assumption.

- An example of clarifying information that could be added to the DBA dose analysis tables is in relation to DCD Table 15.0-9, which gives analysis assumptions and inputs. In addition to stating that calculation inputs change at "time of shutdown cooling entry" or "time to isolate", the assumed time in seconds, minutes or hours (as appropriate) should also be listed in the table.

- Appropriate regulations and major guidance were identified, but not enough information is provided to demonstrate how the plant design meets requirements (Ex: Section 12.3).

- Some recent industry operating experience or NRC generic communications may not have been incorporated into the design and documentation. For example, there was discussion of the use of gland leakoff controls which is no longer current because valve designs have improved.

- Maps regarding radiation protection and vital area access did not include detail on physical barriers and routes to vital equipment.

- NRC staff found some inconsistency between DCD Section 6.4 dose results and both the DCD Chapter 15 dose analysis results and the related calculations that were audited. DCD Table 6.4-1 indicates that the doses are reported in units of rem. However, if that were the case, some of the DBA control room doses as reported would exceed the GDC-19 criterion of five rem TEDE. Even if the results are meant to be reported in mSv, then the values for the LOCA and MSLB are not the same dose values as reported in the related control room dose calculation files.

- Neither DCD Chapter 4, nor DCD Section 12.2 discussed whether the APR1400 has a High Duty Core, as defined by EPRI Report 1008102 “PWR Axial Offset Anomaly (AOA) Guidelines,” and the potential impact this could have on Occupational Radiation Exposure (ORE). DCD Section 4.4.6.2 states that the design uses Rhodium self-powered neutron detectors, but these irradiated sources are not described in DCD section 12.2. Since these irradiated components are periodically replaced, they should also be described in the solid waste section of DCD Chapter 11.

- DCD Chapter 10 does not describe the 10 CFR 20.1406 radiological impact of main steam radioactivity on the operation of interconnected systems such as the Auxiliary Steam System and the Auxiliary Steam Boiler. Chapter 10 does not contain references to RG 8.8, RG 4.21, or DC/COL/ISG-6, so it is not clear to that the staff that radiation protection design features to minimize ORE from steam generator blowdown systems and condensate polisher resins and filters have been considered within the DCD design.

- RG 1.206, Section C.I.12.3.1, states that the applicant should show the location of airborne and area radiation monitors on the plant layout drawings. However, the monitors appeared to be missing from the layout drawings.

- SRP Section 12.2, states that applicant’s should provide radiation source term information and that, “The source descriptions should include all pertinent information required for input to shielding codes used in the design process.” In addition, 10 CFR 52.47(a)(5) requires the applicant to identify the kinds and quantities of radioactive materials expected to be produced in the operation and the means for controlling and limiting radioactive effluents and radiation exposures within the limits set forth in 10 CFR Part 20. However, the DCD and technical reports included no source dimensions for the sources listed in Chapter 12. In addition, staff identified several major sources of radiation that were not included as a Chapter 12 source, including the Boric Acid Concentrator (Evaporator), the Boric Acid Concentrates tank, and the reverse osmosis system. The descriptive information associated with some sources of radiation, such as irradiated fuel, was vague and scattered throughout the DCD. For example, DCD section 12.2 implied that the source content of the irradiated fuel was consistent with DCD chapter 4, without a clear corresponding discussion within chapter 4, or a clear description of which parameters of DCD Chapter 4 were applicable to the fuel description provided within section 12.2. The staff was not able to correlate main steam radioactivity concentrations to the stated radioactive content of the Condensate Polisher demineralizer beds.

- Section 12.3 of the DCD appeared to echo the regulatory guidance rather than providing details about the features provided to reduce ORE. For instance, neither section 12.3 nor the appropriate sections of chapter 5 or chapter 9 discussed the allowable cobalt content of components in contact with reactor coolant system fluids. Cobalt is a major contributor to ORE. Provisions for handling potentially high dose rate components, such as irradiated self-powered neutron detectors, are not described within DCD section 12.3. Design features to prevent personnel access to Very High Radiation Areas (VHRA), such as in the fuel transfer tube, are not discussed in DCD section 12.3. DCD Section 12.3 and the relevant chapters of the DCD do not provide a description of the specific design features provided to minimize contamination of the facility and the environment.

