RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

APR1400 Design Certification

Korea Electric Power Corporation / Korea Hydro & Nuclear Power Co., LTD

Docket No. 52-046

RAI No.:	**108-7973**
SRP Section:	**15.00.03 - Design Basis Accidents Radiological Consequence Analyses for Advanced Light Water Reactors**
Application Section:	**Chapter 15 including 15A**
Date of RAI Issue:	**07/23/2015**

Question No. 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?

Response

For MSLB, the RCS fluid is released to the IRWST in containment through the nozzle of the pilot operated safety and relief valve (POSRV) or through the reactor vessel head vent pipes in the reactor coolant gas vent system (RCGVS) to reduce the increased pressure of the RCS. This is anticipated to occur at 30 minutes after initiation of the event by an operator action. This release could continue for 8 hours until the cold shutdown cooling condition is established.

However, it is conservatively assumed that all the RCS fluid is released to the containment atmosphere, not to the IRWST, at the start of the event. Then, the radioactivity contained in the released RCS fluid is assumed to be instantaneously and homogeneously mixed in the free volume of the containment.

In the radiological consequence analysis, however, the release pathway for the RCS fluid from POSRV or RCGVS to the environment via the containment was not considered since its impact on the containment pressure increase, which is the driving force for containment leakage, is negligible as described in Section 15.1.5.5.1.

Impact on DCD

There is no impact on the DCD.

Impact on PRA

There is no impact on the PRA.

Impact on Technical Specifications

There is no impact on the Technical Specifications.

Impact on Technical/Topical/Environmental Reports

There is no impact on any Technical, Topical, or Environmental Report.

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Question No. 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?

Response

a. An iodine partition coefficient of 1.0 was applied to the affected SG liquid iodine steaming rates during the entire period of the event, where the affected SG is assumed to be empty. Therefore, the parameter ‘SG Liquid Iodine Partition Coefficient’ of 100 in DCD Table 15.1.5-12 will be updated to be 1.0.

For the unaffected SG, an iodine partition coefficient of 100 can be applied if the SG tubes are submerged by secondary liquid as stated in RG 1.183. However, for conservatism, an iodine partition coefficient of 1.0 was also applied to liquid iodine steaming rates for the unaffected SG during the period of 0 – 30 minutes. After 30 minutes, as indicated in Table 15.1.5-12, the release from the unaffected SG is not assumed since the plant is assumed to be cooled down by the affected SG to maximize the radiological consequences.

For clarify, the corresponding descriptions will be updated.

b. Even though the steam generator water level reaches the auxiliary feedwater actuation setpoint, thus leading to an initiation of auxiliary feedwater injection at 7.18 seconds, the secondary coolant in the affected SG is dried out at 287.28 seconds as indicated in Table 15.1.5-5. To maximize the offsite doses due to primary-to-secondary (P-T-S) leakage, however, it was assumed that the affected SG dryout condition occurred instantaneously following the MSLB event and continued for the entire period (0 - 8 hours) of the event. The SG dryout causes the P-T-S leakage to be released from the RCS into the environment without mitigation (i.e., no partitioning factor). Under the dryout conditions, all the P-T-S leakages in liquid are assumed to be instantaneously evaporated to steam.

Impact on DCD

DCD Section 15.1.5.5.1 and Table 15.1.5-12 will be updated as indicated in the Attachment.

Impact on PRA

There is no impact on the PRA.

Impact on Technical Specifications

There is no impact on the Technical Specifications.

Impact on Technical/Topical/Environmental Reports

There is no impact on any Technical, Topical, or Environmental Report.

Even though the affected SG is filled with the auxiliary feedwater to cool down the RCS, the partitioning of iodine in the secondary liquid is not credited in the analysis.

