KHNPDCDRAIsPEm Resource

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Sent:	Thursday, July 23, 2015 9:35 AM
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Subject:	APR1400 Design Certification Application RAI 108-7973 (15.00.03 - Design Basis Accidents Radiological Consequence Analyses for Advanced Light Water Reactors)
Attachments:	APR1400 DC RAI 108 RPAC 7973.pdf; image001.jpg

KHNP

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Please submit your RAI response to the NRC Document Control Desk.

Thank you,

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Hearing Identifier: KHNP_APR1400_DCD_RAI_Public
Email Number: 117

Mail Envelope Properties (fd872a7438134aba9b9f9417f2f26086)

Subject: APR1400 Design Certification Application RAI 108-7973 (15.00.03 - Design Basis Accidents Radiological Consequence Analyses for Advanced Light Water Reactors)
Sent Date: 7/23/2015 9:35:09 AM
Received Date: 7/23/2015 9:35:10 AM
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Files	**Size**	**Date & Time**
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image001.jpg	5040	

Options
Priority: Standard
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Sensitivity: Normal
Expiration Date:
Recipients Received:

REQUEST FOR ADDITIONAL INFORMATION 108-7973

Issue Date: 07/23/2015
Application Title: APR1400 Design Certification Review – 52-046
Operating Company: Korea Hydro & Nuclear Power Co. Ltd.
Docket No. 52-046
Review Section: 15.00.03 - Design Basis Accidents Radiological Consequence Analyses for Advanced Light Water Reactors
Application Section: Chapter 15 including 15A

QUESTIONS

15.00.03-1

10 CFR 52.47(a)(2)(iv) requires that an application for a design certification include a final safety analysis report that provides a description and safety assessment of the facility. The APR1400 design control document (DCD) addresses this requirement. The safety assessment analyses are done, in part, to show compliance with the radiological consequence evaluation factors in 52.47(a)(2)(iv)(A) and 52.47(a)(2)(iv)(B) for offsite doses, 10 CFR 50, Appendix A, general design criterion (GDC) 19 for control room radiological habitability, and the requirements related to the technical support center in 10 CFR 52.47(b)(8) and (b)(11) and Paragraph IV.E.8 of Appendix E to 10 CFR Part 50. The radiological consequences of design basis accidents are evaluated against these regulatory requirements and the dose acceptance criteria given in NUREG-0800, Standard Review Plan (SRP) 15.0.3.

DCD Chapter 15A describes the modeling of the core fission product inventory used as input to the safety assessment analyses in DCD Chapter 15. Provide the following additional information on the maximum core fission product inventory calculation discussed in DCD 15A.1.1 and Table 15A-1:

a. Fuel burnup used to calculate the core inventory is listed as 56.4 GWD/MTU, however DCD 4.2.1 and Table 4.3-1 indicate that the APR-1400 design is based on maximum fuel rod average burnup of 60 GWD/MTU. Explain this discrepancy.
b. What power history assumptions were made in the calculations (e.g., percent time at full power, average power per U mass (MW/MTU))?
c. Is the core inventory listed calculated at beginning of core life (BOL), end of core life (EOL) or some other time? Is the time the same for all nuclides, or were calculations done to maximize the inventory per nuclide?
d. How were the gadolinia-urania burnable absorber rods addressed in the core inventory calculations?

15.00.03-2

10 CFR 52.47(a)(2)(iv) requires that an application for a design certification include a final safety analysis report that provides a description and safety assessment of the facility. The safety assessment analyses are done, in part, to show compliance with the radiological consequence evaluation factors in 52.47(a)(2)(iv)(A) and 52.47(a)(2)(iv)(B) for offsite doses, 10 CFR 50, Appendix A, GDC 19 for control room radiological habitability, and the requirements related to the technical support center in 10 CFR 52.47(b)(8) and (b)(11) and Paragraph IV.E.8 of Appendix E to 10 CFR Part 50. The radiological consequences of design basis accidents are evaluated against these regulatory requirements and the dose acceptance criteria given in SRP 15.0.3.

REQUEST FOR ADDITIONAL INFORMATION 108-7973

DCD Chapter 15A provides a description of the methods used to estimate coolant activity concentrations for input to the DCD Chapter 15 safety assessments. Provide the following information on the primary coolant concentration calculations:

a. Why are the DCD Table 15A-3 iodine concentrations and Table 15A-8 noble gas concentrations for 1% fuel defect are not the same as those in Table 11.1-2?

b. DCD Tables 15A-4 through 15A-7 list information on the accident-specific iodine appearance rates and iodine spiking. Explain the following differences:

   i. Column 2 values for total coolant activity per isotope are not the same between the four accident-specific tables. Clarify how the total isotopic activity was calculated for each accident.