For instance, the location of buried piping containing potentially radioactive fluid is not identified. Neither DCD Section 12.3, nor DCD section 9.1, describe which radiation monitors are provided to satisfy the requirements of 10 CFR 50.68 or 10 CFR 70.24, and which section of the regulations is to be used for these required radiation monitors. DCD Section 12.3 contains erroneous information, such as the use of drip pans to control airborne contamination from noble gases and radio-iodides. DCD Section 12.3 does not contain a description of the design features of the ventilation system provided to minimize ORE. Current industry documents, such as EPRI TR- 1000923 ," Valve Packing Performance Improvement," discourage the use of lantern rings, except where necessary, and when they are used, recommends the use of graphite lantern rings. Emphasis is also placed on the finish of the valve stem to reduce wear and leakage. Similar current guidance exist for the use of check valves, pump seals, internal finishes and polishing, is also not reflected in DCD Section 12.3. This is indicates that current industry operating experience has not been used to update the proposed DCD submittal.

- The DCD utilized the computer codes Microshield and RUNT-G to perform their shielding calculations. The RUNT-G code is not a code recognized by the NRC. The DCD should provide information explaining why this code can be used as an acceptable alternative to codes recognized by the NRC, where this code was used instead of the other stated code, and why.

- It is not clear that DCD section 12.5 contains a full description of the ALARA program and the Ground Water Protection program, in accordance with the guidance contained in SECY 04-0032, "Programmatic Information Needed for Approval of a Combined License Without Inspections, Tests, Analyses and Acceptance Criteria."

- SRP Section 14.3.8 states that, "The reviewer should ensure that Tier 1 identifies and describes, commensurate with their safety significance, those SSCs that provide radiation shielding, confinement or containment of radioactivity, ventilation of airborne contamination, or radiation (or radioactivity concentration) monitoring for normal operations and during accidents." While the Compound Building contained several very high radiation areas and significant high radiation areas, the DCD contained no ITAAC for walls in the Compound Building.

- RG 1.206, Section C.12.1.2, states, "In accordance with the requirements in 10 CFR 20.1406 describe the design approaches implemented to minimize, to the extent practicable, contamination of the facility and the environment, facilitate eventual decommissioning, and minimize, to the extent practicable, the generation of radioactive waste." In reviewing DCD Section 12.3-12.4, staff noted that the applicant provided very little specific design information related to demonstrating compliance with 10 CFR 20.1406. More specific design information should be included in the DCD.

- Technical Specification (TS), Section 3.4.14 (B) refers to the use of the Containment Gaseous Radiation Monitor. DCD Section 5.2.6.1.1.3 states that the gaseous radiation monitor is sensitive enough to detect a 1-gpm leak in one hour. When the vendor representatives were asked what source term was used as the basis for calculating the leakage detection sensitivity, they stated 0.125% fuel cladding defects. This is not consistent with the guidance contained within RG 1.45 "Guidance on Monitoring and Responding to Reactor Coolant System Leakage." RG 1.45 states, "Analysis of the capabilities of leakage monitoring systems that measure radioactivity should use a realistic primary coolant radioactivity concentration assumption consistent with plant normal operations (as opposed to the maximum concentration permitted by technical specifications or used in accident analysis)." Information Notice 2005-24: "Nonconservatism in Leakage Detection Systems," discusses industry experience associated with use of non-conservative isotopic mixture assumptions (i.e. 0.1% failed fuel instead of a "realistic" primary radioactivity concentration). The use of a failed fuel source term indicates that current industry operating experience and generic communications, such as to calculate radiation monitor sensitivity for leakage detection, have not been used to update the proposed DCD submittal.