Release via the Affected Steam Generator

The post-MSLB thermal hydraulic condition in the affected SG is such that the primary-to-secondary (P-T-S) leakage is assumed to flash immediately to vapor in the affected SG, and the radioiodine and noble gases carried from the RCS to the affected SG are directly released to the environment without mitigation concurrently with the initiation of the MSLB accident. During the SG dryout, the radioiodine in the affected SG liquid is assumed to be released to environment with steaming rates. ~~The affected SG is assumed to be filled with the feedwater to cool down the RCS, and an iodine partition coefficient between the secondary liquid in the SG and the steam generated is used for the secondary liquid iodine steaming rates.~~

Release via the Unaffected Steam Generator

In the cases of unaffected SG, in which tubes are fully submerged by the secondary liquid, the P-T-S leakage is assumed to mix with the secondary water without flashing. The radioactivity in the bulk water ~~is~~ can be assumed to become vapor at a rate that is the function of the steaming rate and the partition coefficient. However, the partitioning of iodine in the unaffected SG is not credited for the period of 0 - 30 minutes.

Release via the Condenser

Prior to the LOOP, the contaminated secondary steam in the unaffected SG is released to the condenser. However, the steam release to the condenser is not considered in the post-MSLB activity release to the environment due to the tortuous path to the condenser via the turbines and moisture separators, and the condenser hold-up time.

Figure 15A-1 in Appendix 15A shows the leakage or transport of the activity released to the environment, MCR, and TSC during the MSLB accident.

15.1.5.5.2 Input Parameters and Initial Conditions

The design basis MSLB accident is analyzed using a conservative set of assumptions based on NRC RG 1.183, Appendix E, and the APR1400 design inputs. Input parameter values used for the MSLB radiological consequence evaluation are presented in Table 15.1.5-12.

Table 15.1.5-12 (2 of 3)

Parameter	Value
Secondary System Activity Transport Model	
Primary-to-secondary Leakage Rate through SGs	2.27 L/min (0.6 gpm) for two SGs
Integrated P-T-S Leakage 0 ~ 2 hr 2 ~ 8 hr	 272 kg (601 lbm) 818 kg (1,803 lbm)
Total Mass Release from Affected SG For SLBFPDLOOP 0 ~ 0.5 hr 0.5 ~ 2 hr 2 ~ 8 hr For SLBZPLOOPD 0 ~ 0.5 hr 0.5 ~ 2 hr 2 ~ 8 hr	 196,862 kg (434,000 lbm) 241,315 kg (532,000 lbm) 657,720 kg (1,450,000 lbm) 158,760 kg (350,000 lbm) 226,346 kg (499,000 lbm) 639,576 kg (1,410,000 lbm)
Total Mass Release from Unaffected SG For SLBFPDLOOP 0 ~ 0.5 hr 0.5 ~ 8 hr For SLBZPLOOPD 0 ~ 0.5 hr 0.5 ~ 8 hr	 40,824 kg (90,000 lbm) 0.0 kg (0.0 lbm) [illegible] (86,000 lbm) 0.0 kg (0.0 lbm)
Termination of Release from ~~Affected~~ Unaffected SG	30 min
~~Unaffected~~ Affected SG P-T-S Leak Duration, and Termination of Release from ~~Unaffected~~ Affected SG	8 hr
SG Liquid Iodine Partition Coefficient	~~100~~ 1.0 (for both SGs)
Letdown System F[illegible]	18,100 kg/hr (39,842 lbm/hr)
RCS Fluid Released to IRWST	5,443 kg (12,000 lbm) For SLBFPDLOOP 2,948 kg (6,500 lbm) For SLBZPLOOPD

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Question No. 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 thedose 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.

Response

Table 1 shows the onsite χ/Qs applied to each of the radiological consequence analysis. For the MSLB MCR dose calculation, the onsite χ/Qs for “South Main Steam Valve Room Direct Release to MCR South Intake” provided in DCD Table 2.3-4 and “South Main Steam Valve Room Release to Auxiliary Building South Intake” provided in DCD Table 2.3-6 are used for the MCR HVAC intake and unfiltered inleakage pathways, respectively. Since the current DCD Table 2.3-12 is missing the onsite χ/Q values “FHA vent to Auxiliary Building South intake”, these values will be added in DCD Table 2.3-12.