   ii. Column 4 values for the letdown purification removal rate are not the same between tables. Provide the basis for letdown purification removal rate values used for each accident.

c. DCD 15A.1.2.4 states that alkali metal activities in the primary coolant are ignored because they have a low partition coefficient from the liquid to steam phase and the dose contribution is negligible. Guidance on particulate radionuclide transport from the RCS through the secondary system is given in RG 1.183, Appendix E, position 5.5.4, which states that the retention in the steam generators is limited by moisture carryover from the steam generators. This moisture carryover is applied to the steam release from the steam generators to give the alkali release fraction for those DBAs that model the secondary system release pathway. DCD Table 5.4.2-1, “Steam Generator Design Parameters,” gives the maximum weight percent moisture carryover as 0.25%.

   i. Provide a justification for this difference from RG 1.183 guidance, including the statement that the dose contribution from alkali metals in the RCS (primary coolant) is negligible.

   ii. Alternatively, revise the analyses that include primary-to-secondary leakage through the steam generators (SGs) as a release pathway to include the transport of alkali metals.

d. In technical specification (TS) 3.4.12 the RCS primary-to-secondary leakage is limited to 0.39 L/min through any one SG. The bases for TS 3.4.12 state that the initial condition in the dose analyses assumes 0.39 L/min per SG primary-to-secondary leakage. In the DBA dose analyses, contrary to this, DCD Tables 15.1.5-12, 15.2.8-3, 15.3.3-3, 15.4.8-4, 15.6.2-4 and 15.6.3-5 list the primary-to-secondary leakage as 2.27 L/min total for two SGs. RG 1.183 guidance states that the primary-to-secondary leak rate in the steam generators should be assumed to be the leak rate limiting condition for operation specified in the technical specifications. What is the basis for this dose analysis assumption which greatly exceeds the technical specification limit?

15.00.03-3

10 CFR 52.47(a)(2)(iv) requires that an application for a design certification include a final safety analysis report that provides a description and safety assessment of the facility. The safety assessment analyses are done, in part, to show compliance with the radiological consequence evaluation factors in 52.47(a)(2)(iv)(A) and 52.47(a)(2)(iv)(B) for offsite doses, 10 CFR 50, Appendix A, GDC 19 for control room radiological habitability, and the requirements related to the technical support center in 10 CFR 52.47(b)(8) and (b)(11) and Paragraph IV.E.8 of Appendix E to 10 CFR Part 50. The radiological consequences of design basis accidents are evaluated against these regulatory requirements and the dose acceptance criteria given in SRP 15.0.3.

REQUEST FOR ADDITIONAL INFORMATION 108-7973

The initial reactor coolant system (RCS) liquid mass is not consistent in all DBA dose analyses as expected – explain the differences. The initial reactor coolant mass is listed as input to analyses in the following DCD tables, and the value is different in each: Table 11.1-1 (coolant concentration), Table 15.1.5-12 (main steam line break), Table 15.2.8-3 (feedwater line break), Table 15.3.3-3 (reactor coolant pump rotor seizure), Table 15.4.8-4 (control element assembly ejection), Table 15.6.2-4 (small line break), Table 15.6.3-5 (steam generator tube rupture), and Table 15.6.5-13 (LOCA).

15.00.03-4

10 CFR 52.47(a)(2)(iv) requires that an application for a design certification include a final safety analysis report that provides a description and safety assessment of the facility. The safety assessment analyses are done, in part, to show compliance with the radiological consequence evaluation factors in 52.47(a)(2)(iv)(A) and 52.47(a)(2)(iv)(B) for offsite doses, 10 CFR 50, Appendix A, GDC 19 for control room radiological habitability, and the requirements related to the technical support center in 10 CFR 52.47(b)(8) and (b)(11) and Paragraph IV.E.8 of Appendix E to 10 CFR Part 50. The radiological consequences of design basis accidents are evaluated against these regulatory requirements and the dose acceptance criteria given in SRP 15.0.3.

The initial secondary liquid mass in the steam generators is not consistent in all DBA dose analyses as expected – explain the differences. The initial secondary coolant liquid mass is given as analysis input in the following DCD tables, and the value is different in each: Table 11.1-5 (coolant concentration), Table 15.1.5-12 (main steam line break), Table 15.2.8-3 (feedwater line break), Table 15.3.3-3 (reactor coolant pump rotor seizure), Table 15.4.8-4 (control element assembly ejection), Table 15.6.2-4 (small line break), and Table 15.6.3-5 (steam generator tube rupture).