- TS Section 5.5.2 "Primary Coolant Sources Outside of Containment" does not include all of the systems described within the basis NUREG. TS Section 5.5.8, In Service Testing, does not mention ESF leakage testing. TS Section 5.5.11, "Ventilation Filter Testing," does not mention the TSC ventilation system. TS section 5.5.9(b)(2) discusses allowable leakage from a single steam generator, but does not provide an allowable total leakage rate. TS Bases 5.5.9 and 5.5.10 do not appear to explicitly mention NEI 97-06 "Steam Generator Program Guidelines." NEI 97-06 [Rev 2] section 2.3, "OPERATIONAL LEAKAGE PERFORMANCE CRITERION" states that the "PWR Primary-to-Secondary Leak Guidelines" provide reasonable assurance that the operational leakage performance criterion will be met. It further states that licensees shall establish primary-to-secondary leak monitoring procedures in accordance with the EPRI PWR Primary-to-Secondary Leak Guidelines. EPRI PWR "Primary-to-Secondary Leak Guidelines" Revision 3 states that that station radiation monitoring equipment shall be such that a 30 gallons per day (gpd) leak total can be detected. Insufficient information has been provided by the applicant for the staff to be able to determine if the specified radiation monitor sensitivity value is consistent with the specified SG tube leakage rate threshold of 30 gpd. It is not clear to the staff which radiation monitors are credited by KHNP for this function. DCD Section 11.5 should contain the methods, models, assumptions, and related parameters to determine that the minimum required radiation detection sensitivity values for the radiation monitors designed to monitor the Primary-to-Secondary Leakage Rate satisfy the required SG Tube Leakage Rate detection capability.

- DCD Section 1.2.10.2 states that the fuel pool is designed for 46 percent of the bundles from the core during refueling. This is supported by the description of the SFP Cleanup system provided in section 1.2.11.5. This is inconsistent with current industry practice to do a full core off load during refueling outages.

- Table 1.9-1 (Sheet 22 of 29) states that RG 4.21 "Minimization of Contamination and Radioactive Waste Generation: Life-Cycle Planning," is only applicable to COL applicants, however, the design aspects of RG 4.21 are applicable to both the DCD, and to the site specific portions of the plant described in the COL FSAR).

- RG 4.22 "Decommissioning Planning During Operations," is applicable to the COL applicant. Table 1.9-1 (Sheet 28 of 29) states that RG 8.25 "Air Sampling in the Workplace," is only applicable to the COL, however, RG 8.25 discusses aspects of the ventilation systems that are design specific. There were other inappropriate or inaccurate characterizations of the applicability of regulatory guidance.

- DCD Table 3.2-1 states that the Steam Generator Blowdown System (SGBDS) is classified in accordance with RG 1.143, Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants." However, it does not state to what level and references only Position C.1.1 of RG 1.143.

- DCD Table 1.9-2 Sheet 3 of 33, only includes a reference to 3.2.2, but does not include a reference to DCD Chapter 11, for the radioactive waste systems. In accordance with RG 1.143, Revision 2, which the KHNP DCD references, all radwaste and steam generator blowdown systems and components and structures housing such systems and components should be classified and designed in accordance with the RG. However, the staff could find no classification for the Compound Building, which houses the radwaste processing systems and components. In addition, the staff discovered that only certain components in the solid, liquid, and gaseous waste management systems identified in Section 12.2 were classified in Chapter 11. Finally, the staff could not find any classification for the steam generator blowdown system.

3.5. Severe Accident Analysis and Probabilistic Risk Assessment (PRA)

The staff in the Probabilistic Risk Assessment (PRA) and Severe Accidents Branch in the Division of Safety Systems and Risk Assessment of the Office of New Reactors audited the APR1400 DCD Chapter 19, "Probabilistic Risk Assessment & Severe Accident Evaluation," and DCD Section 17.4 on the reliability assurance program (RAP). The intent of this regulatory audit was to review and evaluate the portions of the DCD containing the high-level descriptions of PRA and RAP to ensure that it contains the information required by regulations and that the information is complete and technically adequate. This included reviewing applicable interface documentation (e.g., detailed PRA notebooks, severe accident evaluation report). In addition, the staff collected risk insights to facilitate the NRC detailed technical review of the design certification application (e.g., use of risk insights to facilitate the allocation of review resources in a manner consistent with the risk significance of the various design features). The staff held numerous meetings with KHNP PRA staff to discuss its review of these documents and provide detailed audit observations associated with this subject area. The staff also participated in daily exit meetings with KHNP to summarize these audit observations.

The following are the staff's observations regarding its initial assessment of the completeness and technically adequacy of APR1400 DCD Chapter 19 and Section 17.4. With the exception of the first item, these items can be resolved through the NRC review process (e.g., requests for additional information, RAIs) without a significant impact on the review schedule or resources.