Table 1 Onsite χ/Qs Used for MCR Habitability Analyses

Accident		Applied Onsite χ/Qs From Release Points To	
		MCR Intake	Auxiliary Building Intake (Unfiltered Inleakage)
MSLB		South MSV Room to South intake (Table 2.3-4)	South MSV Room to South intake (Table 2.3-6)
RCPLR		South ADV to South intake (Table 2.3-7)	South MSV Room to South intake (Table 2.3-6)
CEA	CTMT leak	CB Surface to North intake (Table 2.3-2)	CB Surface to North intake (Table 2.3-3)
	SG Release	South ADV to South intake (Table 2.3-7)	South MSV Room to South intake (Table 2.3-6)
SGTR		South ADV to South intake (Table 2.3-7)	South MSV Room to South intake (Table 2.3-6)
LOCA	CTMT leak	CB Surface to North intake (Table 2.3-2)	CB Surface to North intake (Table 2.3-3)
	ESF	AB South Vent to South intake (Table 2.3-10)	AB South Vent to South intake (Table 2.3-11)
	CLVPS	South MSV Room to South intake (Table 2.3-4)	South MSV Room to South intake (Table 2.3-6)
	External cloud	CB Surface to MCR Roof CL (Table 2.3-2)	-
LDLB	ABCAEES	AB South Vent to South intake (Table 2.3-10)	AB South Vent to South intake (Table 2.3-11)
	ADV	South ADV to South intake (Table 2.3-7)	South MSV Room to South intake (Table 2.3-6)
FLB	CTMT leak	CB Surface to North intake (Table 2.3-2)	CB Surface to North intake (Table 2.3-3)
	SG Release	South ADV to South intake (Table 2.3-7)	South MSV Room to South intake (Table 2.3-6)
FHA		FHA vent to South intake (Table 2.3-12)	FHA vent to South intake (Table 2.3-12)_refer to Attachment

Please refer to the response to RAI 174-8211, Question 02.03.04-5 for the justification of selection of source-to-receptor locations.

Impact on DCD

DCD Table 2.3-12 will be revised as indicated in the Attachment.

Impact on PRA

There is no impact on the PRA.

Impact on Technical Specifications

There is no impact on the Technical Specifications.

Impact on Technical/Topical/Environmental Reports

There is no impact on any Technical, Topical, or Environmental Report.

and Auxiliary Building South Intake

Table 2.3-12

Onsite χ/Q for Fuel Handling Area Exhaust Release
to MCR North and South Intakes

Time Interval (hr)	Onsite χ/Q (s/m^3)	
	Fuel Handling Area To	
	MCR North Intake	MCR South Intake
0–2	1.52E-04	2.59E-04
2–8	1.31E-04	2.04E-04
8–24	6.02E-05	8.98E-05
24–96	4.01E-05	5.93E-05
96–720	3.19E-05	4.58E-05

AB South Intake
1.04E-03
8.18E-04
3.59E-04
2.37E-04
1.83E-04

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Question No. 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

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?

Response

The RCS fluid is released to the IRWST in the containment over two (2) time periods at 28.37 and 455.80 seconds after initiation of the event, respectively. However, it is conservatively assumed that 3,651 lbm of RCS fluid is directly released for a duration of 1 minute to the containment, not released to the IRWST, simultaneously with the event. Therefore, the parameters for IRWST volume and the release rate from the IRWST to the containment were not used in this analysis.

Impact on DCD

There is no impact on the DCD.

Impact on PRA

There is no impact on the PRA.

Impact on Technical Specifications

There is no impact on the Technical Specifications.

Impact on Technical/Topical/Environmental Reports

There is no impact on any Technical, Topical, or Environmental Report.