15.00.03-5

10 CFR 52.47(a)(2)(iv) requires that an application for a design certification include a final safety analysis report that provides a description and safety assessment of the facility. The safety assessment analyses are done, in part, to show compliance with the radiological consequence evaluation factors in 52.47(a)(2)(iv)(A) and 52.47(a)(2)(iv)(B) for offsite doses, 10 CFR 50, Appendix A, GDC 19 for control room radiological habitability, and the requirements related to the technical support center in 10 CFR 52.47(b)(8) and (b)(11) and Paragraph IV.E.8 of Appendix E to 10 CFR Part 50. The radiological consequences of design basis accidents are evaluated against these regulatory requirements and the dose acceptance criteria given in SRP 15.0.3.

DCD Section 15.1.5 provides a description of the design basis accident (DBA) main steam line break (MSLB) outside containment. DCD page 15.1-28 states that RCS fluid is released to the IRWST during the MSLB) outside containment. Provide additional information on this in-containment movement of RCS fluid:

a. From where is this release and how does it get to the IRWST?
b. Is this flow controlled?
c. When does the flow start and stop?

15.00.03-6

10 CFR 52.47(a)(2)(iv) requires that an application for a design certification include a final safety analysis report that provides a description and safety assessment of the facility. The safety assessment analyses are done, in part, to show compliance with the radiological consequence evaluation factors in 52.47(a)(2)(iv)(A) and 52.47(a)(2)(iv)(B) for offsite doses, 10 CFR 50, Appendix A, GDC 19 for control room radiological habitability, and the requirements related to the technical support center in 10 CFR 52.47(b)(8) and (b)(11) and Paragraph IV.E.8 of Appendix E to 10 CFR Part 50. The radiological consequences of design basis accidents are evaluated against these regulatory requirements and the dose acceptance criteria given in SRP 15.0.3.

Clarify the assumptions for MSLB releases through the affected SG, both break flow and steaming, and releases through the unaffected SG.

a. DCD Table 15.1.5-12 states that the release from the affected SG is terminated at 30 minutes, which is coincident with the operator initiating cooldown. However, the table also gives mass releases through 8 hours for the affected SG.
b. DCD Table 15.1.5-12 states that the release through the unaffected SG (both primary-to-secondary leakage and steaming) is terminated at 8 hours. However, the table only gives mass releases from the unaffected SG through 0.5 hours.

15.00.03-7

10 CFR 52.47(a)(2)(iv) requires that an application for a design certification include a final safety analysis report that provides a description and safety assessment of the facility. The safety assessment analyses are done, in part, to show compliance with the radiological consequence evaluation factors in 52.47(a)(2)(iv)(A) and 52.47(a)(2)(iv)(B) for offsite doses, 10 CFR 50, Appendix A, GDC 19 for control room radiological habitability, and the requirements related to the technical support center in 10 CFR 52.47(b)(8) and (b)(11) and Paragraph IV.E.8 of Appendix E to 10 CFR Part 50. The radiological consequences of design basis accidents are evaluated against these regulatory requirements and the dose acceptance criteria given in SRP 15.0.3.

DCD Table 15.1.5-12 gives the SG iodine partition coefficient as 100 for the MSLB dose analysis.

a. Is this factor applied to releases through both SGs? For all time periods?
b. Page 15.1-29 indicates that there is a dryout period for the affected SG. RG 1.183 guidance states that the releases should be to the environment without mitigation (no partitioning factor) during dryout. What is the time and duration of the dryout? How was this modeled in the dose calculation?

15.00.03-8

10 CFR 52.47(a)(2)(iv) requires that an application for a design certification include a final safety analysis report that provides a description and safety assessment of the facility. The safety assessment analyses are done, in part, to show compliance with the radiological consequence evaluation factors in 52.47(a)(2)(iv)(A) and 52.47(a)(2)(iv)(B) for offsite doses, 10 CFR 50, Appendix A, GDC 19 for control room radiological habitability, and the requirements related to the technical support center in 10 CFR 52.47(b)(8) and (b)(11) and Paragraph IV.E.8 of Appendix E to 10 CFR Part 50. The radiological

consequences of design basis accidents are evaluated against these regulatory requirements and the dose acceptance criteria given in SRP 15.0.3.

DCD Table 15.1.5-12 does not specify which onsite χ/Qs from DCD Tables 2.3-2 through 2.3-12 were used in the MSLB dose analyses. Clarify which set of onsite χ/Qs were used for each pair of release point and receptor (both CR HVAC intake and unfiltered inleakage) relevant to the MSLB analyses and document in the DCD.

15.00.03-9

10 CFR 52.47(a)(2)(iv) requires that an application for a design certification include a final safety analysis report that provides a description and safety assessment of the facility. The safety assessment analyses are done, in part, to show compliance with the radiological consequence evaluation factors in 52.47(a)(2)(iv)(A) and 52.47(a)(2)(iv)(B) for offsite doses, 10 CFR 50, Appendix A, GDC 19 for control room radiological habitability, and the requirements related to the technical support center in 10 CFR 52.47(b)(8) and (b)(11) and Paragraph IV.E.8 of Appendix E to 10 CFR Part 50. The radiological consequences of design basis accidents are evaluated against these regulatory requirements and the dose acceptance criteria given in SRP 15.0.3.