- DCD Chapter 19 does not contain the Level 2 PRA results for internal fire and internal flooding at-power in accordance with 10 CFR 52.47(a)(27). These results are necessary in order to confirm that the risk associated with the design compares against the Commission's goal of less than 1E-6/year for large release frequency. This information must be provided in DCD Chapter 19 in order to commence a detailed technical review of this chapter.

- Page 19.1-162 in DCD Chapter 19 states: "Multiple Compartment Analysis (MCA) considers the potential for fire spread from one compartment to an adjacent compartment via a failed fire barrier. There were 1,312 potential MCA scenarios identified….[thought needs to be completed].” This part of the DCD should be completed, because MCA analysis is an important element of the internal fire PRA analysis.

- For the low power and shutdown (LPSD) analysis in DCD Chapter 19, a discussion should be provided on how the design meets the recommended actions in Generic Letter 88-17 on losses of decay heat removal during Pressurized Water Reactor reduced inventory conditions.

- For the LPSD analysis in DCD Chapter 19, additional discussion should be provided to substantiate not performing a quantified shutdown fire and flooding PRA analysis.

- For the internal flooding analysis in DCD Chapter 19, it is assumed that only the fire protection system has an infinite volume of water and is susceptible to double-guillotine break. This conservative assumption should be clarified since this scenario accounts for 90% of the core damage frequency due to internal flooding.

- DCD Chapter 19 should specify the COL action or information items related to PRA and severe accident evaluation, which are to be addressed in a combined license (COL) application. Regulatory Guide 1.206, Section C.III.4, “Combined License Action or Information Items,” provides a definition of COL action or information items and a discussion of how COL applicants should address these items. A definition of COL action item is also provided in NRC Standard Review Plan Chapter 1.0, “Introduction and Interfaces.”

- DCD Chapter 19 identifies the risk insights, but does not disposition each insight. Each risk insight should receive a disposition such as a reference to another portion of the DCD, an ITAAC, or a COL action or information item to ensure that the risk insights remain valid in the as-to-be-built, as-to-be-operated plant.

- The description of the RAP in the APR1400 DCD Section 17.4 and Section 17.4 of the US-APWR DCD, Revision 3, are very similar. Given these similarities, KHNP should consider reviewing for applicability, the question and response in NRC RAI 6268 for US-APWR DCD Section 17.4 provided by Accession Number ML12129A126 (publically available) in the Agencywide Documents Access and Management System [ADAMS].

3.6. Human Factors Engineering (HFE)

Introduction:

The Operator Licensing and Human Factors Engineering Branch (COLP) participated in a pre-licensing audit of the KHNP APR1400 Design Certification Document (DCD) and other materials related to the HFE portion of the proposed submittal. During the audit, the staff provided examples illustrating the team’s findings. The team did not attempt to identify, or share, all discrepancies identified, but instead focused on those that pose a potentially significant risk to the acceptability of the proposed APR1400 design certification application.

The staff reviewed a sample of the KHNP HFE program. The staff reviewed five of the twelve elements of NUREG-0711, focusing on the following elements that are typically the most challenging and complex: Verification and Validation, Task Analysis, and Functional

Requirements Analysis/Function Allocation. The team also reviewed the elements of Operating Experience Review, HFE Program Management, DCD Tier 1 & 2, and the Human-System Interface (HIS) Basic Platform document.

Observations:
The HFE audit team made the following observations:

- There is a substantial amount of technical information missing. Examples include:

    - The process used for determining the sample of operational conditions to be tested during the verification and validation phase of the HFE program.

    - The methodology for conducting Task Analysis was not completely described.

    - Scenarios to test operator performance and the adequacy of human system interfaces were not provided as part of the Integrated System Validation (ISV).

    - Performance measures used during ISV testing to measure plant and personnel performance were not provided.

    - An explanation for the basis for the Basic HSI Design platform was not provided.

    - NUREG-0711 Task Analysis acceptance criterion 6 (related to number of personnel needed to do the task) and criterion 7 (related to knowledge and abilities required to perform the task) were not addressed.

    - NUREG-0711 OER acceptance criterion 4 (related to using interviews to identify operating experience) and criterion 5 for (related to identifying important human actions from predecessor designs) were not addressed.