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Question No. 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)?

Response

The radioactivity released from the secondary liquid in the affected SG and the RCS fluid is conservatively assumed to be released as vapor and mixed in the free air volume of the primary containment instantaneously and homogeneously. As with the LOCA event, the containment is assumed to leak at its maximum technical specification leak rate of 0.1 percent per day (0.1 %/day) during the first 24 hours and at half of this leak rate (i.e., 0.05 %/day) for the remaining duration of the accident. The gravitational deposition of aerosols inside containment and containment spray operation were conservatively not credited in the analysis.

Impact on DCD

There is no impact on the DCD.

Impact on PRA

There is no impact on the PRA.

Impact on Technical Specifications

There is no impact on the Technical Specifications.

Impact on Technical/Topical/Environmental Reports

There is no impact on any Technical, Topical, or Environmental Report.

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Question No. 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?

Response

a. The description "one-half of the total P-T-S leakage entering the affected SG" in Section 15.2.8.5.1 means that the P-T-S leakage of 0.3 gpm (1.135 L/min) for the one affected SG is assumed since the total P-T-S leakage for the two SGs is 0.6 gpm (2.27 L/min). This DCD section will be updated for clarity.

b. The mass release value of 2,810 kg (6,200 lbm) from the affected SG through the MSSV is only for the secondary fluid. The impact of the P-T-S leakage is considered separately.

Impact on DCD

DCD Section 15.2.8.5.1 will be updated as indicated in the Attachment.

Impact on PRA

There is no impact on the PRA.

Impact on Technical Specifications

There is no impact on the Technical Specifications.

Impact on Technical/Topical/Environmental Reports

There is no impact on any Technical, Topical, or Environmental Report.

Normally, the RCP seal is cooled by (1) seal injection water from chemical and volume control system (CVCS) and (2) the component cooling water system through a high-pressure seal cooler. The evaluations of the reactor coolant pumps presented in Subsections 5.4.1.2 and 5.4.1.3 show that the integrity of RCPs is maintained with a loss of CCW for at least 30 minutes.

15.2.8.5 Radiological Consequences

The radiological consequences are performed to determine EAB, LPZ, MCR, and TSC doses due to main feedwater line break (FLB) accident using the AST methodology, TEDE dose criteria, guidance in SRP 15.0.3, and the plant-specific bounding design information applicable to the APR1400.

15.2.8.5.1 Evaluation Model

The following transport models of radioactive materials are applied to evaluate radiological consequences due to an FLB accident.

Release via the Containment

The secondary coolant is released from the affected SG into the containment building through the feedwater line break and from there is released directly to the environment as a result of the containment leakage. The RCS fluid is released to the IRWST through the pilot-operated safety and relief valve (POSRV) or reactor coolant gas vent system (RCGVS) and from there, released directly to the environment due to the containment leakage. The flashing fraction for radioiodine is conservatively assumed to be 1.

Release via Affected Steam Generator

one-half of the total P-T-S leakage (0.6 gpm) is assumed to enter the affected SG and subsequently be released to the environment through the MSSVs.

At the beginning of the FLB event, ~~one-half of the total P-T-S leakage entering the affected SG~~ is released to the environment through the MSSVs. When the MSIV is closed due to low SG pressure after closure of MSSVs, P-T-S leakage is released to the containment through the broken feedwater line of the affected SG. It is conservatively assumed that the P-T-S leakage is released with no mitigation or dilution. During the period of SG dryout due to the FLB event, the radioiodine in one-half of the total P-T-S leakage entering

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Question No. 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.

Response

As addressed in the Response to Question 15.00.03-11, the DCD sentence means that one-half (i.e. 0.3 gpm for one affected SG) of the total P-T-S leakage (i.e. 0.6 gpm for two SG) enters the affected SG. All the P-T-S leakage entering the affected SG is assumed to flash to vapor during periods of dry-out. This is consistent with RG 1.183 Appendix E guidance.