DCD Section 15.2.8 provides a discussion of the safety analysis for the DBA feedwater line break. Provide the following information regarding the feedwater line break (FWLB) dose analysis release to the IRWST:

a. Table 15.2.8-3 states that the duration of the release through the pressurizer pilot-operated safety relief valve (POSRV) or reactor coolant gas vent system (RCGVS) to the IRWST is 1 minute. At what time does this release start?
b. Is the release from the POSRV/RCGVS assumed to be mixed in the IRWST fluid volume? If so, what volume was assumed for the IRWST?
c. What release rate from the IRWST to the containment was assumed?

15.00.03-10

10 CFR 52.47(a)(2)(iv) requires that an application for a design certification include a final safety analysis report that provides a description and safety assessment of the facility. The safety assessment analyses are done, in part, to show compliance with the radiological consequence evaluation factors in 52.47(a)(2)(iv)(A) and 52.47(a)(2)(iv)(B) for offsite doses, 10 CFR 50, Appendix A, GDC 19 for control room radiological habitability, and the requirements related to the technical support center in 10 CFR 52.47(b)(8) and (b)(11) and Paragraph IV.E.8 of Appendix E to 10 CFR Part 50. The radiological consequences of design basis accidents are evaluated against these regulatory requirements and the dose acceptance criteria given in SRP 15.0.3.

Provide the following information regarding the feedwater line break (FWLB) dose analysis containment release pathway discussed in DCD 15.2.8 and Table 15.2.8-3:

a. Are the vapor releases to the containment from the feedwater line break and through the IRWST pathway assumed to be instantaneously mixed in the entire containment volume, a portion of the volume or not mixed in the containment?
b. Is the assumed containment leak rate the same as used for 0-24 hours for the LOCA (0.1% per day)?

15.00.03-11

10 CFR 52.47(a)(2)(iv) requires that an application for a design certification include a final safety analysis report that provides a description and safety assessment of the facility. The safety assessment analyses are done, in part, to show compliance with the radiological consequence evaluation factors in 52.47(a)(2)(iv)(A) and 52.47(a)(2)(iv)(B) for offsite doses, 10 CFR 50, Appendix A, GDC 19 for control room radiological habitability, and the requirements related to the technical support center in 10 CFR 52.47(b)(8) and (b)(11) and Paragraph IV.E.8 of Appendix E to 10 CFR Part 50. The radiological consequences of design basis accidents are evaluated against these regulatory requirements and the dose acceptance criteria given in SRP 15.0.3.

On DCD page 15.2-25, for the FWLB release via the affected SG, its states that one-half of the total primary-to-secondary leakage entering the affected SG is released to the environment through the main steam safety valves (MSSVs), with no mitigation or dilution. DCD Table 15.2.8-3 gives the release from the affected SG through the MSSV as 2,810 kg (6,200 lbm) for 20 seconds.

a. Does the first statement mean that the primary-to-secondary leakage rate directly to the environment through the MSSV for that 20 seconds is assumed to be 1.135 L/min (50% of the total 2.27 L/min rate for both SGs), or is it 0.5675 L/min (50% of the 1.135 L/min per SG)? (Note – See previous question about analysis assumptions on primary-to-secondary leakage through the SGs exceeding the TS limit.)
b. Does the 2,810 kg value include both the secondary fluid mass release and the primary-to-secondary leakage or only the secondary fluid?

15.00.03-12

10 CFR 52.47(a)(2)(iv) requires that an application for a design certification include a final safety analysis report that provides a description and safety assessment of the facility. The safety assessment analyses are done, in part, to show compliance with the radiological consequence evaluation factors in 52.47(a)(2)(iv)(A) and 52.47(a)(2)(iv)(B) for offsite doses, 10 CFR 50, Appendix A, GDC 19 for control room radiological habitability, and the requirements related to the technical support center in 10 CFR 52.47(b)(8) and (b)(11) and Paragraph IV.E.8 of Appendix E to 10 CFR Part 50. The radiological consequences of design basis accidents are evaluated against these regulatory requirements and the dose acceptance criteria given in SRP 15.0.3.

On DCD page 15.2-25, for the FWLB release via the affected SG, it states that during SG dryout, the iodine in 50% of the total primary-to-secondary leakage entering the affected SG is assumed to flash to vapor and be released to the containment through the feedwater line break without credit for holdup. RG 1.183, Appendix E guidance is that all of the primary-to-secondary leakage to that SG is assumed to flash to vapor during periods of dryout. Provide the basis for this difference from the guidance.

