1) The information in FSAR Section 11.2, "Liquid Waste Management System," and 11.3, "Gaseous Waste Management System" regarding 1) the basis for all input design values and assumptions and 2) the input/output files used in the PWR-GALE code, the LADTAP II code, and the GASPAR Code for calculation of effluent releases in DCD TIER 2, Sections 11.2 and 11.3 is incomplete.

   (Response) All input parameters for PWR-GALE, LADTAP II and GASPAR code calculations are provided in Table 11.2-2, 11.2-4 and 11.3-5. Details of the basis, assumptions and I/O files are documented in the calculation notes 1-035-N373-001, 1-035-N373-003 and 1-035-N373-004 for NRC review and audit. Input and output files are available for independent evaluation by NRC when requested.

2) Insufficient information is available for the staff to confirm calculated liquid and gaseous release values and doses pertaining to compliance with 10 CFR Part 50, Appendix I and 10 CFR Part 20.

   (Response) Descriptions and information used to calculate doses and concentrations are provided in Sections 11.2.3.1 and 11.3.3.1 together with associated tables. See response to Item No. (1) above.

3) RG 1.143:

   a) Information in FSAR Sections 11.2, 11.3, and 11.4 regarding the methodology, calculations and classifications of radioactive waste structures, systems and components (SSCs) is insufficient to confirm SSC classifications in accordance with RG 1.143, Revision 2, committed to in FSAR Section 1.9.

      (Response) Methods to determine SSC classifications are provided in Section 11.2.1.4. This section includes information on how to calculate the design basis (1% fuel defect) radioactive source terms in LWMS. The last two paragraphs describe how to determine the classification using the source terms. The design basis 1% fuel defect source terms for the LWMS components are not provided in Chapter 11 in accordance with RG 1.206, which requires including only the expected radioactive inventories in Chapter 11. Calculation notes 1-035-N377-017 through -021 document the details of the information used to determine the SSC classification. This calculation notes are available for NRC review when requested.

   b) In addition, Section 10.4.8.1.2 states that, "The structure, system, and components (SSCs) of the SGBS classified as RW-IIc are designed in compliance with applicable codes and standards, and guidelines provided in NRC RG 1.143." However, RG 1.143, Revision 2, Regulatory Position 5, only provides two classifications for structures (RW-IIa and RW-IIb). Therefore, the classification for the structure (or those portions of the structure housing the SGBS) must be revised. If this classification is less than RW-IIa, appropriate justification must be provided.

(Response) Description of the last paragraph of page 10.4-56 in Section 10.4.8.1.2 will be revised as "...The system and components of the SGBS classified as RW-IIc are ...". And the structure housing the SGBS system and other components in the Auxiliary Building, is classified as RW-IIa. The Auxiliary Building is designed to SC I, and satisfies the intent of RG1.143.

4) FSAR Sections 11.2, 11.3, 11.4, and 11.5 utilize SSC source terms located in Chapter 12 for various design requirements, however source terms are not referenced in Chapter 11. Source terms used in Chapter 11 support SSC classifications in accordance with RG 1.143. Additional information is needed regarding epoxy wall coating criteria and radiation monitor alarms and ranges. This information is necessary to verify and validate compliance calculations.

(Response) As presented in DCD Tier 2, Section 11.1, two source terms are defined and utilized in Chapter 11: (1) Design basis, which is based on 1% fuel defect, and (2) Expected, based on ANSI/ANS-18.1. Another design basis source term used for shielding and HVAC design, as defined in Chapter 12, is based on 0.25% fuel defect in accordance with the guidance in Section C.2 of RG 8.8. As discussed in response to Item No. (3)(a), the SSC classifications for radwaste systems are based on 1% fuel defect design basis source terms. The 1% fuel defect design basis source terms for radwaste systems are not provided in Chapter 11 because descriptions on these design basis source terms are not required by RG 1.206, C.I.11.2.1. It only mandates to include "Expected" concentrations. However, KHNP will provide these 1% fuel defect source terms in DCD Tier 2 Chapter 11 for NRC's confirmation calculations, if requested.