Impact on DCD

There is no impact on the DCD.

Impact on PRA

There is no impact on the PRA.

Impact on Technical Specifications

There is no impact on the Technical Specifications.

Impact on Technical/Topical/Environmental Reports

There is no impact on any Technical, Topical, or Environmental Report.

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Question No. 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.

Response

Table 1 in the response to Question No. 15.00.03-8 provides the onsite χ/Q values used for each of the radiological consequence analysis. The FWLB dose calculation is performed considering two release points: (1) Containment building leakage and (2) Secondary system release. Therefore, two onsite χ/Q sets are used as follows:

(1) Containment Building (CB) Leakage Path

- CB Leakage To MCR North Intake (DCD Table 2.3-2)
- CB Leakage To Auxiliary Building North Intake (DCD Table 2.3-3)

(2) Secondary System Release Path

- South Atmospheric Dump Valve Releases to MCR South Intake (DCD Table 2.3-7)
- South Main Steam Valve Room Release to Auxiliary Building South Intake (DCD Table 2.3-6)

Impact on DCD

There is no impact on the DCD.

Impact on PRA

There is no impact on the PRA.

Impact on Technical Specifications

There is no impact on the Technical Specifications.

Impact on Technical/Topical/Environmental Reports

There is no impact on any Technical, Topical, or Environmental Report.

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APR1400 Design Certification

Korea Electric Power Corporation / Korea Hydro & Nuclear Power Co., LTD

Docket No. 52-046

RAI No.:	**108-7973**
SRP Section:	**15.00.03 - Design Basis Accidents Radiological Consequence Analyses for Advanced Light Water Reactors**
Application Section:	**Chapter 15 including 15A**
Date of RAI Issue:	**07/23/2015**

Question No. 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.

Response

Table 1 in the response to Question No. 15.00.03-8 provides the onsite χ/Q values used for each of the radiological consequence analysis. For the reactor coolant pump rotor seizure accident dose calculation, the onsite χ/Qs for “South Atmospheric Dump Valve Releases to MCR South Intake” in DCD Table 2.3-7 and “South Main Steam Valve Room Release to Auxiliary Building South Intake” in DCD Table 2.3-6 are used for the MCR HVAC intake path and the unfiltered inleakage path, respectively.

Impact on DCD

There is no impact on the DCD.

Impact on PRA

There is no impact on the PRA.

Impact on Technical Specifications

There is no impact on the Technical Specifications.

Impact on Technical/Topical/Environmental Reports

There is no impact on any Technical, Topical, or Environmental Report.

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Question No. 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.

Response

Table 1 in the response to Question No. 15.00.03-8 provides the onsite χ/Q values for each of the radiological consequence analysis. Since the CEA dose calculation is performed for two independent cases, i.e., (1) Containment leakage release and 2) Secondary system release, four onsite χ/Q sets are used as follows:

(1) Containment Leakage Release Case

- Reactor Containment Building Release to MCR North Intake (DCD Table 2.3-2)

- Reactor Containment Building Release to Auxiliary Building North Intake (DCD Table 2.3-3)

(2) Secondary System Release Case

- South Atmospheric Dump Valve Releases to MCR South Intake (DCD Table 2.3-7)

- South Main Steam Valve Room Release to Auxiliary Building South Intake (DCD Table 2.3-6)

Impact on DCD

There is no impact on the DCD.

Impact on PRA

There is no impact on the PRA.

Impact on Technical Specifications

There is no impact on the Technical Specifications.

Impact on Technical/Topical/Environmental Reports

There is no impact on any Technical, Topical, or Environmental Report.

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Question No. 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.