15.00.03-13

10 CFR 52.47(a)(2)(iv) requires that an application for a design certification include a final safety analysis report that provides a description and safety assessment of the facility. The safety assessment analyses are done, in part, to show compliance with the radiological consequence evaluation factors in 52.47(a)(2)(iv)(A) and 52.47(a)(2)(iv)(B) for offsite doses, 10 CFR 50, Appendix A, GDC 19 for control room radiological habitability, and the requirements related to the technical support center in 10 CFR 52.47(b)(8) and (b)(11) and Paragraph IV.E.8 of Appendix E to 10 CFR Part 50. The radiological consequences of design basis accidents are evaluated against these regulatory requirements and the dose acceptance criteria given in SRP 15.0.3.

DCD Table 15.2.8-3 does not specify which onsite χ/Qs from DCD Tables 2.3-2 through 2.3-12 were used in the FWLB dose analysis. Clarify which set of onsite χ/Qs were used for each pair of release point and receptor (both CR HVAC intake and unfiltered inleakage) relevant to the FWLB analysis and document in the DCD.

15.00.03-14

10 CFR 52.47(a)(2)(iv) requires that an application for a design certification include a final safety analysis report that provides a description and safety assessment of the facility. The safety assessment analyses are done, in part, to show compliance with the radiological consequence evaluation factors in 52.47(a)(2)(iv)(A) and 52.47(a)(2)(iv)(B) for offsite doses, 10 CFR 50, Appendix A, GDC 19 for control room radiological habitability, and the requirements related to the technical support center in 10 CFR 52.47(b)(8) and (b)(11) and Paragraph IV.E.8 of Appendix E to 10 CFR Part 50. The radiological consequences of design basis accidents are evaluated against these regulatory requirements and the dose acceptance criteria given in SRP 15.0.3.

With respect to the reactor coolant pump rotor seizure accident dose analysis, on DCD page 15.3-12, it states that during the first 30 minutes both SGs are expected to have the tubes uncovered. It also states that during this period, the primary-to-secondary leakage flashing fraction averages 15 percent. RG 1.183 guidance is that all of the primary-to-secondary leakage to that SG is assumed to flash to vapor during periods of dryout. Provide the basis for this difference from the guidance.

15.00.03-15

10 CFR 52.47(a)(2)(iv) requires that an application for a design certification include a final safety analysis report that provides a description and safety assessment of the facility. The safety assessment analyses are done, in part, to show compliance with the radiological consequence evaluation factors in 52.47(a)(2)(iv)(A) and 52.47(a)(2)(iv)(B) for offsite doses, 10 CFR 50, Appendix A, GDC 19 for control room radiological habitability, and the requirements related to the technical support center in 10 CFR 52.47(b)(8) and (b)(11) and Paragraph IV.E.8 of Appendix E to 10 CFR Part 50. The radiological consequences of design basis accidents are evaluated against these regulatory requirements and the dose acceptance criteria given in SRP 15.0.3.

DCD Table 15.3.3-3 does not specify which onsite χ/Qs from DCD Tables 2.3-2 through 2.3-12 were used in the reactor coolant pump rotor seizure accident dose analysis. Clarify which set of onsite χ/Qs were used for each pair of release point and receptor (both CR HVAC intake and unfiltered inleakage) relevant to the reactor coolant pump rotor seizure accident dose analysis and document in the DCD.

15.00.03-16

10 CFR 52.47(a)(2)(iv) requires that an application for a design certification include a final safety analysis report that provides a description and safety assessment of the facility. The safety assessment analyses are done, in part, to show compliance with the radiological consequence evaluation factors in 52.47(a)(2)(iv)(A) and 52.47(a)(2)(iv)(B) for offsite doses, 10 CFR 50, Appendix A, GDC 19 for control room radiological habitability, and the requirements related to the technical support center in 10 CFR 52.47(b)(8) and (b)(11) and Paragraph IV.E.8 of Appendix E to 10 CFR Part 50. The radiological consequences of design basis accidents are evaluated against these regulatory requirements and the dose acceptance criteria given in SRP 15.0.3.

With respect to the control element assembly (CEA) ejection accident dose analysis, on DCD page 15.4-33, it states that during the first 30 minutes both SGs are expected to have the tubes uncovered. It also states that during this period, the primary-to-secondary leakage flashing fraction averages 15 percent. RG 1.183 guidance is that all of the primary-to-secondary leakage to that SG is assumed to flash to vapor during periods of dryout. Provide the basis for this difference from the guidance.