KHNP's understanding of the issue regarding epoxy coating criteria is that in which conditions the epoxy coating is required for APR1400. Criteria for epoxy coating are specified in the 'APR1400 ALARA Design Guide' which is available for NRC review/audit. Followings are the criteria specified in the ALARA design guide:

- Check that all floors in potentially contaminated areas are coated to facilitate decontamination.
- Check that walls in potentially contaminated areas shall be coated to heights of 150 cm or 240 cm.
- Check that pipe tunnels which are readily accessible and contain radioactive pipes shall be fully coated.
- Check that cubicles containing radioactive processing equipment which are part of high or moderate energy systems shall be coated to full height.

Information on detection ranges and functions of the radiation monitors are provided in Tables 11.5-1 and 12.3-6. However, the setpoints of the process and effluent monitors are determined by the COL applicant since site-specific information are required to determine the setpoints. KHNP will add the COL items related to site-specific RMS setpoint determination in Sections 11.5 and 12.3.

5) ITAAC and Inconsistencies:

   a) There are inconsistencies between Tier 1 and Tier 2 concerning the functions and identification numbers of some radiation monitors. For example, Tier 2, Table 11.5-1 lists the MCR air intake monitors as particulate, iodine, and gaseous monitors. While Tier 1, Table 2.7.6.4-1 lists them only as gaseous monitors. In addition, Tier 1 Table 2.7.6.5-1 indicates that monitor RE-236 is a safety-related monitor. However, RE-236 is not even identified as a monitor in Chapters 11 or 12. Various other apparent inconsistencies between Tier 1 and Tier 2 have also been identified. In addition, there are also apparent inconsistencies within various Tier 2 sections. For example, for some of the ventilation systems in Chapter 9, some radiation monitors appear to be designated with the prefix "RX" while others with the prefix "RE" ("RE" is used in Chapter 11 and 12). With apparent different designations being used, it is difficult or impossible for the staff to accurately evaluate the design. Therefore, the applicant is requested to review all applicable sections of Tier 1 and Tier 2 related to radiation monitors to ensure that 1) tag numbers are consistent between all Tier 1 and Tier 2 Sections. 2) The type, function, and description of all radiation monitors are consistent between all applicable Tier 1 and Tier 2 sections.

      (Response) The MCR air intake (inline) monitors are designed to monitor only the gaseous radioactivity concentrations. Therefore, the column of range of Particulate and Iodine in DCD Tier 2, Table 11.5-1 shall be modified to be 'N/A'. KHNP will modify Table 11.5-1.

      RE-236 is a non-safety related monitor. Safety class denoted as 'S' in DCD Tier 1, Table 2.7.6.5-1 is a typing error. Therefore, it shall be corrected as 'N' from '3'. KHNP will modify Table 2.7.6.5-1 in DCD Tier 1. Description on RE-236 is provided in the Section 12.3.4.1.5.b and Table 12.3-6.

      'RX' and 'RE' stand for sample probe and radiation element, respectively. RX is installed on the HVAC duct which is a part of HVAC system. Representative sample is extracted from HVAC system using RX and transported to radiation monitoring system which includes radiation element (RE), sample pump, sampling tubing and so on. These are shown in Figure 11.5-1 of DCD Tier 2. Radiation element detects the radiation level and transmits the electrical signal to display and processing unit. Chapter 9 describes the HVAC systems, whereas Chapter 11 deals with the radiation monitoring system. Therefore, radiation monitoring component is expressed as 'RX' in Chapter 9 and 'RE' in Chapter 11.

      All chapters relevant to radiation monitoring system will be cross-checked and appropriate correction will be made to be consistent among chapters.

   b) Additional ITAAC should be provided in Tier 1 for the Compound Building, the liquid, gaseous, and solid radioactive waste systems, and the steam generator blowdown systems to confirm that they are designed in accordance with RG 1.143.

(Response) Previous RAIs between NRC and DC applicants were reviewed and it is found that US-APWR DCD Tier 1(Rev.3) was revised to delete ASME B 31.3 requirements in the ITTAC table incorporating the application of RG 1.143. Regarding the incorporation RG 1.143 into ITAAC of radwaste systems including SGBDS, KHNP would like to receive clarification from NRC staff on what level or method need to be incorporated into the ITAAC.

c) The ITAAC provided in Tier 1, Section 2.8 are incomplete or ambiguous. For example, for Table 2.8-2, ITAAC 1, the ITAAC should reference Figures that provide the actual zoning of individual rooms or the actual shield thicknesses should be provided and/or referenced in the ITAAC.