Response

Table 1 in the response to Question No. 15.00.03-8 provides the onsite χ/Q values used for each of the radiological consequence analysis. The small line break dose calculation is performed considering two release paths: (1) Auxiliary Building Exhaust and (2) Secondary System Release. Therefore, two onsite χ/Q sets are used as follow:

(1) Auxiliary Building Exhaust Release

- Auxiliary Building South Exhaust Release to MCR South Intake (DCD Table 2.3-10)
- South Atmospheric Dump Valve Release to MCR South Intake (DCD Table 2.3-7)

(2) Secondary System Release

- Auxiliary Building South Exhaust Release to Auxiliary Building South Intake (DCD Table 2.3-11)

- South Main Steam Valve Room Release to Auxiliary Building South Intake (DCD Table 2.3-6)

Impact on DCD

There is no impact on the DCD.

Impact on PRA

There is no impact on the PRA.

Impact on Technical Specifications

There is no impact on the Technical Specifications.

Impact on Technical/Topical/Environmental Reports

There is no impact on any Technical, Topical, or Environmental Report.

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Question No. 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.

Response

The iodine appearance rate for SGTR dose analysis was calculated by taking the DE I-131 activity concentrations given in Table 15A-3 multiplying by the initial RCS mass of 290,680 kg, which is given in Table 15.6.3-5. However, since the values in Columns 2 and 4 have some typographical errors, they will be updated. This will not affect the radiological consequence for the SGTR event.

Impact on DCD

DCD Table 15A-5 will be updated as indicated in the Attachment.

Impact on PRA

There is no impact on the PRA.

Impact on Technical Specifications

There is no impact on the Technical Specifications.

Impact on Technical/Topical/Environmental Reports

There is no impact on any Technical, Topical, or Environmental Report.

Table 15A-5

Iodine Appearance Rates for Event-generated Iodine Spike
(Steam Generator Tube Rupture)

Nuclides	3.7×10^4 Bq/g (1.0 μCi/g) DE I-131 Activity (Bq)	Decay Constant (sec^{-1})	Letdown Purification Removal Rate (sec^{-1})	335 Times of Iodine Appearance Rate (Bq/sec)
I-131	~~8.80×10^{12}~~	9.98×10^{-7}	1.90×10^{-5}	~~5.91×10^{10}~~
I-132	~~2.36×10^{12}~~	8.37×10^{-5}	1.90×10^{-5}	~~8.14×10^{10}~~
I-133	~~1.25×10^{13}~~	9.26×10^{-6}	1.90×10^{-5}	~~1.18×10^{11}~~
I-134	~~1.44×10^{12}~~	2.20×10^{-4}	1.90×10^{-5}	~~1.15×10^{11}~~
I-135	~~7.10×10^{12}~~	2.91×10^{-5}	1.90×10^{-5}	~~1.15×10^{11}~~

8.53×10^{12}
2.29×10^{12}
1.21×10^{13}
1.40×10^{12}
6.88×10^{12}

5.72×10^{10}
7.88×10^{10}
1.15×10^{11}
1.12×10^{11}
1.11×10^{11}

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Question No. 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 technical 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.

Response

Since the MCR and TSC dose contribution from the containment shine is negligible compared to other paths shown in Table 15.6.5-14, the TEDE value for this path is assumed to be zero.

The gamma activity above the operating floor (EL 156'-0") in containment is the only portion of the source which can contribute to the direct shine dose to the MCR after the LOCA. Sources in the lower locations are strongly shielded by the primary concrete structure (7 feet thick) surrounding the reactor, the secondary shield wall (4 feet thick), other concrete structures in containment, and the containment structure (4 feet thick).The elevation of the MCR floor is EL 157'-9" and there are two 3-feet thick concrete walls located between the MCR floor and the upper portion of the containment building. Therefore, a total of 10 feet (= 2 × 3 feet + 4 feet) of concrete shielding exists between the MCR and the containment. These shield walls provide ample shielding for totally protecting the MCR operator from the post-LOCA containment shine dose.

Based on a simple shielding calculation which assumes a 1 MeV gamma source with 10 feet of concrete shielding, an attenuation factor of 8.3E-17 is calculated. Therefore, it is not necessary to calculate the direct shine dose from the post-LOCA source in the containment.