15.00.03-17

10 CFR 52.47(a)(2)(iv) requires that an application for a design certification include a final safety analysis report that provides a description and safety assessment of the facility. The safety assessment analyses are done, in part, to show compliance with the radiological consequence evaluation factors in 52.47(a)(2)(iv)(A) and 52.47(a)(2)(iv)(B) for offsite doses, 10 CFR 50, Appendix A, GDC 19 for control room radiological habitability, and the requirements related to the technical support center in 10 CFR 52.47(b)(8) and (b)(11) and Paragraph IV.E.8 of Appendix E to 10 CFR Part 50. The radiological consequences of design basis accidents are evaluated against these regulatory requirements and the dose acceptance criteria given in SRP 15.0.3.

DCD Table 15.4.8-4 does not specify which onsite χ/Qs from DCD Tables DCD Tables 2.3-2 through 2.3-12 were used in the control element assembly (CEA) ejection accident dose analysis. Clarify which set of onsite χ/Qs were used for each pair of release point and receptor (both CR HVAC intake and unfiltered inleakage) relevant to the CEA ejection accident dose analysis and document in the DCD.

15.00.03-18

10 CFR 52.47(a)(2)(iv) requires that an application for a design certification include a final safety analysis report that provides a description and safety assessment of the facility. The safety assessment analyses are done, in part, to show compliance with the radiological consequence evaluation factors in 52.47(a)(2)(iv)(A) and 52.47(a)(2)(iv)(B) for offsite doses, 10 CFR 50, Appendix A, GDC 19 for control room radiological habitability, and the requirements related to the technical support center in 10 CFR 52.47(b)(8) and (b)(11) and Paragraph IV.E.8 of Appendix E to 10 CFR Part 50. The radiological consequences of design basis accidents are evaluated against these regulatory requirements and the dose acceptance criteria given in SRP 15.0.3.

In the DCD 15.6.2 analysis of the radiological consequences of failure of small lines carrying primary coolant outside containment, the letdown line break was analyzed as being conservative. The letdown line has three isolation valves in series inside containment; therefore the break was modeled as manually isolated at 30 minutes. Are there other small lines penetrating containment that carry primary coolant and are not able to isolate a break outside containment and also may result in larger integrated releases?

15.00.03-19

10 CFR 52.47(a)(2)(iv) requires that an application for a design certification include a final safety analysis report that provides a description and safety assessment of the facility. The safety assessment analyses are done, in part, to show compliance with the radiological consequence evaluation factors in 52.47(a)(2)(iv)(A) and 52.47(a)(2)(iv)(B) for offsite doses, 10 CFR 50, Appendix A, GDC 19 for control room radiological habitability, and the requirements related to the technical support center in 10 CFR 52.47(b)(8) and (b)(11) and Paragraph IV.E.8 of Appendix E to 10 CFR Part 50. The radiological consequences of design basis accidents are evaluated against these regulatory requirements and the dose acceptance criteria given in SRP 15.0.3.

DCD Table 15.6.2-4 does not specify which onsite χ/Qs from DCD Tables 2.3-2 through 2.3-12 were used in the small line break dose analysis. Clarify which set of onsite χ/Qs were used for each pair of release point and receptor (both CR HVAC intake and unfiltered inleakage) relevant to the small line break analysis and document in the DCD.

15.00.03-20

10 CFR 52.47(a)(2)(iv) requires that an application for a design certification include a final safety analysis report that provides a description and safety assessment of the facility. The safety assessment analyses are done, in part, to show compliance with the radiological consequence evaluation factors in 52.47(a)(2)(iv)(A) and 52.47(a)(2)(iv)(B) for offsite doses, 10 CFR 50, Appendix A, GDC 19 for control room radiological habitability, and the requirements related to the technical support center in 10 CFR 52.47(b)(8) and (b)(11) and Paragraph IV.E.8 of Appendix E to 10 CFR Part 50. The radiological consequences of design basis accidents are evaluated against these regulatory requirements and the dose acceptance criteria given in SRP 15.0.3.

DCD Section 15.6.3 provides a description of the steam generator tube rupture accident. Clarify the following apparent discrepancy with regard to the DCD Table 15A-5 iodine appearance rate calculation for the steam generator tube rupture (SGTR) dose analysis. Assuming that the RCS initial coolant isotopic activities are calculated by taking the Table 15A-3 3.7E+04 Bq/g DE I-131 isotopic activity concentrations in Bq/g and multiply them by the initial RCS mass of 290,680 kg given in Table 15.6.3-5 (ensuring unit agreement), the results do not match the values given in column 2 of Table 15A-5, which are higher. If instead the RCS initial mass of 2.92E+05 kg given in Table 11.1-1 is used, the values do not match either.