(Response) The ITAAC Section 2.8 will be revised incorporating NRC comments. References will be provided in the ITAAC table to refer to radiation zoning and shield thicknesses of individual rooms.

6) The range of some of the monitors provided appear to be inadequate to detect the radiation levels expected in the area they are provided. For example, the instrument calibration facility area radiation monitor only has a range up to 100 mSv/hr. However, the instrument calibration facility is designated as containing dose rates over 5000 mSv/hr on FSAR Figure 12.3-10. Please ensure that the radiation monitors have a range consistent with the area they are located and the functions that they are to perform.

(Response) Radiation zone for calibration room is determined assuming that the calibrator is working, in which radiation level reaches up to Zone 8. However, normally it is classified as Zone 2 since it is not working. The area radiation monitor in that area (RE-286) is installed to monitor abrupt rise of the local dose rate to alert operators. The warning setpoint of the monitor is typically set to the upper limit of Zone 2 (0.025 mSv/hr) and the alarm is set to 5 times the value, which is 0.125 mSv/hr. This is why the range of the RE-286 is limited to 100 mSv/hr. However, incorporating NRC's comment, KHNP will change the range of this monitor to cover the dose rate greater than 5,000 mSv/hr.

7) Dose Assessment:

a) Some of the information regarding reducing occupational radiation exposure (ORE) is generic instead of plant specific details. More plant specific information is necessary. For example, FSAR Section 12.4.1.2.1 states, "Improvement of the plant layout, access provisions, and operational procedures can reduce exposure time in RCAs for surveillance, inspection, and testing works, thereby minimizing occupational radiation exposure." Instead of providing this general statement, the applicant should indicate how the APR1400 is designed to reduce occupational exposure; as well as provide any examples they may have of how access provisions and operational procedures will reduce exposure.

(Response) Specific design features of APR1400 to minimize ORE are described in Subsection 12.3.1 “Facility Design Features” of DCD TIER 2. Related paragraph in Section 12.4.1.2.1 will be revised to refer Section 12.3.1 for the specific design improvement to minimize ORE.

b) While the applicant indicates that they comply with RG 8.19, they do not provide all of the information suggested in RG 8.19, such as providing the number of workers estimated to be working on a given task or the frequency of each work activity.

(Response) The “Working Time (man-hrs)” in Tables 12.4-1 through 12.4-6 means the combination of “Exposure time per event (hr)”, “Number of workers” and “Number of events per year” defined in RG 8.19. Because the number of events per year for some activities such as routine maintenance does not necessarily have to be identified with respect to collective dose estimation, the resulting working time with the unit of “man-hours” were used. However, these tables will be revised to include the “Number of workers” and “Exposure Time”, if requested.

8) Source Terms, Shielding, zoning, and access controls:

a) FSAR Table 12.3-4 provides minimum shield wall thicknesses for various rooms throughout the plant. However, this table is incomplete as it does not provide shield wall thicknesses for many significant radiation sources. Two of the more significant sources noted include the volume control tank room and spent resin long term storage area which are listed as very high radiation areas during normal operation in Figures 12.3-5 and 12.3-10, yet no information is provided regarding the shielding for these areas. Standard Review Plan (SRP) Section 12.3-12.4 indicates that shielding thicknesses should be provided for all shielded spaces in the plant. Therefore, please update the FSAR to specify the shielding thicknesses for all walls, floors, and ceilings credited in reducing radiation exposure throughout the plant.

(Response) Per response to this NRC’s comments, Table 12.3-4 will be revised to provide all the shield thicknesses.

b) FSAR Section 12.3.2.3 provides access control information for certain significant high radiation areas (areas greater than 1 Gy/hr (100 Rad/hour)). However, specific access control information is not provided for other areas identified as significant high radiation areas or very high radiation areas. Therefore, please update FSAR Section 12.3.2.3 to provide specific access control information for all significant high radiation areas, especially, those areas that are considered very high radiation areas in accordance with 10 CFR 20.1602.

(Response) Very high radiation areas greater than 1 Gy/hr are identified in Table 12.3-5. Information on control of access to high radiation areas are presented in DCD TIER 2 Section 12.3.1.1.h. As addressed in that section, high radiation areas greater than 1 mSv/hr are provided with locked doors to prevent inadvertent access. And these areas are marked with signage to alert personnel as part of the operational radiation protection program.

c) VHRA boundaries are not identified on the Chapter 12 figures. This information is necessary to ensure compliance with 10 CFR 20.1602.