Impact on DCD

There is no impact on the DCD.

Impact on PRA

There is no impact on the PRA.

Impact on Technical Specifications

There is no impact on the Technical Specifications.

Impact on Technical/Topical/Environmental Reports

There is no impact on any Technical, Topical, or Environmental Report.

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Question No. 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?

Response

In the analysis for the ESF leakage release path, any reduction in release activity by mixing or holdup within the Auxiliary Building was not credited for conservatism. The DCD will be revised to clarify the use of this assumption.

Impact on DCD

DCD Section 15.6.5.5.1.2 will be updated as indicated in the Attachment.

Impact on PRA

There is no impact on the PRA.

Impact on Technical Specifications

There is no impact on the Technical Specifications.

Impact on Technical/Topical/Environmental Reports

There is no impact on any Technical, Topical, or Environmental Report.

IRWST water pH remains at greater than 7.0 for duration of the accident including the effect of acids and bases created during the LOCA event and the radiolysis products. Consequently, the re-evolution of dissolved iodine from the IRWST is not credible and is therefore not considered in the analysis.

15.6.5.5.1.2 Engineered Safety Feature (ESF) System Leakage

The ESF systems that recirculate IRWST water outside containment are assumed to leak during their intended operation. This release source includes leakage through valve packing glands, pump shaft seals, flanged connections, and other similar components. The radiological consequences from the postulated ESF leakage are analyzed and combined with consequences postulated for other fission product release paths to determine the total radiological consequences from the LOCA.

Post-LOCA Sump Water Iodine Source Term

NRC RG 1.183 requires that, with the exception of noble gases, all of the fission products released from the fuel to the containment are assumed to instantaneously and homogeneously mix in the IRWST water. Consistent with this guidance, a total of 40 percent of the core iodine released during the gap and early in-vessel phases is assumed to mix in the IRWST water.

ESF Leakage Release Path

The ESF pumps including the containment spray (CS), safety injection (SI), and component cooling water (CCW) pumps are located in the auxiliary building (AB). The ESF leakage is assumed to be retained on the floor of the equipment compartments in the AB and the iodine in the ESF leakage flashes and becomes airborne in the AB and the iodine is released to the environment through the AB ventilation exhaust system.

Reduction in release activity by mixing or holdup within the AB is not credited for conservatism

Flashing of Iodine from ESF Leakage

NRC RG 1.183 requires that if the temperature of the ESF leakage exceeds 100 °C (212 °F), the fraction of total iodine in the liquid that becomes airborne is assumed to be equal to the fraction of the leakage that flashes to vapor. This flash fraction (FF) is determined using a

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Question No. 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 thedose 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.

Response

Table 1 in the response to Question No. 15.00.03-8 provides the onsite χ/Q values for each of the radiological consequence analysis. As shown in the table, four sets of onsite χ/Qs are used for each of the different release paths for the MCR dose analysis due to LOCA.

Impact on DCD

There is no impact on the DCD.

Impact on PRA

There is no impact on the PRA.

Impact on Technical Specifications

There is no impact on the Technical Specifications.

Impact on Technical/Topical/Environmental Reports

There is no impact on any Technical, Topical, or Environmental Report.

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Question No. 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 APR1400 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?

Response

The APR1400 shielding design is based on a minimum spent fuel decay time of 100 hours as indicated in DCD Section 12.3.2.3 (page 12.3-28). In the safety analysis for fuel handling accident, however, a shorter decay time of 72 hours is assumed for conservatism based on previous TS 3.9.3 (Containment penetrations) and TS 3.9.6 (Refueling Operations).

However, as indicated in the response of RAI-8152 Question 12.02-14, the bases for TS 3.9.3 (Closure of containment penetrations) and TS 3.9.6 (Maintaining the minimum water level of 23 feet) was updated such that the fuel handling accident analysis is based on 100 hours of minimum decay time. However, the decay time used in the radiological consequence analysis will not be changed, since the source term for 72 hours is more conservative than that for 100 hours.