15.00.03-21

10 CFR 52.47(a)(2)(iv) requires that an application for a design certification include a final safety analysis report that provides a description and safety assessment of the facility. The safety assessment analyses are done, in part, to show compliance with the radiological consequence evaluation factors in 52.47(a)(2)(iv)(A) and 52.47(a)(2)(iv)(B) for offsite doses, 10 CFR 50, Appendix A, GDC 19 for control room radiological habitability, and the requirements related to the technical support center in 10 CFR 52.47(b)(8) and (b)(11) and Paragraph IV.E.8 of Appendix E to 10 CFR Part 50. The radiological consequences of design basis accidents are evaluated against these regulatory requirements and the dose acceptance criteria given in SRP 15.0.3.

With regard to the information in DCD 15.6.3 and Table 15.6.3-5, what is the alkali metal partition rate assumed in the steam generator tube rupture (SGTR) dose analysis? How does it compare to the steam generator moisture carryover?

15.00.03-22

10 CFR 52.47(a)(2)(iv) requires that an application for a design certification include a final safety analysis report that provides a description and safety assessment of the facility. The safety assessment analyses are done, in part, to show compliance with the radiological consequence evaluation factors in 52.47(a)(2)(iv)(A) and 52.47(a)(2)(iv)(B) for offsite doses, 10 CFR 50, Appendix A, GDC 19 for control room radiological habitability, and the requirements related to the technical support center in 10 CFR 52.47(b)(8) and (b)(11) and Paragraph IV.E.8 of Appendix E to 10 CFR Part 50. The radiological consequences of design basis accidents are evaluated against these regulatory requirements and the dose acceptance criteria given in SRP 15.0.3.

DCD Table 15.6.3-5 states that the onsite χ/Qs used in the SGTR analysis are given in Tables 2.3-6 and 2.3-7.

a. DCD Table 2.3-6 gives the onsite χ/Qs for releases from the south main steam safety valve (MSSV) room to the south auxiliary building intake.
    i. Verify that these are the values to use used for the control room unfiltered inleakage from a release from the MSSVs.
    ii. If so, which are the values to use for the CR HVAC intake for a release from the MSSVs, and did your analysis use those values?
b. DCD Table 2.3-7 gives the onsite χ/Qs for releases from the atmospheric dump valves (ADVs) to the closest CR HVAC intake.
    i. Which are the values to use for the control room unfiltered inleakage for a release from the ADV, and did your analysis use those values?
c. Clarify which set of onsite χ/Qs were used for each pair of release point and receptor (both CR HVAC intake and unfiltered inleakage) relevant to the SGTR dose analysis and document in the DCD.

15.00.03-23

10 CFR 52.47(a)(2)(iv) requires that an application for a design certification include a final safety analysis report that provides a description and safety assessment of the facility. The safety assessment analyses are done, in part, to show compliance with the radiological consequence evaluation factors in

52.47(a)(2)(iv)(A) and 52.47(a)(2)(iv)(B) for offsite doses, 10 CFR 50, Appendix A, GDC 19 for control room radiological habitability, and the requirements related to the technical support center in 10 CFR 52.47(b)(8) and (b)(11) and Paragraph IV.E.8 of Appendix E to 10 CFR Part 50. The radiological consequences of design basis accidents are evaluated against these regulatory requirements and the dose acceptance criteria given in SRP 15.0.3.

The safety analysis for the LOCA is described in DCD Section 15.6.5, and includes a discussion of the direct dose to personnel in the control room and techncial support center from DBAs. In DCD Table 15.6.5-14, the dose results for the main control room and technical support center (TSC) have a line item for direct dose from containment shine, but the value given is 0 mSv. Provide a detailed description of the direct dose analyses for containment shine from a LOCA, including inputs, assumptions, methods and results to provide a basis.

15.00.03-24

10 CFR 52.47(a)(2)(iv) requires that an application for a design certification include a final safety analysis report that provides a description and safety assessment of the facility. The safety assessment analyses are done, in part, to show compliance with the radiological consequence evaluation factors in 52.47(a)(2)(iv)(A) and 52.47(a)(2)(iv)(B) for offsite doses, 10 CFR 50, Appendix A, GDC 19 for control room radiological habitability, and the requirements related to the technical support center in 10 CFR 52.47(b)(8) and (b)(11) and Paragraph IV.E.8 of Appendix E to 10 CFR Part 50. The radiological consequences of design basis accidents are evaluated against these regulatory requirements and the dose acceptance criteria given in SRP 15.0.3.

For the LOCA dose analysis engineered safety features (ESF) leakage pathway discussed in DCD 15.6.5 and Table 15.6.5-13, did the model use mixing or holdup in the auxiliary building ESF areas?

15.00.03-25

10 CFR 52.47(a)(2)(iv) requires that an application for a design certification include a final safety analysis report that provides a description and safety assessment of the facility. The safety assessment analyses are done, in part, to show compliance with the radiological consequence evaluation factors in 52.47(a)(2)(iv)(A) and 52.47(a)(2)(iv)(B) for offsite doses, 10 CFR 50, Appendix A, GDC 19 for control room radiological habitability, and the requirements related to the technical support center in 10 CFR 52.47(b)(8) and (b)(11) and Paragraph IV.E.8 of Appendix E to 10 CFR Part 50. The radiological consequences of design basis accidents are evaluated against these regulatory requirements and the dose acceptance criteria given in SRP 15.0.3.