    - HFE process elements mentioned in the DCD are not included in the corresponding Verification and Validation Implementation Plan (e.g., sampling of operational conditions, operational sequence diagrams).

- Numerous grammatical and sentence structure issues contribute to making the meaning of the material unclear.

- Where acceptance criteria are addressed, the level of detail (specificity, measurability) does not support use of the implementation plan as the source of acceptance criteria for the accompanying "DAC ITAAC." Examples include:

    - The task analysis IP defines Hierarchal Task Analysis and Task Decomposition but does not describe how the methods will be implemented (i.e. who is responsible, how these will be integrated).

    - The description for the APR1400 HFE design team placement and authority lacks sufficient detail to determine where the team fits into the overall engineering organization, its reporting relationship, and authority.

- o Although the FRA/FA mentions “success paths” for accomplishing functions, the concept of success paths is merely mentioned but not explained or illustrated.

- o The current ITAAC format is not sufficient to determine element completion. The acceptance criteria do not reflect that a Results Summary Report must be provided to document the final design product and evaluation results must be presented that demonstrate the HFE design process was implemented as described in the DCD.

The observations are considered generic because they occurred repeatedly in the HFE program elements sampled by the staff and the associated documentation. Previous staff experience with issues similar to those identified by the audit has caused numerous and unnecessary schedule delays because multiple rounds of requests for information (RAIs) were needed for the staff to arrive at a safety finding. It is expected that KHNP will provide a complete and technically sufficient HFE program at DC submittal, with a level of detail that will allow the staff to apply the regulatory review guidance in NUREG-0711. KHNP needs to provide more information in the DCD and other HFE documentation in order to meet this expectation at the time of their DC application.

3.7. Environmental Qualification (section 3.11)

Section 3.11 of the DCD was reviewed with respect to the radiological qualification requirements. These are the observations identified during that review. Per 10 CFR 50.49(d) the applicant is required to describe the environmental conditions, including temperature, pressure, humidity, radiation, chemicals, and submergence at the location where the equipment must perform as specified in accordance with paragraphs (d)(1) and (2) of 10 CFR 50.49.

<u>Observations</u>

- DCD Chapter 1 contains incorrect or inconsistent information. For instance, Table 1.9-1 (sheet 4 of 29) states that RG 1.40 “Qualification of Continuous Duty Safety-Related Motors for Nuclear Power Plants,” is not applicable because there are no safety-related motors inside of containment, but it does not address the applicable regulatory guidance for safety-related motors located outside of containment.

- DCD Section 3.11 regarding Environmental Qualification of equipment does not contain entries that separately describe the radiation environment, including beta, gamma, and neutron exposure. The section does not appear to adequately describe the environmental conditions for equipment, including which equipment is subjection to emersion, degradation due to loss of ventilation and which equipment is in a harsh environment, strictly due to radiation exposure. This section does not provide a description of the methods, models, and assumptions used to calculate the Total Integrated Dose to equipment. This section does not appear to provide a clear description of the required operational duration of each piece of equipment. The lack of this information makes it infeasible for the staff to assess the adequacy of the post accident vital area mission doses required to be available in DCD section 12.4.

3.8. Additional DCD Chapter 3 Review

NRC staff performed a high-level review of DCD Tier 2, Chapter 3 sections and related technical reports during the APR1400 audit and has the following observations.

DCD Tier 2 Sections 3.9.3, 3.9.4 and 3.9.5
Design information lacks load combinations, sufficient design methodologies on major components (e.g., reactor vessel and internals, pressurizer, steam generators, safety relief valves, ECCS debris screens, component supports, and core support), and preparation of design specifications and design reports for ASME components. The GSI-191 technical report lacks the design methodology and structural analysis of the debris screens.

DCD Tier 2 Section 3.12:
KHNP states that their intent is to use design acceptance criteria (DAC) for the APR1400 piping design. KHNP provided examples of detailed piping design and stress reports for two piping subsystems. Feedback on the design and reports was provided during the audit.

4.0 CONCLUSION

NRC staff successfully completed its audit of the KHNP pre-application DCD and provided significant feedback on its completeness and technical sufficiency. NRC staff recommends that KHNP address the gaps identified in this report prior to submitting the APR1400 DCD application for NRC review.