Impact on DCD

There is no impact on the DCD.

Impact on PRA

There is no impact on the PRA.

Impact on Technical Specifications

There is no impact on the Technical Specifications.

Impact on Technical/Topical/Environmental Reports

There is no impact on any Technical, Topical, or Environmental Report.

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Question No. 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?

Response

a. Since the description "from the top of the SFP racks to the SFP surface" in DCD Section 15.7.4.1 could lead to misunderstanding, it will be updated to be consistent with the TS 3.7.14.

b. If a dropped fuel assembly lies in a horizontal position on top of a spent fuel rack in the spent fuel pool, the depth of water above the damaged fuel assembly would be less than 23 ft due to the width of the fuel bundle, thereby not meeting the conditions for use of the pool decontamination factors from RG 1.183, Appendix B, Section 2. However, Technical Specification B 3.7.14 in APR1400 DCD states:

> "In the case of a single bundle which is dropped and lying horizontally on top of the spent fuel racks, there could be less than 23 ft above the top of the fuel bundle and the surface by the width of the bundle. To offset this small non-conservatism, the analysis assumes that all fuel rods fail."

Although an overall effective decontamination factor of 200 for a fuel handling accident over the spent fuel pool is reduced due to a water depth less than 23 ft, this effect is more than offset by the reduction in the number of damaged fuel rods when considering material properties and pool conditions. Therefore, the decontamination factor of 200 specified in RG 1.183, Appendix B, Section 2 could be applied to the case of a dropped fuel assembly falling in the horizontal position.

Impact on DCD

DCD Section 15.7.4.1 will be updated as indicated in the Attachment.

Impact on PRA

There is no impact on the PRA.

Impact on Technical Specifications

There is no impact on the Technical Specifications.

Impact on Technical/Topical/Environmental Reports

There is no impact on any Technical, Topical, or Environmental Report.

signal of high airborne radiation. The containment purge system is also designed to close the isolation valve of the low volume exhaust system with shorter time than the transit time of radioactive materials through the inner damper of low volume exhaust system. These requirements are applicable when irradiated fuel is moved in the containment (i.e., during a refueling outage) to confine the post-FHA release inside the containment and eliminate any potential activity release to the environment. Even if LOOP is assumed in the FHA analysis, the radioactive materials do not escape to the environment because the isolation valves of the purge system is designed to be closed when the narmal power is lost. Therefore, it is not required to analyze the radiological consequence of FHA in the containment.

FHA Outside Containment

The spent fuel pool (SFP) is located in the fuel handling area inside the auxiliary building. After the FHA in the SFP, the fission products released from the breached fuel assembly are scrubbed in the SFP water with a depth of 7 m (23 ft) from the top of ~~the SFP racks~~ irradiated fuel assemblies seated in the storage racks to the SFP surface. Escaped radioactivity is detected by the fuel handling area radiation monitors so that the fuel handling area emergency ventilation actuation signal (FHAEVAS) is actuated. The post-FHA activity from the SFP is then drawn by the safety-grade fuel handling area emergency ventilation system equipped with HEPA and charcoal prior to being released to the environment. The release from the FHA in the SFP is terminated when all the radioactivities released from the breached fuel assembly are discharged to the environment with the flow capacity of the emergency fuel handling area ventilation system.

15.7.4.2 Input Parameters and Initial Conditions

The fractions of the core inventory assumed to be in the gap for the various radionuclides are given in NRC RG 1.183. The release fractions are used in conjunction with the core fission product inventory with the maximum core radial peaking factor of 1.80.

It is assumed that all gap activity in the damaged rods is instantaneously released to the pool water. The gap radionuclides included are xenons, kryptons, and iodines. It is further assumed that the irradiated fuel is not removed from the reactor until the unit has been shut down for at least 72 hours.