(Response) Locations of the VHRA are identified in Table 12.3-5 instead of figures in Chapter 12. The radiation zone drawings in Figures 12.3-1 through 12.3-16 will be modified to identify the VRHA in the same drawings.

d) FSAR Table 12.3-5 provides a listing of all areas potentially greater than 1 Gy/hr during normal operating conditions and anticipated operational occurrences. However, there are areas listed as greater than 5000 mSv/hr (approximately 5 Gy/hr) during normal operation that are not listed in Table 12.3-5. Please ensure that both Table 12.3-5 and the chapter 12 figures are accurate and consistent.

(Response) Missing areas such as volume control tank and spent resin storage area that are designated as Zone 8 but are not listed as VRHA will be added in Table 12.3-5.

e) Section 12.3-12.4 of the SRP indicates that the staff will evaluate shielding and dose rates during fuel transfer. However, there is no source term information provided for the maximum activity fuel assembly to be transferred to the spent fuel pool. This information is used by the staff to evaluate shielding during fuel transfer.

(Response) Shielding calculation for fuel transfer tube is based on the spent fuel source term provided in Table 12.2-9. Gamma source strengths for the two (2) 100 hour decayed fuel assemblies are multiplied by the radiation peaking factor of 1.55.

9) FSAR Table 1.9-2 indicates that instead of using RG 1.183 for determining the source terms for equipment qualification, the applicant is using RG 1.4. This is inconsistent with the SRP and no information is provided in the FSAR justifying this alternative approach. Justification of this alternative approach is necessary for the staff to fully conduct its technical review. Since RG 1.4 only considers the design basis LOCA, it is unclear if the applicant considered all design basis accidents for equipment qualification. In addition, source terms may be non-conservative using RG 1.4.

(Response) APR1400 applies RG 1.183 and RG 1.89 for determining the source terms for equipment qualification. As indicated in Table 1.9-1, RG 1.4 is no longer used for either radiological consequence analysis or equipment qualification. Table 1.9-2 (Item 3.11) will be revised as follows: "… NRC RGs 1.183 and 1.89 are applied for radiological consequence analysis and equipment qualification".

10) SRP Section 12.2 states that the applicant should provide shielding source terms for post-accident shielding for vital area access. While Table 12.2-24 provides initial core release

percentages for different nuclide groups, no post-accident source terms are provided. Source terms for post-accident sources that require shielding to limit worker dose when accessing vital areas should be provided. For example, the filters of the main control room ventilation system located above the main control room will likely be a significant radiation source, requiring shielding to limit the dose to control room personnel to less than 5 rem in accordance with GDC 19. If this is the case, the post-accident source term for these filters should be provided (as well as the thickness of the floor below these filters, which shields the main control room).

(Response) For the post-accident shielding design for vital area, detailed analyses are performed to determine source terms in the safety-related components such as safety injection and containment spray systems. Post-accident source terms for these components are determined using the nuclide-specific release fractions in Table 12.2-24 and the core inventory data given in Table 15A-1. In addition, the buildup activities in the MCR filters are also determined using the filter efficiencies and intake flow rates based on the amount of radionuclides released from the containment leakage, ESF leakage and leakage prior to containment purge isolation. These source terms are voluminous and cumbersome to be included in the DCD. However, KHNP plans to include the post-accident source terms in the DCD. KHNP would like to clarify in what level of depth the post-accident source terms need to be provided.

11) For 10 CFR Part 52 new reactor applications (DC, ESP, COL), NEI 05-01A (ML060530203) is not the principal staff guidance document for severe accidents and SAMDAs, rather the staff guidance is contained in Sections 7.2 (ML071690007) and 7.3 (ML071780656) of NUREG-1555, Standard Review Plan for Environmental Reviews for Nuclear Power Plants, Revision 1 - July 2007 (ESRP). For the KHNP APR1400 design certification acceptance review, the accident appendix of the environmental acceptance review checklist (ML072250354, publicly available) was applied. The following significant gaps are related to severe accidents and severe accident mitigation alternatives (SAMDAs) as provided in ESRP Sections 7.2 and 7.3 which should be associated with the KHNP APR1400 Environmental Report (APR1400 ER) for design certifications:

a) ER must rely upon technical information as provided from the Level 2 PRA. Technical deficiencies identified in Chapter 19, “Probabilistic Risk Assessment and Severe Accident Evaluation” directly affect the quality and technical sufficiency of the ER.