DCD Table 15.6.5-13 does not specify which onsite χ/Qs were used in the LOCA dose analysis. Clarify which set of onsite χ/Qs were used for each pair of release point and receptor (both CR HVAC intake and unfiltered inleakage), for each LOCA release pathway and document in the DCD.

15.00.03-26

10 CFR 52.47(a)(2)(iv) requires that an application for a design certification include a final safety analysis report that provides a description and safety assessment of the facility. The safety assessment analyses are done, in part, to show compliance with the radiological consequence evaluation factors in 52.47(a)(2)(iv)(A) and 52.47(a)(2)(iv)(B) for offsite doses, 10 CFR 50, Appendix A, GDC 19 for control room radiological habitability, and the requirements related to the technical support center in 10 CFR 52.47(b)(8) and (b)(11) and Paragraph IV.E.8 of Appendix E to 10 CFR Part 50. The radiological consequences of design basis accidents are evaluated against these regulatory requirements and the dose acceptance criteria given in SRP 15.0.3.

10 CFR 50.36 provides requirements for technical specifications, including criteria for establishment of technical specification limiting conditions for operation.

The APR-1400 does not propose a decay time technical specification. What is the basis for not providing such a technical specification? Without such a technical specification how is the 72 hour decay time assumed in the fuel handling accident (FHA) dose analysis, as discussed in DCD 15.7.4 and Table 15.7.4-1, ensured?

15.00.03-27

10 CFR 52.47(a)(2)(iv) requires that an application for a design certification include a final safety analysis report that provides a description and safety assessment of the facility. The safety assessment analyses are done, in part, to show compliance with the radiological consequence evaluation factors in 52.47(a)(2)(iv)(A) and 52.47(a)(2)(iv)(B) for offsite doses, 10 CFR 50, Appendix A, GDC 19 for control room radiological habitability, and the requirements related to the technical support center in 10 CFR 52.47(b)(8) and (b)(11) and Paragraph IV.E.8 of Appendix E to 10 CFR Part 50. The radiological consequences of design basis accidents are evaluated against these regulatory requirements and the dose acceptance criteria given in SRP 15.0.3.

10 CFR 50.36 provides requirements for technical specifications, including criteria related to technical specification limiting conditions for operation.

With respect to the FHA dose analysis described in DCD 15.7.4, TS 3.7.14 states that the spent fuel pool (SFP) water shall be maintained at least 7 m (23 ft) above the top of irradiated fuel assemblies in the storage racks. The FHA dose analysis assumes scrubbing of the fission product release using decontamination factors from RG 1.183, which states that the water depth above the damaged fuel should be 23 ft or greater.

a. The FHA dose analysis in DCD 15.7.4 states that the water level is 7 m (23 ft) from the top of the SFP racks to the SFP surface. Compare this dose analysis assumption to the water depth assured by TS 3.7.14. Is there additional water above the top of the fuel and below the top of the storage racks, therefore making the assumption used in the FHA dose analysis not bounded by the TS?
b. For the FHA, the fuel assembly that is dropped is assumed to be damaged with release of fission products. This dropped fuel assembly would not be seated in the storage racks, but instead may come to rest lying atop the storage racks in a horizontal position. If this is the case, is the depth of water above the damaged fuel assembly, as controlled by TS 3.7.14, less than 7 m (23 ft), thereby not meeting the conditions for use of the pool decontamination factors from RG 1.183?

15.00.03-28

10 CFR 52.47(a)(2)(iv) requires that an application for a design certification include a final safety analysis report that provides a description and safety assessment of the facility. The safety assessment analyses are done, in part, to show compliance with the radiological consequence evaluation factors in 52.47(a)(2)(iv)(A) and 52.47(a)(2)(iv)(B) for offsite doses, 10 CFR 50, Appendix A, GDC 19 for control room radiological habitability, and the requirements related to the technical support center in 10 CFR 52.47(b)(8) and (b)(11) and Paragraph IV.E.8 of Appendix E to 10 CFR Part 50. The radiological consequences of design basis accidents are evaluated against these regulatory requirements and the dose acceptance criteria given in SRP 15.0.3.

DCD Table 15.7.4-1 provides a list of limiting onsite χ/Qs for the unfiltered inleakage to the control room envelope from the FHA. These values do not seem to be taken from DCD Tables 2.3-2 through 2.3-12. How were these values developed? Is the release point the fuel handling area vent (point 13 on DCD Figure 2.3-1)?


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