   (Response) The Level 2 PRA of APR1400 DCD is described in the subsection 19.1.4.2. The technical adequacy of APR1400 PRA is described in subsection 19.1.2.3 of DCD Tier 2. The technical quality of APR1400 PRA is consistent with the guidance provided in the ASME/ANS PRA Standard (ASME/ANS RA-Sa–2009) and the NRC RG 1.200, “An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities.”

b) Appropriate detailed technical documentation of the offsite consequence analysis (i.e., a Level 3 PRA) was not provided and results in the negative responses for the Section 7.2

segment of the environmental sufficiency checklist.

(Response) Technical documentation of the offsite consequence analysis is described in the Appendices of "Level I & II Sensitivity Analysis for SAMDA" (1-037-N419-301, proprietary), which contains benefit-cost analysis as well as the Level 3 PRA results for the ER. The analysis report is not included in DCD package, but the report is available at the KHNP Washington D.C. Center for NRC review.

c) The list of SAMDAs is solely based on the generic list in support of license renewal contained in NEI 05-01A, Severe Accident Mitigation Alternatives (SAMA) Analysis (ML060530203). Therefore, The APR1400 ER and Section 19.2.6 lack a design-specific SAMDA identification process (see Section 5 of NEI 05-01A, Standard Review Plan Section 19.2.6 of NUREG-0800, and ESRP Section 7.3).

(Response) The design-specific SAMDA list for the APR1400 is combined with the generic SAMDA list in Table 4 of the ER (e.g., SAMA ID #1: provides an additional battery capacity, SAMA ID #9: provides additional diesel generators). If necessary, KHNP would like to clarify this issue during the conference call and provide appropriate responses as necessary.

d) Appropriate justification and documentation were not provided for a number of screened out SAMDAs (e.g., SAMDAs screened out due to excessive implementation cost and low benefit).

(Response) The brief justifications for screened out SAMDAs are provided in Table 4 of the ER. Additional justification and documentation can be provided upon NRC request.

e) In a number of places, there are statements that a function or requirement is satisfied, but no information about how. For example:

i) Page 3: "The APR1400 PRA model also quantified internal fire, internal flooding, and low-power & shutdown events" but no reference to a technical report is provided.

(Response) The reference to the technical report will be described in the Environmental Report as below:

3.0 BASE RISK

~

The APR1400 PRA model (Reference 1) also quantified internal fire, internal flooding, and low-power & shutdown events. Risk from ~

12.0 REFERENCES

4. The APR1400 PRA Summary Report, APR1400-E-P-NR-13001-P, submitted on Nov. 30, 2013.

ii) Page 3: No justification in the ER or through a reference is provided for the statement: “This factor can be used in later calculations to adjust benefits that are calculated using only the internal events STCs.”

(Response) KHNP would like to clarify our approach during the conference call and then will provide appropriate responses as necessary.

iii) Page 3: Level 1, 2, and 3 PRAs are mentioned several times but no reference to a technical report is provided for the sources of these PRAs.

(Response) Level 1 and 2 PRAs are provided in the PRA Summary Report (APR1400-E-P-NR-13001-P, submitted on Nov. 30, 2013). Level 3 PRA for the ER is described in the Appendices of “Level I & II Sensitivity Analysis for SAMDA” (1-037-N419-301, proprietary), which is not included in DCD package. The report is available at the KHNP Washington D.C. Center for NRC review.

iv) Page 3 and 4: The consequence code used for the Level 3 PRA is not cited and a technical report for the quantification of the code’s input parameters and subsequence analysis report with input and output listings were not provided.

(Response) The consequence code (WinMACCS), the code’s input parameters and subsequence analysis are described in the Appendices of “Level I & II Sensitivity Analysis for SAMDA” (1-037-N419-301, proprietary), which is not included in DCD package. The report is available at the KHNP Washington D.C. Center for NRC review.

v) Page 16: Justification is not provided nor referenced for assessing individual SAMDA items that were screened because they would not be feasible to implement.

(Response) The brief justifications for screened out SAMDAs which would not be feasible to implement are provided in Table 4 of the ER. Additional justification can be provided upon NRC request.